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SAFETY REVIEW OF WPPSS NUCLEAR PROJECT NO. 2 AT CORE FLOW CONDITIONS ABOVE RATED FLOW THROUGHOUT CYCLE 1 AND FINAL FEEDWATER TEMPERATURE REDUCTION

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ABSTRACT

A safety evaluation has been performed to show that Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 (WNP-2 or Hanford 2) can increase core flow to operate within the region of the operating map bounded by the line between 100% power, 100% core flow (100,100) and 100% power, 106% core flow (100, 106) throughout Cycle 1. WNP-2, after reaching End-of-Cycle 1 (EOC1) exposure (depletion of full-power reactivity under standard feedwater conditions) with all control rods out, can continue to operate in the region of the operating map bounded by the 106% core flow line between 100% power and the cavitation interlock power with or without the last-stage feedwater heaters valved out-of-service (Final Feedwater Temperature Reduction of $\leq$ 65°F at rated power).

The minimum critical power ratio (MCPR) operating limits will be changed from the values established by the Final Safety Analysis Report licensing submittal, to the appropriate values (Table 2-2) for Increased Core Flow (ICF) and Final Feedwater Temperature Reduction (FFWTR) operating conditions. All other operating limits established in the Cycle 1 licensing basis have been found to be bounding for the ICF and FFWTR operations as defined above.

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1. INTRODUCTION AND SUMMARY

This evaluation supports the operation of the Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 (WNP-2 or Hanford 2), within the increased core flow (ICF) region of the operating map as illustrated in Figure 1-1. This report presents the results of a safety evaluation for operation with ICF for Cycle 1 [up to and including End-of-Cycle 1 (EOC1) exposure]. The safety evaluation also covers operation for exposure beyond standard EOC1* with ICF and/or last-stage feedwater heaters valved out, followed by a natural reactivity coastdown bounded by 106% core flow. Final feedwater temperature reduction (FFWTR) from a normal rated power temperature of 420°F to a feedwater temperature of 355°F at 100% power and reactivity coastdown to a minimum feedwater temperature of approximately 321°F (about 65% power) should occur only at the end-of-cycle. The extended region of operation with increased core flow followed by FFWTR at end-of-cycle is bounded by the ICF region marked on the operating map in Figure 1-1.

In order to evaluate operation with ICF and FFWTR, the limiting abnormal operational transients reported in the Final Safety Analysis Report (FSAR), Reference 1, for rated flow operation were reevaluated at EOC1 at 106% core flow with and without FFWTR. The loss-of-coolant accident (LOCA), fuel loading error accident, rod drop accident, and rod withdrawal error event were also reevaluated for increased core flow operation. These events were also reevaluated for end-of-cycle operation with ICF and the last-stage feedwater heaters valved out.

*EOC1 is defined as the core average exposure at which there is no longer sufficient reactivity to achieve rated thermal power with rated core flow, all control rods withdrawn (beyond Rod Position 24), all feedwater heaters in service and equilibrium xenon.

In addition, the effect of the increased pressure differences (due to the increased core flow) on the reactor internals components, fuel channels, and fuel bundles was also analyzed to show that the design limits will not be exceeded. The effect of the increased core flow rate on the flow-induced vibration response of the reactor internals was also evaluated to ensure that the response is within acceptable limits. The thermal-hydraulic stability was evaluated for ICF/FFWTR operation, and the increase in the feedwater nozzle and feedwater sparger usage factors due to the feedwater temperature reduction was determined. The impact of feedwater temperature reduction and increased core flow on the containment LOCA response was also analyzed.

The results of the safety evaluation show that the current technical specifications with incorporation of the MCPR limits of Table 2-2 are adequate to preclude the violation of any safety limits during operation of WNP-2 within the increased core flow region of the operating map as illustrated in Figure 1-1 for Cycle 1 and for exposures beyond EOC1 with the conditions assumed in the analysis. The ΔCPRs and the minimum critical power ratio (MCPR) operating limits for plant operation are given in Tables 2-1 and 2-2. The EOC1 Option A and Option B MCPR limits (Reference 1) will be increased to the appropriate values as shown in Table 2-2.

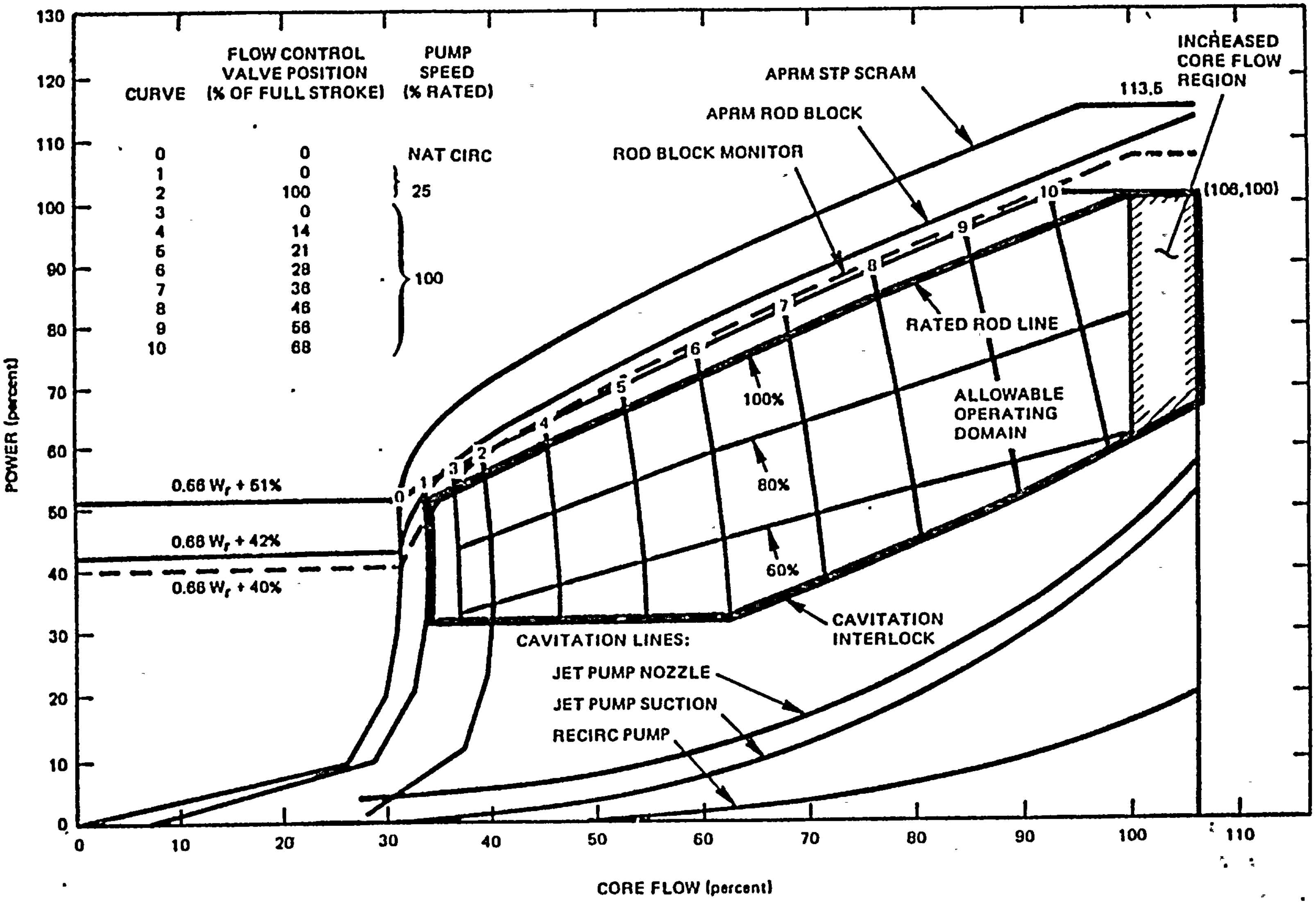

Figure 1-1. Operating Map

2. SAFETY ANALYSIS

2.1 ABNORMAL OPERATIONAL TRANSIENTS

2.1.1 Limiting Transients

The limiting abnormal operational transients analyzed in the Cycle 1 FSAR licensing submittal (Reference 1) were reevaluated for increased core flow and/or FFWTR.

Nuclear transient data for 104.5% power*, 106% core flow (104.5, 106) with and without the last-stage feedwater heaters out were developed based on the Haling method at rated power for EOC1. The nuclear data was then used to analyze the load rejection with bypass failure (LRNBP) event and the feedwater controller failure to maximum demand (FWCF) event at the (104.5, 106) conditions.

The results of the transient analyses are presented in Tables 2-1 and 2-2 with the limiting transient results previously submitted in the FSAR licensing submittal (Reference 1). The transient performance responses are presented in Figures 2-1 through 2-4. The results demonstrate that the ΔCPR values and the critical power ratio operating limits for the LRNBP and FWCF events increase compared with the corresponding FSAR values. However, the FSAR licensing submittal (Reference 1) OLCPR = 1.24 for either Option A or Option B based on the rod withdrawal error (RWE) transient is bounding for both the LRNBP and FWCF events for ICF with or without FFWTR. The current evaluation of the RWE event is presented in Section 2.1.3.

*All transients were analyzed using 105% steam flow. The power level corresponding to this condition will vary from 104.5% to 104.2%, depending on whether final feedwater heaters are in service. The 104.5 power level provides a 5% steam flow margin to the 100% power operating conditions to simulate eventual stretch power operation, similar to the original FSAR analyses.

Decreasing the power from the 100% rated condition along the 106% core flow line will result in an increase in transient ΔCPR for some events. This increase is less than the increase in operating CPR due to the power decrease, and, hence, such operation will not result in violation of the safety limit MCPR due to a transient (Reference 2, p. 2-12).

2.1.2 Overpressurization Analysis

The limiting transient for ASME code overpressurization analysis, main steam isolation valve (MSIV) closure with flux scram (direct scram failure), was evaluated for the extended EOC1 conditions with ICF without FFWTR (Table 2-3 and Figure 2-5). For this evaluation ICF without FFWTR is more severe than ICF with FFWTR. The ICF for the LRNBP event results in a less severe overpressure transient than MSIV closure with flux scram. The overpressurization analysis (Table 2-3) for the ICF region produced a peak vessel pressure of 1264 psig, which is below the upset code limit of 1375 psig and is, therefore, acceptable.

2.1.3 Rod Withdrawal Error

The rod withdrawal error transient was evaluated under ICF and/or FFWTR conditions. When ICF is employed, the rod block monitor (RBM) setpoint (which is flow biased) increases, giving an unacceptably high MCPR limit. Thus, the RBM should be clipped at flows greater than 100% of rated so that the ΔCPR values (Reference 1) determined without ICF apply.

2.2 FUEL LOADING ERROR

This event is not adversely affected by the increased core flow mode of operation with the last-stage feedwater heaters removed from service. The impact of ICF and/or FFWTR on ΔCPR is expected to be very small compared with the margin to the OLCPR. Thus, the FSAR ΔCPR would not be affected by this event under ICF and/or FFWTR conditions.

2.3 ROD DROP ACCIDENT

WNP-2 uses banked position withdrawal sequence (BPWS) for control rod movement. Control Rod Drop Accident (CRDA) results from BPWS plants have been statistically analyzed. The results show that, in all cases, the peak fuel enthalpy in an RDS would be much less than the corresponding design limit even with a maximum incremental rod worth corresponding to 95% probability at the 95% confidence level. Based on these results, it was proposed to the US NRC, and subsequently found acceptable, to delete the CRDA from the standard GE-BWR reload package for the BPWS plants (Reference 2, Section S.2.5.1.3 (1), Page 2-53). Hence, the CRDA is not specifically analyzed for WNP-2.

2.4 LOSS-OF-COOLANT ACCIDENT (LOCA) ANALYSIS

LOCA analysis performed for WNP-2 shows that operation with ICF without FFWTR bounds operation with ICF and FFWTR.

The effect of increased core flow on LOCA analyses is not significant because the parameters which most strongly affect the calculated peak cladding temperature (PCT), i.e., high power node boiling transition time and core reflooding time, have been shown to be relatively insensitive to increased core flow.

Results of the LOCA analysis performed show that the PCT for ICF increases by less than 5°F throughout the break spectrum compared to the rated core flow condition.

Therefore, it is concluded that the LOCA PCT is acceptable and that the current maximum average planar linear heat generation rates (MAPLHGRs) for WNP-2 are applicable for ICF.

2.5 THERMAL-HYDRAULIC STABILITY

The General Electric Company has established stability criteria to demonstrate compliance to requirements set forth in 10CFR50 Appendix A, General Design Criteria (GDC). 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 all licensed GE BWR fuel designs including those fuels contained in the General Electric Standard Application for Reactor Fuel (GESTAR, Reference 2) is demonstrated on a generic basis in Reference 3 (for operation in the normal as well as the extended operating domain with ICF and FFWTR). The NRC has reviewed and approved this in Reference 4; therefore, a specific analysis for each cycle is not required. The WNP-2 Cycle 1 core contains licensed GE BWR initial core and, hence, the generic evaluation in Reference 3 is applicable to WNP-2.

For operation in the ICF region, the stability margin (defined by the core decay ratio) is increased as flow increases for a given power. ICF operation is bounded by the fuel integrity analyses in Reference 3.

Similarly, operation in the FFWTR mode is bounded by the fuel integrity analyses in Reference 3. In general, the effect of reduced feedwater temperature results in a higher initial CPR which yields even larger margins than those reported in Reference 3. The fuel integrity analyses are independent of the stability margin, since the reactor is already assumed to be in limit cycle oscillations. Reference 3 also demonstrates that even if neutron flux limit cycle oscillations did occur just below the neutron flux scram setpoint, fuel design limits are not exceeded for those GE BWR fuel designs contained in General Electric Standard Application for Reactor Fuel (GESTAR, Reference 2). These evaluations demonstrate that substantial thermal/mechanical margin is available for the GE BWR fuel designs even in the unlikely event of very large oscillations.

To provide assurance that acceptable plant performance is achieved during operation in the least stable region of the power/flow map, as well as during all plant maneuvering and operating states, a generic set of operating recommendations has been developed as set forth in Reference 5 and communicated to all GE BWRs. These recommendations instruct the operator on how to reliably detect and suppress limit cycle neutron flux oscillations should they occur. The recommendations were developed to conservatively bound the expected performance of all current product lines and are applicable to operation with FFWTR (feedwater temperature of approximately 355°F at rated power).

Table 2-1

CORE-WIDE TRANSIENT ANALYSIS RESULTS AT ICF AND/OR FFWTR

Transient Description[a]	Figure Number	Power (% NBR)	Flow (% NBR)	Rated Feedwater Temperature Reduction[b] (°F)	Maximum Neutron Flux (% NBR)	Maximum Core Ave. Surface Heat Flux (% Initial)	Maximum Dome Press (psig)	Maximum Vessel Press (psig)	Maximum Steam Line Press (psig)	ΔCPR[c]
LRNBP	Ref. 1	104.4	100	0	236.4	107.8	1173	1202	1168	0.09
LRNBP	2.1	104.2	106	0	252.4	108.8	1172	1203	1168	0.11
LRNBP	2.2	104.5	106	65	243.2	108.8	1160	1191	1157	0.11
FWCF	Ref. 1	104.4	100	0	154.3	108.7	1148	1177	1140	0.08
FWCF	2.3	104.2	106	0	163.7	109.1	1145	1177	1141	<0.13
FWCF	2.4	104.5	106	65	174.7	113.9	1138	1166	1135	0.13

a. LRNBP = Load rejection with bypass failure, FWCF = feedwater controller failure to maximum demand,

b. Reduction of feedwater temperature from nominal rated feedwater temperature (420°F) and at rated conditions.

c. ΔCPR based on initial CPR which yields MCPR = 1.06; uncorrected for Options A and B.

Table 2-2

REQUIRED MCPR OPERATING LIMITS AT ICF AND/OR FFWTR

Transient Description[a]	Initial Core Power (% NBR)	Initial Core Flow (% NBR)	ΔCPR^{b}	$OLCPR_A^{c,e}$	$OLCPR_B^{d,e}$
LRNBP (FSAR)	104.4	100	0.09	1.20	1.12
LRNBP[f]	104.2	106	0.11	1.22	1.14
FWCF (FSAR)	104.4	100	0.08	1.19	1.16
FWCF[g]	104.5	106	0.13	1.24	1.21
			ΔCPR	OLCPR	
RWE (FSAR)	104.4	100	0.18	1.24[h]	

a. LRNBP = Load rejection with bypass failure, FWCF = feedwater controller failure at maximum demand, RWE = rod withdrawal error.

b. ODYN results without adjustment factors, based on initial CPR which yields an MCPR = 1.06.

c. Includes Option A adjustment factors.

d. Includes Option B adjustment factors.

e. Option A and B adjustment factors are specified in the NRC safety evaluation report on ODYN (NEDO-24154 and NEDE-24154P).

f. For load rejection with bypass failure, ICF w/o FFWTR bounds ICF with FFWTR.

g. For feedwater controller failure to maximum flow demand, ICF with FFWTR bounds ICF w/o FFWTR.

h. Required OLCPR using either Option A or Option B adjustment factor with rod block monitor of 106% at rated flow

Table 2-3

OVERPRESSURIZATION ANALYSIS RESULTS

Transient	Initial Power (%)	Initial Flow (%)	Maximum Vessel Pressure (psig)	Figure No.
MSIV Closure - Flux Scram (FSAR)	104.3	100	1266	Reference 1
MSIV Closure - Flux Scram (ICF w/o FFWTR)	104.2	106	1264	Figure 2-5

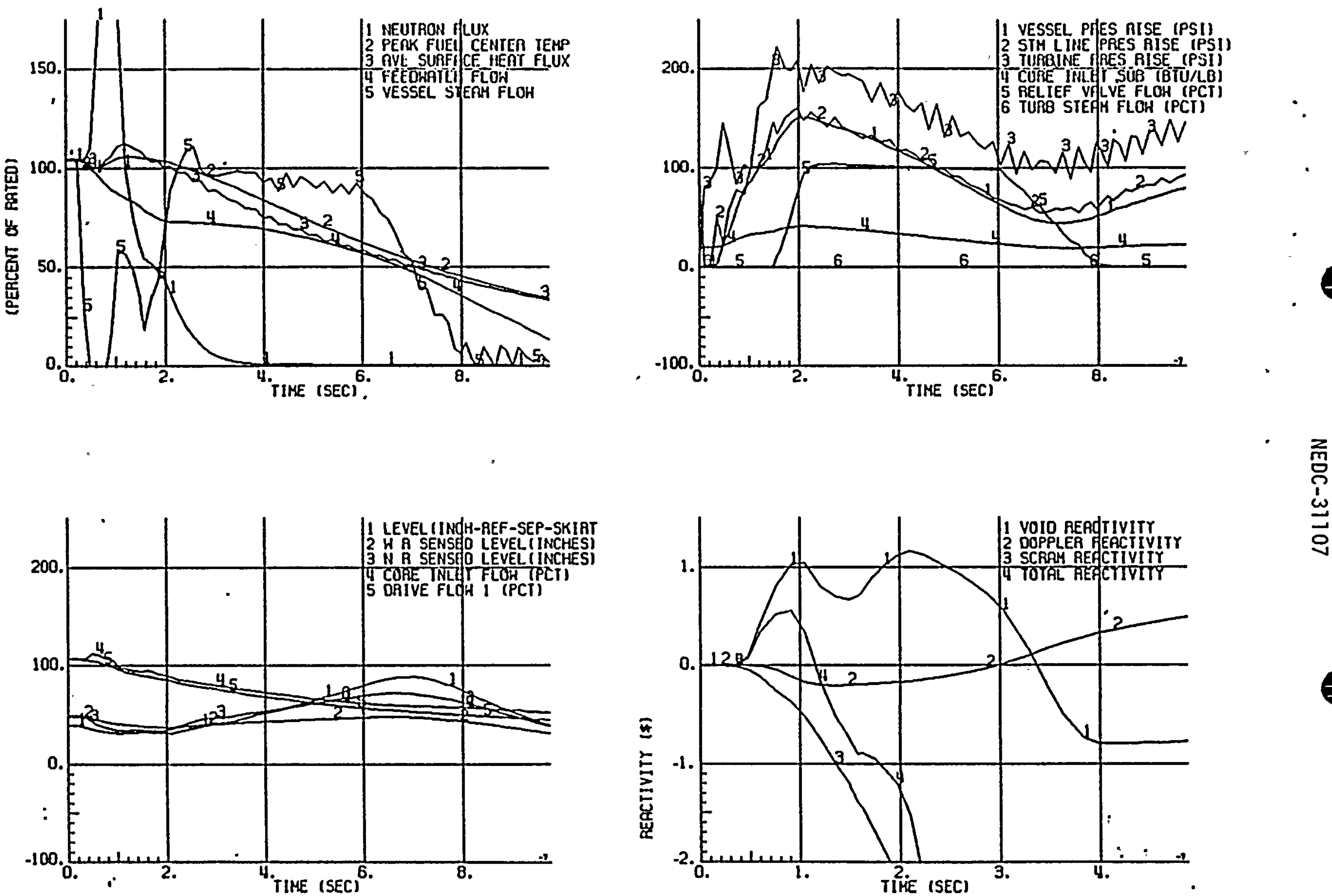

Figure 2-1. Generator Load Rejection with Bypass Failure at 104.2% Power, 106% Flow and Normal Feedwater Temperature

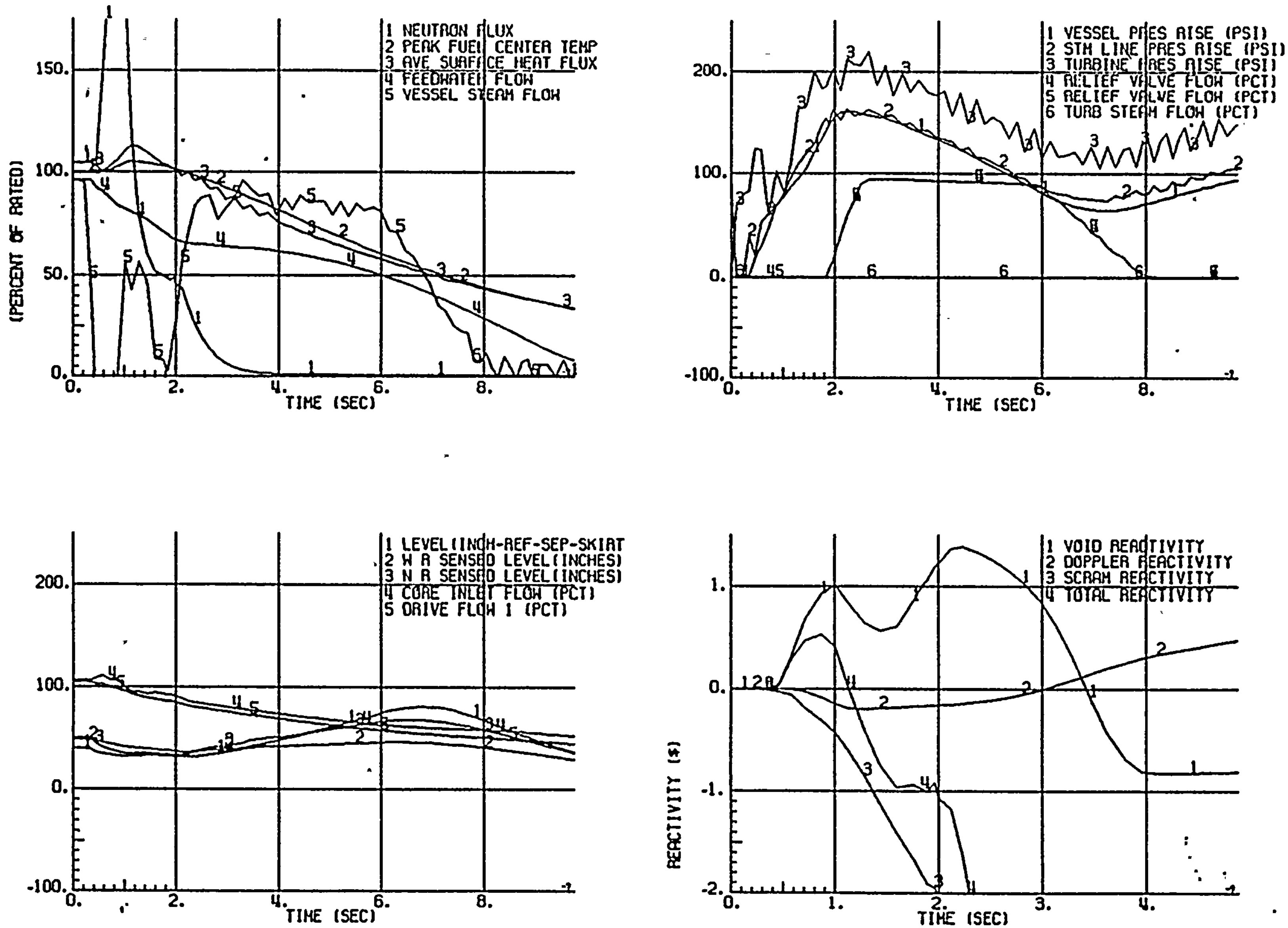

Figure 2.2. Generator Load Rejection With Bypass Failure at 104.5% Power, 106% Flow and Reduced Feedwater Temperature of 65°F at Rated Power.

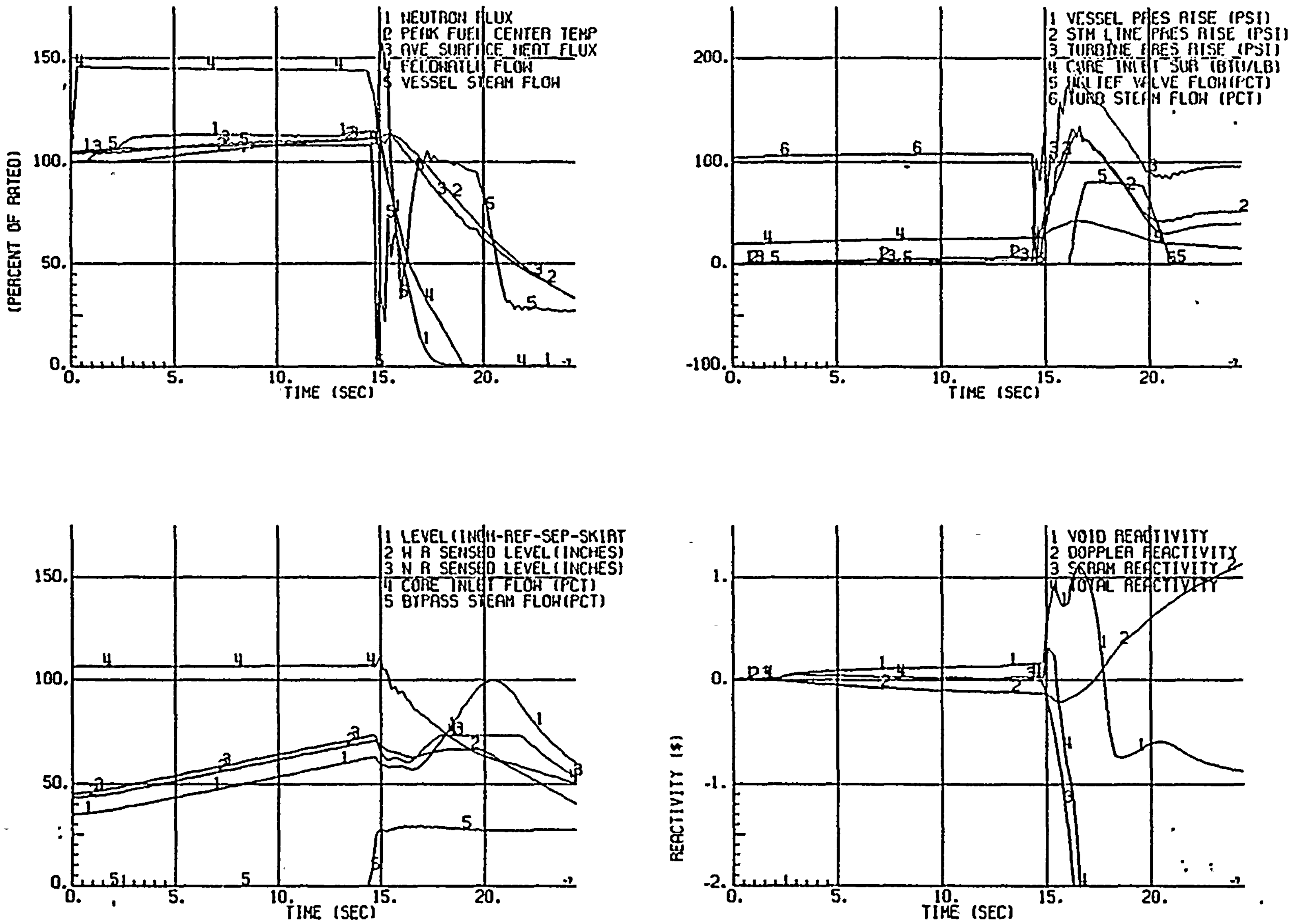

Figure 2-3. Feedwater Controller Failure, Maximum Demand at 104.2% Power, 106% Flow and Normal Feedwater Temperature

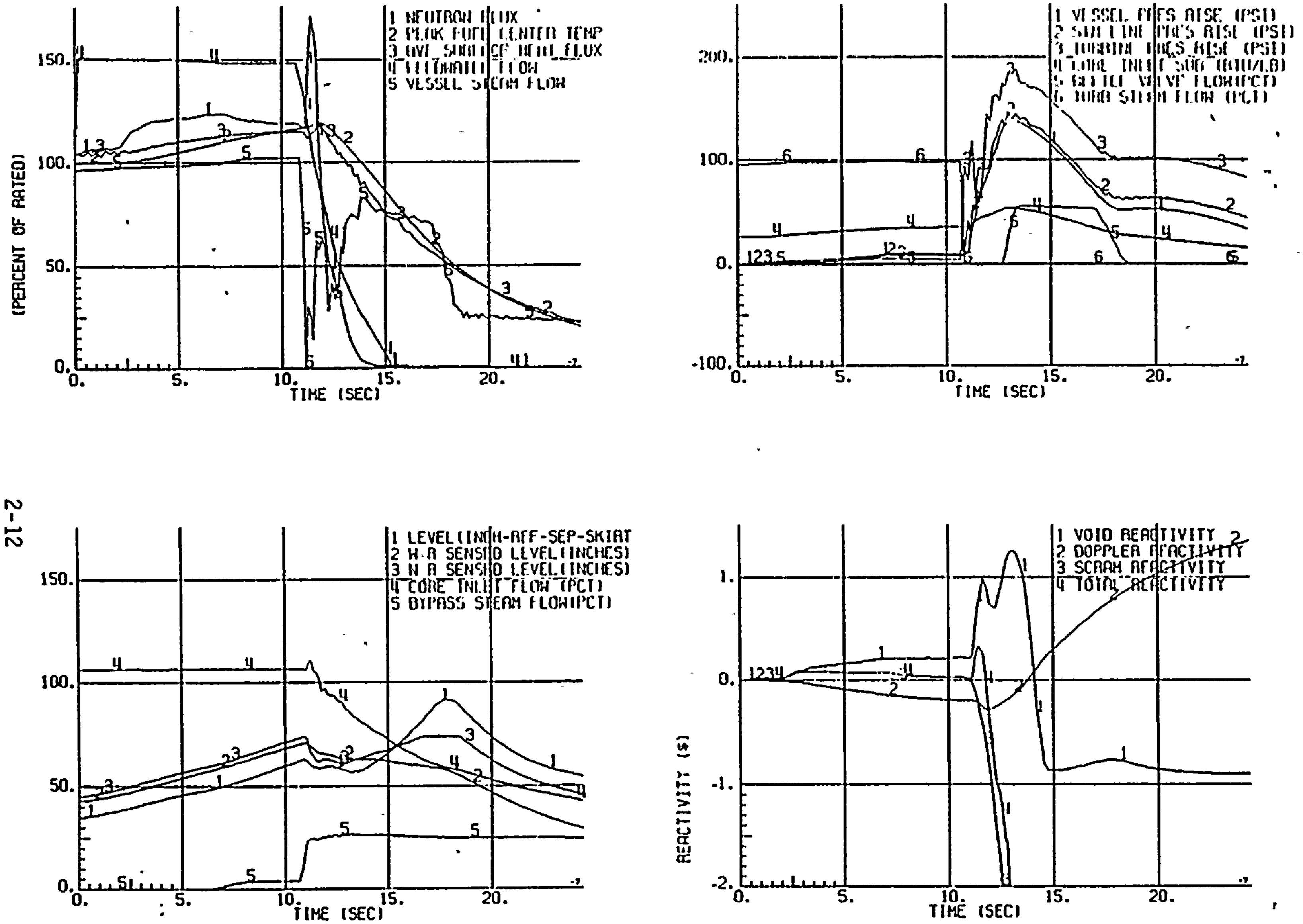

Figure 2-4. Feedwater Controller Failure, Maximum Demand, at 104.5% Power, 106% Flow and Reduced Feedwater Temperature of 65°F at Rated Power

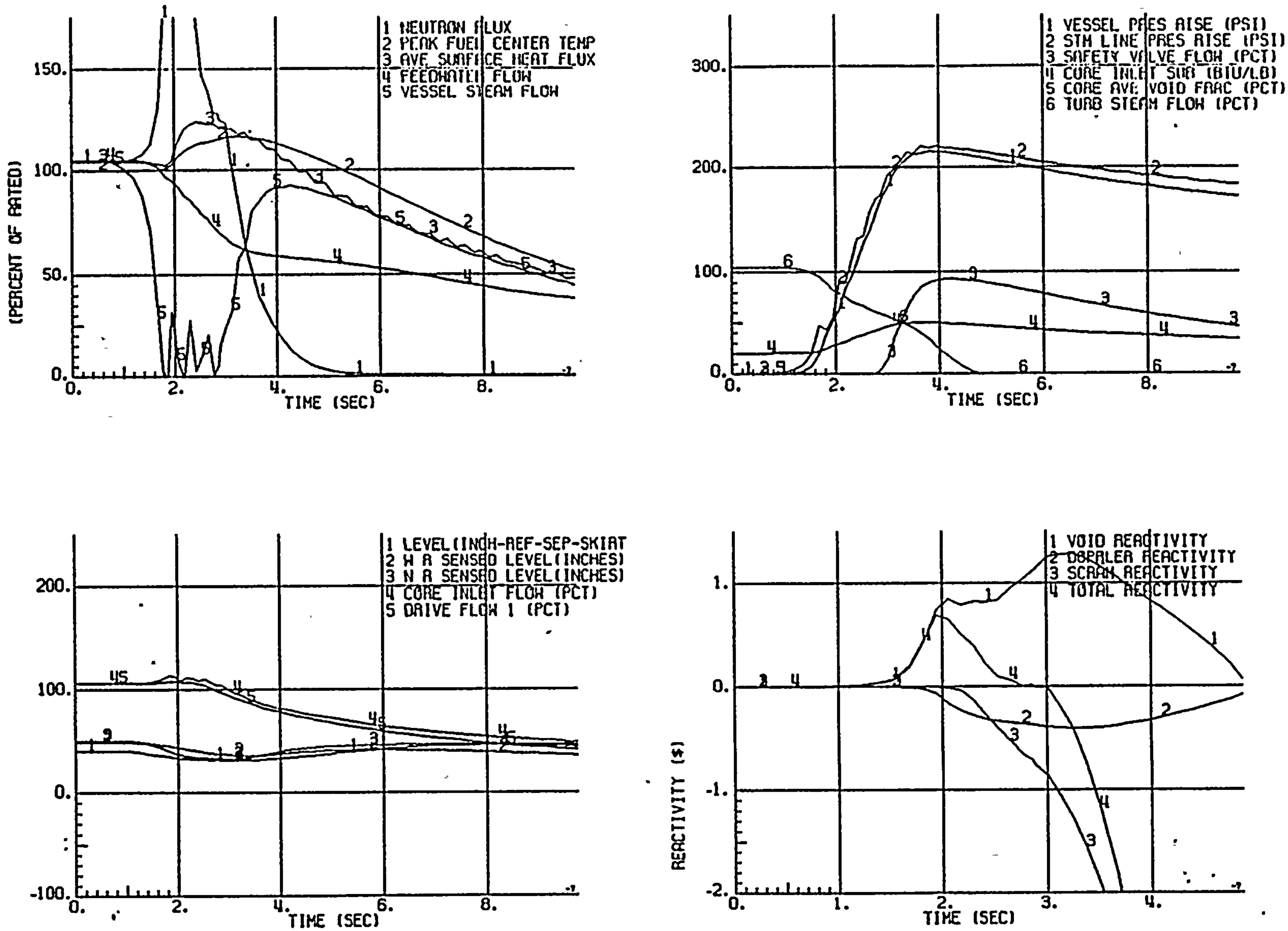

Figure 2-5. MSIV Closure, Flux Scram, at 104.2% Power, 106% Flow and Normal Feedwater Temperature

3. MECHANICAL EVALUATION OF REACTOR INTERNALS AND FUEL ASSEMBLY

3.1 LOADS EVALUATION

Evaluations were performed to determine bounding acoustic and flow-induced loads, reactor internal pressure difference loads and fuel-support loads for ICF and/or FFWTR operation.

Acoustic loads are lateral loads on the vessel internals that result from propagation of the decompression wave created by a sudden recirculation suction line break. The acoustic loading on vessel internals is proportional to the total pressure wave amplitude in the vessel recirculation outlet nozzle. The total pressure amplitude is the sum of the initial pressure subcooling plus the experimentally determined pressure undershoot below saturation pressure. FFWTR operation increases the expected acoustic loads because this downcomer subcooling increases and, therefore, the total pressure wave amplitude increases. The high velocity flow patterns in the downcomer resulting from a recirculation suction line break also create lateral loads on the reactor vessel internals. These loads are proportional to the square of the critical mass flow rate out of the break. The additional subcooling in the downcomer resulting from FFWTR operation leads to an increase in the critical flow and, therefore, to a corresponding increase in the flow-induced loads. The reactor internals most impacted by acoustic and flow-induced loads are the shroud, shroud support and jet pumps.

A reactor internals pressure difference analysis was performed for the ICF region. The increased reactor internal pressure differences across the reactor internals were generated for the maximum core flow at normal, upset, and faulted conditions for the reactor internal impact evaluation.

Fuel-support loads and fuel bundle lift for WNP-2 were evaluated based on results from probabilistic fuel lift analyses performed at 106% of rated core flow following the procedures of Reference 6. Fuel-support loads and fuel bundle lift were evaluated for upset, faulted and fatigue load combinations.

It was shown that the fuel bundle lift is a small fraction of the applicable design criteria (established in the NRC Safety Evaluation Report to Reference 6) for the faulted event.

3.2 LOADS IMPACT

3.2.1 Reactor Internals

The reactor internals most affected by ICF and/or FFWTR operation are the core plate, shroud support, shroud, top guide, shroud head, steam dryer, control rod guide tube, control rod drive housing and jet pump. These and other components were evaluated using the bounding loads, discussed in Section 3.1, under normal, upset, emergency and faulted conditions. It is concluded that the stresses produced in these and other components are within the allowable design limits given in the Final Safety Analysis Report (Chapter 3 and 4) or the ASME Code, Section III, Subsection NG.

3.2.2 Fuel Assemblies

The fuel assemblies, including fuel bundles and channels, were evaluated for increased core flow operation considering the effects of loads discussed in Section 3.1 under normal, upset,faulted and fatigue load combinations. Results of the evaluation demonstrate that the fuel assemblies are adequate to withstand ICF effects to 106% rated flow.

The fuel channels were also evaluated under normal, upset, emergency and faulted conditions for increased core flow (Reference 7). The channel wall pressure differentials were found to be within the allowable design values .

4. FLOW-INDUCED VIBRATION

To ensure that the flow-induced vibration response of the reactor internals is acceptable, a single reactor of each product line and size undergoes an extensive vibration test during initial plant startup. After analyzing the results of such tests and assuring that all responses fall within acceptable limits of the established criteria, the reactor is classified as a valid prototype in accordance with Regulatory Guide 1.20. All other reactors of the same product line and size undergo a less rigorous confirmatory test to assure similarity to the base test. The acceptance criteria used for vibration assessment is based on a maximum allowable alternating stress intensity of 10,000 psi.

The increased core flow vibration analysis was performed by analyzing the startup test vibration data for the valid prototype plant (BWR/5-251 Tokai 2). Based on the results of the analysis and a review of the test data, the reactor internals response to flow-induced vibration is expected to be within acceptable limits for plant operation in the ICF region (region bounded as shown on the power flow map, Figure 1-1).

5. FEEDWATER NOZZLE AND FEEDWATER SPARGER FATIGUE USAGE

5.1 METHOD AND ASSUMPTIONS

The fatigue experienced by the feedwater nozzle and feedwater sparger results from two phenomena: system cycling and rapid cycling. System cycling is caused by major temperature changes associated with system transients. The system cycle stresses are based on limiting cycles that use the maximum temperature range possible to show expected worst conditions. These transients are identified on thermal cycle diagrams. Thermal stresses due to these transients are calculated by determining inner and outer metal surface temperatures using finite element analysis. Fatigue usage is determined by dividing the number of design cycles for each transient by the number of allowable cycles for each stress calculated. Cumulative system fatigue usage is determined by summing all of the respective transient fatigue usage factors.

Rapid cycling is caused by small, high frequency temperature fluctuations caused by mixing of relatively colder nozzle annulus water with the reactor coolant. The colder water impinging the nozzle bore originates from the boundary layer of colder water formed by heat transfer through the thermal sleeve. The mixing region extends from the feedwater nozzle surface region to the feedwater sparger surface; therefore, rapid cycling applies to both of these components. Once thermal stress due to rapid cycling is determined, fatigue usage is calculated and the results are added to the cumulative system cycling usage factor to obtain the total usage factor.

The introduction of FFWTR will cause a change in calculated rapid cycling fatigue only. This is because the system transient is very mild (small temperature change and relatively long duration) and is bounded by the original design basis thermal stress analysis. General Electric has developed standardized rapid cycling duty maps for each BWR plant that cover the design basis rapid cycles in the same manner that thermal cycle diagrams cover the design basis thermal transients (system cycling). The methodology used to develop the duty maps is based on the results of extensive testing of feedwater

nozzles by General Electric. FFWTR is analyzed by modifying the design cycles in order to gauge its effect on fatigue usage. The reduced feedwater temperature will tend to increase fatigue usage due to an increase in thermal stress.

An evaluation of the effect of FFWTR on the feedwater nozzle and feedwater sparger fatigue was performed for the following conditions:

As the last step in a 12-month fuel cycle, FFWTR to a feedwater temperature of 355°F (65°F reduction from nominal rated feedwater temperature) at rated power for 18 days was followed by a 3% per week coastdown over 12 weeks to a final power of 65%. The coastdown was initiated from a reduced feedwater temperature of 55°F. The associated feedwater temperature at the end of the coastdown was 321°F.

The analysis was performed by simulating the feedwater temperature reduction during the coastdown period in four equal increments. An appropriate maximum feedwater flow rate was assumed for each of the four increments to provide conservative results.

5.2 FEEDWATER NOZZLE FATIGUE

The original stress analysis of the feedwater nozzle showed that the maximum system cycling fatigue usage factor for the nozzle blend radius region was 0.6524 for emergency and faulted conditions (Reference 8). The usage factor for rapid cycling using the design basis (unmodified) duty map is 0.2047, providing a total 40-year usage factor of 0.8571. The usage factor for rapid cycling including FFWTR operation is 0.2796 providing a total 40-year usage factor of 0.9320. This result is based on FFWTR operation during every 12 month cycle for the life of the plant. This is equivalent to 0.0019 fatigue damage per cycle of FFWTR operation. The 40-year total usage factor remains below the ASME Code Limit of 1.0 with FFWTR operation and is thus considered acceptable. The results are summarized in Table 5-1.

The results of this analysis are based on cycling correlations developed during testing of various nozzle configurations. The fatigue results are intended to be a conservative best-estimate for the expected plant operation. A more accurate evaluation of fatigue usage could be made by considering actual plant performance.

5.3 FEEDWATER SPARGER FATIGUE

Feedwater sparger fatigue usage is calculated in the same manner as feedwater nozzle fatigue usage. However, since the feedwater sparger is not an ASME Boiler and Pressure Vessel Class 1 Code component, a fatigue analysis was not originally required. WNP-2 has a welded single thermal sleeve design which does not allow leakage of feedwater flow to occur at the safe end as do other thermal sleeve designs. This leakage flow is the primary contributor to sparger fatigue usage. Therefore, sparger fatigue usage is much less affected by changes in feedater flow and temperarture for the welded single sleeve design. The sparger is made from stainless steel material which is less susceptible to high cycle fatigue than the low alloy steel of the nozzle as evidenced by the differences in their respective fatigue curves. Small changes in flat the (high cycle) portion of the fatigue curve can cause very significant changes in fatigue usage (i.e., a relatively small change in stress can cause a very significant change in the allowable number of cycles). Thus, it becomes evident that the sparger fatigue damage is much less severe than nozzle fatigue damage during feedwater condition changes like FFWTR for the welded single sleeve design. Since the nozzle fatigue damage is so low (0.0019 per cycle), the sparger damage will be insignificant and, therefore, can be neglected.

Table 5-1

FEEDWATER NOZZLE FATIGUE USAGE

Condition	Fatigue Usage Due to FFWTR (Over Normal Operation)* Per Cycle	40-Year Fatigue Usage Factor*
Normal Operation	------	0.8571
FFWTR	0.0019	0.9320

*The total fatigue usage factor includes a system cycling usage factor of 0.6524 due to emergency and faulted conditions as given in the original stress analysis of the nozzle (Reference 8).

6. CONTAINMENT ANALYSIS

The impact of feedwater temperature reduction and increased core flow operation on the containment LOCA response was evaluated.

The results show that the containment LOCA response for ICF operation alone is bounded by the corresponding FSAR results (Reference 1). Operation with FFWTR causes a slight increase in the initial drywell pressurization rate over the rate reported in the FSAR. The calculated peak values for drywell pressure and wetwell pressure under ICF and/or FFWTR are bounded by the corresponding values for the FSAR (Chapter 6) conditions. The peak value for drywell floor differential presure (download) is bounded by the appropriate design limit of 25 psid. All other containment parameters are bounded by the results reported in the FSAR.

The LOCA-related pool swell, condensation oscillation and chugging loads were evaluated at the worst power/flow conditions during ICF/FFWTR operation. Pool boundary pressure load during pool swell under ICF/FFWTR conditions exceeds the load calculated based on FSAR conditions by less than 2.2%. However, this load and all other pool swell loads are bounded by the appropriate design loads. The condensation oscillation and chugging loads with ICF/FFWTR conditions are also bounded by the appropriate design loads.

7. OPERATING LIMITATION

Restrictions/limitations which are unique to ICF/FFWTR operation are identified below.

7.1 FEEDWATER HEATERS

The FFWTR analyses have assumed that the last-stage feedwater heater is valved out-of-service in each string of feedwater heaters (Final Feedwater Temperature Reduction $\leq$ 65°F at rated power) for exposures beyond EOC1. This may be done at any time after EOC1 whether or not ICF is used. This is done to help increase or maintaine rated power after all control rods have been withdrawn at EOC1 and was accounted for in the safety analyses in Sections 2.

7.2 OPERATING MAP

The allowable operating domain of the normal power-flow map has been increased to allow operation at 100% power up to 106% core flow. The minimum allowable power in this increased core flow region is bounded by the jet pump cavitation protection interlock as shown in Figure 1-1. The increased core flow reactor internal pressure differences and fuel bundle lift calculations were analyzed and are applicable only for reactor operation within the ICF region shown on the power flow map in Figure 1-1.

7.3 MCPR OPERATING LIMITS

Required MCPR operating limits applicable to ICF/FFWTR have been determined for WNP-2 as given in Table 2-2.

7.4 K_f FACTOR

For core flows greater than or equal to rated core flow, the K_f factor is equal to 1.0.

7.5 CONTROL RODS

The safety evaluation for ICF with FFWTR operation was performed with the assumption of an all-rods-out condition. This is defined as the condition of operation in which all control rods are fully withdrawn from the core or inserted no deeper than rod position 24.

8. REFERENCES

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SAFETY REVIEW OF WPPSS NUCLEAR PROJECT NO. 2 AT CORE FLOW CONDITIONS ABOVE RATED FLOW THROUGHOUT CYCLE 1 AND FINAL FEEDWATER TEMPERATURE REDUCTION

S. WOLF

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WPPSS NUCLEAR PROJECT NO. 2
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AND FINAL FEEDWATER TEMPERATURE REDUCTION

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ILLUSTRATIONS

ABSTRACT

A safety evaluation has been performed to show that Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 (WNP-2 or Hanford 2) can increase core flow to operate within the region of the operating map bounded by the line between 100% power, 100% core flow (100,100) and 100% power, 106% core flow (100, 106) throughout Cycle 1. WNP-2, after reaching End-of-Cycle 1 (EOC1) exposure (depletion of full-power reactivity under standard feedwater conditions) with all control rods out, can continue to operate in the region of the operating map bounded by the 106% core flow line between 100% power and the cavitation interlock power with or without the last-stage feedwater heaters valved out-of-service (Final Feedwater Temperature Reduction of ≤ 65°F at rated power).

The minimum critical power ratio (MCPR) operating limits will be changed from the values established by the Final Safety Analysis Report licensing submittal, to the appropriate values (Table 2-2) for Increased Core Flow (ICF) and Final Feedwater Temperature Reduction (FFWTR) operating conditions. All other operating limits established in the Cycle 1 licensing basis have been found to be bounding for the ICF and FFWTR operations as defined above.

ACKNOWLEDGMENTS

The analyses reported in this report were performed by the combined efforts of many individual contributors, including:

C. S. Chen, G. G. Chen, D. A. Copinger, S. K. Dhar, M. L. Gensterblum, J. K. Garrett, B. Haaberg, B. H. Koepke, M. O. Lenz, H. X. Nghiem, J. R. Pallette, R. Seetharaman, G. L. Stevens, M. W. Thompson, S. Wolf and C. T. Young

1. INTRODUCTION AND SUMMARY

This evaluation supports the operation of the Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 (WNP-2 or Hanford 2), within the increased core flow (ICF) region of the operating map as illustrated in Figure 1-1. This report presents the results of a safety evaluation for operation with ICF for Cycle 1 [up to and including End-of-Cycle 1 (EOC1) exposure]. The safety evaluation also covers operation for exposure beyond standard EOC1* with ICF and/or last-stage feedwater heaters valved out, followed by a natural reactivity coastdown bounded by 106% core flow. Final feedwater temperature reduction (FFWTR) from a normal rated power temperature of 420°F to a feedwater temperature of 355°F at 100% power and reactivity coastdown to a minimum feedwater temperature of approximately 321°F (about 65% power) should occur only at the end-of-cycle. The extended region of operation with increased core flow followed by FFWTR at end-of-cycle is bounded by the ICF region marked on the operating map in Figure 1-1.

In order to evaluate operation with ICF and FFWTR, the limiting abnormal operational transients reported in the Final Safety Analysis Report (FSAR), Reference 1, for rated flow operation were reevaluated at EOC1 at 106% core flow with and without FFWTR. The loss-of-coolant accident (LOCA), fuel loading error accident, rod drop accident, and rod withdrawal error event were also reevaluated for increased core flow operation. These events were also reevaluated for end-of-cycle operation with ICF and the last-stage feedwater heaters valved out.

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*EOC1 is defined as the core average exposure at which there is no longer sufficient reactivity to achieve rated thermal power with rated core flow, all control rods withdrawn (beyond Rod Position 24), all feedwater heaters in service and equilibrium xenon.

In addition, the effect of the increased pressure differences (due to the increased core flow) on the reactor internals components, fuel channels, and fuel bundles was also analyzed to show that the design limits will not be exceeded. The effect of the increased core flow rate on the flow-induced vibration response of the reactor internals was also evaluated to ensure that the response is within acceptable limits. The thermal-hydraulic stability was evaluated for ICF/FFWTR operation, and the increase in the feedwater nozzle and feedwater sparger usage factors due to the feedwater temperature reduction was determined. The impact of feedwater temperature reduction and increased core flow on the containment LOCA response was also analyzed.

The results of the safety evaluation show that the current technical specifications with incorporation of the MCPR limits of Table 2-2 are adequate to preclude the violation of any safety limits during operation of WNP-2 within the increased core flow region of the operating map as illustrated in Figure 1-1 for Cycle 1 and for exposures beyond EOC1 with the conditions assumed in the analysis. The ΔCPRs and the minimum critical power ratio (MCPR) operating limits for plant operation are given in Tables 2-1 and 2-2. The EOC1 Option A and Option B MCPR limits (Reference 1) will be increased to the appropriate values as shown in Table 2-2.

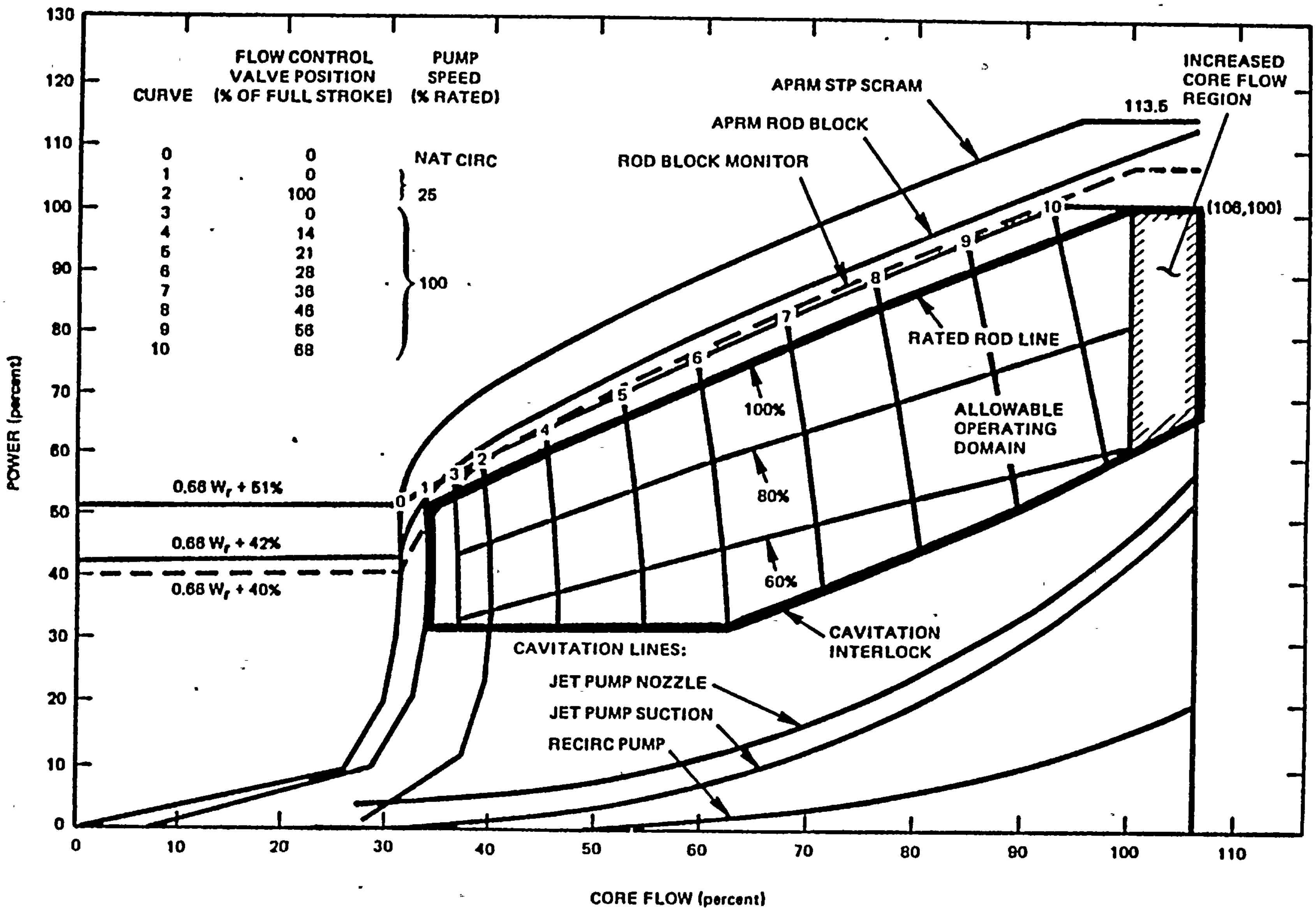

Figure 1-1. Operating Map

2. SAFETY ANALYSIS

2.1 ABNORMAL OPERATIONAL TRANSIENTS

2.1.1 Limiting Transients

The limiting abnormal operational transients analyzed in the Cycle 1 FSAR licensing submittal (Reference 1) were reevaluated for increased core flow and/or FFWTR.

Nuclear transient data for 104.5% power*, 106% core flow (104.5, 106) with and without the last-stage feedwater heaters out were developed based on the Haling method at rated power for EOC1. The nuclear data was then used to analyze the load rejection with bypass failure (LRNBP) event and the feedwater controller failure to maximum demand (FWCF) event at the (104.5, 106) conditions.

The results of the transient analyses are presented in Tables 2-1 and 2-2 with the limiting transient results previously submitted in the FSAR licensing submittal (Reference 1). The transient performance responses are presented in Figures 2-1 through 2-4. The results demonstrate that the ΔCPR values and the critical power ratio operating limits for the LRNBP and FWCF events increase compared with the corresponding FSAR values. However, the FSAR licensing submittal (Reference 1) OLCPR = 1.24 for either Option A or Option B based on the rod withdrawal error (RWE) transient is bounding for both the LRNBP and FWCF events for ICF with or without FFWTR. The current evaluation of the RWE event is presented in Section 2.1.3.

---

*All transients were analyzed using 105% steam flow. The power level corresponding to this condition will vary from 104.5% to 104.2%, depending on whether final feedwater heaters are in service. The 104.5 power level provides a 5% steam flow margin to the 100% power operating conditions to simulate eventual stretch power operation, similar to the original FSAR analyses.

Decreasing the power from the 100% rated condition along the 106% core flow line will result in an increase in transient ΔCPR for some events. This increase is less than the increase in operating CPR due to the power decrease, and, hence, such operation will not result in violation of the safety limit MCPR due to a transient (Reference 2, p. 2-12).

2.1.2 Overpressurization Analysis

The limiting transient for ASME code overpressurization analysis, main steam isolation valve (MSIV) closure with flux scram (direct scram failure), was evaluated for the extended EOC1 conditions with ICF without FFWTR (Table 2-3 and Figure 2-5). For this evaluation ICF without FFWTR is more severe than ICF with FFWTR. The ICF for the LRNBP event results in a less severe overpressure transient than MSIV closure with flux scram. The overpressurization analysis (Table 2-3) for the ICF region produced a peak vessel pressure of 1264 psig, which is below the upset code limit of 1375 psig and is, therefore, acceptable.

2.1.3 Rod Withdrawal Error

The rod withdrawal error transient was evaluated under ICF and/or FFWTR conditions. When ICF is employed, the rod block monitor (RBM) setpoint (which is flow biased) increases, giving an unacceptably high MCPR limit. Thus, the RBM should be clipped at flows greater than 100% of rated so that the ΔCPR values (Reference 1) determined without ICF apply.

2.2 FUEL LOADING ERROR

This event is not adversely affected by the increased core flow mode of operation with the last-stage feedwater heaters removed from service. The impact of ICF and/or FFWTR on ΔCPR is expected to be very small compared with the margin to the OLCPR. Thus, the FSAR ΔCPR would not be affected by this event under ICF and/or FFWTR conditions.

2.3 ROD DROP ACCIDENT

WNP-2 uses banked position withdrawal sequence (BPWS) for control rod movement. Control Rod Drop Accident (CRDA) results from BPWS plants have been statistically analyzed. The results show that, in all cases, the peak fuel enthalpy in an RDS would be much less than the corresponding design limit even with a maximum incremental rod worth corresponding to 95% probability at the 95% confidence level. Based on these results, it was proposed to the US NRC, and subsequently found acceptable, to delete the CRDA from the standard GE-BWR reload package for the BPWS plants (Reference 2, Section S.2.5.1.3 (1), Page 2-53). Hence, the CRDA is not specifically analyzed for WNP-2.

2.4 LOSS-OF-COOLANT ACCIDENT (LOCA) ANALYSIS

LOCA analysis performed for WNP-2 shows that operation with ICF without FFWTR bounds operation with ICF and FFWTR.

The effect of increased core flow on LOCA analyses is not significant because the parameters which most strongly affect the calculated peak cladding temperature (PCT), i.e., high power node boiling transition time and core reflooding time, have been shown to be relatively insensitive to increased core flow.

Results of the LOCA analysis performed show that the PCT for ICF increases by less than 5°F throughout the break spectrum compared to the rated core flow condition.

Therefore, it is concluded that the LOCA PCT is acceptable and that the current maximum average planar linear heat generation rates (MAPLHGRs) for WNP-2 are applicable for ICF.

2.5 THERMAL-HYDRAULIC STABILITY

The General Electric Company has established stability criteria to demonstrate compliance to requirements set forth in 10CFR50 Appendix A, General Design Criteria (GDC). 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 all licensed GE BWR fuel designs including those fuels contained in the General Electric Standard Application for Reactor Fuel (GESTAR, Reference 2) is demonstrated on a generic basis in Reference 3 (for operation in the normal as well as the extended operating domain with ICF and FFWTR). The NRC has reviewed and approved this in Reference 4; therefore, a specific analysis for each cycle is not required. The WNP-2 Cycle 1 core contains licensed GE BWR initial core and, hence, the generic evaluation in Reference 3 is applicable to WNP-2.

For operation in the ICF region, the stability margin (defined by the core decay ratio) is increased as flow increases for a given power. ICF operation is bounded by the fuel integrity analyses in Reference 3.

Similarly, operation in the FFWTR mode is bounded by the fuel integrity analyses in Reference 3. In general, the effect of reduced feedwater temperature results in a higher initial CPR which yields even larger margins than those reported in Reference 3. The fuel integrity analyses are independent of the stability margin, since the reactor is already assumed to be in limit cycle oscillations. Reference 3 also demonstrates that even if neutron flux limit cycle oscillations did occur just below the neutron flux scram setpoint, fuel design limits are not exceeded for those GE BWR fuel designs contained in General Electric Standard Application for Reactor Fuel (GESTAR, Reference 2). These evaluations demonstrate that substantial thermal/mechanical margin is available for the GE BWR fuel designs even in the unlikely event of very large oscillations.

To provide assurance that acceptable plant performance is achieved during operation in the least stable region of the power/flow map, as well as during all plant maneuvering and operating states, a generic set of operating recommendations has been developed as set forth in Reference 5 and communicated to all GE BWRs. These recommendations instruct the operator on how to reliably detect and suppress limit cycle neutron flux oscillations should they occur. The recommendations were developed to conservatively bound the expected performance of all current product lines and are applicable to operation with FFWTR (feedwater temperature of approximately 355°F at rated power).

Table 2-1

CORE-WIDE TRANSIENT ANALYSIS RESULTS AT ICF AND/OR FFWTR

Transient[a] Description	Figure Number	Power (% NBR)	Flow (% NBR)	Rated Feedwater Temperature Reduction[b] (°F)	Maximum Neutron Flux (% NBR)	Maximum Core Ave. Surface Heat Flux (% Initial)	Maximum Dome Press (psig)	Maximum Vessel Press (psig)	Maximum Steam Line Press (psig)	ΔCPR[c]
LRNBP	Ref. 1	104.4	100	0	236.4	107.8	1173	1202	1168	0.09
LRNBP	2.1	104.2	106	0	252.4	108.8	1172	1203	1168	0.11
LRNBP	2.2	104.5	106	65	243.2	108.8	1160	1191	1157	0.11
FWCF	Ref. 1	104.4	100	0	154.3	108.7	1148	1177	1140	0.08
FWCF	2.3	104.2	106	0	163.7	109.1	1145	1177	1141	<0.13
FWCF	2.4	104.5	106	65	174.7	113.9	1138	1166	1135	0.13

a. LRNBP = Load rejection with bypass failure, FWCF = feedwater controller failure to maximum demand,

b. Reduction of feedwater temperature from nominal rated feedwater temperature (420°F) and at rated conditions.

c. ΔCPR based on initial CPR which yields MCPR = 1.06; uncorrected for Options A and B.

Table 2-2

REQUIRED MCPR OPERATING LIMITS AT ICF AND/OR FFWTR

Transient Description[a]	Initial Core Power (% NBR)	Initial Core Flow (% NBR)	ΔCPR[b]	$OLCPR_A$[c,e]	$OLCPR_B$[d,e]
LRNBP (FSAR)	104.4	100	0.09	1.20	1.12
LRNBP[f]	104.2	106	0.11	1.22	1.14
FWCF (FSAR)	104.4	100	0.08	1.19	1.16
FWCF[g]	104.5	106	0.13	1.24	1.21
			ΔCPR	OLCPR	
RWE (FSAR)	104.4	100	0.18	1.24[h]	

a. LRNBP = Load rejection with bypass failure, FWCF = feedwater controller failure at maximum demand, RWE = rod withdrawal error.

b. ODYN results without adjustment factors, based on initial CPR which yields an MCPR = 1.06.

c. Includes Option A adjustment factors.

d. Includes Option B adjustment factors.

e. Option A and B adjustment factors are specified in the NRC safety evaluation report on ODYN (NEDO-24154 and NEDE-24154P).

f. For load rejection with bypass failure, ICF w/o FFWTR bounds ICF with FFWTR.

g. For feedwater controller failure to maximum flow demand, ICF with FFWTR bounds ICF w/o FFWTR.

h. Required OLCPR using either Option A or Option B adjustment factor with rod block monitor of 106% at rated flow

Table 2-3

OVERPRESSURIZATION ANALYSIS RESULTS

Transient	Initial Power (%)	Initial Flow (%)	Maximum Vessel Pressure (psig)	Figure No.
MSIV Closure - Flux Scram (FSAR)	104.3	100	1266	Reference 1
MSIV Closure - Flux Scram (ICF w/o FFWTR)	104.2	106	1264	Figure 2-5

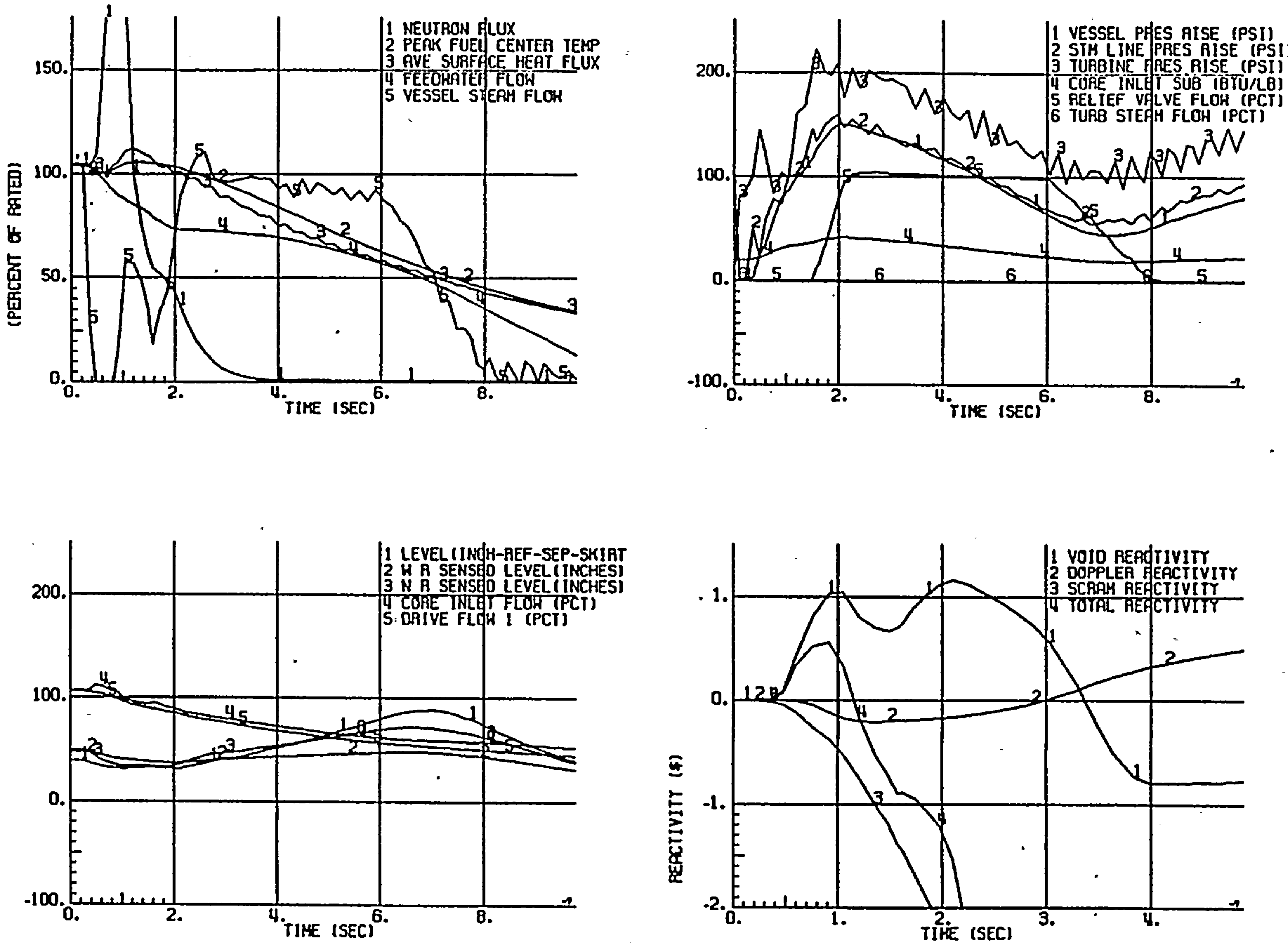

Figure 2-1. Generator Load Rejection with Bypass Failure at 104.2% Power, 106% Flow and Normal Feedwater Temperature

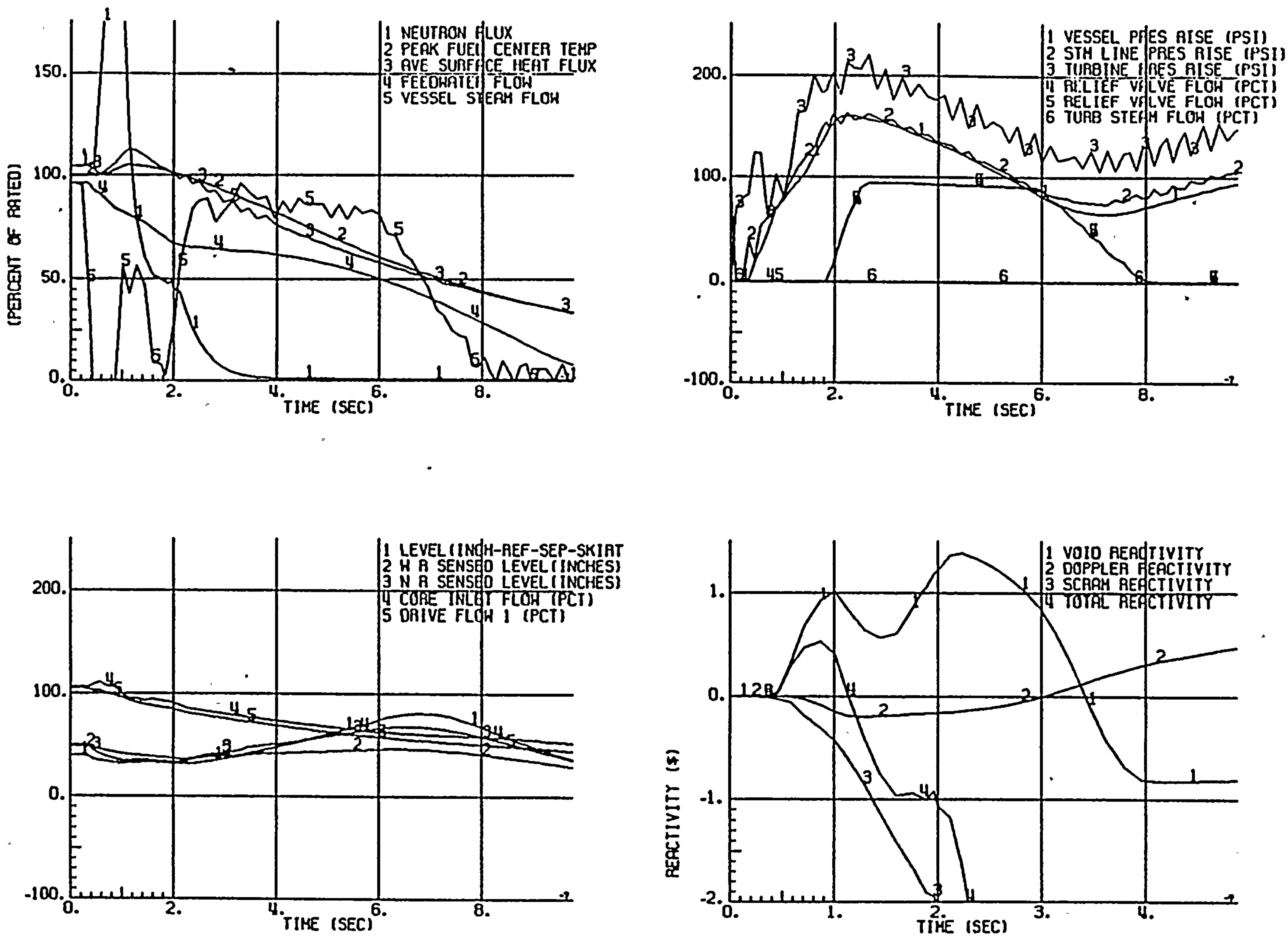

Figure 2.2. Generator Load Rejection With Bypass Failure at 104.5% Power, 106% Flow and Reduced Feedwater Temperature of 65^{O}F at Rated Power.

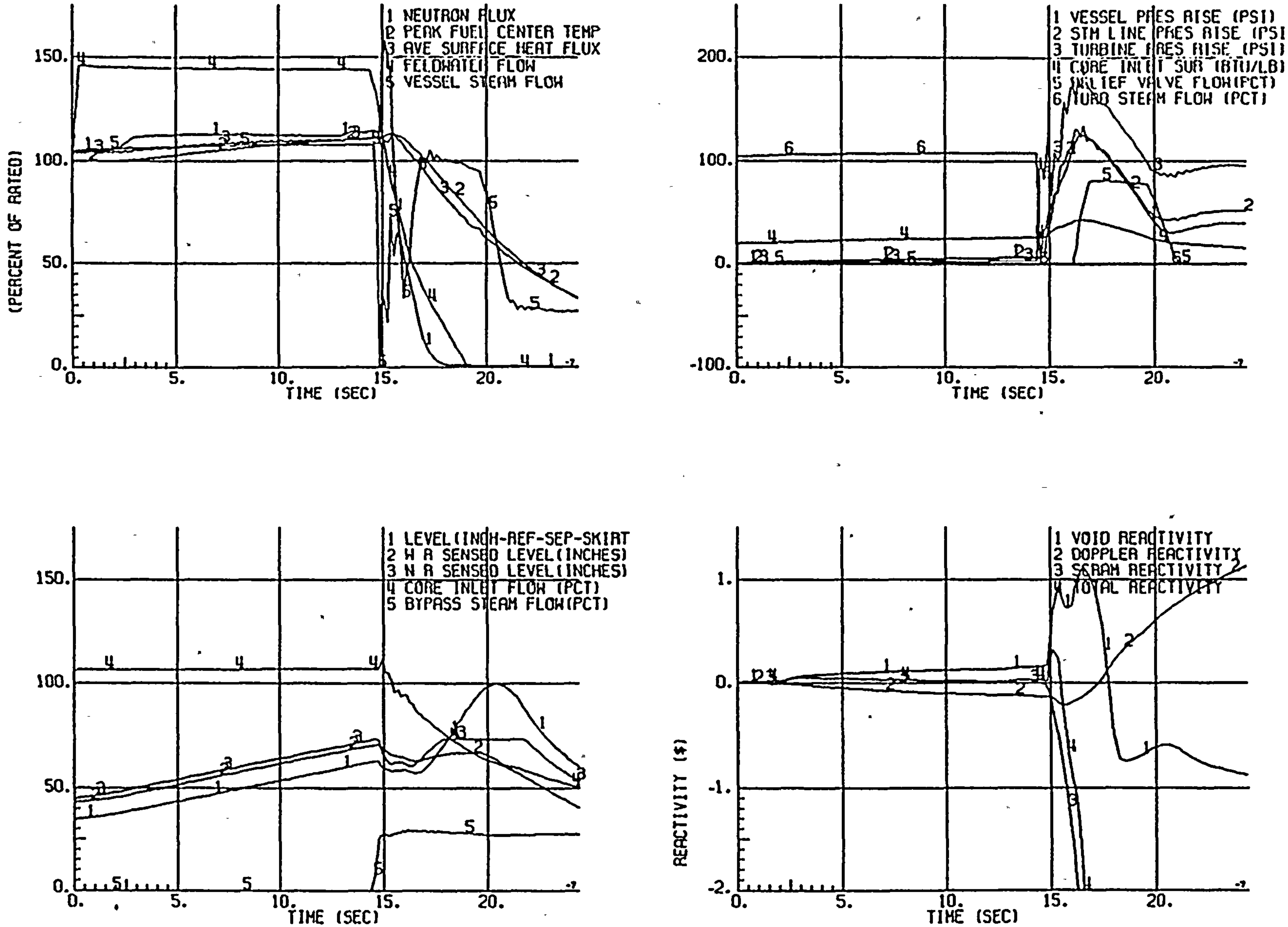

Figure 2-3. Feedwater Controller Failure, Maximum Demand at 104.2% Power, 106% Flow and Normal Feedwater Temperature

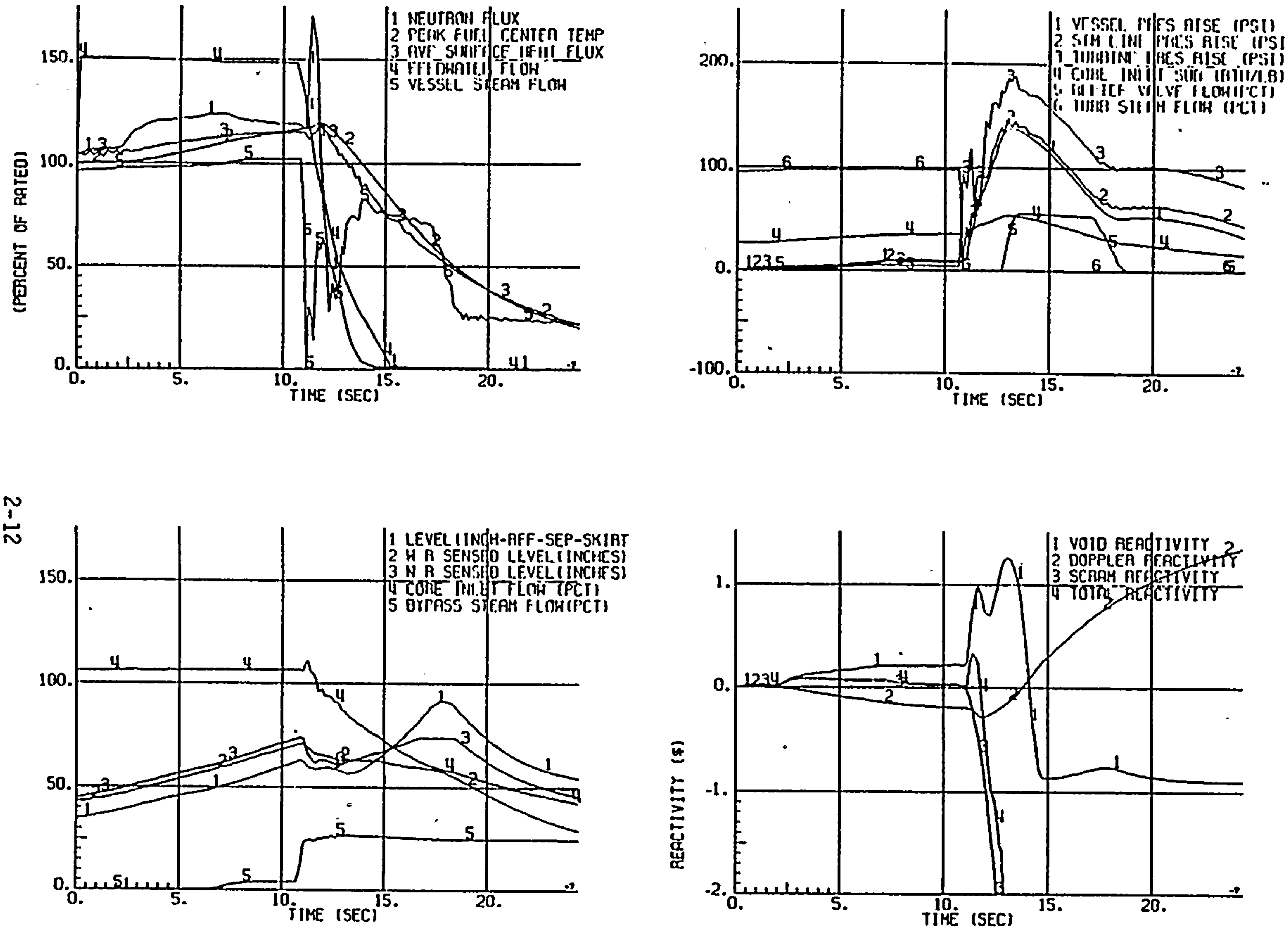

Figure 2-4. Feedwater Controller Failure, Maximum Demand, at 104.5% Power, 106% Flow and Reduced Feedwater Temperature of 65°F at Rated Power

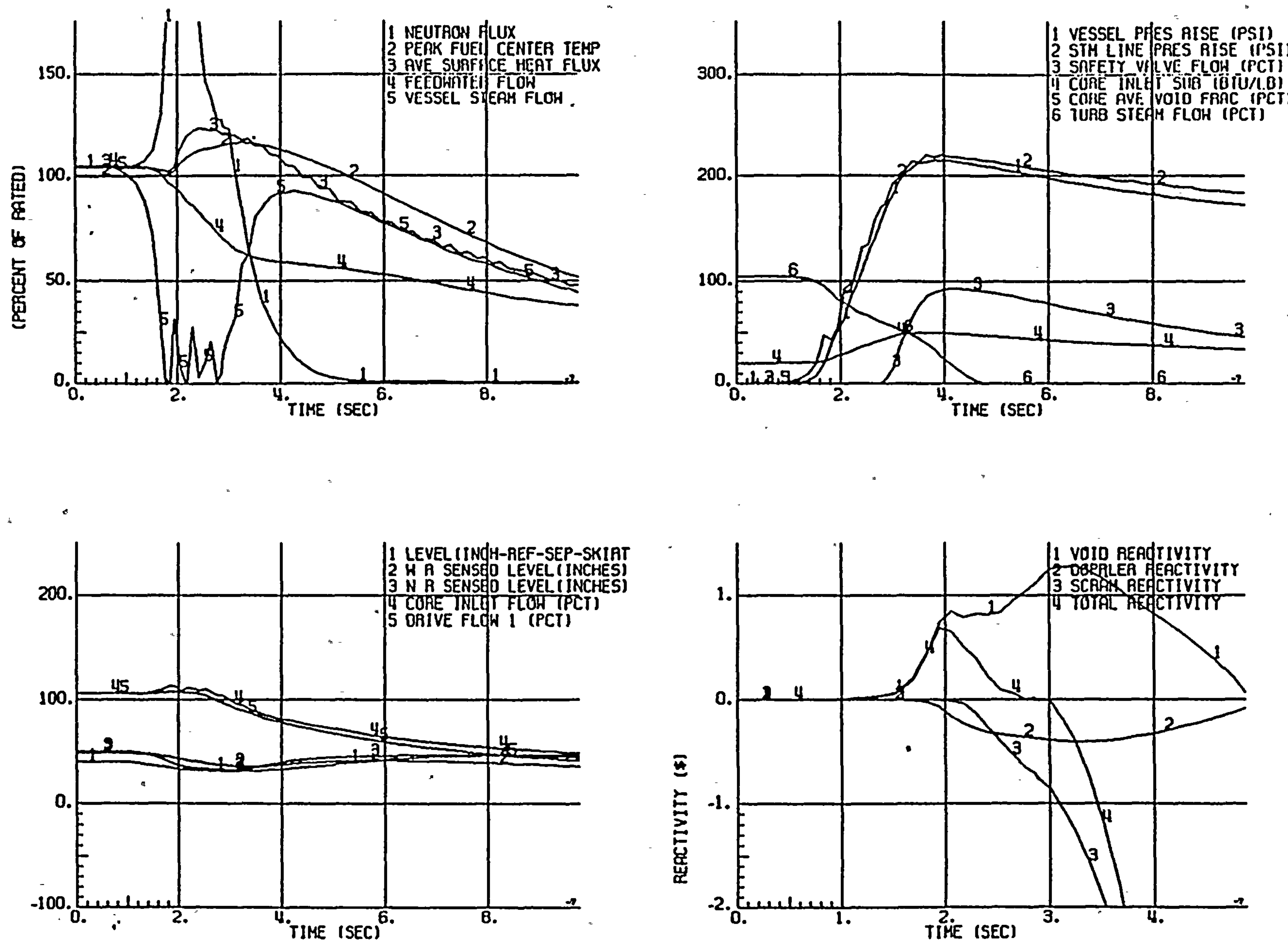

Figure 2-5. MSIV Closure, Flux Scram, at 104.2% Power, 106% Flow and Normal Feedwater Temperature

3. MECHANICAL EVALUATION OF REACTOR INTERNALS AND FUEL ASSEMBLY

3.1 LOADS EVALUATION

Evaluations were performed to determine bounding acoustic and flow-induced loads, reactor internal pressure difference loads and fuel-support loads for ICF and/or FFWTR operation.

Acoustic loads are lateral loads on the vessel internals that result from propagation of the decompression wave created by a sudden recirculation suction line break. The acoustic loading on vessel internals is proportional to the total pressure wave amplitude in the vessel recirculation outlet nozzle. The total pressure amplitude is the sum of the initial pressure subcooling plus the experimentally determined pressure undershoot below saturation pressure. FFWTR operation increases the expected acoustic loads because this downcomer subcooling increases and, therefore, the total pressure wave amplitude increases. The high velocity flow patterns in the downcomer resulting from a recirculation suction line break also create lateral loads on the reactor vessel internals. These loads are proportional to the square of the critical mass flow rate out of the break. The additional subcooling in the downcomer resulting from FFWTR operation leads to an increase in the critical flow and, therefore, to a corresponding increase in the flow-induced loads. The reactor internals most impacted by acoustic and flow-induced loads are the shroud, shroud support and jet pumps.

A reactor internals pressure difference analysis was performed for the ICF region. The increased reactor internal pressure differences across the reactor internals were generated for the maximum core flow at normal, upset, and faulted conditions for the reactor internal impact evaluation.

Fuel-support loads and fuel bundle lift for WNP-2 were evaluated based on results from probabilistic fuel lift analyses performed at 106% of rated core flow following the procedures of Reference 6. Fuel-support loads and fuel bundle lift were evaluated for upset, faulted and fatigue load combinations.

It was shown that the fuel bundle lift is a small fraction of the applicable design criteria (established in the NRC Safety Evaluation Report to Reference 6) for the faulted event.

3.2 LOADS IMPACT

3.2.1 Reactor Internals

The reactor internals most affected by ICF and/or FFWTR operation are the core plate, shroud support, shroud, top guide, shroud head, steam dryer, control rod guide tube, control rod drive housing and jet pump. These and other components were evaluated using the bounding loads, discussed in Section 3.1, under normal, upset, emergency and faulted conditions. It is concluded that the stresses produced in these and other components are within the allowable design limits given in the Final Safety Analysis Report (Chapter 3 and 4) or the ASME Code, Section III, Subsection NG.

3.2.2 Fuel Assemblies

The fuel assemblies, including fuel bundles and channels, were evaluated for increased core flow operation considering the effects of loads discussed in Section 3.1 under normal, upset,faulted and fatigue load combinations. Results of the evaluation demonstrate that the fuel assemblies are adequate to withstand ICF effects to 106% rated flow.

The fuel channels were also evaluated under normal, upset, emergency and faulted conditions for increased core flow (Reference 7). The channel wall pressure differentials were found to be within the allowable design values .

4. FLOW-INDUCED VIBRATION

To ensure that the flow-induced vibration response of the reactor internals is acceptable, a single reactor of each product line and size undergoes an extensive vibration test during initial plant startup. After analyzing the results of such tests and assuring that all responses fall within acceptable limits of the established criteria, the reactor is classified as a valid prototype in accordance with Regulatory Guide 1.20. All other reactors of the same product line and size undergo a less rigorous confirmatory test to assure similarity to the base test. The acceptance criteria used for vibration assessment is based on a maximum allowable alternating stress intensity of 10,000 psi.

The increased core flow vibration analysis was performed by analyzing the startup test vibration data for the valid prototype plant (BWR/5-251 Tokai 2). Based on the results of the analysis and a review of the test data, the reactor internals response to flow-induced vibration is expected to be within acceptable limits for plant operation in the ICF region (region bounded as shown on the power flow map, Figure 1-1).

5. FEEDWATER NOZZLE AND FEEDWATER SPARGER FATIGUE USAGE

5.1 METHOD AND ASSUMPTIONS

The fatigue experienced by the feedwater nozzle and feedwater sparger results from two phenomena: system cycling and rapid cycling. System cycling is caused by major temperature changes associated with system transients. The system cycle stresses are based on limiting cycles that use the maximum temperature range possible to show expected worst conditions. These transients are identified on thermal cycle diagrams. Thermal stresses due to these transients are calculated by determining inner and outer metal surface temperatures using finite element analysis. Fatigue usage is determined by dividing the number of design cycles for each transient by the number of allowable cycles for each stress calculated. Cumulative system fatigue usage is determined by summing all of the respective transient fatigue usage factors.

Rapid cycling is caused by small, high frequency temperature fluctuations caused by mixing of relatively colder nozzle annulus water with the reactor coolant. The colder water impinging the nozzle bore originates from the boundary layer of colder water formed by heat transfer through the thermal sleeve. The mixing region extends from the feedwater nozzle surface region to the feedwater sparger surface; therefore, rapid cycling applies to both of these components. Once thermal stress due to rapid cycling is determined, fatigue usage is calculated and the results are added to the cumulative system cycling usage factor to obtain the total usage factor.

The introduction of FFWTR will cause a change in calculated rapid cycling fatigue only. This is because the system transient is very mild (small temperature change and relatively long duration) and is bounded by the original design basis thermal stress analysis. General Electric has developed standardized rapid cycling duty maps for each BWR plant that cover the design basis rapid cycles in the same manner that thermal cycle diagrams cover the design basis thermal transients (system cycling). The methodology used to develop the duty maps is based on the results of extensive testing of feedwater

nozzles by General Electric. FFWTR is analyzed by modifying the design cycles in order to gauge its effect on fatigue usage., The reduced feedwater temperature will tend to increase fatigue usage due to an increase in thermal stress.

An evaluation of the effect of FFWTR on the feedwater nozzle and feedwater sparger fatigue was performed for the following conditions:

As the last step in a 12-month fuel cycle, FFWTR to a feedwater temperature of 355°F (65°F reduction from nominal rated feedwater temperature) at rated power for 18 days was followed by a 3% per week coastdown over 12 weeks to a final power of 65%. The coastdown was initiated from a reduced feedwater temperature of 55°F. The associated feedwater temperature at the end of the coastdown was 321°F.

The analysis was performed by simulating the feedwater temperature reduction during the coastdown period in four equal increments. An appropriate maximum feedwater flow rate was assumed for each of the four increments to provide conservative results.

5.2 FEEDWATER NOZZLE FATIGUE

The original stress analysis of the feedwater nozzle showed that the maximum system cycling fatigue usage factor for the nozzle blend radius region was 0.6524 for emergency and faulted conditions (Reference 8). The usage factor for rapid cycling using the design basis (unmodified) duty map is 0.2047, providing a total 40-year usage factor of 0.8571. The usage factor for rapid cycling including FFWTR operation is 0.2796 providing a total 40-year usage factor of 0.9320. This result is based on FFWTR operation during every 12 month cycle for the life of the plant. This is equivalent to 0.0019 fatigue damage per cycle of FFWTR operation. The 40-year total usage factor remains below the ASME Code Limit of 1.0 with FFWTR operation and is thus considered acceptable. The results are summarized in Table 5-1.

The results of this analysis are based on cycling correlations developed during testing of various nozzle configurations. The fatigue results are intended to be a conservative best-estimate for the expected plant operation. A more accurate evaluation of fatigue usage could be made by considering actual plant performance.

5.3 FEEDWATER SPARGER FATIGUE

Feedwater sparger fatigue usage is calculated in the same manner as feedwater nozzle fatigue usage. However, since the feedwater sparger is not an ASME Boiler and Pressure Vessel Class 1 Code component, a fatigue analysis was not originally required. WNP-2 has a welded single thermal sleeve design which does not allow leakage of feedwater flow to occur at the safe end as do other thermal sleeve designs. This leakage flow is the primary contributor to sparger fatigue usage. Therefore, sparger fatigue usage is much less affected by changes in feedater flow and temperarture for the welded single sleeve design. The sparger is made from stainless steel material which is less susceptible to high cycle fatigue than the low alloy steel of the nozzle as evidenced by the differences in their respective fatigue curves. Small changes in flat the (high cycle) portion of the fatigue curve can cause very significant changes in fatigue usage (i.e., a relatively small change in stress can cause a very significant change in the allowable number of cycles). Thus, it becomes evident that the sparger fatigue damage is much less severe than nozzle fatigue damage during feedwater condition changes like FFWTR for the welded single sleeve design. Since the nozzle fatigue damage is so low (0.0019 per cycle), the sparger damage will be insignificant and, therefore, can be neglected.

Table 5-1

FEEDWATER NOZZLE FATIGUE USAGE

Condition	Fatigue Usage Due to FFWTR (Over Normal Operation)* Per Cycle	40-Year Fatigue Usage Factor*
Normal Operation	------	0.8571
FFWTR	0.0019	0.9320

*The total fatigue usage factor includes a system cycling usage factor of 0.6524 due to emergency and faulted conditions as given in the original stress analysis of the nozzle (Reference 8).

6. CONTAINMENT ANALYSIS

The impact of feedwater temperature reduction and increased core flow operation on the containment LOCA response was evaluated.

The results show that the containment LOCA response for ICF operation alone is bounded by the corresponding FSAR results (Reference 1). Operation with FFWTR causes a slight increase in the initial drywell pressurization rate over the rate reported in the FSAR. The calculated peak values for drywell pressure and wetwell pressure under ICF and/or FFWTR are bounded by the corresponding values for the FSAR (Chapter 6) conditions. The peak value for drywell floor differential presure (download) is bounded by the appropriate design limit of 25 psid. All other containment parameters are bounded by the results reported in the FSAR.

The LOCA-related pool swell, condensation oscillation and chugging loads were evaluated at the worst power/flow conditions during ICF/FFWTR operation. Pool boundary pressure load during pool swell under ICF/FFWTR conditions exceeds the load calculated based on FSAR conditions by less than 2.2%. However, this load and all other pool swell loads are bounded by the appropriate design loads. The condensation oscillation and chugging loads with ICF/FFWTR conditions are also bounded by the appropriate design loads.

7. OPERATING LIMITATION

Restrictions/limitations which are unique to ICF/FFWTR operation are identified below.

7.1 FEEDWATER HEATERS

The FFWTR analyses have assumed that the last-stage feedwater heater is valved out-of-service in each string of feedwater heaters (Final Feedwater Temperature Reduction $\leq$ 65°F at rated power) for exposures beyond EOC1. This may be done at any time after EOC1 whether or not ICF is used. This is done to help increase or maintaine rated power after all control rods have been withdrawn at EOC1 and was accounted for in the safety analyses in Sections 2.

7.2 OPERATING MAP

The allowable operating domain of the normal power-flow map has been increased to allow operation at 100% power up to 106% core flow. The minimum allowable power in this increased core flow region is bounded by the jet pump cavitation protection interlock as shown in Figure 1-1. The increased core flow reactor internal pressure differences and fuel bundle lift calculations were analyzed and are applicable only for reactor operation within the ICF region shown on the power flow map in Figure 1-1.

7.3 MCPR OPERATING LIMITS

Required MCPR operating limits applicable to ICF/FFWTR have been determined for WNP-2 as given in Table 2-2.

7.4 K_f FACTOR

For core flows greater than or equal to rated core flow, the K_f factor is equal to 1.0.

7.5 CONTROL RODS

The safety evaluation for ICF with FFWTR operation was performed with the assumption of an all-rods-out condition. This is defined as the condition of operation in which all control rods are fully withdrawn from the core or inserted no deeper than rod position 24.

8. REFERENCES

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