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SAFETY REVIEW OF PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3 AT CORE FLOW CONDITIONS ABOVE RATED FLOW THROUGHOUT CYCLE 6

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SAFETY REVIEW
OF
PEACH BOTTOM ATOMIC POWER STATION
UNIT NO. 3
AT CORE FLOW CONDITIONS ABOVE RATED FLOW
THROUGHOUT CYCLE 6

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ABSTRACT

A safety evaluation has been performed to show that Peach Bottom Unit 3 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, 105% core flow (100,105) throughout Cycle 6. Peach Bottom Unit 3, 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, 105% core flow (100,105) and 70% power, 110% core flow (70,110) with or without 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 (Y1003J01A54, December 1982), 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 evaluation supports the operation of the Peach Bottom Atomic Power Station, Unit 3, within the region of the operating map bounded by ABCDE on the operating map in Figure 1-1. This report presents the results of a safety evaluation for operation with increased core flow (ICF) for Cycle 6 [up to and including end-of-cycle 6 (EOC6) exposure]. The safety evaluation also covers operation for exposure beyond standard EOC6* with ICF and/or last stage feedwater heaters valved out, followed by a natural reactivity coastdown to 70% power under conditions bounded by 110% core flow. Final feedwater temperature reduction (FFWTR) to approximately 328°F and reactivity coastdown 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 BCDE 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 at EOC6-2000 MWd/t and end of rated power reactivity at 105% 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.

In addition, 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 core 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 increased core flow operation, and the increase in the feedwater nozzle and feedwater sparger usage factors due to the feedwater

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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 (beyond Rod Position 24), all feedwater heaters in service and equilibrium xenon.

temperature reduction were 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 2-3b are adequate to preclude the violation of any safety limits during operation of Peach Bottom, Unit 3, within the region bounded by BCDEF 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 (EOC6 only), 2-3a, and 2-3b. The EOC6 Option A and Option B MCPR limits (Reference 1) will be increased to the appropriate values as shown in Table 2-3b.

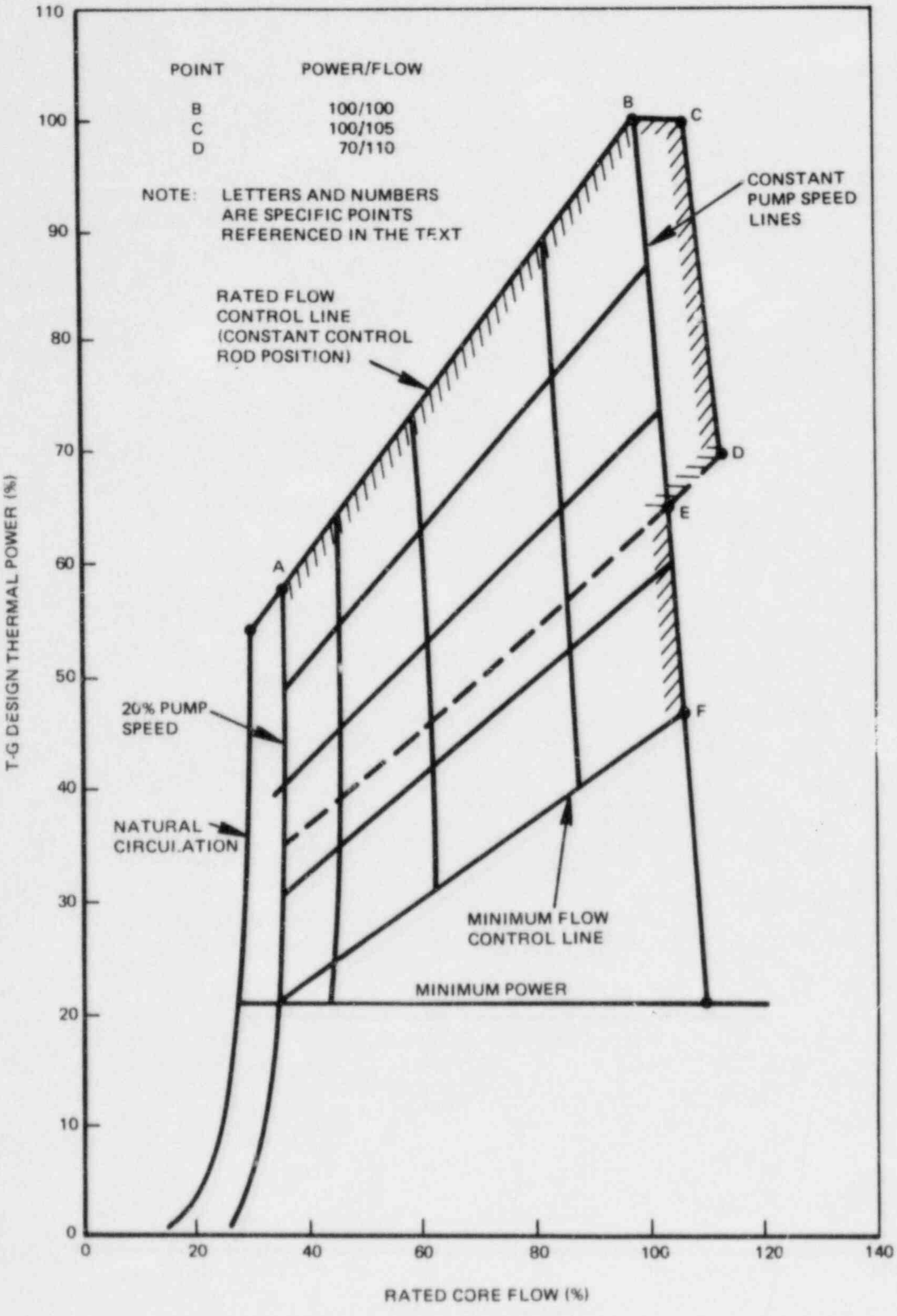

Figure 1-1. Operating Map

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 and/or FFWTR as follows.

Nuclear transient data for 104.7% power,* 105% core flow (104.7,105) with and without the last stage feedwater heaters out 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 (104.7,105) conditions.

The results of the transient analyses are presented in Tables 2-1, 2-2, 2-3a and 2-3b with the transient results contained in the Reload 5 licensing submittal (Reference 1). The licensing submittal is bounding for all cases up to the EOC6-2000 MWd/t exposure point. Beyond EOC6-2000 MWd/t, this analysis is bounding for ICF/FFWTR operation.

After EOC6-2000 MWd/t exposure, for the LR w/o BP event, FFWTR decreases steamflow thereby improving the LR w/o BP response. Thus, LR w/o BP with ICF is the most limiting transient, with the exception of the FWCF event under Option B for the PBLTA2 bundles. As shown in Tables 2-1 and 2-2, the ΔCPR for the (104.7,105) condition with and without feedwater temperature reduction exceeds the license basis ΔCPR used to set the operating limits for EOC6. 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 2-3.

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*All transients were analyzed using 105% steam flow. The power level corresponding to this condition will vary from 104.7% to 104.5%, depending on whether final feedwater heaters are in service. The 104.7% power level provides a 5% power margin to the 100% power operating condition.

The results of core-wide ΔCPR for Options A and B with fuel types of PBLTA1 and 2 and P8x8R are shown in Table 2-2. The analyses demonstrate that the final MCPR must be increased for Option A and Option B operation beyond EOC6-2000 MWd/t exposure.

Increasing the core flow from 105% to 110% of rated along the constant pump speed line as power decreases (line CD 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 without FFWTR (Table 2-4 and Figure 2-3). The ICF without FFWTR (which is more severe than the ICF with FFWTR case) will not result in a more severe overpressure transient for the MSIV closure event compared to the Reference 1 basis. The ICF for the LR w/o BP event results in a less severe overpressure transient. The overpressurization analysis (Table 2-4) for the ICF region produced a peak vessel pressure of 1276 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 recalculated under ICF conditions. When ICF is employed, the rod block monitor (RBM) block (which is flow biased) increases, giving an unacceptably high MCPR limit (around 1.35). Thus, the RBM should be clipped at 107%, giving a ΔCPR of 0.18.

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 lower initial steam flow and inlet enthalpy, due to ICF, results in a less

severe event than the non-ICF case. 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 2 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 Peach Bottom 3 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 consluded that the LOCA analysis and maximum average planar linear heat generation rates (MAPLHGRs) determined for the Peach Bottom, Unit 3, Reload-5 core (Reference 4) remain unchanged for use in the increased core flow region of the operating map.

2.5 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.

The results of this analysis, given in Table 2-5, show Peach Bottom 3, Cycle 6, to be in compliance with the ultimate performance criteria, including the least stable condition.

Table 2-1

CORE-WIDE TRANSIENT ANALYSIS RESULTS

								ΔCPR[a]			
Transient	Exposure	Power (%)	Flow (%)	ϕ (%)	Q/A (%)	P_{SL} (psig)	P_V (psig)	PBLTA2	PBLTA1	PTA/ P8x8R	Plant Response
LR w/o BP (Reference 1)	EOC6	104.5	100	695	128	1223	1243	0.26	0.27	0.26	(Reference 1)
LR w/o BP[b]	EOC6-2000 MWd/t	104.7	105	562	122	1203	1227	0.18	0.18	0.18	Figure 2-1
LR w/o BP[b]	EOC6	104.7	105	717	128	1222	1244	0.26	0.27	0.27	Figure 2-2
FWCF (Reference 1)	EOC6	104.5	100	322	124	1154	1197	0.19	0.19	0.19	(Reference 1)
FWCF[c]	EOC6	104.7	105	344.1	128.6	1144	1186	0.22	0.22	0.22	Figure 2-3

[a] Uncorrected for Options A and B.

[b] Feedwater heaters in service; this case bounds for LR w/o BP + Feedwater Heater Out-of-Service.

[c] Last-stage feedwater heater valved out-of-service; this case bounds FWCF with Feedwater Heaters In Service.

Table 2-2

EOC6 CORE-WIDE ΔCPR^{a} RESULTS

Transient	Option A PBLTA2	Option A PBLTA1	Option A PTA/P8x8R	Option B PBLTA2	Option B PBLTA1	Option B PTA/P8x8R
LR w/o BP (104.5% power, 100% flow, Reference 1)	0.32	0.33	0.32	0.20	0.21	0.20
LR w/o BP[b] (104.7% power, 105% flow)	0.32	0.33	0.33	0.20	0.21	0.21
FWCF (104.5% power, 100% flow, Reference 1)	0.25	0.25	0.25	0.18	0.18	0.18
FWCF[c] (104.7% power, 105% flow)	0.28	0.28	0.28	0.21	0.21	0.21

[a] 104.7% power, 105% flow at EOC6 exposure conditions: corrected for Options A and B.

[b] Feedwater heaters in service; this case bounds for LR w/o BP + Feedwater Heater Out-of-Service.

[c] Last-stage feedwater heater valved out-of-service (FFWTR); this case bounds FWCF with Feedwater Heaters In Service.

Table 2-3a

MCPR OPERATING LIMITS AT INCREASED CORE FLOW
FOR PEACH BOTTOM UNIT 3, EOC6-2000 MWd/t

Transient	Option A			Option B		
	PBLTA2	PBLTA1	PTA/ P8x8R	PBLTA2	PBLTA1	PTA/ P8x8R
LR w/o BP (104.5% power, 100% flow, Reference 1)	1.33	1.33	1.33	1.12	1.12	1.12
LR w/o BP (104.7% power, 105% flow, FW heater in service)	1.30	1.30	1.30	1.11	1.11	1.11
FWCF (104.5% power, 100% flow, Reference 1)	1.24	1.24	1.24	1.18	1.18	1.18
FWCF (104.7% power, 105% flow, FW heater in service)	1.22	1.22	1.22	1.16	1.16	1.16

Table 2-3b

MCPR OPERATING LIMITS AT INCREASED CORE FLOW AND/OR
FEEDWATER TEMPERATURE REDUCTION FOR PEACH BOTTOM
UNIT 3, EXPOSURES GREATER THAN EOC6-2000 MWd/t[a]

Transient	Option A			Option B		
	PBLTA2	PBLTA1	PTA/ P8x8R	PBLTA2	PBLTA1	PTA/ P8x8R
LR w/o BP (104.5% power, 100% flow, Reference 1)	1.39	1.40	1.39	1.27	1.28	1.27
LR w/o BP (104.7% power, 105% flow, FW heater in service)	1.39	1.40	1.40	1.27	1.28	1.28
FWCF (104.5% power, 100% flow, Reference 1)	1.32	1.32	1.32	1.25	1.25	1.25
FWCF (104.7% power, 105% flow, FW heater out-of-service)	1.35	1.35	1.35	1.28	1.28	1.28

[a]Assumes RBM setpoint clipped at 107%.

Table 2-4

OVERPRESSURIZATION ANALYSIS

Transient	Power (%)	Flow (%)	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure - Flux Scram (Licensing Submittal)	104.5	100	1245	1276	(Reference 1)
MSIV Closure - Flux Scram (ICF w/o FFTWR)	104.7	105	1245	1276	Figure 2-3

Table 2-5

PEACH BOTTOM 3
BOUNDING STABILITY DECAY RATIO VALUES

Rod Line Analyzed: 105% Rod Line

Reactor Core Stability Decay Ratio, X_2/X_0: 0.95

Channel Hydrodynamic Performance Decay Ratio, X_2/X_0:

Channel Type	Decay Ratio (X_2/X_0)
P8x8R	0.29
PBLTA1	0.12
PBLTA2	0.27

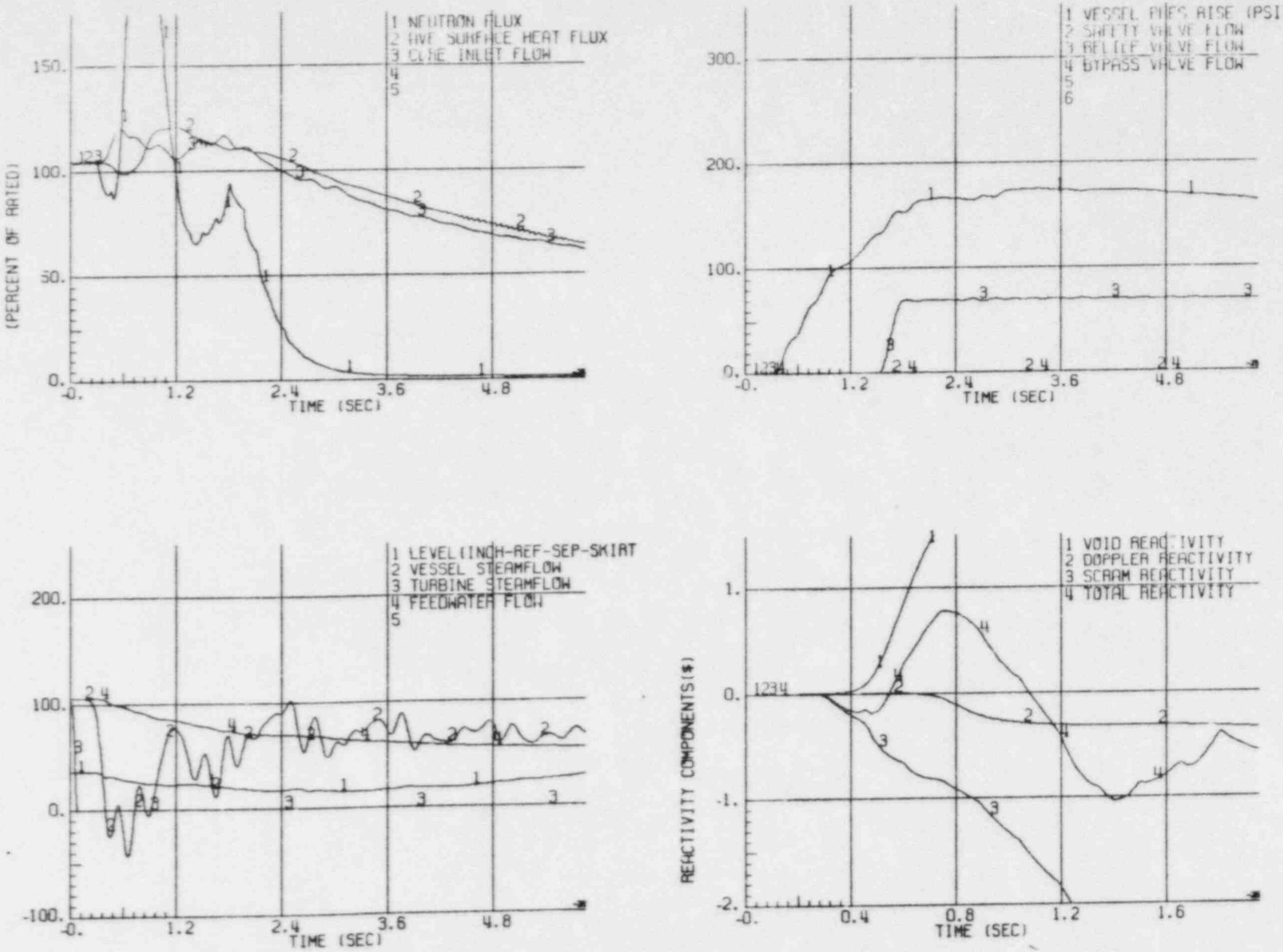

Figure 2-1. Generator Load Rejection, Without Bypass (104.7% Power, 105% Flow, with Normal Feedwater Temperature)

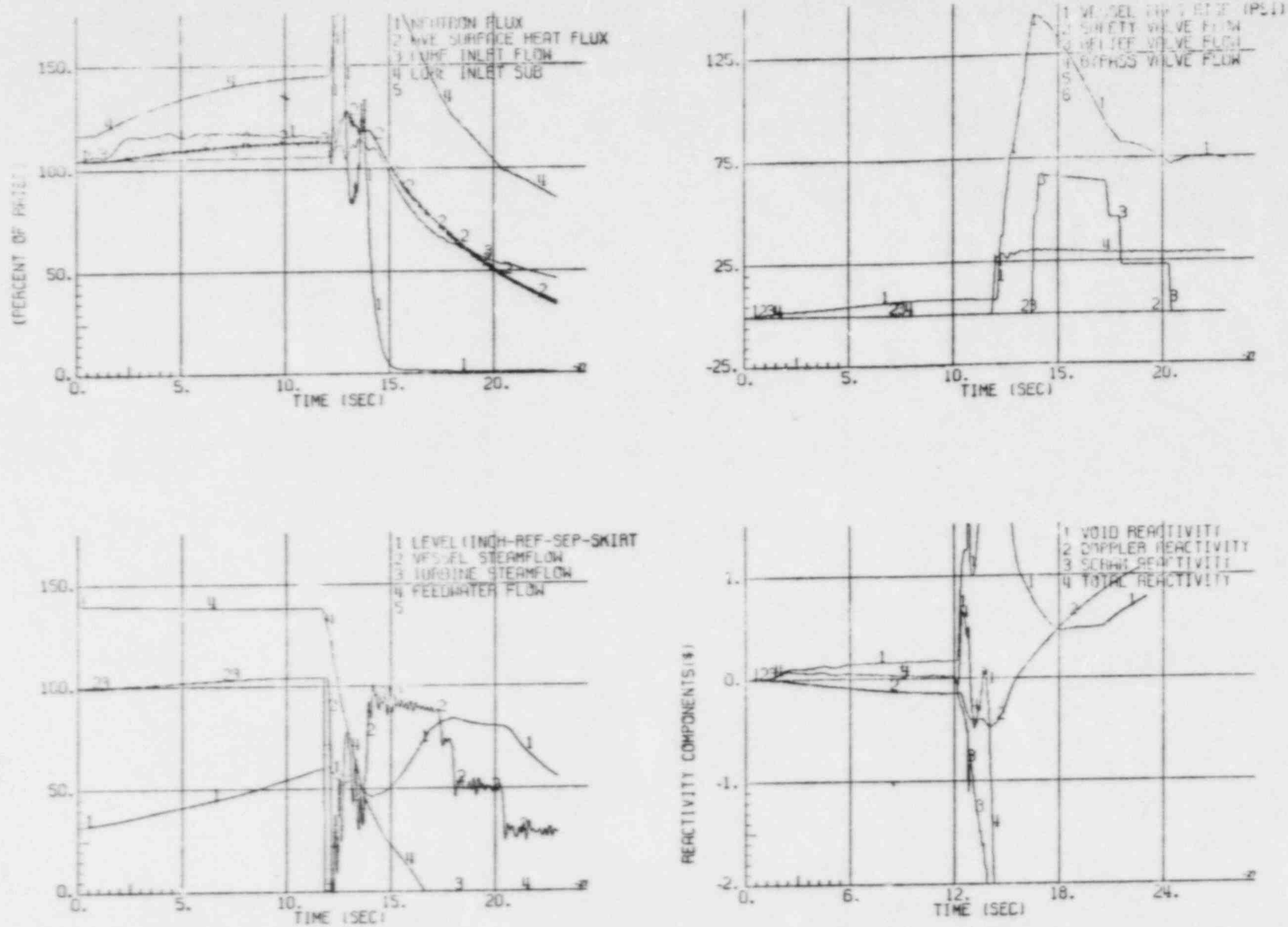

Figure 2-2. Feedwater Controller Failure, Maximum Demand (104.7% Power, 105% Flow, with Feedwater Temperature Reduction)

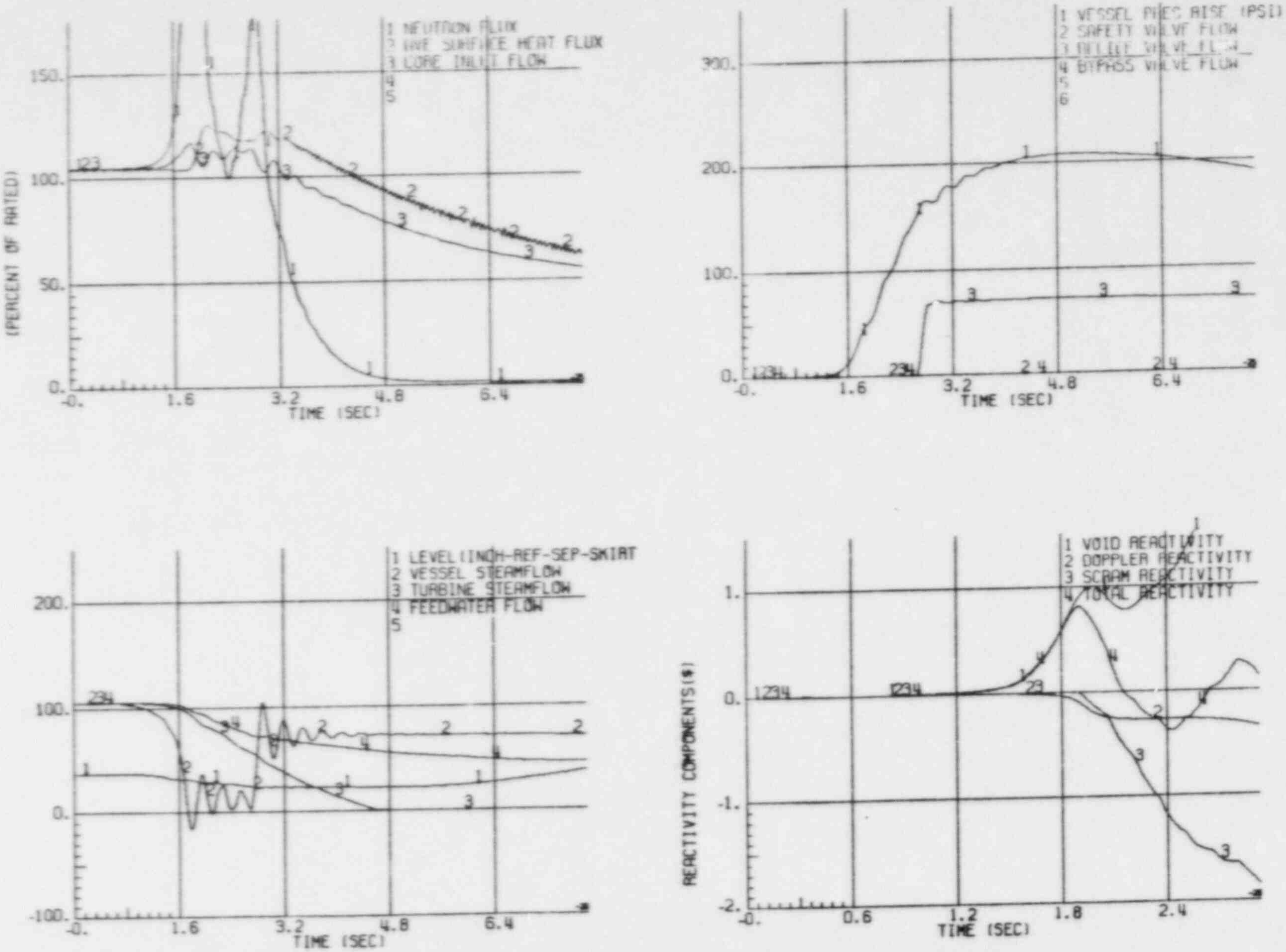

Figure 2-3. MSIV Closure, Flux Scram (104.7% Power, 105% Flow with Normal Feedwater Temperature)

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 Appendix C 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 (Reference 5).

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 analysis shows that the fuel bundle lift margin is sufficient for operation in the increased flow region bounded by ABCDEF in Figure 1-1.

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. The confirmatory test performed at Peach Bottom 3 showed that the flow-induced vibration response was similar to the base test BWR/4 251 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 Peach Bottom 3. 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.

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 ABCDEF on the power flow map, Figure 1-1).

5. FEEDWATER NOZZLE AND FEEDWATER SPARGER USAGE FATIGUE

5.1 FEEDWATER NOZZLE FATIGUE

An evaluation of the effects of the feedwater temperature reduction on feedwater nozzle and feedwater sparger fatigue was performed for the planned coastdown. The reduced feedwater temperature was calculated to be 328°F for the 100% power, 100% flow condition at EOC6, and 304°F for the worst case 70% power, 110% flow condition.

Peach Bottom 3 has the General Electric final fix feedwater nozzle thermal sleeve which was evaluated in Reference 6 and shown to have a maximum 40-yr usage factor of no greater than 0.82 under normal operating conditions with a feedwater temperature of 376°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 6 and 7. This analysis was for an FFWTR to 328°F for 26 days followed by a coastdown to 70% power and a feedwater temperature of 304°F over a period of 12 weeks at the end of each cycle.

The results of this analysis show that if the refurbishment schedule specified in Reference 6 is followed, the average additional fatigue usage due to rapid cycling that will occur on the feedwater nozzle for 26 days at 328°F and 12 weeks at a temperature of 304°F is 0.01774/year. Operation at these conditions on a continued basis after every cycle would produce a usage factor greater than 1.0 in 28 to 29 years, assuming 15-yr refurbishment intervals as determined in the Reference 6 report. The refurbishment period of 15 years can be reduced to 14 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 6.

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.

5.2 FEEDWATER SPARGER FATIGUE

Startup for Peach Bottom 2 and 3 occurred in February and September of 1974, respectively. The spargers were replaced after approximately 6 years of service (April 1980 and May 1981). Only the fatigue usage for the remaining 34 years is considered in this analysis.

Since the feedwater sparger is not an ASME Boiler Pressure Vessel code safety class component, a fatigue analysis was not originally required. To gauge the effect of the FFWTR/ICF on the sparger, the fatigue usage factor in the sparger for the original thermal duty cycles was calculated. The fatigue usage for the system cycles is 0.23 for the remaining 34 years. The total usage including both system and rapid cycling is 0.77 which, because it is less than one, means that the sparger will last for 34 years if FFWTR/ICF is not incorporated. The changes in thermal duty due to the FFWTR/ICF were then incorporated and the new fatigue usage was found to be 1.13. This corresponds to a life of 28 years. The effect of increasing the core flow to 105% or 110% is negligible.

These calculations are based on an 18-month refueling cycle with rapid cycling for the following conditions added to the thermal duty to simulate the FFWTR/ICF:

a. 26.5 days at 328°F, 100% feedwater flow.
b. 12 weeks at 304°F, 100% feedwater flow.

The calculated fatigue usage is mainly due to high cycle thermal fatigue duty at stress ranges approaching the endurance limit of the material (at 10^8 cycles). Because of this, the calculated usage is strongly dependent on assumed conservatisms in the fatigue curve, leakage rate and temperature ranges. Nevertheless, the comparison of results with and without FFWTR/ICF suggests that the impact of FFWTR is not significant from the viewpoint of sparger fatigue usage.

6. CONTAINMENT ANALYSIS

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

The results show no appreciable impact on the containment LOCA response. The drywell pressurization rate remains bounded by the value tested in plant unique tests for defining LOCA-related pool swell loads. Therefore, the current containment LOCA response analyses results are adequate for the extended operating conditions stated above.

7. OPERATING LIMITATIONS

The operation limits established in the existing Reload-5 licensing basis (Reference 1) have been shown to be bounding for plant operation in the increased core flow region of the operating map (Figure 1-1). Restrictions/limitations which are unique to increased core flow operation are identified below.

7.1 FEEDWATER HEATERS

The increased core flow analyses have assumed that the fifth-stage (top) feedwater heater is valved out-of-service in each string of feedwater heaters for exposures beyond EOC6. This is done to help increase power in the increased core flow region of the operating map, and was accounted for in the safety analyses in Subsections 2.1 and 2.2.

7.2 OPERATING MAP

The increased core flow reactor internal pressure differences and fuel bundle lift calculations were analyzed and are applicable only for reactor operation within the region bounded by ABCDEF on the power flow map (Figure 1-1).

7.3 K_f FACTOR

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

7.4 CONTROL RODS

The safety evaluation for increased core flow 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 notch position 24.

8. REFERENCES

1. "Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit 3 Reload No. 5," General Electric Company, December 1982 (Y1003J01A54 Revision 2).

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

3. "Safety Review of Peach Bottom Atomic Power Station Unit No. 3 at Core Flow Conditions Above Rated Flow at the End of Cycle 3," General Electric Company, June 1979 (NEDO-24039-3).

4. "Loss-of-Coolant Accident Analysis Report for Peach Bottom Nuclear Power Station, Unit 3," General Electric Company, December 1977 (NEDO-24082, as amended).

5. "BWR Fuel Channel Mechanical Design and Deflection," General Electric Company, September 1976 (NEDE-21354-P).

6. "Feedwater Nozzle Rapid Cycling Fatigue Analysis - Peach Bottom Atomic Power Station Units 2 and 3," General Electric Company, June 1983 (NSEO-38-0483).

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

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