NEDO-30242
DRF L12-00634
83NED119S1
CLASS I
SEPTEMBER 1983
SUPPLEMENT 1

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

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

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CONTENTS

ABSTRACT

A safety evaluation has been performed to show that Pilgrim can reduce feedwater temperature followed by increased 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) for EOC6. Pilgrim, after reaching EOC6 exposure (depletion of full-power reactivity under standard feedwater conditions) with all power rods out, can continue to operate at rated power by using FFWTR first and then using ICF to 107.5%. The analyzed region of the operating map is 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 not be changed from the values established by the Reload-5, Cycle 6 reload licensing submittal (Y1003J01A28, Rev. 2, Feb. 1983) during the operation of reduced feedwater temperature. However, the MCPR operating limits must be increased from the values established by the Reload-5, Cycle 6 reload licensing submittal to the appropriate values (Table 2-1) depending on the operating conditions for increased core flow after feedwater temperature reduction.

1. INTRODUCTION AND SUMMARY

This report presents the results of a safety evaluation for operation of the Pilgrim Nuclear Power Station with last stage feedwater heaters valved out-of-service at end-of-cycle 6 (EOC6)* and for exposure beyond standard (EOC6) followed by increased core flow (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 a reduction of approximately 43°F in the feedwater temperature followed by an increase of 107.5% core flow (100,107.5) and 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 final feedwater temperature reduction (FFWTR) followed by increased core flow (FFWTR/ICF) is bounded by ABCDE on the operating map in Figure 1-1.

In order to evaluate operation with FFWTR followed by ICF, the limiting abnormal operational transients reported in References 1 and 2 were reevaluated. The loss-of-coolant accident (LOCA), fuel loading error accident, rod drop accident, and rod withdrawal error event were analyzed in References 1 and 2 and the results are applicable for FFWTR/ICF operation.

The effect of the increased pressure differences (due to the increased core flow) on the reactor internal components, fuel channels, and fuel bundles was analyzed in Reference 2 and the results are applicable for FFWTR/ICF operation. The effect of the increased flow rate on the flow-induced vibration response of the reactor internals was also evaluated in Reference 2 and the results are adequate for FFWTR/ICF operation. The thermal-hydraulic stability was reanalyzed for FFWTR/ICF operation, and the increase in the feedwater nozzle usage factor due to the feedwater temperature reduction was reanalyzed. The

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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.

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

The results of the safety evaluation show that the current technical specifications are adequate for FFWTR at EOC6 conditions. However, the minimum critical power ratio (MCPR) limits must be increased to the appropriate values in Table 2-1 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 FFWTR/ICF conditions. The MCPR operating limits given in Reference 2 and repeated in Table 2-1 are applicable to plant operation for ICF operation which follows FFWTR.

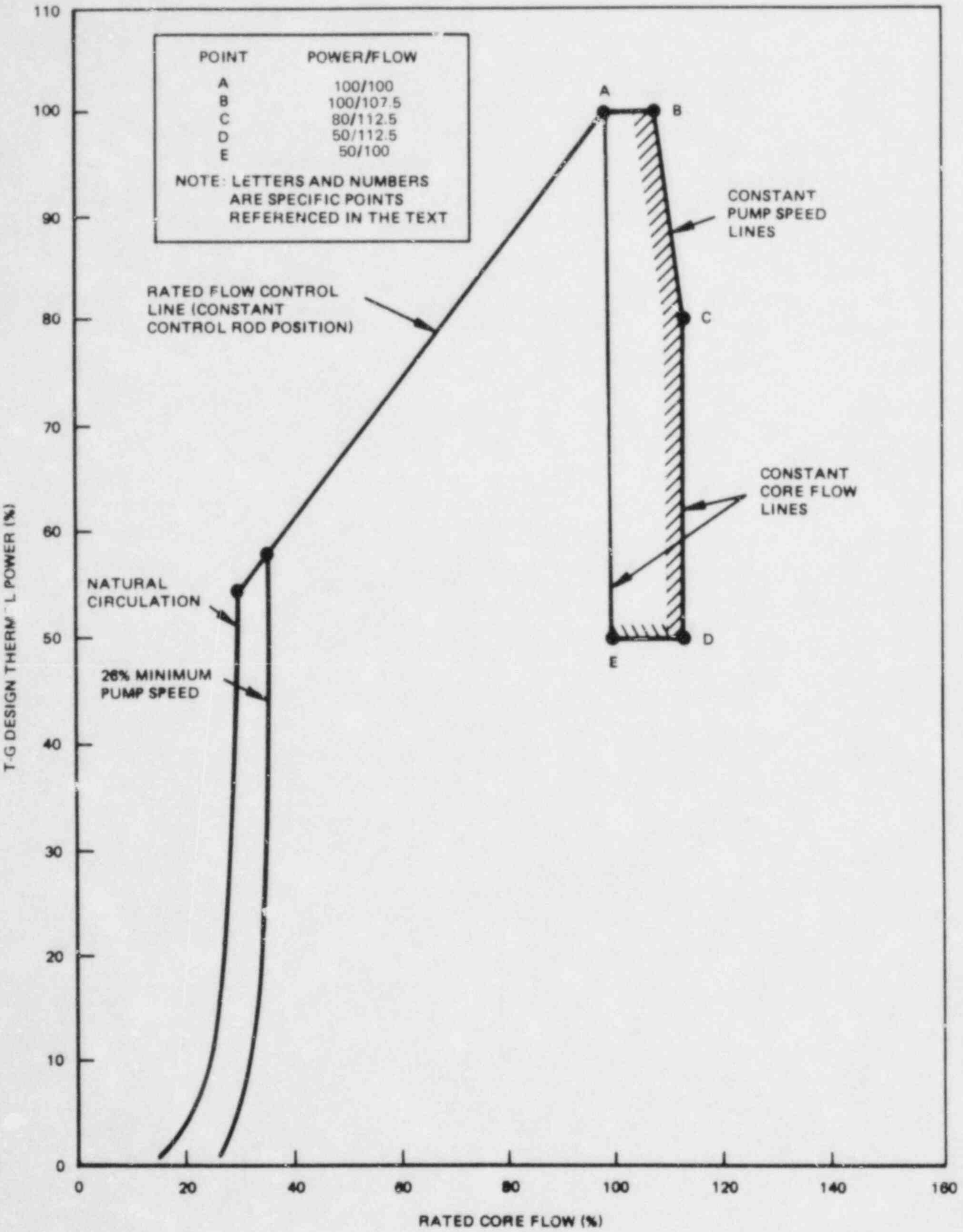

Figure 1-1. Operating Map

2. SAFETY ANALYSIS

The limiting abnormal operational transients analyzed in the Reload-5, Cycle 6 reload licensing submittal (Reference 1) and the analysis for increased core flow followed by final feedwater temperature reduction (ICF/FFWTR) (Reference 2) were reevaluated for FFWTR/ICF.

Table 2-1 shows the results of MCPR operating limits for the case of ICF/FFWTR (Reference 2). As shown in Table 2-1, FFTWR has the effect of reducing MCPR for both options A and B at EOC6 conditions because of improved transient response, which is the result of improved scram effectiveness and reduced steam flow at rated thermal power output. Therefore, the current operating MCPR limit established in the Reload-5 licensing submittal (Reference 1) is adequate for EOC6 with only FFWTR in operation. Analysis of ICF operation subsequent to FFWTR indicated no appreciable change in transient input data (and therefore transient results) relative to the reference to ICF/FFWTR results. Therefore, the bounding reference to MCPR limits (ICF heater in) conservatively bounds FFWTR/ICF operation.

The results of other abnormal operational transient analysis (Reference 2), namely overpressurization analysis, rod withdrawal error analysis, fuel loading error analysis, rod drop accident, and loss-of-coolant accident (LOCA) analysis, were reevaluated. The results of this evaluation show that the conclusion derived in Reference 2 for ICF/FFWTR is applicable to the present case of FFWTR/ICF.

Table 2-1

MCPR OPERATING LIMITS AT ICF/FFWTR
FOR PILGRIM UNIT 1, EOC6 (Reference 2)

Transient	Option A		Option B	
	8x8	P8x8R	8x8	P8x8R
LR w/o BP[a] (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

[a] LR w/o BP = Load rejection without bypass event

3. FEEDWATER NOZZLE USAGE FATIGUE

An evaluation of the effects of the feedwater temperature reduction on feedwater nozzle fatigue was recalculated for FFWTR and for the planned coastdown. The reduced feedwater temperature was calculated to be 320°F for the 100% power, 100% flow condition at EOC6, and 296°F for the worst case 70% power, 112.5% flow condition.

Pilgrim 1 has the General Electric final fix feedwater nozzle thermal sleeve which was evaluated in Reference 3 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 3 and 4. This analysis was for a final feedwater temperature reduction to 320°F for 26 days followed by a coastdown to 70% power and a feedwater temperature of 296°F over a period of 13 weeks at the end of each cycle.

The results of this analysis show that if the refurbishment schedule specified in Reference 3 is followed, the average additional fatigue usage due to rapid cycling that will occur on the feedwater nozzle for 26 days at 320°F and 13 weeks at a temperature of 296°F is 0.0209/year (0.0103/year for the case of ICF/FFWTR, Reference 2). Operation at these conditions on a continued basis after every cycle would produce a usage factor greater than 1.0 in 25 to 26 years, assuming 13-year refurbishment intervals as determined in the Reference 3 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 3.

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.

4. THERMAL-HYDRAULIC STABILITY ANALYSIS

The channel hydrodynamic stability and the reactor core stability were reevaluated for the last stage feedwater heaters valved out-of-service followed by increased core flow operation. 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 will be slightly increased from 0.59 of standard reload analysis to 0.66 of FFWTR in the worst case. However, this increase in decay ratio is still within the design limits of 1.0 with a comfortable margin. The overall results indicate that the thermal-hydraulic stability is acceptable for FFWTR/ICF.

5. 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 5), indicating no impact on pool swell loads. The drywell peak pressure and temperature with FFWTR and ICF 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.

6. 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. "Safety Review of Pilgrim Nuclear Power Station, Unit No. 1 at Core Flow Conditions Above Rated Flow Throughout Cycle 6", NEDO-30242, August 1983.

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

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

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

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