August 22, 2000

Mr. Randall K. Edington
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
P. O. Box 220
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: FINAL FEEDWATER TEMPERATURE REDUCTION (TAC NO. MA1594)

Dear Mr. Edington:

The Commission has issued the enclosed Amendment No. 112 to Facility Operating License (FOL) No. NPF-47 for the River Bend Station, Unit 1 (RBS). The amendment consists of changes to paragraph 2.C(13) of the FOL in response to your application dated April 9, 1998, as supplemented by letters dated January 13, 1999, and June 28, 2000.

The amendment allows RBS to operate with final feedwater temperature reduction (FFWTR) in order to extend the fuel cycle by maintaining the core thermal power at or close to rated power thus delaying the start of normal coastdown.

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA by N. Kalyanam for/

Jefferey F. Harold, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures: 1. Amendment No. 112 to NPF-47
2. Safety Evaluation

cc w/encls: See next page

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Accession No.:

*No major changes made to Safety Evaluation

OFFICE	PDIV-1/PM	PDIV-1/LA	SRXB/SC	EMEB/SC	SPLB/SC	OGC	PDIV-1/SC
NAME	JHarold	DJohnson	RCaruso*	KManoly	~~G~~Hubbard	[illegible]	RGramm
DATE	8/10/00	8/9/00	8/16/00	8/10/00	8/15/00	8/21/00	8/22/00

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY GULF STATES, INC.

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 112
License No. NPF-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc.* (the licensee) dated April 9, 1998, as supplemented by letters dated January 13, 1999, and June 28, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

* Entergy Operations, Inc. is authorized to act as agent for Entergy Gulf States, Inc, and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. NPF-47 is amended as indicated in the attachment to this license amendment.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to Facility Operating License

Date of Issuance: August 22, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 112

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following pages of Facility Operating License NPF-47 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
-6-	-6-

(13) Partial Feedwater Heating (Section 15.1. SER)

During power operation, the facility shall not be operated with a feedwater heating capacity which would result in a rated thermal power feedwater temperature less than 320 °F.

(14) Emergency Response Capabilities (Generic Letter 82-33. Supplement 1 to NUREG-0737. Section 7.5.2.4. SER and SSER 3. Section 18. SER. SSER 2 and SSER 3)

EOI shall complete the requirements of NUREG-0737 Supplement #1 as specified in Attachment 5. Attachment 5 is hereby incorporated into this license.

(15) Salem ATWS Events. Generic Letter 83-28 (Section 7.2.2.5. SSER 3)

EOI shall submit responses to and implement the requirements of Generic Letter 83-28 on a schedule which is consistent with that given in its letters dated August 3. 1984 and May 30. 1985.

(16) Merger Related Reports

Entergy Gulf States. Inc. shall inform the Director. NRR:

a. Sixty days prior to a transfer (excluding grants of security interests or liens) from Entergy Gulf States. Inc. to Entergy or any other entity of facilities for the production. transmission or distribution of electric energy having a depreciated book value exceeding one percent (1%) of Entergy Gulf States. Inc.'s consolidated net utility plant. as recorded on Entergy Gulf States. Inc.'s books of account.

b. Of an award of damages in litigation initiated against Entergy Gulf States. Inc. by Cajun Electric Power Cooperative regarding River Bend within 30 days of the award.

(17) Primary containment air lock doors may be open during CORE ALTERATIONS. except when moving recently irradiated fuel. (i.e.. fuel that has occupied part of a critical reactor core within the previous 11 days). provided the following conditions exist:

1) One door in each air lock is capable of being closed.

2) Hoses and cables running through the air lock employ a means to allow safe. quick disconnect and are tagged at both ends with specific instructions to expedite removal.

3) There is a minimum of 23 feet of water over the core.

4) The air lock doors are not blocked open to allow expeditious closure.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By application dated April 9, 1998 (Reference 1), as supplemented by letters dated January 13, 1999, and June 28, 2000, Entergy Operations, Inc. (EOI or the licensee) requested changes to Facility Operating License (FOL) No. NPF-47 for the River Bend Station, Unit 1 (RBS). The proposed changes would revise License Condition 2.C(13) and allow RBS to operate with final feedwater temperature reduction (FFWTR) in order to extend the fuel cycle by maintaining the core thermal power at or close to rated power by delaying the start of normal coastdown. The January 13, 1999, letter provided a revised proprietary version of the licensee's analysis submitted in its original April 9, 1998, application and the June 28, 2000, letter provided additional information to support staff review of the original application, and thus did not affect the initial finding of no significant hazards consideration determination dated May 20, 1998 (63 FR 27762).

2.0 BACKGROUND

Section 15.1, "Decrease in Core Coolant Temperature," of NUREG-0989, "SER [Safety Evaluation Report] Related to the Operation of River Bend Station," issued on May 1984, states the following:

> "The applicant was asked to justify that operation with partial feedwater heating to extend the cycle beyond the normal end of cycle condition would not result in a more limiting change in minimum critical power ratio than that obtained using the assumption of normal Feedwater heating. The staff requires that analysis be provided prior to operation in this mode, if a decision is made to operate in this mode. Until such analyses are provided, the staff will condition the license to prohibit operation in this mode."

Amendment No. 37 (Reference 2), issued on July 6, 1989, re-instituted Licensing Condition 2.C(13) prohibiting operation of RBS with reduced feedwater (FW) temperature beyond the end of the normal cycle (EOC) without a written approval from the staff. In the amendment request, EOI submitted safety analyses consistent with the Chapter 15 requirements of the Updated Safety Analysis Report (USAR). General Electric Nuclear Energy (GENE) performed safety analyses, which justifies operating with reduced FW temperature at EOC. The licensee seeks to operate RBS with reduced FW temperature at EOC in order to delay the normal coastdown

period while maintaining the licensed core thermal power. The licensee evaluated the impact of the FFWTR operation on the thermal limits, fuel thermal-mechanical performances, and the applicable Chapter 15 accident analysis for loss of coolant accidents (LOCAs) and anticipated transients without scram (ATWS). The amendment proposes that License Condition 2.C(13) should state,

> "During power operation, the facility shall not be operated with a feedwater heating capacity which would result in a rated thermal power feedwater temperature less than 320 °F."

RBS is licensed to operate with an increased core flow (ICF) of 107%, and the licensee will initially extend the cycle by operating with ICF. The licensee indicated that after cycle extension from ICF is exhausted, FFWTR would then provide a sufficient reactivity increase to maintain the thermal power at 100% and delay the normal power coastdown.

The licensee states that, if the FFWTR operation at EOC is authorized and implemented, the facility could operate for 14 effective full-power days beyond the current cycle. The licensee also expects to gain similar economic benefits from future cycles under FFWTR operation. Cycle 7 is the reference cycle for evaluating the impact of FFWTR during the cycle extension operation. GENE designates operation at the 100% power, 100% core flow, at EOC exposure, with all-rods-out (ARO) as "EOC." The EOC operating condition with increased core flow (107% core flow), is referred to as extended EOC or "EEOC." Similarly, further cycle extension with reduced FW temperature at EEOC is designated as "EEEOC" (100% power, 107% core flow, ARO, at EOC exposure point). All of the transient analyses that are used to determine the impact of the FFWTR operation were performed under EEEOC operating conditions and for simplicity, this condition will be referred to as FFWTR condition. This SER reviews the impact of FFWTR operation on the safe operation of RBS and determines if the analyses are consistent with the applicable regulations, standard review plan, and the Nuclear Regulatory Commission (NRC)-approved methodologies.

SUMMARY OF TERMS

Term	Cycle Exposure	Reactivity/ Control Rods	Reactor Thermal Power	Core Flow	FW Temperature
EOC	End-of-Cycle	ARO	100%	100%	420 °F
EEOC	End-of-Cycle	ARO	100%	**107%**	420 °F
EEEOC	End-of-Cycle	ARO	100%	**107%**	**320 °F**

3.0 <u>EVALUATION</u>

3.1 <u>Anticipated and Abnormal Operation Analysis</u>

The licensee states that the impact of the FFWTR operation on the operating limits and the fuel thermal-mechanical performance is cycle-specific, and similar analysis will be performed to support FFWTR for future cycles. The FFWTR analysis is based on a FW temperature of 320 °F with a deviation of +/- 3%.

3.1.1 Limiting Operational Occurrences

The licensee analyzed the following limiting anticipated operational occurrence (AOO) transients in justifying FFWTR operation for the RBS reference Cycle 7 (Reference 4).

- feedwater controller failure (FWCF) maximum demand
- load reject with no bypass (LRNBP)
- turbine trip with no bypass
- pressure regulator failure downscale

Table 1 (below) summarizes the RBS Cycle 7 bounding operating limit minimum critical power ratio (OLMCPR) for each fuel type, and the corresponding limiting AOO transient for GE8X8EB and GE11 fuel at normal cycle and rated FW temperature (EEOC) in comparison with the EEEOC operating condition (ICF/FFWTR, ARO, at EOC).

TABLE 1

FUEL TYPE	**OLMCPR @100% Power, 107% Core Flow, FW = 420 °F, @EEOC**	**OLMCPR @ 100% Power, 107% Core Flow, FW = 320 °F, @EEEOC****
GE8X8EB	1.25 (Fuel Loading Error (FLE))	1.22 (FWCF)
GE11	1.32 (LRNBP)	1.30 (FWCF)
GE8X8EB (100 °F LFWH)	1.21 (Loss of FW Heating (LFWH))	not analyzed
GE11 (100 °F LFWH)	1.22 (LFWH)	not analyzed

** For RBS reference Cycle 7, the safety limit minimum critical power ratio is 1.10 for two loops and 1.12 for single loop operation.

Licensee's Analysis

According to the licensee, the FWCF is the most limiting transient for FFWTR operation with an OLMCPR of 1.30 and 1.25 for GE11 and GE8X8EB fuel, respectively. At rated FW temperature, the limiting transient for GE11 is LRNBP with an OLMCPR of 1.32. The limiting transient for GE8X8EB under rated FW temperature is set by the FLE-mislocated analysis with an OLMCPR of 1.25. Therefore, the normal fuel cycle transients bound the FFWTR operation transients in the reference Cycle 7.

The licensee also analyzed the 100 °F LFWH transient during rated FW temperature operation (EEOC) and the Δ minimum critical power ratio (ΔMCPR) for the LFWH transient remains bounded by the LRNBP and the FLE transients for GE11 and GE8X8EB fuel types, respectively. EOI did not evaluate the LFWH transient with FFWTR initial condition; however, the licensee stated that LFWH event is expected to yield milder results during FFWTR operation. With an initial FW temperature of 320 °F, the additional subcooling due to the

100 °F LFWH will not have a significant impact. The licensee pointed out that similar trends were observed in the Grand Gulf Nuclear Power Station (Grand Gulf) LFWH transient analysis. In the Grand Gulf analysis, the LFWH event during reduced FW temperature operation resulted in a less severe thermal response than the LFWH event occurring under rated FW temperature operation. EOI reiterated that the OLMCPR for LRNBP (GE11) and FLE (GE8X8EB) events bound the LFWH event under the normal FW operating condition. Similarly, FWCF transient will bound LFWH event during FFWTR operation.

Staff's Review

The staff reviewed the licencee's RBS FFWTR safety analysis submittal, the supplemental Reload Licensing Report (SRLR), and the partial FW temperature reduction safety analyses for RBS (NUREG-0989), Grand Gulf, and the Perry Nuclear Power Plant (Perry). The Cycle 7 SRLR shows LRNBP and FWCF as the most limiting transients during FFWTR operation. The SRLR indicates that for FFWTR operation, FWCF and LRNBP transients for GE11 both yield a ΔMCPR of 0.18. However, the licensee reported in the submittal that only the FWCF will be evaluated for future reload cycles in evaluating the FFWTR operation. In communications with the licensee (Reference 3), the staff requested a justification for the discrepancy in the limiting and dominant transients during FFWTR operation. EOI responded (Reference 3) that the SRLR presents rounded values and that, in fact, the FWCF is bounding for the FFWTR operating condition. EOI's conclusion is consistent with the results reported in both Grand Gulf and Perry safety analyses, which identify the FWCF as the most limiting transient. Since, the actual OLMCPR of 1.32 for RBS Cycle 7 is conservative relative to the FFWTR operating limit of 1.30, the licensee's justification is acceptable.

The licensee also concluded that the LFWH event during FFWTR operation has a less severe impact on the operating limit than the LFWH event initiated at the rated FW temperature of 420 °F. This finding is also consistent with the evaluation done for Perry, a boiling water reactor (BWR)/6 plant (3539 MWe). Perry is one of the plants that is licensed to operate with reduced FW temperature, including the EOC exposure, with ARO operation. Cleveland Electric Illuminating Company (CEIC), the licensee for Perry, analyzed the AOO transients for a range of FW temperatures up to 250 °F. The Perry analysis was based on 104% power, 100% core flow, with FW temperatures of 370 °F, 320 °F, and 250 °F for EOC with ARO. CEIC stated that the LFWH results show that the effect of the colder FW is less as the temperature of the moderator decreases. For Perry, the Δcritical power ratio (ΔCPR) for the LFWH transient during rated FW temperature (420 °F) is greater than the LFWH ΔCPR for a FW temperature of 320 °F. Therefore, the Perry analysis supports the licensee's findings that the effect of FW temperature on the LFWH event is less severe at colder moderator temperatures.

However, the Perry safety analysis indicated that for the FWCF transients, the moderator temperature has a significant effect on the ΔMCPR. The highest ΔMCPR occurred at 250 °F FW temperature for the FWCF transient analysis at different FW temperatures. The ΔMCPR at the FW temperature of 250 °F was 0.04 higher than the ΔMCPR for 370 °F. EOI confirmed to the staff that RBS is not expected to operate at FW temperatures below 320 °F and the modified license condition will explicitly state that the minimum FW temperature is 320 °F. In the event of a loss of stator cooling where a FW temperature drop of greater than 100 °F may occur, RBS will automatically scram on turbine trip or high pressure because RBS has a steam

bypass capacity of 10%. Since the license condition will prohibit operation with a FW temperature of less than 320 °F, the staff finds the justification acceptable.

In the review of the AOO transient analyses, the staff agrees that the actual Cycle 7 OLMCPR of 1.32 (LRNBP) is conservative relative to the FFWTR operation OLMCPR (FWCF with an OLMCPR of 1.30). The staff also finds EOI's justifications and findings for the LFWH events, and the limiting transient for the FFWTR operation consistent with results obtained in the analyses done for other BWR/6 plants.

Fuel Integrity Evaluation:

In reviewing the impact of the FFWTR on the GENE fuel integrity, the licensee evaluated the fuel-specific and transient-specific thermal overpower (TOP) and mechanical overpower (MOP). The licensee stated that the peak TOP and MOP responses for each transient fuel complied with the TOP and MOP design criteria, and the staff finds this acceptable.

Offrated Limits:

In Reference 3, GENE stated that the RBS uses both power-dependent and flow-dependent offrated limits. The power-dependent offrated limits consider both high-flow and low-flow conditions as well as FFWTR and normal temperature. For given power/flow operating combinations, the most limiting of the high-flow, low-flow, FFWTR or normal FW temperature sets the offrated power limits. When operating under a given power/flow condition, the most limiting of the flow offrated limit or the power offrated limit will determine and set the thermal limit for that specific operating condition. In single-loop operation, these offrated power/flow limits are adjusted to account for the change in the safety limits.

The licensee also pointed out that the critical heat flux correlation database covers a subcooling range of 0 - 70 Btu/lbm as given in Amendment 22 to General Electric Standard Application for Reactor Fuel (GESTAR II) (Reference 5) and the 320 °F FW temperature falls within the database range. The staff finds this acceptable.

3.1.2 Vessel Overpressure Performance Analysis

The licensee reported that the main steam isolation valve closure with flux scram (MSIVF) is the limiting vessel overpressure transient for RBS. The MSIVF event during the rated FW temperature operation is more conservative than during FFWTR operation. The reduced steam generation rate during the FFWTR operation yields a milder vessel pressurization transient. Therefore, the MSIVF transient under the rated FW temperature condition yields higher peak vessel pressure than under the FFWTR operation. This finding is also consistent with the results obtained in a number of BWR/6 plants and the staff also finds the licensee's justification acceptable.

3.1.3 Rod Withdrawal Error

The GENE generic rod withdrawal error (RWE) analysis did not bound the RBS reference Cycle 7 RWE transient during normal operation. However, the licensee did not analyze the RWE event during FFWTR operation, stating that the inlet subcooling affect on the RWE transient is insignificant. According to the licensee, the dominant parameters affecting the

RWE transient response are the control rod pattern and the position of the error rod. Thus, FFWTR operation does not affect the RWE response and EOI concluded that analysis of the RWE during FFWTR operation is not necessary.

CEIC did analyze a RWE during Partial FW temperature operation (104.2% power, 100% core flow) at EOC. This analysis used an initial FW temperature range up to 250 °F and the results indicated that reduced FW temperatures had an insignificant effect on the ΔCPRs in a random RWE analyses. CEIC concluded that the 420 °F FW temperature operation RWE analysis adequately bounds the partial feedwater temperature operation RWE. Therefore, the RBS RWE evaluation is consistent with the Perry findings. Moreover, the Cycle 7 SRLR indicates that the RWE OLMCPR is bounded by an overall Cycle 7 OLMCPR of 1.32 for rated FW operation. The staff finds the licensee's evaluation acceptable.

3.1.4 Transient Event for Future Reloads

The licensee proposes that for future cycles, only the FWCF transient will be analyzed for the FFWTR operation. EOI pointed out that for every reload, all of the limiting transients will be analyzed for normal cycle operation and rated FW temperature. For FFWTR operation, the FWCF transient will be included in the reload analysis since the FWCF transient is the only FFWTR transient that yields a higher OLMCPR value than if the same transient occurred during rated conditions. The staff accepts EOI's conclusion since FWCF is the most limiting transient during FFWTR condition and this finding is consistent with reduced FW temperature analyses done for other BWR/6 facilities.

3.2 Rod Drop Accident

RBS uses a banked position withdrawal sequence (BPWS), and GENE had performed generic BPWS analysis. As part of compliance with Amendment 25, GENE verified the applicability of the new fuel designs up to GE13 to the generic BPWS analysis. The licensee stated that RBS plant specific control rod drop accident (CRDA) analysis is bounded by the generic analysis, which covered a wide range of initial moderator temperatures. In addition, the generic analysis conservatively did not account for the negative voids feedback effect and the FFWTR operating condition does not have an adverse impact on the worth of the rod and the corresponding analysis. Consequently, the control rod worth is bounded by the CRDA generic analysis. The staff finds this acceptable.

3.3 Fuel Loading Error

EOI stated that the FLE is predominantly affected by the R-factor uncertainty change, and the R-factor change due to mis-oriented fuel bundle is not affected by the FW temperature reduction. The severity of the misoriented and mislocated bundle events are determined by the ΔCPR characteristics of the fuel bundles involved and that the FW temperature affects both bundles to the same degree. Therefore, the resulting impact is small and the FLE need not be analyzed for FFWTR. The staff finds the licensee's explanation adequate.

3.4 Loss-of-Coolant-Accident

The licensee did not analyze LOCA during FFWTR operation, but instead references the findings of other BWR units analysis. The licensee stated that the increased subcooling, due to the FFWTR condition, or the lower moderator temperature raising the critical mass flow rate, affects negatively the core uncovering time. However, this effect is counterbalanced by the higher initial coolant mass and the delay in the lower plenum flashing due to the increased subcooling. The licensee concluded that the impact of the FFWTR operation on a LOCA consequence was insignificant and the peak cladding temperature (PCT) remains bounded by the current PCT for rated FW temperature operation. In Reference 6, CEIC reported that the LOCA analysis for Perry at a partial feedwater temperature of 250 °F resulted in a lower PCT than for rated FW temperature operation. CEIC also attributes the reduced PCT during partial FW temperature operation to the increase in the total system mass and the delay in the lower plenum flashing. In Reference 7, a Grand Gulf LOCA analysis with 50 °F FW temperature reduction also resulted in lower PCT, in comparison with the rated FW temperature LOCA. This finding is also consistent with the licensee's qualitative review, and the staff finds EOI's LOCA evaluation acceptable.

3.5 Anticipated Transient without SCRAM

The licensee also evaluated qualitatively the impact of the FFWTR operation on ATWS. According to EOI, the colder moderator yields lower steam flow and void fraction. For Cycle 7, the steam flow rate under normal FW temperature is 12.46 Mlbm/hr; the steam flow rate reduces to 11 Mlbm/hr with a 100 °F reduction in the FW temperature (See Appendix A for RBS Cycle 7, Reload 6 of the SRLR). Consequently, during an ATWS event during FFWTR operation, the peak heat flux, vessel bottom pressure, and suppression pool temperature are all lower. Because of the reduced steam generation, the relief valves release a higher percentage of the steam flow and the vessel pressurization rate becomes lower. The mass/energy released into the wetwell would be lower than at rated FW temperature.

The licensee also noted that in previous sensitivity studies for an ATWS event, the PCT for MSIV closure with no scram event at a FW temperature of 170 °F, resulted in a PCT of 1320 °F in comparison to 1542 °F for rated FW temperature. EOI concluded that an ATWS during rated feedwater temperature bounds the results of ATWS under FFWTR conditions. This is consistent with other BWR plants licensed to operate with reduced FW temperature at EOC, and the staff finds the licensee's evaluation to be acceptable.

3.6 Thermal-Hydraulic Stability

The licensee states that the RBS operational procedures include thermal-hydraulic stability operational recommendations, which are consistent with the GENE Service Information Letter (SIL) 380 as well as the NRC Bulletin 88-07, Supplement 1. According to the licensee, the guidelines presented by both SIL 380 and Bulletin 88-07 are applicable to rated and reduced FW and both documents have been incorporated into RBS Cycle 7 reload licensing basis. Therefore, plant- and cycle-specific stability analysis is not required. The staff finds the licensee's justification acceptable.

The staff reviewed the Chapter 15 analysis presented in the current amendment request. The safety analysis and the qualitative justifications provided by the licensee provide sufficient basis

to conclude that RBS can be safely operated with FW temperature of 320 °F, at the EOC exposure point with ARO conditions.

3.7 Reactor Vessel

3.7.1 Feedwater Nozzle and Sparger Fatigue

Amendment No. 37 approved RBS to operate at a reduced feedwater temperature with the FW heaters out of service (FWHOS). The allowable limiting conditions for the FWHOS operation were determined based on the cumulative fatigue usage factors (CUFs) of the feedwater nozzle and the feedwater sparger not exceeding the allowable limit of unity. The licensee has determined, and the staff has accepted, that, on average, during forty years of operation, the permissible limit in feedwater operational temperature reduction is 100 °F for a duration of 61 days per year, or 50 °F for a duration of 256 days per year. Currently, these limits are protected by implementing the RBS plant procedures that track the number of permissible days in the FWHOS operation. These plant procedures and evaluation results are applicable to both FWHOS and FFWTR since the same amount of feedwater temperature reduction of up to 100 °F is assumed for each mode of operation. The licensee also committed in its April 9, 1998, submittal to maintaining these existing plant procedures for both FFWTR and FWHOS conditions. The staff finds the licensee's evaluation to be adequate to maintain the feedwater nozzle and feedwater sparger CUFs within the Code allowable limit of 1.0 since the addition of the FFWTR operation will be within the current monitoring program.

3.7.2 Reactor Internals Mechanical Integrity

The licensee performed a design basis analysis in support of Amendment No. 37 and concluded that sufficient design margin exists for the reactor internals relative to the loads in FWHOS operation. This conclusion is also applicable to the FFWTR operation with a 100 °F feedwater temperature reduction, similar to the FWHOS. The limiting reactor internal components in this operation (i.e., the shroud, shroud support, and the jet pumps) were evaluated for the critical acoustic and flow-induced loads under the FWHOS condition. The licensee further performed thermal hydraulic analysis to evaluate the effects of the FFWTR operation with the increased core flow (ICF/FFWTR). The licensee indicated that the acoustic and flow-induced loads for the ICF/FFWTR condition remain bounded by the existing analysis for the FWHOS operation. Therefore, the staff concludes that the current design of reactor internals is adequate to maintain its structural integrity for the FFWTR and ICF/FFWTR conditions.

3.8 Containment Response

The impact of FWHOS operation on the containment LOCA response was evaluated for both main steamline and recirculation line breaks over the power/flow range for FWHOS operation in support of Amendment No. 37. The peak drywell and wetwell pressure and temperature, pool swell, condensation oscillation, and chugging loads during peak FWHOS operation were evaluated.

The peak drywell-to-wetwell differential pressure during FWHOS operation was calculated to occur for a recirculation line break at the maximum vessel subcooling condition on the power to flow map. The licensee's submittal for Amendment No. 37 stated that the Moody slip flow

correlation (NEDO-20533) was used in the analysis of the recirculation line break. The peak differential pressure increased by 0.2 psi compared to the main steamline break design basis accident; however, the resulting differential pressure (18.8 psid) is still below the design differential pressure of 25 psid.

The pool swell, condensation oscillation, and chugging loads evaluated at the worst power to flow map condition during FWHOS operation vary slightly over the peak values as presented in Section 6 of the USAR. The analysis concluded that this variation is not significant and that adequate design margin exists with regard to these loads.

The mass and energy flow from a recirculation line break into the containment is dependent upon the core inlet enthalpy. A lower core inlet enthalpy (or more subcooled condition) will result in a higher mass/energy flow, and therefore more severe containment LOCA loads. Since the approval of Amendment No. 37, RBS has been authorized for operation in the ICF domain. Heat balance calculations performed by GENE have shown that the core inlet enthalpy for the ICF/FFWTR condition is higher than that for a FFWTR (or FWHOS) condition. Since the FWHOS analysis performed in support of Amendment No. 37 assumed the same 100 °F FW temperature reduction as for the FFWTR analysis, the ICF/FFWTR condition will be applicable and thus bounded by the containment LOCA load results obtained in the FWHOS analysis. The staff finds that the results of the licensee's analysis are acceptable.

3.9 Staff Conclusions

The staff reviewed the licensee's request to amend its FOL and revise RBS License Condition 2.C(13). The licensee analyzed the impact of FFWTR operation on the thermal limits, fuel thermal-mechanical performances, and the applicable Chapter 15 accident analysis (LOCAs) and ATWS. All the Chapter 15 reload analyses are required to comply with the NRC-approved methodologies specified in the latest revision of the topical report, GESTAR II. On the basis of its review, the staff finds the proposed license amendment request to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 27762, dated May 20, 1998). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no

environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Entergy Operations, Inc., "License Amendment Request (LAR) 97-16, Final Feedwater Temperature Reduction Analysis," dated April 9, 1998 (NRC Document Control Accession No. 9804160462)

2. U.S. Nuclear Regulatory Commission, "Correction to Amendment No. 37 to Facility Operating License No. NPF-47 - River Bend Station, Unit 1," dated July 6, 1989

3. E-mail Communication Between Entergy Operations, Inc. and the NRC staff, August 24, 1999, and September 21, 1999.

4. General Electric, "Supplemental Reload Licensing Report for River Bend Station, Reload 6 - Cycle 7," dated July 1997

5. General Electric Nuclear Energy, "General Electric Standard Application for Reactor Fuel," GESTAR II, NEDE-24011-P-A

6. Cleveland Electric Illuminating Co., "PNPP [Perry Nuclear Power Plant] Partial Feedwater Heating Operation Analysis," 1985, Perry Nuclear Power Plant, Units 1 and 2

7. Entergy Operations, Inc., "Amendment Request to the Operating License (PCOL-90/07)," dated June 8, 1990

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