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SAFETY REVIEW OF PILGRIM NUCLEAR POWER STATION, UNIT NO. 1 AT CORE FLOW CONDITIONS ABOVE RATED FLOW THROUGHOUT CYCLE 6

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OF
PILGRIM NUCLEAR POWER STATION
UNIT NO. 1
AT CORE FLOW CONDITIONS ABOVE RATED FLOW
THROUGHOUT CYCLE 6

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

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ABSTRACT

A safety evaluation has been performed to show that Pilgrim 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, 107.5% core flow (100,107.5) throughout Cycle 6. Pilgrim, after reaching EOC6 exposure (depletion of full-power reactivity under standard feedwater conditions) with all power rods out, can continue to operate in the region of the operating map bounded by the constant recirculation pump speed line between 100% power, 107.5% flow (100,107.5), and 80% power, 112.5% flow (80,112.5), and constant core flow line to 50% power, 112.5% flow (50,112.5), with the last-stage feedwater heaters valved out-of-service.

The minimum critical power ratio (MCPR) operating limits will be changed from the values established by the Reload-5, Cycle 6 reload licensing submittal (Y1003J01A28, Rev. 2, Feb. 1983), to the appropriate values (Table 2-3) depending on the operating conditions. All other operating limits established in the Reload-5 licensing basis have been found to be bounding for the increased core flow region.

1. INTRODUCTION AND SUMMARY

This report presents the results of a safety evaluation for operation of the Pilgrim Nuclear Power Station with increased core flow (ICF) for Cycle 6, and for exposure beyond standard end-of-cycle 6 (EOC6)* with last stage feedwater heaters valved out subsequent to ICF. This evaluation supports the operation within the region of the operating map bounded by ABCDE on the operating map in Figure 1-1. The conditions of operation which were evaluated were those of continued 100% power operation beyond the standard EOC6 conditions with 107.5% core flow followed by a reduction of approximately 43°F in the feedwater temperature followed by a natural reactivity coastdown to 80% power under conditions bounded by 112.5% core flow. The evaluation also includes continued operation in the region of the operating map bounded by the constant core flow line between 80% power, 112.5% core flow (80,112.5) and 50% power, 112.5% core flow (50,112.5). The extended region of operation with increased core flow followed by final feedwater temperature reduction (FFWTR) is bounded by ABCDE on the operating map in Figure 1-1.

In order to evaluate operation with ICF and FFWTR, the limiting abnormal operational transients reported in Reference 1 for rated flow operation were reevaluated. 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.

The effect of the increased pressure differences (due to the increased core flow) on the reactor internal components, fuel channels, and fuel bundles was also analyzed to show that the design limits will not be exceeded. The effect of the increased flow rate on the flow-induced vibration response of the reactor internals was also evaluated to ensure that the response was within acceptable limits. The thermal-hydraulic stability was evaluated for

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*EOC6 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, all feedwater heaters in service and equilibrium xenon.

increased core flow operation, and the increase in the feedwater nozzle usage factor 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 2-3 are adequate to preclude the violation of any safety limits during operation of Pilgrim Unit 1 within the region bounded by ABCDE on the operating map in Figure 1-1 for Cycle 6 and for exposures beyond EOC6 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 2-2 and 2-3, respectively. The MCPR limits will be raised from 1.46 (8x8) and 1.49 (P8x8R) in Reference 1 to the appropriate values (Table 2-3) depending on the operating conditions.

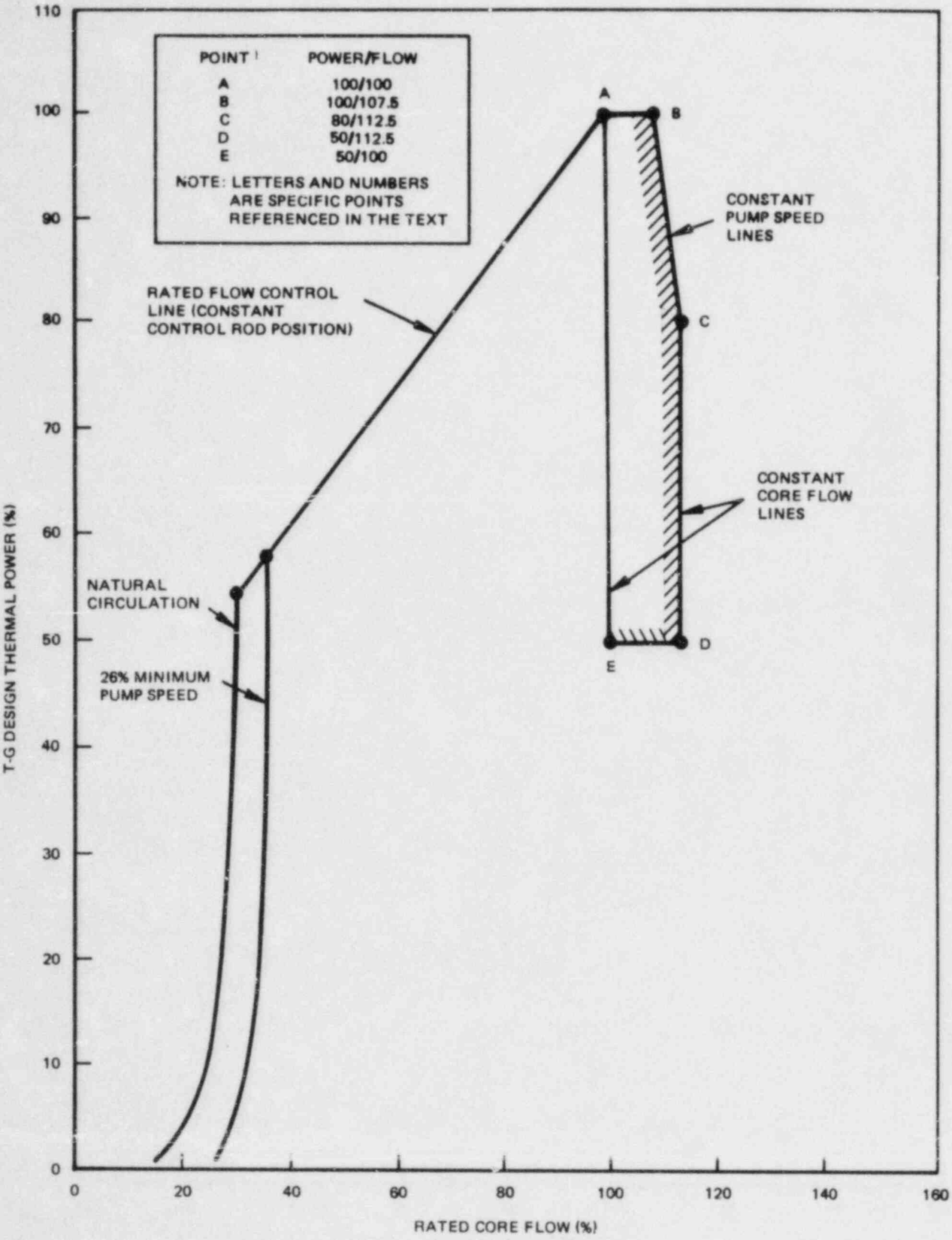

Figure 1-1. Operating Map

2. SAFETY ANALYSIS

2.1 ABNORMAL OPERATIONAL TRANSIENTS

2.1.1 Limiting Transients

The limiting abnormal operational transients analyzed in the Reload-5, Cycle 6 reload licensing submittal (Reference 1) were reevaluated for increased core flow followed by final feedwater temperature reduction as follows.

Nuclear transient data for 100% power, 107.5% core flow (100,107.5) with and without the last stage feedwater heaters out (approximately 43°F reduction in feedwater temperature) were developed based on recent EOC6 as-burned core projections. This nuclear data was then used to analyze the load rejection without bypass event (LR w/o BP) and the feedwater controller failure (FWCF) event at the (100,107.5) conditions.

The results of the transient analyses are presented in Tables 2-1, 2-2 and 2-3 along with the transient results contained in the Reload-5 licensing submittal (Reference 1). As shown in Tables 2-1 and 2-2, the ΔCPR for the (100,107.5) condition with and without feedwater temperature reduction exceeds the license basis ΔCPR (Reference 1) for the LR w/o BP event used to set the operating limits. Therefore, the current technical specification MCPR operating limits described in Reference 1 should be modified to incorporate these changes. The transient responses are presented in Figures 2-1 through 2-5.

The results of core-wide ΔCPR for Options A and B with fuel types of 8x8 and P8x8R are shown in Table 2-2. The analyses demonstrate that the final MCPR must be increased from 1.46 (8x8) and 1.49 (P8x8R) in the Reload-5, Cycle 6 reload submittal (Reference 1) to 1.49 (8x8) and 1.52 (P8x8R), respectively, for Option A operation.

Increasing the core flow from 107.5% to 112.5% of rated along the constant pump speed line as power decreases (line BC in Figure 1-1) may result in a slight increase in transient ΔCPR. This increase is insignificant compared to the increase in operating MCPR 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 overpressurization analysis, main steam isolation valve (MSIV) closure with flux scram, was evaluated for the extended EOC6 conditions with ICF and with and without FFWTR (Table 2-4 and Figures 2-4 and 2-5). The ICF without FFWTR will result in a slightly more severe overpressure transient for the MSIV closure event compared to the Reference 1 basis. The ICF for the LR w/o BP events results in a less severe overpressure transient (compared to the MSIV closure event) as shown in Table 2-1. The overpressurization analysis (Table 2-4) for the ICF region produced a peak vessel pressure of 1365 psig, which is below the upset code limit of 1375 psig and is, therefore, not limiting.

2.1.3 Rod Withdrawal Error

The rod withdrawal error transient evaluated as part of the Reference 1 analysis was performed at conservative operating conditions (maximum core reactivity with the maximum worth rod being the error rod). Complete withdrawal of the error rod would yield a ΔCPR which is bounded by the Reference 1 Option B MCPR limits. Consequently, the limits specified in Table 2-3 are bounding for this event.

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 resulting lower initial steam flow and inlet enthalpy results in a less severe event. Thus, the results reported in the Reload-5 licensing submittal (Reference 1) are bounding for operation in the increased core flow region.

2.3 ROD DROP ACCIDENT

This event is a startup accident evaluated at minimum core flow, and thus the increased core flow operation is a second-order effect. The results reported in the Reference 1 licensing submittal are bounding for operation in the increased core flow region.

2.4 LOCA ANALYSIS

A discussion of the LOCA calculations performed for increased core flow operation for Pilgrim 1 is presented in Reference 3.

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.

This LOCA analysis is documented in Reference 3, which concludes that PCT, for an increased core flow condition, varies by <10°F throughout the break spectrum compared to the rated core flow condition.

Therefore, it is concluded that the LOCA analysis and maximum average planar linear heat generation rates (MAPLHGRs) determined for the Pilgrim Unit 1 Reload-5 core (Reference 3) are unchanged for use in the increased core flow region of the operating map.

Table 2-1

CORE-WIDE TRANSIENT ANALYSIS RESULTS

Transient	Exposure	Power (%)	Flow (%)	ϕ (%)	Q/A (%)	P_{SL} (psig)	P_V (psig)	ΔCPR[a] 8x8	ΔCPR[a] P8x8R	Plant Response
LR w/o BP (Cycle 6 Reload)	EOC6	100	100	597	123	--	--	0.33	0.36	(Reference 1)
LR w/o BP[b]	EOC6+114 MWd/t	100	107.5	645.1	124.2	1305	1317	0.36	0.39	Figure 2-1
LR w/o BP[c]	EOC6+384 MWd/t	100	107.5	608.7	122.9	1293	1306	0.34	0.37	Figure 2-2
FWCF (Licensing Submittal)	EOC6	100	100	385	123	--	--	0.28	0.30	(Reference 1)
FWCF[c]	EOC6+384 MWd/t	100	107.5	384.3	123.4	1186	1217	0.28	0.31	Figure 2-3

NOTES:

[a]Uncorrected for Options A and B.

[b]Feedwater heaters in service.

[c]Last-stage feedwater heater valved out-of-service.

Table 2-2

EOC6 CORE-WIDE ΔCPR[a] RESULTS

Transient	Option A 8x8	Option A P8x8R	Option B 8x8	Option B P8x8R
LR w/o BP (100% power, 100% flow, Reference 1)	0.39	0.42	0.34	0.37
LR w/o BP[b]	0.42	0.45	0.37	0.40
LR w/o BP[c]	0.40	0.43	0.35	0.38
FWCF (100% power, 100% flow, Reference 1)	0.34	0.36	0.26	0.28
FWCF[c] (100% power, 107.5% flow)	0.34	0.37	0.25	0.28

[a]100% power, 107.5% flow at EOC6 exposure conditions.

[b]Feedwater heaters in service.

[c]Last-stage feedwater heater valve out-of-service (FFWTR).

Table 2-3

MCPR OPERATING LIMITS AT INCREASED CORE FLOW
FOR PILGRIM UNIT 1, EOC6

	Option A		Option B	
Transient	8x8	P8x8R	8x8	P8x8R
LR w/o BP (100% power, 100% flow, Reference 1)	1.46	1.49	1.41	1.44
LR w/o BP (100% power, 107.5% flow, FW heater in service)	1.49	1.52	1.44	1.47
LR w/o BP (100% power, 107.5% flow, FW heater out-of-service)	1.47	1.50	1.42	1.45

Table 2-4

OVERPRESSURIZATION ANALYSIS

Transient	Power (%)	Flow (%)	P_{SL} (psig)	P_V (psig)	Plant Response
MSIV Closure - Flux Scram (Licensing Submittal)	100	100	1346	1360	(Reference 1)
MSIV Closure - Flux Scram (ICF w/o FFTWR)	100	107.5	1352	1365	Figure 2-4
MSIV Closure - Flux Scram (ICF with FFTWR)	100	107.5	1336	1349	Figure 2-5

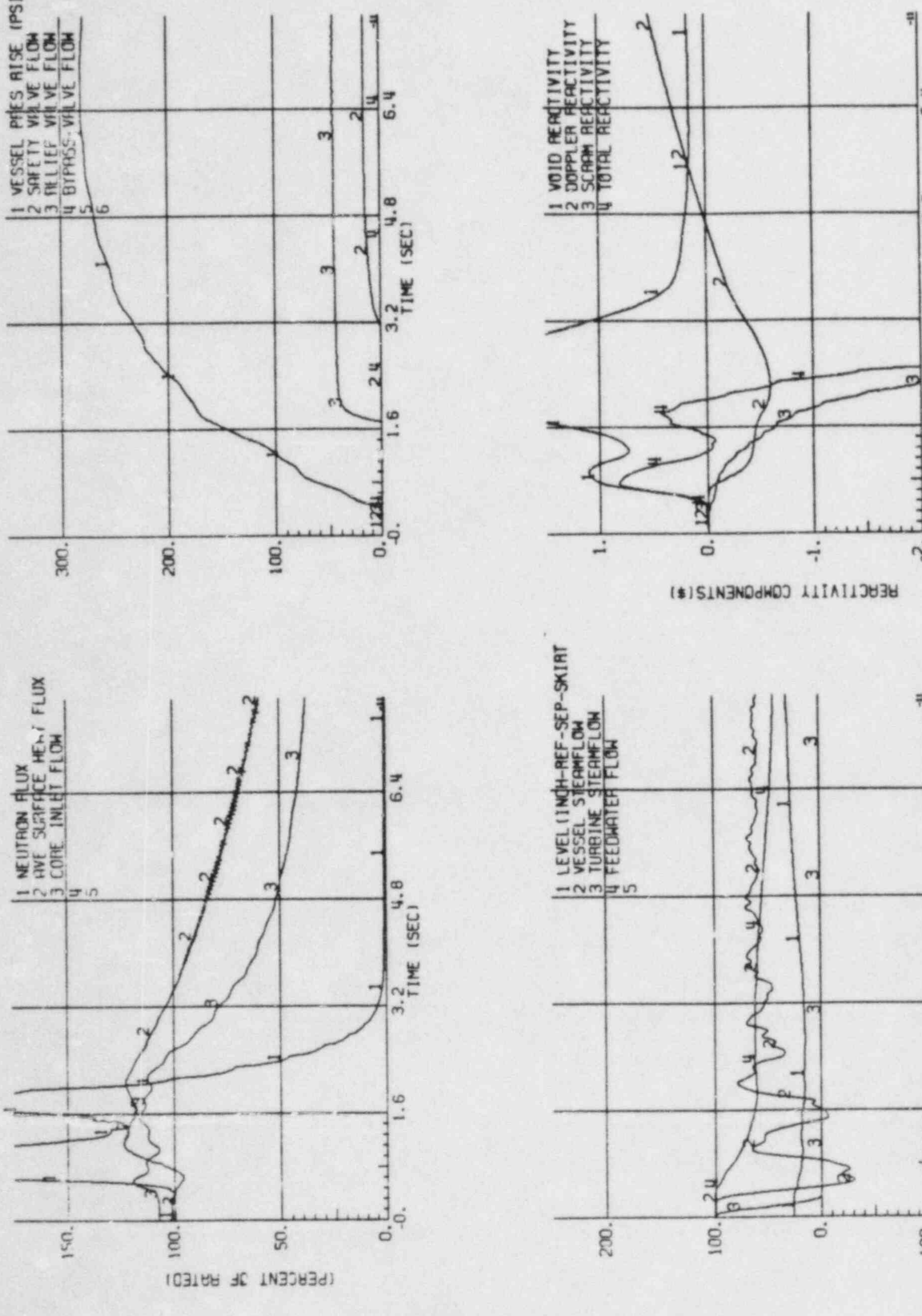

Figure 2-1. Generator Load Rejection, Without Bypass (100% Power, 107.5% Flow, with Normal FW Temperature)

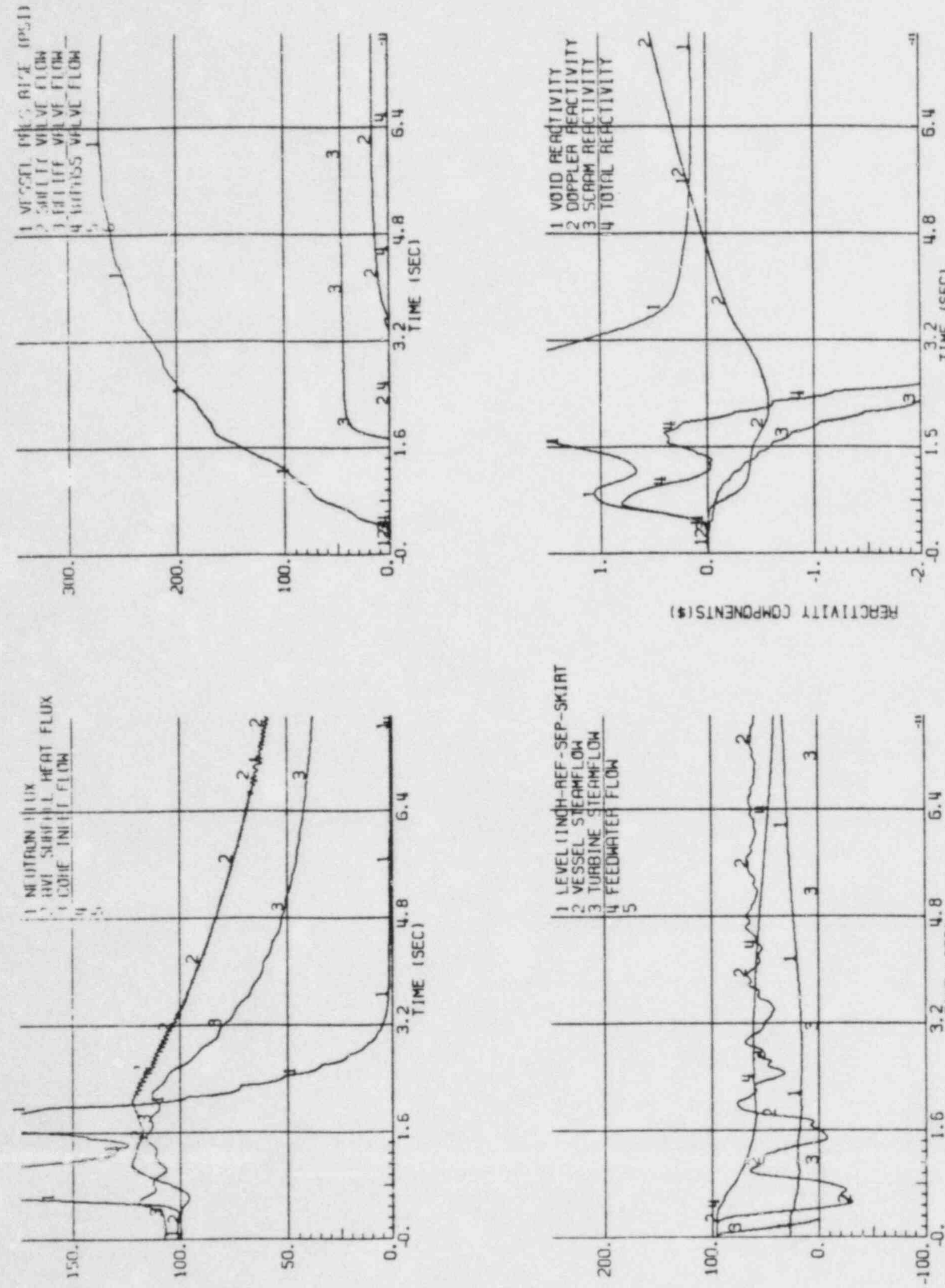

Figure 2-2. Generator Load Rejection, Without Bypass (100% Power, 107.5% Flow, with FW Temperature Reduction)

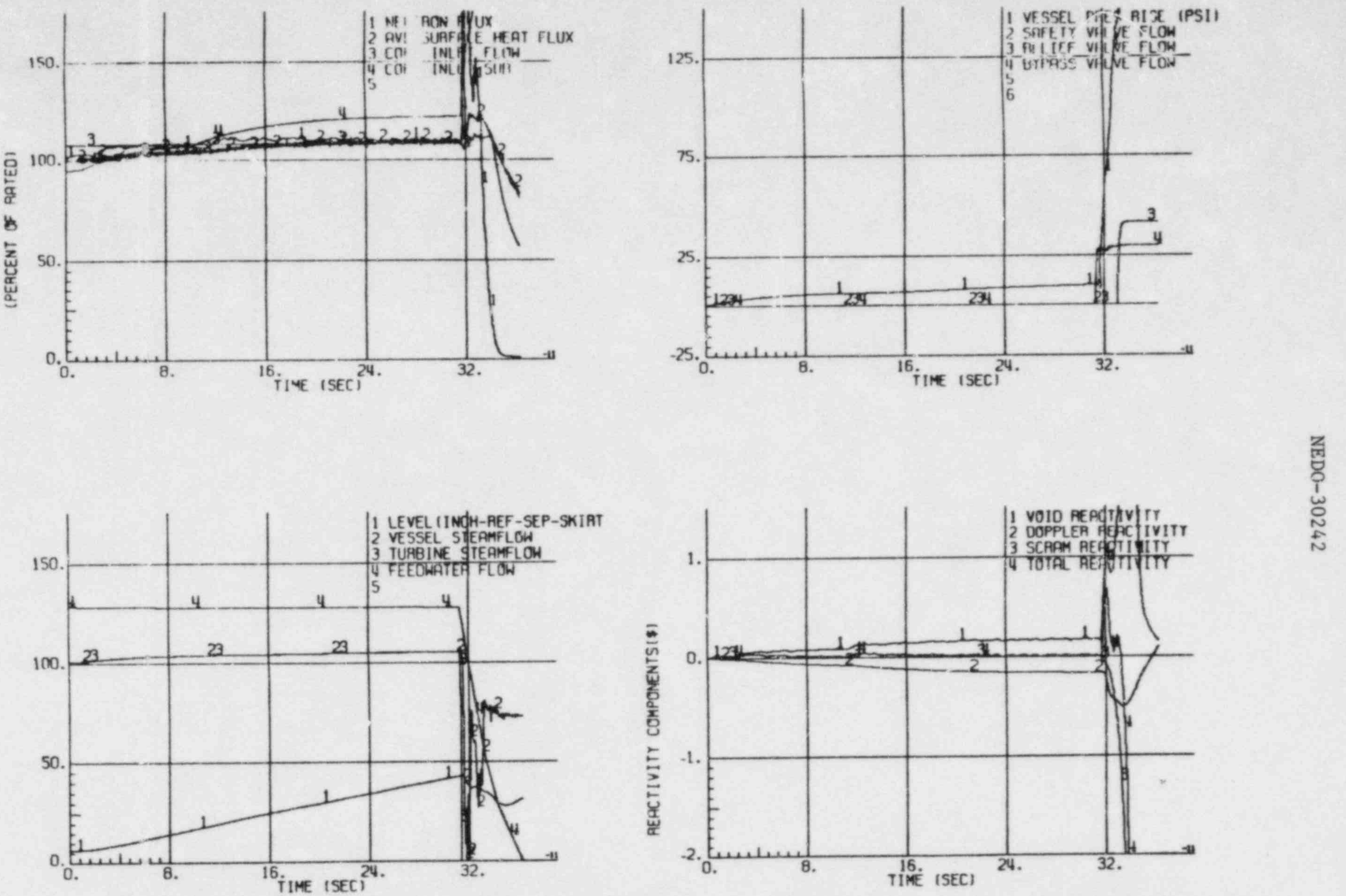

Figure 2-3. Feedwater Controller Failure, Maximum Demand (100% Power, 107.5% Flow, with FW Temperature Reduction)

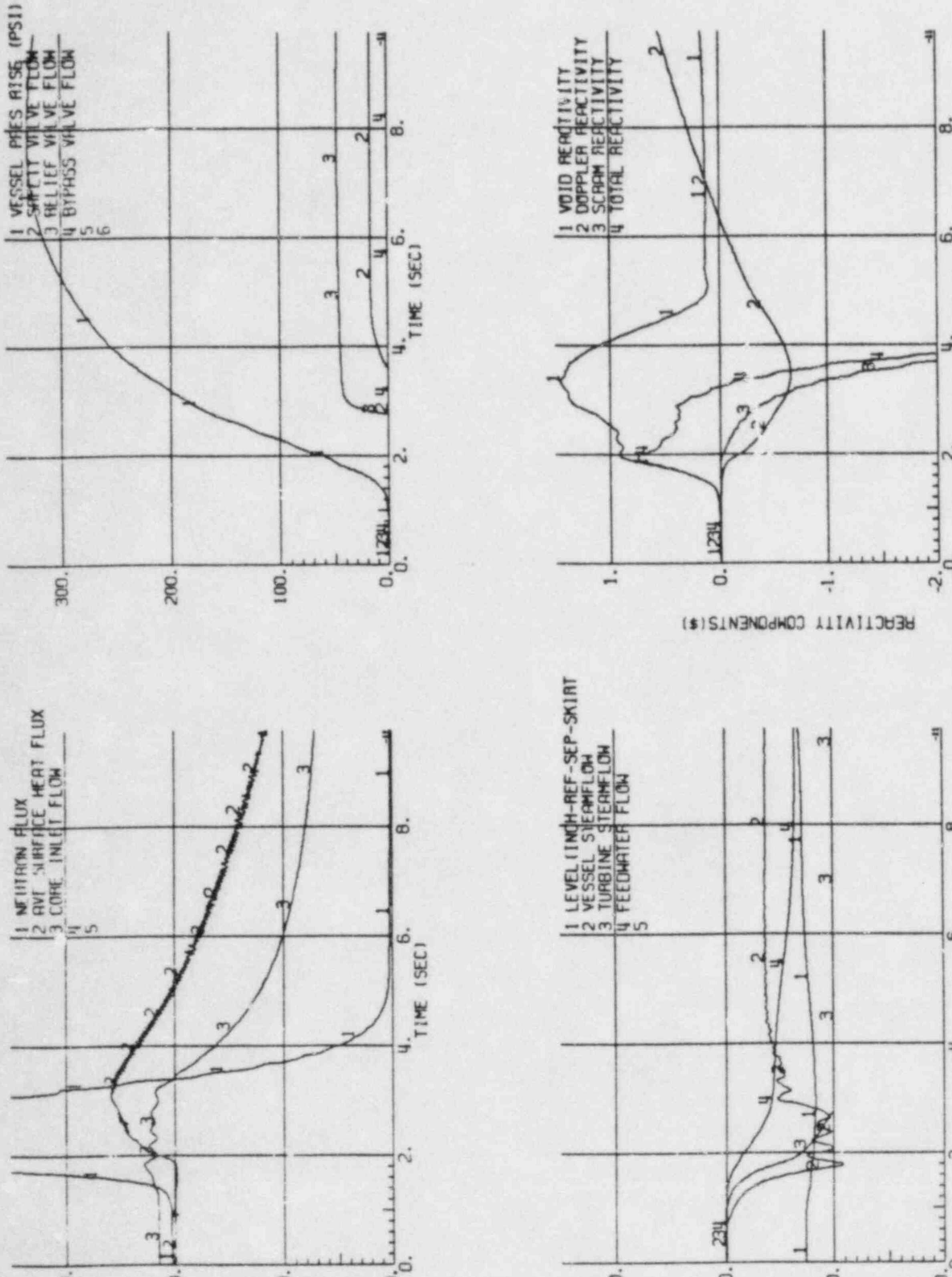

Figure 2-4. MSIV Closure, Flux Scram (100% Power, 107.5% Flow with Normal FW Temperature)

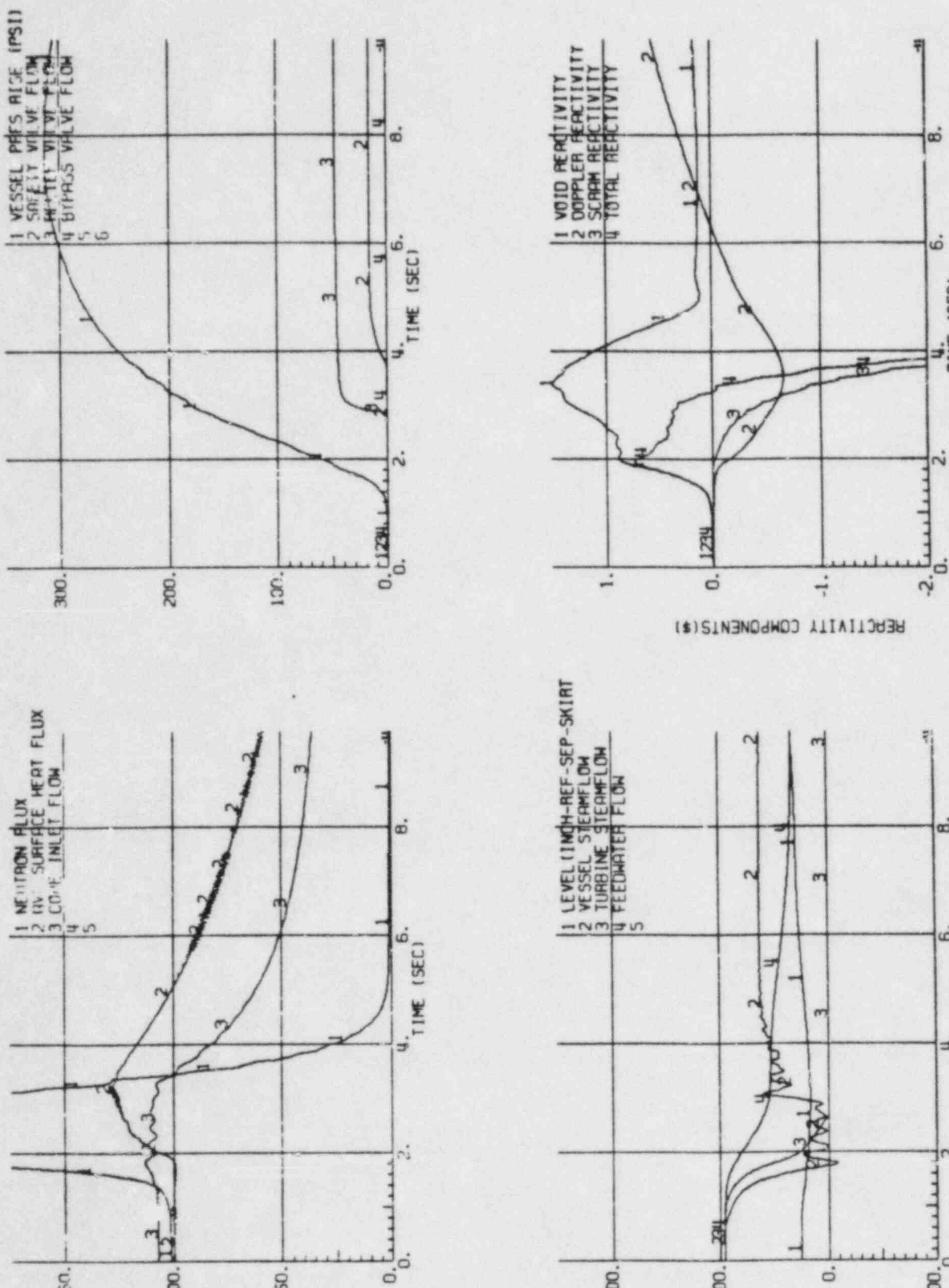

Figure 2-5. MSIV Closure, Flux Scram (100% Power, 107.5%) ow with FW Temperature Reduction)

3. REACTOR INTERNALS PRESSURE DROP

Reactor internals pressure differences have been calculated for the increased core flow condition and evaluated against allowable limits. The evaluation included consideration of upset, emergency, and faulted conditions, in addition to conditions during normal operation.

3.1 REACTOR INTERNALS

The reactor internals most affected by pressure differences under increased core flow conditions are the core plate, guide tube, shroud support, shroud, and top guide. These components were evaluated under normal, upset, emergency, and faulted conditions. The pressure differentials for these components during increased core flow operation were found to produce stresses that are within the allowable limits given in the Final Safety Analysis Report.

3.2 FUEL CHANNELS

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

3.3 FUEL BUNDLES

The margin to fuel bundle lift was reevaluated for increased core flow operation. The analysis considered the added bundle lift component due to increased core flow, in addition to the effect of the design basis LOCA, the control rod friction force due to scram, and the Design Basis Earthquake. The fuel bundle minimum lift margin is 135 pounds (net downward force on fuel bundle), during the worst-case faulted event from rated operating conditions (100% power, 107.5% flow) following by a steamline break at 102% rated steam flow and 107.5% recirculation flow. Thus, the effect of increased core flow is clearly acceptable in terms of avoiding fuel bundle lift.

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 (endurance limit) of 10,000 psi. Based on the valid prototype plant vibration results, Pilgrim 1 had the shroud, jet pumps and jet pump riser braces instrumented. The confirmatory test performed at Pilgrim 1 showed that the flow-induced vibration response was similar to the base test BWR/3 224 size reactor and within design requirements.

The increased core flow vibration analysis was performed by analyzing the startup test vibration data for the valid prototype plant and for Pilgrim 1. The vibration levels for normal 100% power, 100% flow operation were conservatively extrapolated by the ratio of flow velocity squared for each of the instrumented reactor internal components. The jet pump riser braces showed the highest vibration response (32.4% of acceptance criteria) at 112.5% rated core flow for two-pump operation. In addition to analyzing the startup test data, an evaluation of the riser brace structural natural frequency was also performed to determine if an excitation phenomena would exist because of increased recirculation pump speed (blade passing frequency). The results showed that the riser brace natural frequency is high enough (169% of blade passing) to avoid such an excitation. This riser brace excitation would be the most limiting as a result of an increase in pump speed and flow.

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 by ABCDE on the power flow map, Figure 1-1).

5. FEEDWATER NOZZLE USAGE FATIGUE

An evaluation of the effects of the feedwater temperature reduction on feedwater nozzle fatigue was performed for the planned coastdown. The reduced feedwater temperature was calculated to be 320°F for the 100% power, 107.5% flow condition at EOC6, and 305°F for the worst case 80% power, 112.5% flow condition.

Pilgrim 1 has the General Electric final fix feedwater nozzle thermal sleeve which was evaluated in Reference 4 and shown to have a maximum 40-yr usage factor of no greater than 0.96 under normal operating conditions with a feedwater temperature of 365°F.

To evaluate the additional fatigue usage that will occur due to the feedwater temperature reduction, a new calculation was performed using the methods documented in References 4 and 5. This analysis was for a final feedwater temperature reduction to 320°F for 16 days followed by a coastdown to 80% power and a feedwater temperature of 305°F over a period of 8 weeks at the end of each cycle.

The results of this analysis show that if the refurbishment schedule specified in Reference 4 is followed, the average additional fatigue usage due to rapid cycling that will occur on the feedwater nozzle for 16 days at 320°F and 8 weeks at a temperature of 305°F is 0.0103/year. Operation at these conditions on a continued basis after every cycle would produce a usage factor greater than 1.0 in 36 to 37 years, assuming 13-year refurbishment intervals as determined in the Reference 4 report. The refurbishment period of 13 years can be reduced to 12 years in order to keep the 40-yr usage factor below 1.0. Note that those refurbishment intervals are based on the leakage flow estimates used in Reference 4.

Although the assumptions made in this analysis make it conservative in nature, actual refurbishment intervals should be established by actual plant performance and monitored secondary seal leakage. Therefore, it is concluded that if FFWTR is desired on a continuing basis, the actual seal refurbishment

period as determined by monitored secondary seal leakage will be impacted by 1 year.

6. THERMAL-HYDRAULIC STABILITY ANALYSIS

The channel hydrodynamic stability and the reactor core stability were evaluated for increased core flow operation with the last stage feedwater heaters valved out-of-service. From the stability standpoint of view, both channel and core decay ratios for the increased core flow operation would be less severe than the standard reload analysis because the reactor core initially operates at a higher core flow. The FFWTR could improve the channel decay ratio because of the increased subcooling effect. The core decay ratio for FFWTR alone would be slightly increased. However, the combined effect of operating the reactor core with ICF first and then FFWTR would have a lower core decay ratio and lower channel decay ratio because of a lower final power level.

The reactor core stability and the channel hydrodynamic stability decay ratios reported in the Reload-5 licensing submittal (Reference 1) are, therefore, bounding for increased core flow operation beyond EOC6.

7. CONTAINMENT ANALYSIS

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

The results show no appreciable impact on the containment LOCA response. The drywell pressurization rate is lower than the Mark I containment plant unique load definition value (Reference 6), indicating no impact on pool swell loads. The drywell peak pressure and temperature with ICF and FFWTR are slightly higher, but they are still below the Mark I containment limits. Therefore, the current containment LOCA response analyses results are adequate for the extended operating conditions stated above.

8. REFERENCES

1. "Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 Reload No. 5," Y1003J01A28, Revision 2, General Electric Company, February, 1983.

2. "General Electric Standard Application for Reactor Fuel (Supplement for United States)", NEDE-24011-P-A-US-6, April 1983.

3. "Loss-of-Coolant Accident Analysis Report for Pilgrim Nuclear Power Station," General Electric Company, August 1977 (NEDO-21696, as amended).

4. "Feedwater Nozzle Rapid Cycling Fatigue Analysis - Pilgrim Nuclear Power Station," NSEO-18-0383, General Electric Company, March 1983.

5. "Boiling Water Reactor Feedwater Nozzle/Sparger Final Report," General Electric Company, NEDE-21821-02, January 1980.

6. "Mark I Containment Program, Plant Unique Load Definition, Pilgrim Nuclear Power Station," General Electric Company, NEDO-24565, May 1982.

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DATE September 1983

NOTE: Correct all copies of the applicable publication as specified below.

ITEM	REFERENCES (SECTION, PAGE PARAGRAPH, LINE)	INSTRUCTIONS (CORRECTIONS AND ADDITIONS)
1.	Page 4-1/4-2	Replace with revised page 4-1/4-2

4. FLOW-INDUCED VIBRATION

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Based on the results of the analysis and a review of the test data, the reactor internals response to flow-induced vibration is concluded to be within acceptable limits for plant operation in the ICF region (region bounded by ABCDE on the power flow map, Figure 1-1).