WNP-2 SINGLE-LOOP OPERATION ANALYSIS

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Prepared for

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

WNP-2 NUCLEAR POWER STATION

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Attachment 1

APPENDIX 6.A

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6.A RECIRCULATION SYSTEM SINGLE-LOOP OPERATION

6.A.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 Section 6.2 and 6.3 and Chapter 15.0, were reviewed for the single-loop case with only one pump in operation.

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 flow dependent MCPR limit (K_f) factors 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. The least stable power/flow condition, achieved by tripping both recirculation pumps, is not affected by one-pump operation.

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 0.84.

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.

6.A.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 6.A.7-1. 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 6.A.2.2. This revision resulted in a single-loop operation process computer 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 incremental increase in the required MCPR fuel cladding integrity safety limit.

6.A.2.1 Core Flow Uncertainty

6.A.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 40%). 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.

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

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

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 the 100% flow control line and calculating the correct value of C based on the core support plate ΔP and the loop flow indicator readings.

6.A.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 6.A.7-1. 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 6.A.2-1):

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*The analytical expected value of the "C" coefficient for WNP-2 is ~ 0.89.

$$W_C = W_A - W_I$$

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 $\sigma_{W_{sys}}$, $\sigma_{W_{A_{rand}}}$, $\sigma_{W_{I_{rand}}}$ 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 above 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 the 6% core 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.

6.A.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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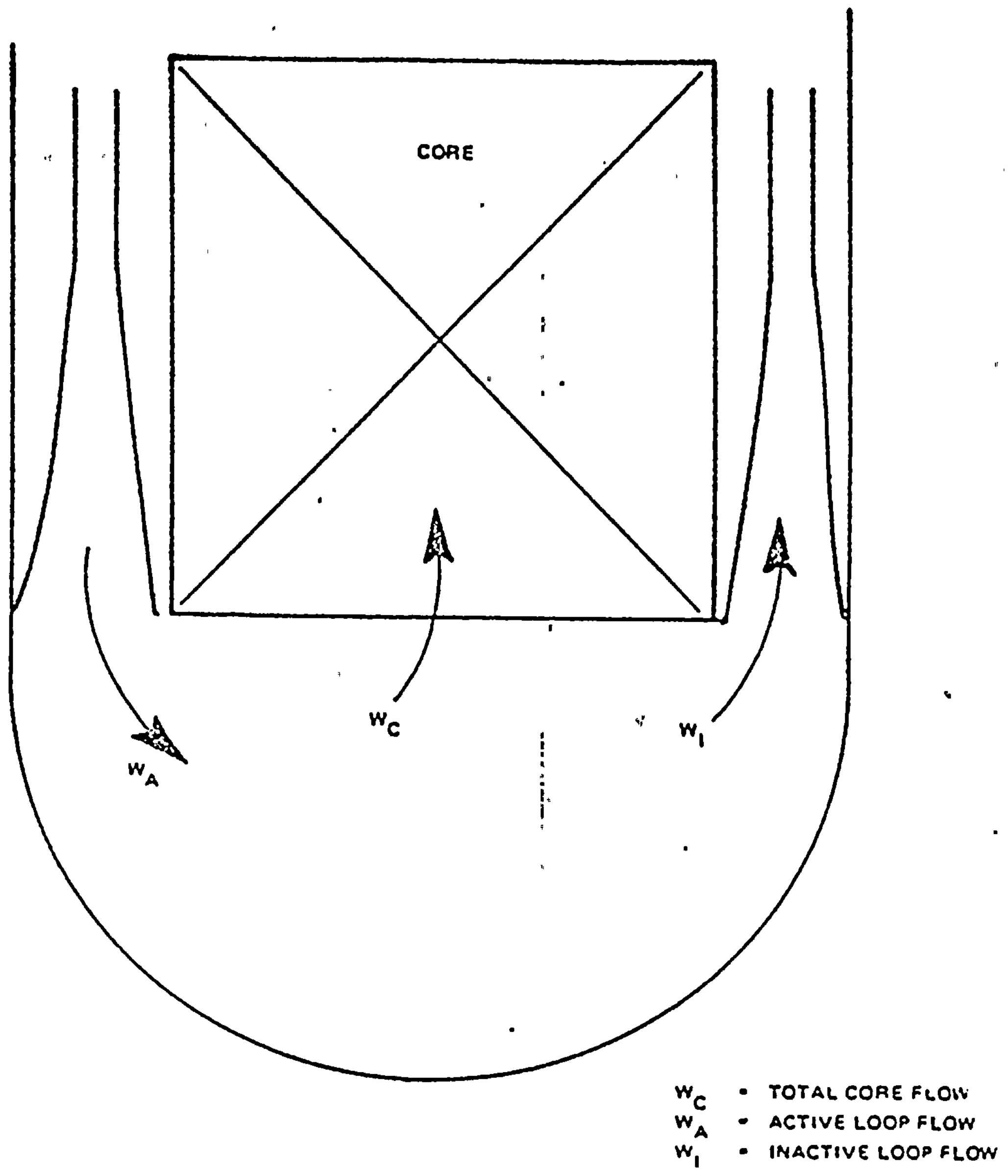
*This flow split ratio varies from about 0.13 to 0.36. The 0.36 value is a conservative bounding value. The analytical expected value of the flow split ratio for WNP-2 is ~ 0.23.

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 uncertainty value for single-loop operation of 6.8% for initial cores and 9.1% for reload cores.



WASHINGTON PUBLIC POWER
SUPPLY SYSTEM
NUCLEAR PROJECT NO 2

ILLUSTRATION OF SINGLE RECIRCULATION LOOP OPERATION FLOWS

FIGURE 6.A.2-1

6.A.3 MCPR OPERATING LIMIT

6.A.3.1 ABNORMAL OPERATING TRANSIENTS

Operation with one recirculation loop results in a maximum power output which is 20% to 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 decrease, and cold water increase transients, results presented in the FSAR bound both the thermal and overpressure consequences of one-loop operation.

Figure 6.A.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.

Cold water increase transients can result from either recirculation flow controller failure, or introduction of colder water into the reactor vessel by events such as loss of feedwater heater. For the former, the K_f factors are derived assuming both recirculation loop controllers fail. 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 K_f factors derived with the two-pump assumption are 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 would result in a neutron flux transient which would exceed the flow reference scram. The resulting scram is expected to be less severe than the rated power/flow case documented in the FSAR.

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 coldwater 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 6.A.3-1.

6.A.3.1.1 Feedwater Controller Failure - Maximum Demand

6.A.3.1.1.1 Identification of Causes and Frequency Classification

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

This event is considered to be an incident of moderate frequency.

6.A.3.1.1.2 Sequence of Events and Systems Operation

With excess feedwater flow, the water level rises to the high-level reference point at which time the feedwater pumps and the main turbine are tripped and a scram is initiated. Table 6.A.3-2 lists the sequence of events for Figure 6:A.3-2. The figure shows the changes in important variables during this transient.

Identification of Operator Actions

a. Observe high feedwater pump trip has terminated the failure event.

b. Switch the feedwater controller from auto to manual control to try to regain a correct output signal.

c. Identify causes of the failure and report all key plant parameters during the event.

Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. Important system operational actions for this event are high level tripping of the main turbine, feedwater turbine, turbine stop valve scram trip initiation, recirculation pump trip (RPT), and low-water level initiation of the reactor core isolation cooling system and the high-pressure core spray system to maintain long-term water level control following tripping of feedwater pumps (not simulated).

6.A.3.1.1.3 Effect of Single Failures and Operator Errors

In Table 6.A.3-2, the first sensed event to initiate corrective action to the transient is the vessel high-water level (L8) trip. Multiple level sensors are used to sense and detect when the water level reaches the L8 setpoint. At this point in the logic, a single failure will not initiate or prevent a turbine trip signal. Turbine trip signal transmission, however, is not built to single-failure criterion. The result of a failure at this point would have the effect of delaying the pressurization "signature". However, high moisture levels entering the turbine will be detected by high levels in the moisture separators which are designed to trip the unit. In addition, excessive moisture entering the turbine will cause vibration to the point where it, too, will trip the unit.

Scram trip signals from the turbine are designed such that a single failure will neither cause nor impede a reactor scram trip.

6.A.3.1.1.4 Core and System Performance

Mathematical Model

The computer model described in Reference 6.A.7-2 was used to simulate this event.

Input Parameters and Initial Conditions

The analysis has been performed with the plant condition tabulated in Table 6.A.3-1, except the initial vessel water level is at level setpoint L4 for conservatism. By lowering the initial water level, more feedwater will get in, hence higher neutron flux will be attained before Level 8 is reached.

The same void reactivity coefficient used for the pressurization transient is applied since a more negative value conservatively increases the severity of the power increase. 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 135% of rated feedwater flow occurs at the design pressure of 1060 psig. Since the reactor is initially operating at a lower power level, the feedwater sparger experiences a pressure which is much lower than the design pressure, hence the feedwater runout capacity reaches 173% of initial flow.

Results

The simulated feedwater controller transient is shown in Figure 6.A.3-2 for the case of 78.7% power 64.3% core flow. The high-water level turbine trip and feedwater pump trip are initiated at approximately 10.64 seconds. Scram occurs simultaneously from stop valve closure, and limits the

neutron flux peak and fuel thermal transient so no fuel damage occurs. MCPR is considerably above the safety limit. The turbine bypass system opens to limit peak pressure in the steamline near the safety valves to 1112 psig and the pressure at the bottom of the vessel to about 1124 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, and worth characteristics). Expected plant behavior is, therefore, expected to lead to a less severe transient.

6.A.3.1.1.5 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.

6.A.3.1.1.6 Radiological Consequences

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

6.A.3.1.2 Generator Load Rejection Without Bypass with Recirculation Pump Trip (RPT)

6.A.3.1.2.1 Identification of Causes and Frequency Classification

Fast closure of the turbine control valves (TCV) is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent overspeed of the turbine-generator rotor. Closure of the main turbine control valves will increase system pressure.

This event is categorized as an infrequent incident with the following characteristics:

Frequency:	0.0036/plant-year
MTBE:	278 years
(Mean Time Between Events)	

Frequency basis: thorough searches of domestic plant operating records have revealed three instances of bypass failure during 628 bypass system operations. This gives a probability of bypass failure of 0.0048. Combining the actual frequency of a generator load rejection with the failure rate of the bypass yields a frequency of a generator load rejection with bypass failure of 0.0036 event/plant year.

6.A.3.1.2.2 Sequence of Events and Systems Operation

Sequence of Events

A loss of generator electrical load at 78.7% and 64.3% flow under single recirculation loop operation produces the sequence of events listed in Table 6.A.3-3.

Identification of Operator Actions

a. Verify proper bypass valve performance.

b. Observe that the feedwater/level controls have maintained the reactor water level at a satisfactory value.

c. Observe the pressure regulator is controlling reactor pressure at the desired value.

d. Record peak power and pressure.

e. Verify relief valve operation.

System Operation

Turbine control valve fast closure initiates a scram trip signal for power levels greater than 30% NB rated. In addition, recirculation pump trip is initiated. Both of these trip signals satisfy single failure criterion and credit is taken for these protection features.

The pressure relief system which operates the relief valves independently when system pressure exceeds relief valve instrumentation setpoints is assumed to function normally during the time period analyzed.

All plant control systems maintain normal operation unless specifically designated to the contrary.

Mitigation of pressure increase during this transient is accomplished by the reactor protection system functions. Turbine control valve trip scram and RPT are designed to satisfy the single failure criterion.

Mathematical Model

The computer model described in Reference 6.A.7-2 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 6.A.3-1.

The turbine DIGITAL electro-hydraulic control system (~~EHC~~ DEH) 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 would normally be independent of any turbine generator overspeed effect. It is continuously supplied at rated frequency as automatic fast transfer to auxiliary power supplies normally occurs. For the purposes of worst case analysis, the recirculation pumps are assumed to remain tied to the main generator and thus increase in speed with the turbine generator overspeed until tripped by the recirculation pump trip system (RPT).

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.

6.A.3.1.2.3 Results

The simulated generator load rejection without bypass is shown in Figure 6.A.3-3.

Table 6.A.3-4 shows for the case of bypass failure, peak neutron flux reaches about 128.8% of rated and average surface heat flux peaks at 103.6% of its initial value. Peak pressure at the valves reaches 1142 psig. The peak nuclear system pressure reaches 1158 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig. The calculated MCPR is 1.29, 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, and worth characteristics). Expected plant behavior is, therefore, expected to reduce the actual severity of the transient.

6.A.3.1.2.4 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.

6.A.3.1.2.5 Radiological Consequences

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

6.A.3.1.3 Recirculation Pump Seizure Accident

6.A.3.1.3.1 Identification of Causes and Frequency Classification

The case of recirculation pump seizure represents the extremely unlikely event of instantaneous stoppage of the pump motor shaft of one recirculation pump. This produces a very rapid decrease of core flow as a result of the large hydraulic resistance introduced by the stopped rotor.

This event is considered to be a limiting fault.

6.A.3.1.3.2 Sequence of Events and Systems Operation

Table 6.A.3-5 lists the sequence of events for this recirculation pump seizure accident.

<u>Identification of Operator Actions</u>

The operator must verify that the reactor scrams with the turbine trip resulting from reactor water level swell. The operator should regain control of reactor water level through RCIC operation or by restart of a feedwater pump, and must monitor reactor water level and pressure control after shutdown.

6.A.3.1.3.3 Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection, and reactor protection systems.

Operation of HPCS and RCIC systems, though not included in this simulation, are expected to occur to maintain adequate water level.

6.A.3.1.3.4 Core and System Performance

Mathematical Model

The computer model described in Reference 6.A.7-3 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 6.A.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 78.7% 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 6.A.3-1.

6.A.3.1.3.5 Results

Figure 6.A.3-4 presents the results of the accident. Core coolant flow drops rapidly, reaching a minimum value of 25% rated at about 1.4 seconds. The level swell produces a trip of both the main and feedwater turbines which, in turn, results in stop valve closure scram. The turbine trip, occurring after the time at which MCPR results, does not significantly retard the heat flux decrease and imposes no threat to fuel thermal limits.. Considerations of uncertainties are included in the GETAB analysis.

6.A.3.1.3.6 Barrier Performance

The bypass valves and momentary opening of some of the safety/relief valves limit the pressure to well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

6.A.3.1.3.7 Radiological Consequences

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

6.A.3.1.4 Summary and Conclusions

The transient peak value results are summarized in Table 6.A.3-4. The Critical Power Ratio (CPR) results are summarized in Table 6.A.3-6. 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 6.A.3-4 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.

6.A.3.2 ROD WITHDRAWAL ERROR

The rod withdrawal error at rated power is given in the FSAR. These analyses are performed to demonstrate, even if the operator ignores all instrument indications and the alarm which could occur during the course of the transient, the rod block system will stop rod withdrawal at a minimum critical power ratio (MCPR) which is higher than the fuel cladding integrity safety limit. Modification of the rod block equation (below) and lower power assures the MCPR safety limit is not violated.

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. 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 40% core flow without correction.

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

The two-pump rod block equation is:

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

The one-pump equation becomes:

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

where

ΔW = difference between two-loop and single-loop effective drive flow at the same core flow. This value is expected to be 5% of rated (to be determined DURING POWER ASCENSION TESTING); ~~by the Supply System);~~

RB = power at rod block in %;

m = flow reference slope for the rod block monitor (RBM), and

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

6.A.3.3 OPERATING MCPR LIMIT

For single-loop operation, the rated condition steady-state 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 6.A.2), the limiting transients have been analyzed. These analyses indicated there is more than enough MCPR margin during single-loop operation to compensate for this increase in safety limit. At lower flows, the steady-state operating MCPR limit is established by multiplying the rated flow steady-state limit by the same K_f factor. This ensures the 99.9% statistical limit requirement is always satisfied for any postulated abnormal operational occurrence.

TABLE 6.A.3-1

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

1. Thermal Power Level Analysis Value, MWt	2616 (78.7% Rated)
2. Steam Flow, lb/hr	11.01×10^6 (77.0% NBR)
3. Core Flow, lb/hr	69.80×10^6 (64.3% Rated)
4. Feedwater Flow Rate, lb/sec	3059
5. Feedwater Enthalpy, Btu/lb	370
6. Vessel Dome Pressure, psig	986
7. Vessel Core Pressure, psig	991
8. Turbine Bypass Capacity, % NBR	25
9. Core Coolant Inlet Enthalpy, Btu/lb	518
10. Turbine Inlet Pressure, psig	954
11. Fuel Lattice	P8x8R
12. Core Average Gap Conductance, Btu/sec-ft^2-°F	0.1744
13. Core Leakage Flow, %	11.84
14. Required MCPR Operating Limit	1.37[(a)]
15. MCPR Safety Limit	1.07
16. Doppler Coefficient (-)¢/°F Analysis Data	0.215[(b)]
17. Void Coefficient (-)¢/% Rated Voids Analysis Data for Power Decrease Events	9.82[(b)]
18. Core Average Void Fraction, %	41.65[(b)]
19. Jet Pump Ratio, M	3.23

csc:rm/I05052*-27

TABLE 6.A.3-1
(Continued)

20. Safety/Relief Valve Capacity, % NBR	
@1164 psig	107.1
Manufacturer	CROSBY
Quantity Installed	18
21. Relief Function Delay, Seconds	0.4
22. Relief Function Response, Seconds	0.1
23. Setpoints for Safety/Relief Valves	
Safety Function, psig	1177, 1187, 1197, 1207 1217
Relief Function, psig	1106, 1116, 1126, 1136 1146
24. Number of Valve Groupings Simulated	
Safety Function, No.	5
Relief Function, No.	5
25. High Flux Trip, % NBR	
Analysis Setpoint (1.21 x 1.043), % NBR	126.2
26. High Pressure Scram Setpoint, psig	1071
27. Vessel Level Trips, Feet Above ~~Separator Skirt Bottom~~ INSTRUMENT ZERO	
Level 8 - (L8), Feet	4.542 ~~6.083~~(c)
Level 4 - (L4), Feet	~~3.750~~ 2.625
Level 3 - (L3), Feet	~~1.750~~ 1.083
Level 2 - (L2), Feet	(-)~~4.708~~ 4.167
28. APRM Thermal Trip	
Setpoint, % NBR @ 100% Core Flow	122.03
29. RPT Delay, Seconds	0.19
30. RPT Inertia for Analysis, lb/ft^2	24500

(a) Two-loop operation operating limit for 64.3% core flow, obtained by applying K_f-curve to operating limit CPR at rated condition, i.e., 1.24.

(b) Parameters used in Reference 6.A.7-3 analysis only. Reference 6.A.7-2 values are calculated within the code for end of Cycle 1 condition.

(c) 6 inches lower than FSAR L8 setpoint ~~(i.e., 5.583')~~ was used for pump seizure case only to get turbine trip.

csc:rm/I05052*-28

TABLE 6.A.3-2

SEQUENCE OF EVENTS FOR FEEDWATER CONTROLLER FAILURE WITH BYPASS (FIGURE 6.A.3-2)

Time (sec)	Event
0	Initiate simulated an upper limit failure of 173% initial feedwater flow
10.64	L8 vessel level setpoint trips main turbine and feedwater pumps
10.65	Reactor scram trip actuated from main turbine stop valve position switches.
10.65	Recirculation pump trip (PRT) actuated by stop valve position switches.
10.80	Turbine stop valves closed and main turbine bypass valves start to open
14.1	Group 1 relief valves actuated on high pressure
2I.1 (est)	Group 1 relief valve closed

TABLE 6.A.3-3

SEQUENCE OF EVENTS FOR LOAD REJECTION WITHOUT BYPASS (FIGURE 6.A.3-3

Time (sec)	Event
(-)0.015 (approx.)	Turbine generator detection of loss of electrical load
0	Turbine generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure
0	Turbine bypass valves fail to operate
0	Fast control valve closure (FCV) initiates scram trip
0	Fast control valve closure (FCV) initiates a recirculation pump trip (RPT)
0.07	Turbine control valves closed
0.19	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation
2.17	Group 1 relief valves actuated
2.36	Group 2 relief valves actuated
2.61	Group 3 relief valves actuated
2.97	Group 4 relief valves actuated
6.40 (est)	Group 4 relief valves close
6.70 (est)	Group 3 relief valves close
7.10 (est)	Group 2 relief valves close
8.80 (est)	Group 1 relief valves close

TABLE 6.A.3-4
SUMMARY OF TRANSIENT PEAK VALUE RESULTS

SINGLE-LOOP OPERATION

PARA-GRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON ~~FLOW~~ FLUX (% NBR)	MAXIMUM DOME PRESSURE (psig)	MAXIMUM VESSEL PRESSURE (psig)	MAXIMUM STEAMLINE PRESSURE (psig)	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX (% of Initial)	FREQUENCY* Category
		Initial Condition	78.7	986	994	979	100.0	N/A
6.A.3.1.1	6.A.3-2	Feedwater flow Controller Failure (Maximum Demand)	105.7	1113	1124	1112	109.5	a
6.A.3.1.2	6.A.3-3	Generator Load Rejection	128.8	1142	1158	1142	103.6	b
6.A.3.1.3	6.A.3-4	Seizure of Active Recirculation Pump	78.7	1045	1055	1044	100.2	c

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

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TABLE 6.A.3-5

SEQUENCE OF EVENTS FOR PUMP SEIZURE (FIGURE 6.A.3-4)

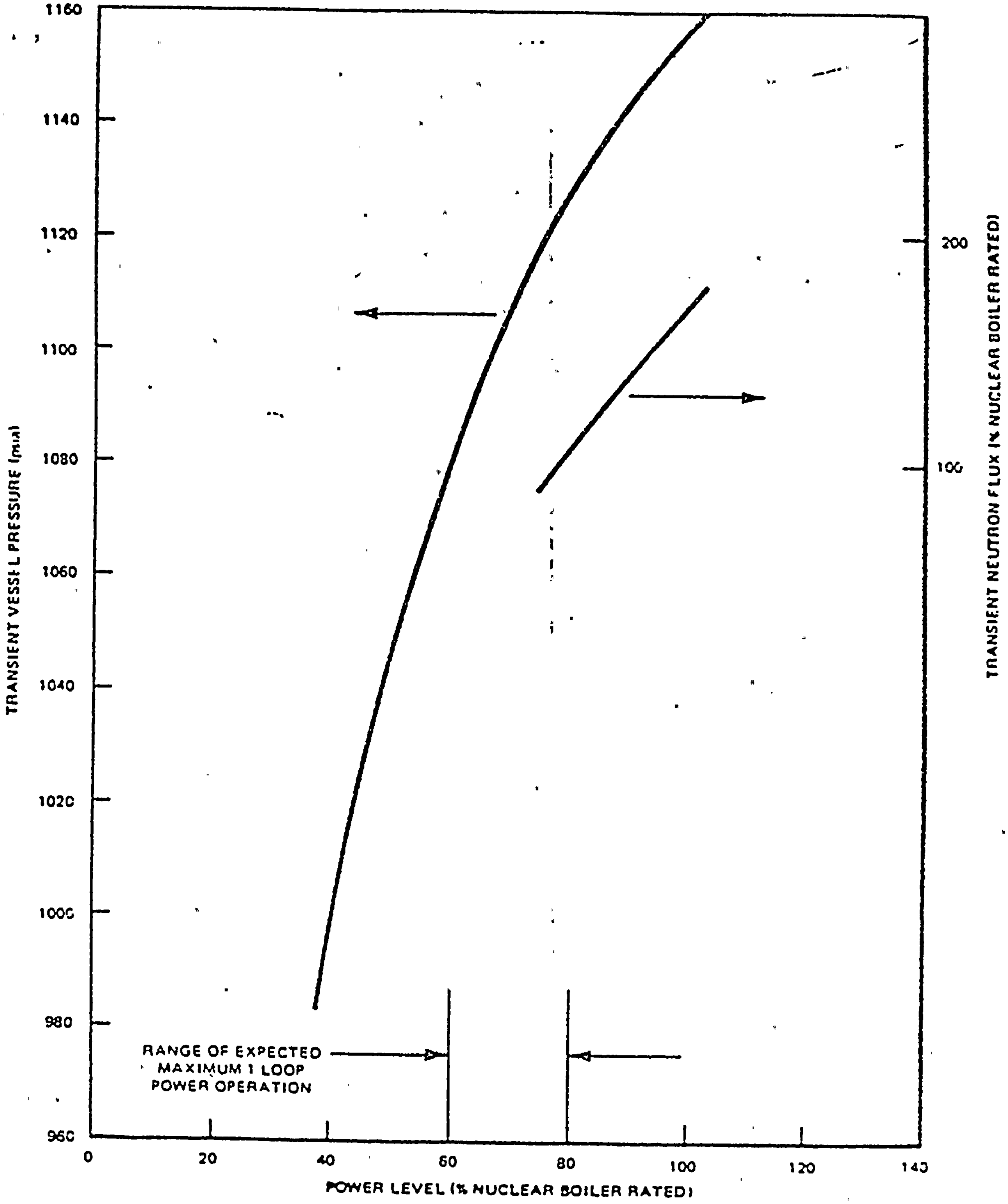
Time (sec)	Event
0	Single pump seizure was initiated, core flow decreases to natural circulation
1.08	Reverse flow ceases in the idle loop
2.72	High vessel water level (L8) trip initiates main turbine trip
2.72	High vessel water level (L8) trip initiates feedwater turbine trip
2.72	High turbine trip initiates bypass operation
2.75 (est)	Main turbine valves reach 90% open position and initiate reactor scram trip
2.85	Turbine stop valves closed and turbine bypass valves start to open to regulate pressure
10.2	Turbine bypass valves start to close
24.5	Turbine bypass valves closed
44.6	Turbine bypass valves reopen on pressure increase at turbine inlet.

TABLE 6.A.3-6

SUMMARY OF CRITICAL POWER RATIO RESULTS - SINGLE-LOOP OPERATION

	FWCF	LRNBT	PS
Rated Operating Limit MCPR	1.24	1.24	1.24
Required Initial MCPR Operating Limit at SLO	1.37	1.37	1.37
ΔCPR	0.12	0.08	0.27
Transient MCPR at SLO	1.25	1.29	1.10
SLMCPR at SLO	1.07	1.07*	1.07*
Margin Above SLMCPR	0.18	0.22	0.03
Frequency Category	Limiting fault	Infrequent incident	Moderate frequent incident

*It is not necessary for these events to meet SLMCPR requirements due to the frequency of occurrence category.

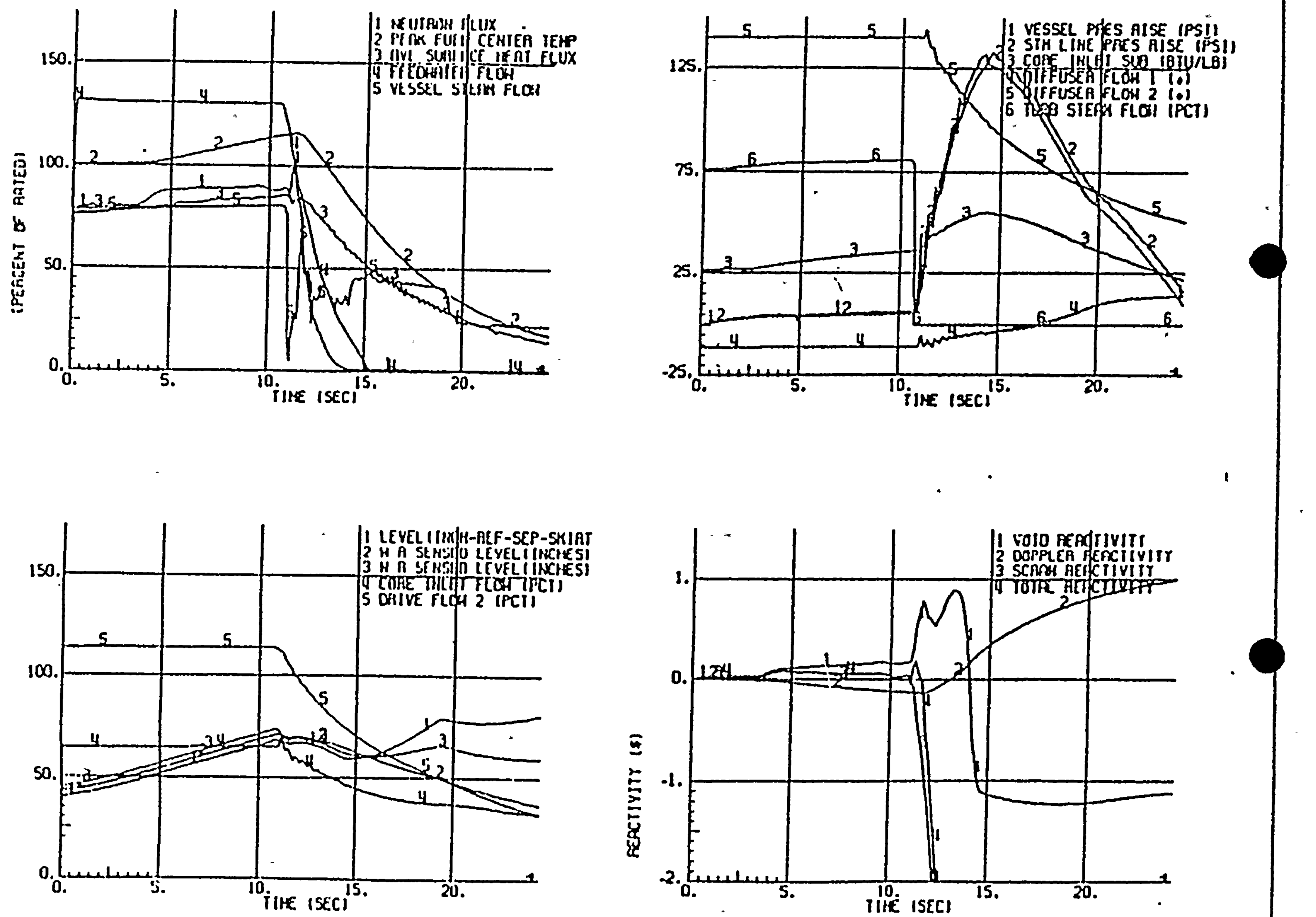


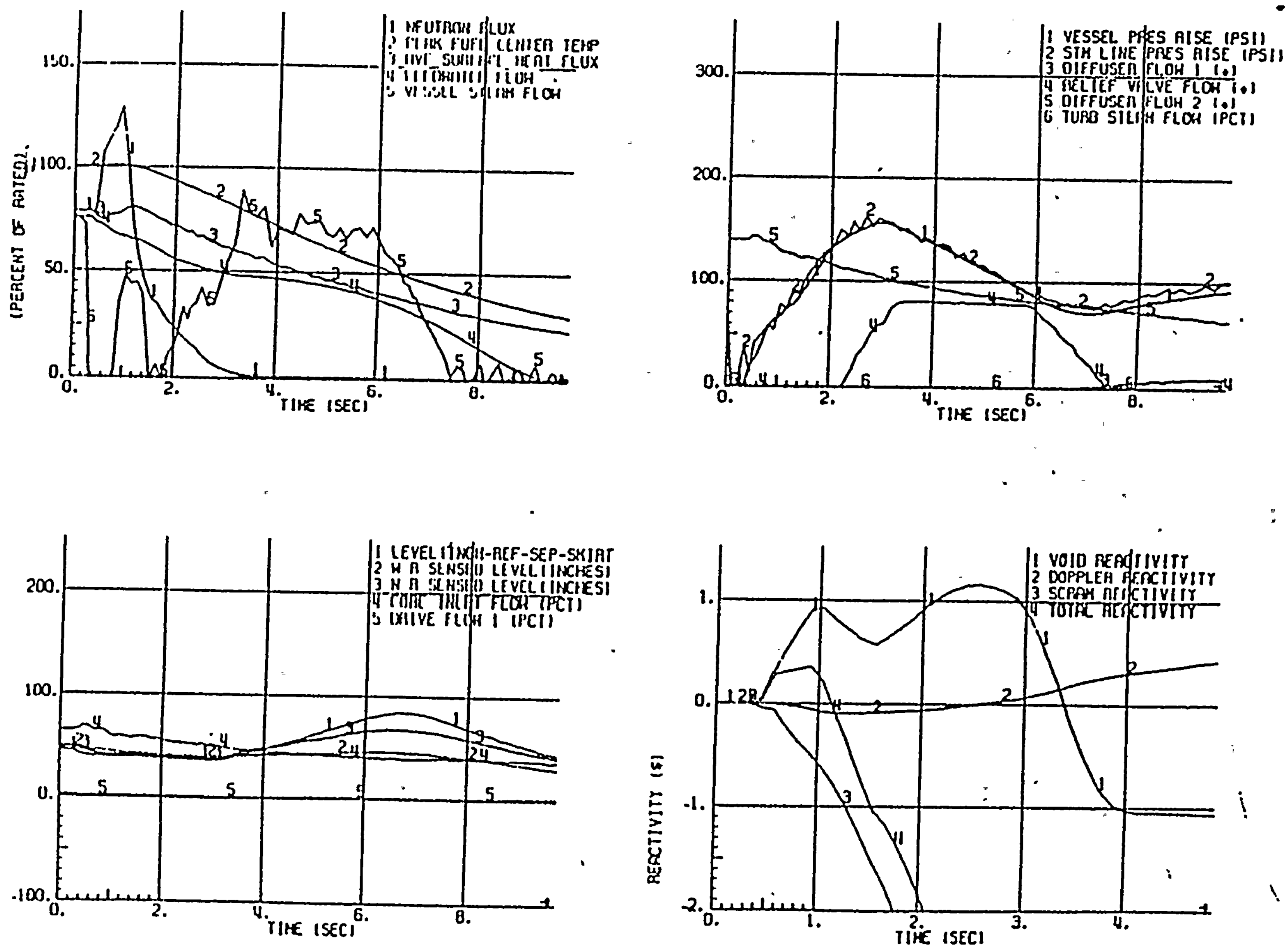
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO 2	MAIN TURBINE TRIP WITH BYPASS MANUAL FLOW CONTROL	FIGURE 6.A.3-1

WASHINGTON PUBLIC POWER
SUPPLY SYSTEM
NUCLEAR PROJECT NO 2

FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND
SINGLE-LOOP OPERATION

FIGURE
6.A.3-2

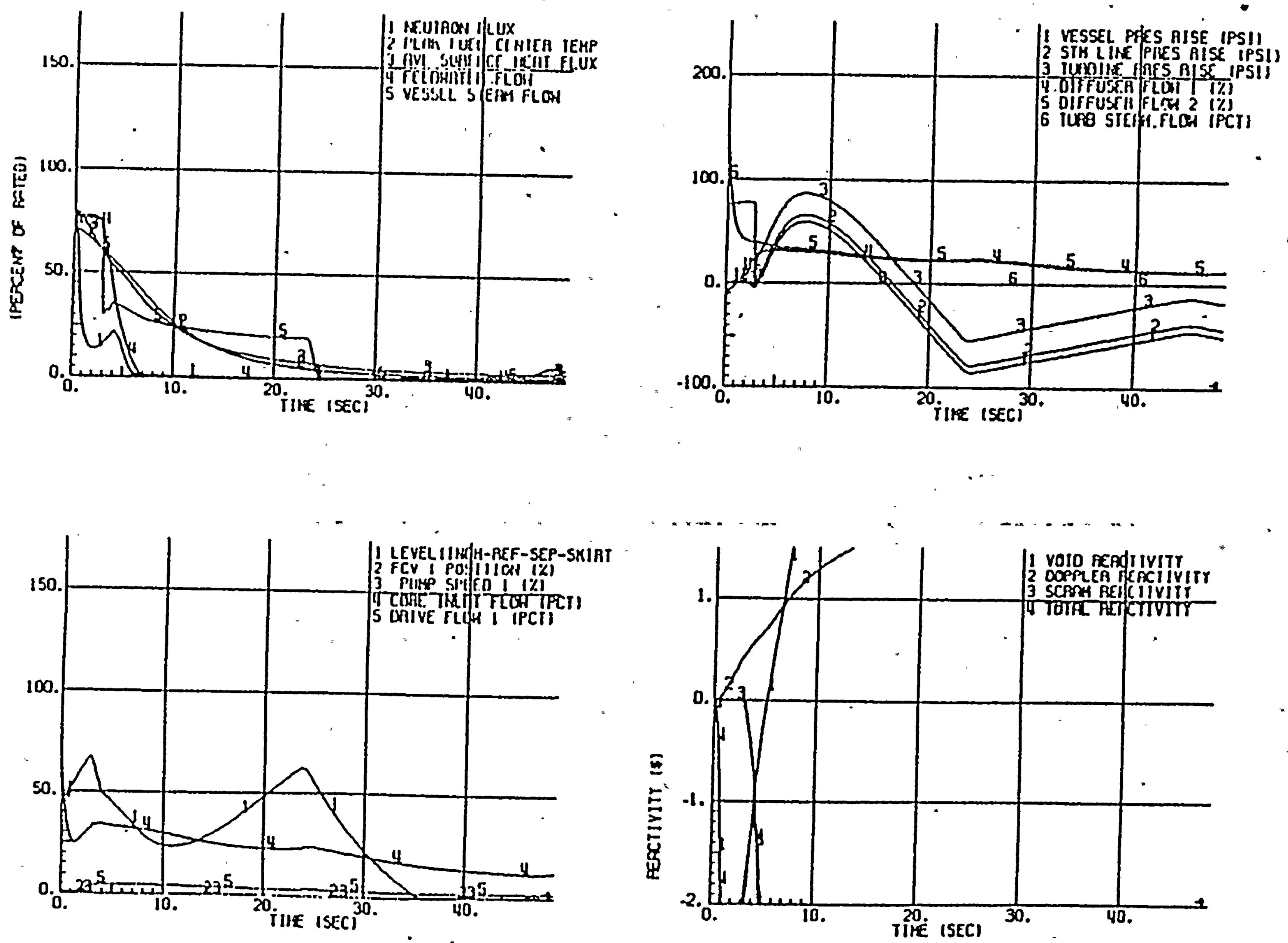




WASHINGTON PUBLIC POWER
SUPPLY SYSTEM
NUCLEAR PROJECT NO 2

LOAD REJECTION W/O BYPASS
SINGLE-LOOP OPERATION

FIGURE
6.A.3-3



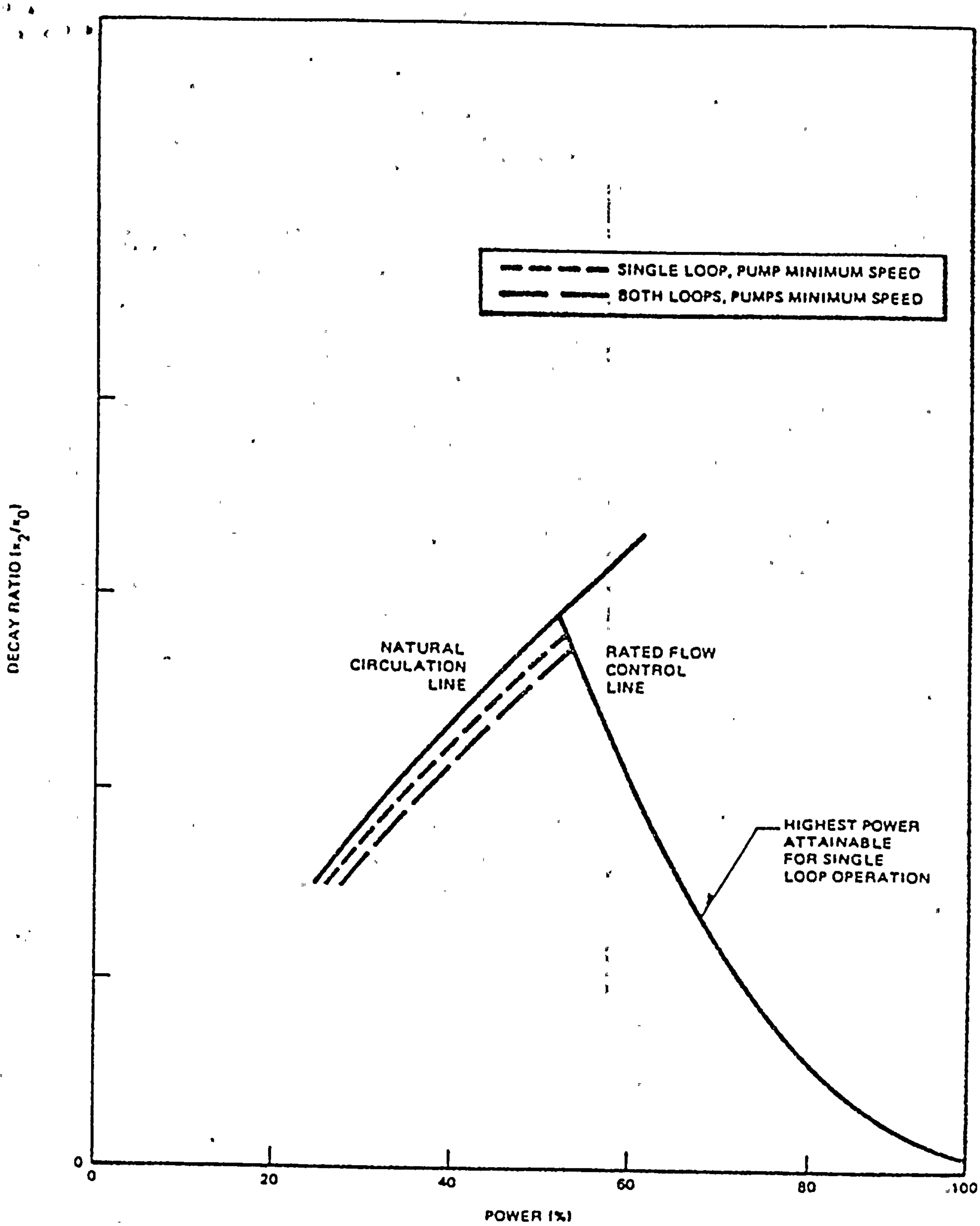
WASHINGTON PUBLIC POWER
SUPPLY SYSTEM
NUCLEAR PROJECT NO 2

PUMP SEIZURE
SINGLE-LOOP OPERATION

FIGURE
6.A.3-4

6.A.4 STABILITY ANALYSIS

The least stable power/flow condition attainable under normal conditions occurs at natural circulation with the control rods set for rated power and flow. This condition may be reached following the trip of both recirculation pumps. As shown in Figure 6.A.4-1, operation along the minimum forced recirculation line with one pump running, at minimum speed, is more stable than operating with natural circulation flow only, but is less stable than operating with both pumps operating at minimum speed. Because of the increased flow fluctuation during one recirculation loop operation, the flow control should be left in manual operation to preclude unnecessary wear on the automatic controls.



WASHINGTON PUBLIC POWER
SUPPLY SYSTEM
NUCLEAR PROJECT NO 2

TYPICAL DECAY RATIO VERSUS POWER TREND FOR TWO-LOOP AND SINGLE-LOOP OPERATION

FIGURE 6.A.4-1

6.A.5 LOSS-OF-COOLANT ACCIDENT ANALYSIS

An analysis of single recirculation loop operation using the models and assumptions documented in Reference 6.A.7-4 was performed for WNP-2. Using this method, SAFE/REFLOOD computer code runs were made for a full spectrum of break sizes for only the suction size breaks (most limiting for WNP-2). 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. WNP-2 does not have equalizer lines. The situation of "equalizer valve open" does not apply to this analysis.

6.A.5.1 BREAK SPECTRUM ANALYSIS

SAFE/REFLOOD calculations were performed using assumptions given in Section II.A.7.3.1 of Reference 6.A.7-4. Hot node uncovered time (time between uncovery and reflood) for single-loop operation is compared to that for two-loop operation in Figure 6.A.5-1.

The maximum uncovered time for two-loop operation is 131 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 132 seconds and occurs also at 100% DBA suction break. This is the most limiting break for single-loop operation.

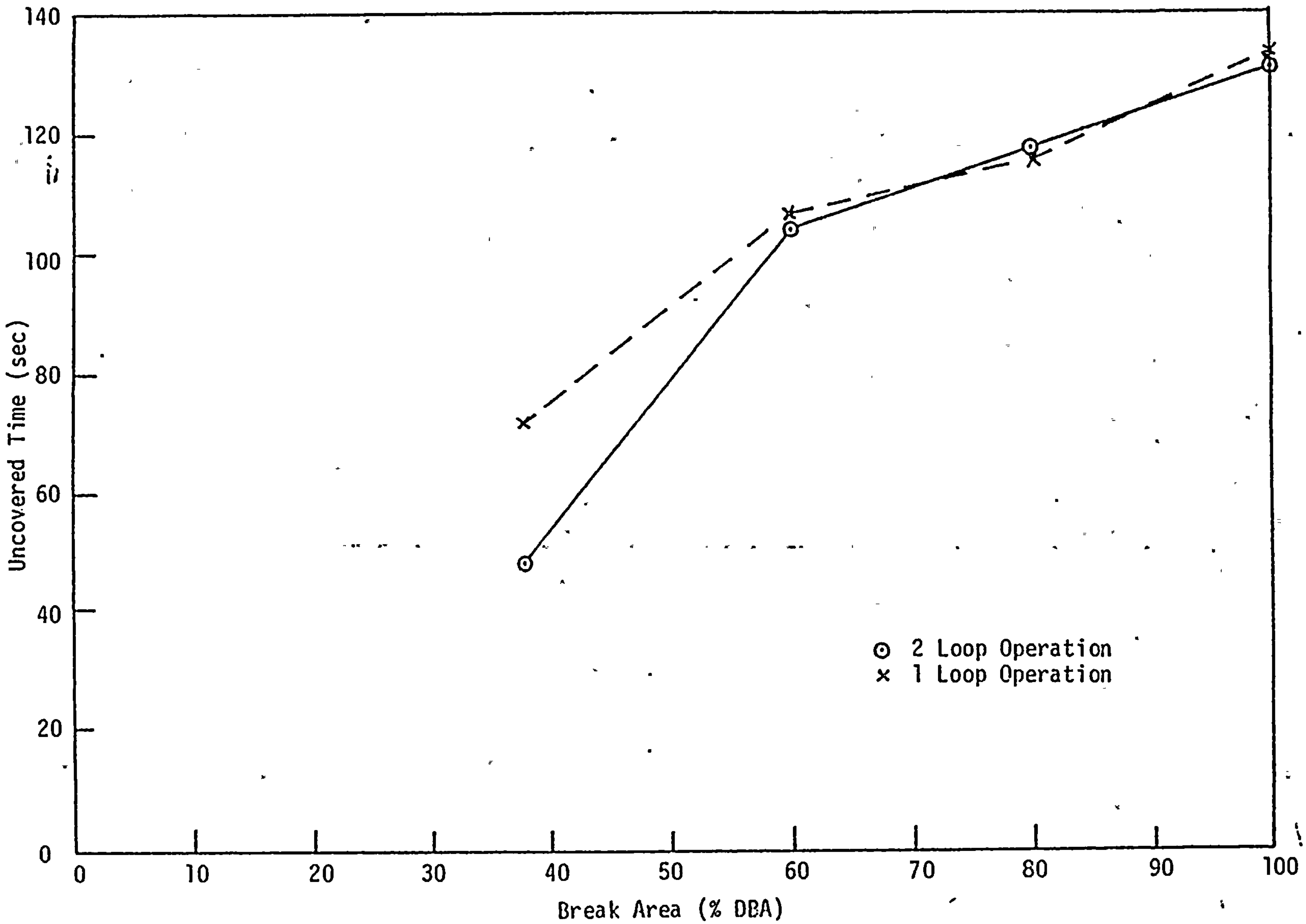
6.A.5.2 SINGLE-LOOP MAPLHGR DETERMINATION

CHASTE heatup calculations were performed in accordance with Section II.A.7.3 of Reference 6.A.7-4 to determine the single-loop MAPLHGR reduction factor for single-loop operation. This analysis was performed for the most limiting case (100% DBA suction break). The most limiting single-loop operation MAPLHGR reduction factor (i.e., yielding the lowest MAPLHGR) for 8 x 8 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 6.A.7-4, single recirculation loop MAPLHGR values are conservative when calculated in this manner.

6.A.5.3 SMALL BREAK PEAK CLADDING TEMPERATURE

Section II.A.7.4.4.2 of Reference 6.A.7-4 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° to 500°F PCT) for one-pump operation, the calculated PCT values for small breaks will be well below the 1456°F small break PCT value previously reported for WNP-2, and significantly below the 2200°F 10CFR50.46 cladding temperature limit.



WASHINGTON PUBLIC POWER
SUPPLY SYSTEM
NUCLEAR PROJECT NO 2

UNCOVERED TIME VS. BREAK AREA
SUCTION BREAK, LPCS FAILURE

FIGURE 6.A.5-1

6.A.6 CONTAINMENT ANALYSIS

A single-loop operation containment analysis was performed for WNP-2. The peak wetwell pressure, diaphragm download 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 78.7% power/64.3% core flow. This peak wetwell pressure decreased by about one percent (0.5 psi) compared to the rated two-loop operation pressure given in Section 6.2. The diaphragm floor download and pool swell velocity evaluated at the worst power/flow condition during single-loop operation were found to be bounded by the rated power analysis presented in Section 6.2.

6.A.7 REFERENCES

6.A.7-1 "General Electric BWR Thermal Analysis Basis (GETAB); Data, Correlation, and Design Application", NEDO-10958-A, January 1977.

6.A.7-2 "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", NEDO-24154, October 1978.

6.A.7-3 R. B. Linford, "Analytical Methods of Plant Transients Evaluation for the General Electric Boiling Water Reactor", NEDO-10802, April 1973.

6.A.7-4 "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.