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Feedwater Heater(s) Out-of-Service Analysis For River Bend Station

GE Nuclear Energy

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FEEDWATER HEATER(S) OUT-OF-SERVICE ANALYSIS

FOR

RIVER BEND STATION

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1. INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report presents the results of a safety and impact evaluation which was performed to support continued operation of River Bend Station (RBS) with a number of feedwater heater(s) out of service (FWHOS). Continued operation with FWHOS is desirable in the event that certain feedwater heater(s) or string(s) of heater(s) become inoperable during a reactor fuel cycle. Operational flexibility and plant capacity factor are improved if the plant is able to continue operating until full heating can be restored or until the next convenient outage occurs.

Design evaluations reported in the RBS Updated Safety Analysis Report (USAR) and the RBS Cycle 2 Reload Licensing Submittal (Reference 1) justify operation with full feedwater heating which corresponds to a rated feedwater temperature* of 420°F. Operation with FWHOS will result in lower feedwater temperatures with increased subcooling in the core downcomer region and at the core inlet. Loss of feedwater heating from the highest pressure heaters would result in the highest temperature reduction. Loss of heating from the low-pressure heaters would result in only a slight reduction of feedwater temperature. It is estimated that, at the worst, feedwater temperature loss during an operating cycle due to inoperable, out-of-service or unavailable heater stages is less than 100°F. Therefore, evaluations presented herein assume a 100°F reduction in temperature. Reference 1 has already evaluated the consequence of the transient with a sudden feedwater temperature loss of 100°F when initiated from the 420°F rated feedwater temperature. This report will justify the continued operation at the steady-state condition with FWHOS corresponding to a range of rated feedwater temperature from 420°F to 320°F during the operating cycle. Certain design bases were reevaluated to determine

*To simplify discussion on FWHOS operation, the term "rated feedwater temperature" is used in this report to mean "feedwater temperature at 100% core thermal power (100% steam flow) and 100% core flow conditions." The feedwater temperature at any given core power and flow is dependent upon the combination of operable heater(s) in each of the two strings of heaters and the core power level.

the effects of FWHOS operation. Table 1-1 lists potentially affected design evaluations along with a brief description of the technical issues that arise for FWHOS operation.

1.2 SUMMARY

Evaluations were performed to justify FWHOS operating conditions for a rated feedwater temperature range of 420°F to 320°F. Results of these evaluations, as discussed in this report, indicate that RBS is capable of safe operation with partial feedwater heating corresponding to a feedwater temperature reduction of up to 100°F below normal condition, provided that applicable limits are observed. Table 1-2 lists the limits that have changed as a result of this reduced feedwater temperature operation. The scope of this analysis does not include the coastdown portion of the fuel cycle.

Table 1-1

ITEMS POTENTIALLY AFFECTED BY OPERATION WITH FWHOS

Item	Comment
Transient Response[a]	Increased core inlet subcooling can change calculated ΔCPR and peak pressures for limiting transients.
Stability Margins[a]	Increased core subcooling can cause an increase in calculated decay ratio.
ECCS Thermal-Hydraulic Analysis[a]	Increased mass release from a design basis loss-of-coolant accident (LOCA) can cause changes in the reactor blowdown response.
Acoustic Loads During Postulated LOCA Events	The increased subcooling in the reactor downcomer region could affect predicted acoustic loads on the shroud and jet pumps.
Annulus Pressurization Loads During Postulated LOCA Events	Increased mass release rates from a design basis LOCA can cause increased loads on the reactor vessel.
Containment Responses and Loads During Postulated LOCA Events	Drywell pressurization rate, and peak drywell pressure could be affected by higher blowdown release which could impact containment calculations.
Feedwater Nozzle, Sparger and Piping Fatigue	Reduced feedwater temperature can affect fatigue usage.

[a]These items may be cycle and fuel-type dependent. (The other items are generic and cycle-independent.)

Table 1-2
OPERATING LIMIT CHANGES

40-year Average Number of Days of Allowble Usage During an Operating Year for FWHOS Operation without Exceeding the Feedwater Sparger Fatigue Usage Factor Limit	Rated FW Temperature between 420°F and 370°F - 256 Days Rated FW Temperature between 370°F and 320°F - 61 Days

2. TRANSIENT EVENT ANALYSIS

2.1 ABNORMAL OPERATING TRANSIENTS

All core-wide transients described in RBS USAR Chapter 15 and RBS Cycle 2 Reload Licensing Submittal (Reference 1) were examined for FWHOS operation. Three limiting abnormal operating transients reported in Reference 1 were reevaluated in detail for the FWHOS operation:

a. Generator Load Rejection with Bypass Failure (LRBPF)
b. Feedwater Flow Controller Failure (FWCF)
c. Loss of 100°F Feedwater Heating (LFWH)

The reevaluations for the LRBPF and FWCF events were performed at 100% power/100% core flow condition with a rated feedwater temperature of 320°F for Cycle 2. Reactor heat balance, core coolant hydraulics and nuclear transient parameter data were generated and used in the transient analysis. The initial conditions for the FWHOS transient analysis are listed in Table 2-1. The GEMINI/ODYN procedure described in Reference 2 was used for analyses of both the LRBPF and FWCF transients. The transient peak values and minimum critical power ratio (MCPR) results for the two cases analyzed are summarized in Tables 2-2 and 2-3 respectively, with the licensing values included for comparison. The transient responses are presented in Figures 2-1 and 2-2. The results show that the FWCF ΔCPR for the FWHOS operation increases slightly compared to the licensing value, but is still below the Technical Specfication OLMCPR.

The RBS plant-specific analysis for the 100°F LFWH transient was performed at 102% power/100% core flow condition with 320°F rated feedwater using the GE Three-Dimensional BWR Core Simulator described in Reference 3. The results show that the ΔCPR for the 100°F loss initiated from 320°F is 0.09 compared to 0.11 for 420°F initiation case, indicating that the 100°F LFWH is less severe if initiated from 320°F than 420°F. Therefore, the LFWH for FHWOS is adequately bounded by the normal feedwater temperature case.

2.2 ROD WITHDRAWAL ERROR

The rod withdrawal error (RWE) transient documented in USAR Chapter 15 was analyzed using a statistical evaluation of the minimum critical power ratio (MCPR) and Linear Heat Generation Rate (LHGR) response to the withdrawal of ganged control rods from both rated and off-rated conditions over the entire operating region. Therefore, this analysis covers a wide variety of feedwater temperatures and core subcooling as different off-rated conditions are included in the database. The 95% probability 95% confidence values from this statistical database are used to develop the Rod Withdrawal Limiter (RWL) system setpoints to protect against a rod withdrawal error.

The rod withdrawal error analysis does not need to be evaluated for FWHOS at end of cycle because all control rods will be fully withdrawn. A RWE analysis was performed at 2000 MWD/T before end of equilibrium cycle to examine the effect of the initial feedwater temperature. An initial condition of 250°F was used to bound all FWHOS operation. Results show that ΔCPR values resulting from the worst two feet of withdrawal for the 420°F and 250°F feedwater temperature are identical. Therefore, the ΔCPR values initiating from 250°F feedwater temperature condition fall within the statistical database used to establish the RWL system setpoints. It is concluded that operating limit MCPR does not need to be increased due to RWE for FWHOS operation.

2.3 OPERATING LIMIT MCPR

The MCPR results for the FWHOS operation are summarized in Table 2-3. The Cycle 2 licensing analysis results are included for comparison. The results show that although the FWCF MCPR increases slightly for the FWHOS operation, it is bounded by the RWE MCPR which is not affected by the FWHOS operation. Based on these results, it is concluded that t current RBS Technical Specification (Reference 6) OLMCPR is adequate for FWHOS operation for a rated feedw - temperature range of 420°F to 320°F.

The off-rated power-dependent MCPR ($MCPR_p$) limits are not affected by FWHOS operation since the full power MCPR limit is not affected by FWHOS operation.

The current off-rated flow-dependent MCPR ($MCPR_f$) limit curve is based on the steepest power/flow rod line to protect against the recirculation flow runout transient. A power/flow rod line was generated for the 320°F rated feedwater temperature. It shows that the slope of this rod line is bounded by the current design basis rod line. Therefore, the current $MCPR_f$ limits are valid for FWHOS operation.

2.4 ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

An evaluation was performed which shows that reducing feedwater temperature helps to reduce the consequences of an ATWS event. Reduced feedwater temperature results in a reduction of steam flow and core average void fraction. The lower steam flow rate is produced because more of the core heat is needed to heat up the colder moderator in the core. Therefore, less steam is generated at its rated power as feedwater temperature decreases.

It is concluded that if an ATWS event were initiated at RBS from the FWHOS operation conditions, the results would be less severe than if it were initiated from rated feedwater temperature at 420°F.

2.5 OVERPRESSURIZATION ANALYSIS

Lower initial operating pressure and steam flow rate provide better overpressure protection for the most limiting Main Steam Line Isolation Valve event during FWHOS operation. Table 2-2 also indicates that the peak vessel pressures for the LRBPF and FWCF events analyzed for FWHOS are below those for all feedwater heaters operating case. Hence, it is concluded that the pressure barrier integrity is maintained under the FWHOS operation.

Table 2-1

INITIAL CONDITIONS FOR
FWHOS TRANSIENT ANALYSIS*

		Rated T_{FW} - 320°F
1.	Thermal Power Level (MWt)	2894
2.	Steam Flow (lb/sec)	3049
3.	Core Flow (Mlb/hr)	84.5
4.	Feedwater Flow Rate (lb/sec)	3049
5.	Feedwater Temperature (°F)	320
6.	Vessel Dome Pressure (psig)	1005
7.	Core Exit Pressure (psig)	1015
8.	Core Coolant Inlet Enthalpy (Btu/lb)	514.2
9.	Turbine Inlet Pressure (psig)	979

*The other input data used for the transient analysis are the same as those used in Reference 1.

Table 2-2

SUMMARY OF TRANSIENT PEAK VALUE RESULTS

Transient	Rated FW Temperature (°F)	Peak Neutron Flux (% NBR)	Peak Heat Flux (% NBR)	Peak Vessel Pressure (psig)	ΔCPR[b]
LRBPF	420[a]	286.2	107.8	1213	0.07
	320	284.9	107.6	1192	0.07
FWCF	420[a]	230.2	108.2	1202	0.06
	320	253.8	110.6	1177	0.08

(a) Licensing Basis (Reference 1)

(b) CPR based on an initial CPR which yields a safety limit MCPR of 1.07; uncorrected for ODYN adjustment.

Table 2-3

SUMMARY OF MCPR RESULTS

Transient	Cycle 2 Licensing Basis	FWHOS
LRBPF	1.15	1.15
FWCF	1.14	1.16
LFWH	1.18	1.16
RWE	1.18*	1.18

*The RBS Cycle 2 Licensing Basis OLMCPR

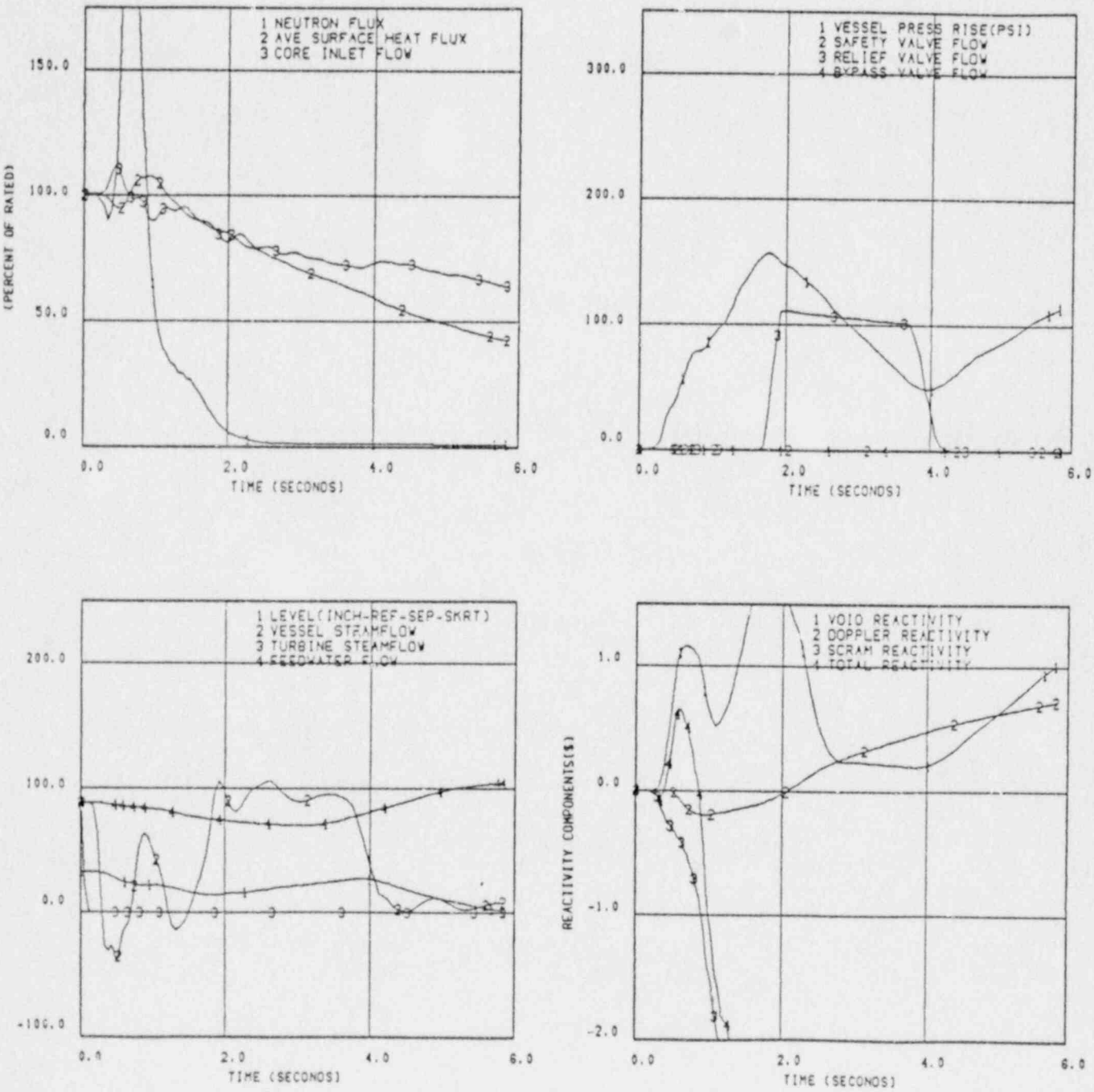

Figure 2-1. Plant Response to Generator Load Rejection with Bypass Failure, 100% Power/100% Core Flow, 320°F Rated Feedwater Temperature

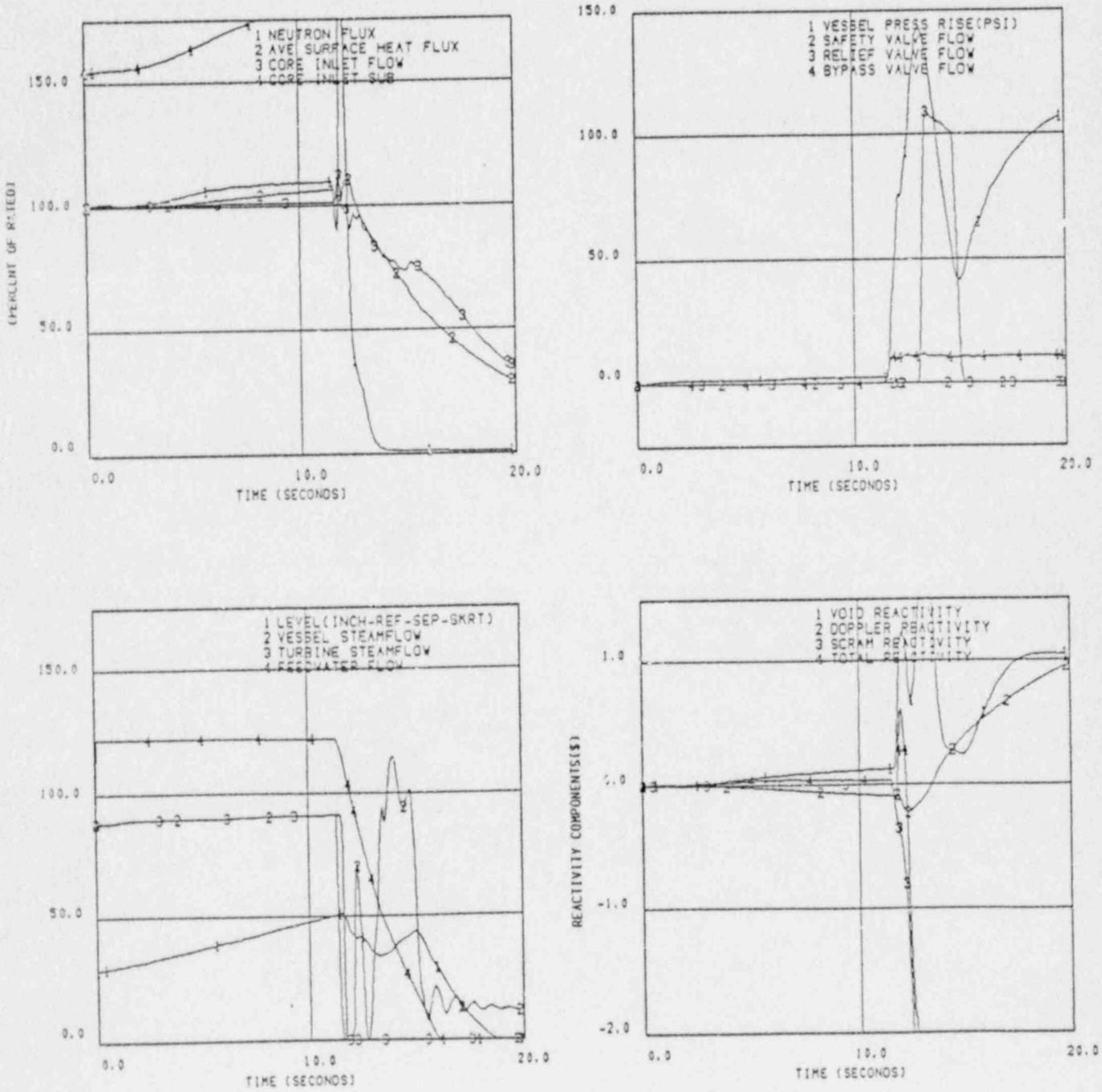

Figure 2-2. Plant Response to Feedwater Controller Failure, 100% Power/100% Core Flow, 320°F Rated Feedwater Temperature

3. THERMAL-HYDRAULIC STABILITY ANALYSIS

General Design Criterion 12 (10CFR50, Appendix A) states that power oscillations which result in exceeding specified acceptable fuel design limits be either not possible or can be readily and reliably detected and suppressed. Historically, compliance to GDC-12 was demonstrated by assuring that neutron flux oscillations would not occur. This eliminated the need to perform fuel integrity calculations under limit cycle conditions. As a result of stability tests at operating BWRs and extensive development and qualification of GE analytical models, stability criteria have been developed which also demonstrate compliance to GDC-12. Reference 4 provides these stability compliance criteria for GE fueled BWRs operating in the vicinity of limit cycles. The NRC has reviewed and approved this in Reference 5; therefore, a specific analysis for each cycle is not required.

Operation in the FWHOS mode is bounded by the fuel integrity analyses in Reference 4. In general, the effect of reduced feedwater temperature results in a higher initial CPR, which yields even larger margins than those reported in Reference 4. The analyses are independent of the stability margin, since the reactor is already assumed in limit cycle oscillations. Reference 4 also demonstrates that for neutron flux limit cycle oscillations just below the 120% 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 5). These evaluations demonstrate that substantial thermal/mechanical margin is available for the GE BWR fuel designs even in the unlikely event of very large oscillation.

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 operator recommendations has been developed 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.

When operating in FWHOS mode during a cycle, the colder feedwater flow increases the core inlet subcooling and will also result in power distribution changes. These changes result in reduced stability margin when operating in the high-power/low-flow region of the operating domain. Tests performed at an overseas BWR/6 in October 1984 evaluated the effects of reduced feedwater temperature during a cycle on stability margins. It was determined that the reduction in stability margin is within the conservative basis of the operator recommendations and, therefore, the recommendations are applicable for FWHOS during the cycle.

4. IMPACT ON LOSS-OF-COOLANT ACCIDENT AND RELATED ANALYSES

4.1 ECCS THERMAL-HYDRAULIC PERFORMANCE

A Loss-of-Coolant Accident Analysis (LOCA) was performed for RBS with FWHOS operation. Reduction of feedwater temperature results in increased subcooling in the vessel, thus increasing the mass flow rate out of a LOCA break. However, an increase in initial total system mass and a delay in lower plenum flashing also occur. They act together to decrease the impact of increased flow out of the recirculation line break. As a result of this offsetting effect, the peak cladding temperature was shown to be lower than the 2144°F value reported for RBS and below the 2200°F 10CFR50.46 cladding temperature limit.

4.2 ACOUSTIC AND FLOW-INDUCED LOADS ON REACTOR VESSEL INTERNALS

The acoustic loads are lateral loads on the vessel internals that result from propagation of the decompression wave created by a postulated recirculation suction line break. The acoustic loading on the vessel internal is proportional to the total pressure wave amplitude in the vessel recirculation outlet nozzle. FWHOS increases subcooling in the downcomer. This results in a lower saturation pressure, thereby having a larger total pressure amplitude and resulting in larger acoustic loads.

The flow-induced loads are additional lateral loads on the vessel internals that result from high-velocity flow in the downcomer in a postulated recirculation line break. These loads are proportional to the square of the critical mass flux rate out of the break. Higher subcooling in the downcomer under FWHOS increases the critical flow and flow-induced loads.

The reactor internals most impacted by acoustic and flow-induced loads under FWHOS operation are the shroud, shroud support and jet pump. The impact on these components was evaluated throughout the FWHOS operating power flow region. The analyses concluded that these components have enough design margin to handle the loading during FWHOS.

4.3 ANNULUS PRESSURIZATION (AP) LOADS

A study has been performed to assess the impact of FWHOS operation on the annulus pressurization (AP) loads for River Bend. A review of RBS USAR Figures 6.2-39 through 6.2-55 indicates that the feedwater line break results in the greatest forces upon the RPV and the greatest pressure differentials across the biological shield wall. Therefore, an evaluation of the feedwater line break flow has been performed in this study. The break flow for the feedwater line break with FWHOS were determined to be less than those presented in the USAR during the inventory depletion period of the feedwater line when the peak AP loads occur. Therefore, the normal operation AP loads calculated in the RBS USAR bound those expected to result under FWHOS operation.

4.4 CONTAINMENT RESPONSE

The impact of FWHOS on the containment LOCA response was evaluated. Both main steamline break and recirculation line break were analyzed over the FWHOS operation power/flow region. The peak drywell and wetwell pressure and temperature, pool swell, condensation oscillation and chugging load during FWHOS operation were evaluated.

The peak drywell-to-containment differential pressure during the FWHOS operation occurred under recirculation line break at the maximum vessel subcooling condition in the power/flow map. This peak differential pressure increased by 0.2 psi compared to the design basis accident main steamline break; however, this differential pressure (18.8 psid) is still below the design differential pressure of 25 psid reported in USAR Section 6. Also, the peak pool swell, condensation oscillation and chugging loads evaluated during FWHOS operation vary slightly over the peak values presented in USAR Section 6. However, the analysis concluded that the variation is insignificant and there is enough design margin to handle these loads during FWHOS operation.

5. FEEDWATER NOZZLE, SPARGER AND PIPING FATIGUE USAGE

5.1 FEEDWATER NOZZLE

An evaluation was performed on the feedwater nozzle in RBS for FWHOS operation. Assuming a full, single 18-month cycle operation with feedwater heater out of service based on an 80% capacity factor would result in 438 full power days operation per cycle. This will result in an additional 0.0214 fatigue usage factor over 40 years of continuous FWHOS operation. Thus, the fatigue usage factor will still be less than 0.8, which is below the limit of 1.0.

5.2 FEEDWATER SPARGER

An evaluation was performed to examine the impact of FWHOS operation on the feedwater sparger for RBS. Two cases were analyzed to determine the number of days allowable per year (for 40 years) for FWHOS operation without exceeding the feedwater sparger fatigue usage factor limit of 1.0. The results show that the 40-year average number of days allowable during an operating year for FWHOS operation decreases with lower feedwater temperature; 256 days and 61 days for rated feedwater temperatures of 370°F and 320°F, respectively.

5.3 FEEDWATER SYSTEM PIPING

A standard stress analysis was performed on the feedwater system piping up to the first feedwater guide lug outside the containment for feedwater temperature at 250°F to bound the 320°F rated feedwater temperature case. Results of the study show that with the additional FWHOS operations, the feedwater piping fatigue usage factor still meets the allowable limit of 1.0.

6. REACTOR PROTECTION SYSTEM LOW POWER SETPOINT

At reactor power levels where significant amounts of steam are being generated the fast closure of turbine stop or control valves will result in rapid reactor vessel pressurization. When pressure increases, power increases especially if the bypass valves fail to open. For this reason, scram occurs on turbine stop valve position and control valve fast closure to provide margin to the core thermal-hydraulic safety limit.

However, at sufficiently low initial thermal power levels, steam flow is within the turbine bypass system capability and only a mild core transient occurs without a need for automatic shutdown. Therefore, automatic shutdown on stop valve closure and fast control valve closure is bypassed at low power. On BWR/6 systems the required lower bound for stop valve and control valve fast closure scram is 40% of rated thermal power. Turbine first stage pressure (TFSP) is the parameter used to initiate the turbine stop valve closure scram bypass functions. At normal feedwater heating operating conditions, this 40% power is equivalent to approximately 30% of the TFSP (in psia) that would exist at turbine valves wide-open steam flow conditions. Below 40% power, the turbine stop valve or control valve scram functions are disabled. At these low power levels, high neutron flux scram and vessel pressure scram and other scram functions are sufficient to provide the safety limit margin even with stop valve or control valve sudden closures.

Under operation with reduced feedwater temperature the relationship between vessel steam flow (and therefore TFSP) and core thermal power changes. Less steam flow is generated at the same thermal power and TFSP is reduced. Therefore the effect of reduced feedwater temperature is to raise the scram bypass power level (by approximately three percent for 320°F rated feedwater temperature). Thus, it is necessary to review the turbine stop and control valve scram bypass setpoints to determine if adjustments are necessary for FWHOS operation to maintain the required 40% bypass setpoint.

Conservatism in current RBS Technical Specification scram bypass TFSP nominal setpoint was assessed by comparing it to the RBS startup test data for "TFSP vs Reactor Power". The nominal TFSP setpoint corresponding to 40% rated power (using measured plant data where TFSP is 754.6 psia at turbine control valves wide open) is estimated to be 235 psia for normal feedwater temperature case and 218 psia for 320°F rated feedwater temperature case. The RBS Technical Specification nominal setpoint is 191 psia. This means that the current Technical Specification setpoint is conservative in the scram bypass power level by approximately 4% of rated power compared to that actually required for 320°F rated feedwater temperature operation including all required uncertainties and allowances. Therefore, the existing conservatism in the Technical Specification setpoint justifies no adjustment of current TFSP setpoint for the FWHOS operation.

It is concluded that the current TFSP setpoint has enough margin to accommodate effects of the FWHOS operation. Therefore, the RPS TFSP setpoints will remain adequate for safe operation of RBS with FWHOS operation.

7. REFERENCES

1. Supplemental Reload Licensing Submittal for River Bend Station Reload 1, 23A5819, Rev. 0, Class I, July 1987.

2. J.S. Charnley to H.N. Berkow (NRC), "Revised Supplementary Information Regarding Amendment 11 to GE Licensing Topical Report NEDE-24011-P-A", January 16, 1986.

3. "Three-Dimensional BWR Core Simulator", NEDO-20953-A, January 1977.

4. G.A. Watford, "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria", General Electric Company, October 1984 (NEDE-22277-P-1).

5. "General Electric Standard Application for Reactor Fuel", General Electric Company, May 1986 (NEDE-24011-P-A-8).

6. Technical Specification for River Bend, NUREG 1172, Docket No. 50-458, Appendix A to License No. NPF-47, Amendment #12.

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