August 26, 1997

Mr. John R. McGaha, Jr.
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
P. O. Box 220
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NO. M97835)

Dear Mr. McGaha:

The Commission has issued the enclosed Amendment No. 97 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 20, 1997, as supplemented by letter dated July 7, 1997.

The amendment revises the TSs to allow the use of flow control spectral shift strategies to increase cycle energy. The request was based on a Maximum Extended Load Line Limit (MELLL) analysis for the River Bend Station.

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,

ORIGINAL SIGNED BY:

David L. Wigginton, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures: 1. Amendment No. 97 to NPF-47
2. Safety Evaluation

cc w/encls: See next page

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

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Entergy Operations, Inc.

River Bend Station

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

ENTERGY GULF STATES, INC. **

CAJUN ELECTRIC POWER COOPERATIVE AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 97
License No. NPF-47

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

   A. The application for amendment by Entergy Gulf States, Inc.* (the licensee) dated January 20, 1997, as supplemented by letter dated July 7, 1997, 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;

---

* EOI is authorized to act as agent for Entergy Gulf States, Inc, which has been authorized to act as agent for Cajun Electric Power Cooperative, and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

** Entergy Gulf States, Inc., which owns a 70 percent undivided interest in River Bend, has merged with a wholly owned subsidiary of Entergy Corporation. Entergy Gulf States, Inc. was the surviving company in the merger.

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

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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment; and Paragraph 2.C.(2) of Facility Operating License No. NPF-47 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 97 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David L. Wigginton, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 26, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 97

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. Revised bases pages and Technical Requirement Manual pages are also included for completeness.

REMOVE	INSERT
3.3-7	3.3-7
3.4-1	3.4-1
3.4-2	3.4-2
B 3.4-3	B 3.4-3
B 3.4-5	B 3.4-5

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux — High	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
	5(a)	3	I	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
b. Inop	2	3	H	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	3	I	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux — High, Setdown	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 20% RTP
b. Flow Biased Simulated Thermal Power — High	1	3	G	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 0.66 W + 67% RTP and ≤ 113% RTP(b)

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Allowable Value is ≤ 0.66 W + 61% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1

A. Two recirculation loops shall be in operation with:

1. Matched flows; and
2. Total core flow and THERMAL POWER within limits.

OR

B. One recirculation loop shall be in operation with:

1. THERMAL POWER ≤ 83% RTP;
2. Total core flow and THERMAL POWER within limits;
3. Required limits modified for single recirculation loop operation as specified in the COLR; and
4. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power—High), Allowable Value of Table 3.3.1.1-1 reset for single loop operation.

---------------------------NOTE---------------------------
Required limit and setpoint modifications for single recirculation loop operation may be delayed for up to 12 hours after transition from two recirculation loop operation to single recirculation loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Recirculation loop jet pump flow mismatch not within limits.	A.1 Shutdown one recirculation loop.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. THERMAL POWER > 83% RTP during single loop operation.	B.1 Reduce THERMAL POWER to ≤ 83% RTP.	1 hour
C. Total core flow as a function of THERMAL POWER within Region II of Figure 3.4.1-1.	C.1 Determine APRM and LPRM neutron flux noise levels.	Once per 8 hours <u>AND</u> 30 minutes after an increase of ≥ 5% RTP
D. Total core flow as a function of THERMAL POWER within Region II of Figure 3.4.1-1. <u>AND</u> APRM or LPRM neutron flux noise level > 3 times established baseline noise level.	D.1 Restore APRM and LPRM neutron flux noise level to ≤ 3 times established baseline levels.	2 hours
E. Total core flow as a function of THERMAL POWER within Region III of Figure 3.4.1-1.	E.1 Restore total core flow as a function of THERMAL POWER to within Region I or II of Figure 3.4.1-1.	4 hours

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the USAR.

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3).

The transient analyses of Chapter 15 of the USAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR limits for single loop operation are specified in the COLR. The APRM flow biased simulated thermal power setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement.

LCO

Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. In addition, the total core flow must be ≥ 45% of rated core flow or total core flow expressed as a function of THERMAL POWER must be in Region I as identified in Figure 3.4.1-1, "THERMAL POWER/Core Flow Stability Regions." Alternatively, with only one recirculation loop in operation, THERMAL POWER must be ≤ 83% RTP, the total core flow limitations identified above must be met, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM

(continued)

BASES

ACTIONS

A.1 (continued)

Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The 2 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

B.1

Should a LOCA or tranisent occur with THERMAL POWER > 83% RTP, during single loop operation the core response may not be bounded by the safety analyses. Therefore, only a limited time is allowed to reduce THERMAL POWER to ≤ 83% RTP.

The 1 hour Completion Time is based on the low probability of an accident occuring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing changes in THERMAL POWER to be quickly detected.

C.1, D.1, and E.1

Due to thermal hydraulic stability concerns, operation of the plant is divided into three regions based on THERMAL POWER and core flows. Region III is a power/flow ratio with core flow < 39% of the rated core flow. Region II is a power/flow ratio with core flow ≥ 39% and < 45% of the rated core flow. Deliberate entry into Region III is not permitted, and if it occurs, immediate action is required to exit the region within 4 hours by reducing THERMAL POWER through control rod insertion or by increasing recirculation loop flow by opening the flow control valve. Operation in Region II is also more susceptible to instability than normal operating parameters. However, operation in this region is allowed with the exception that if evidence of

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 97 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 January 20, 1997, Entergy Operations, Inc. (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-47) for the River Bend Station (RBS), Unit 1. The proposed changes would revise the Technical Specifications (TS) to allow the use of flow control spectral shift strategies to increase cycle energy. The request would allow incorporation of the Maximum Extended Load Line Limit (MELLL) Analysis for RBS. The licensee also provided additional information for the staff's review. The licensee's letter dated July 7, 1997, provided clarifying information and did not change the staff's initial no significant hazards consideration determination.

2.0 BACKGROUND

In their request dated January 20, 1997, the licensee also submitted a safety evaluation for the request, including NEDC-32611P, "Maximum Extended Load Line Limit Analyses for River Bend Station Reload 6 Cycle 7," dated November 1996.

The Maximum Extended Load Line Limit (MELLL) operation mode and the associated TS changes expand the operating domain along the 121% rod line to the power/flow point of 100% power and 75% core flow. An enlarged power/flow map for RBS would permit improved power ascension capability by extending plant operation at rated power with less than rated core flow during the fuel cycle. The average power range monitoring (APRM) flow-biased scram and rod-block setpoints would also be increased to accommodate the MELLL region on power/flow map. Additionally, TS would be revised to accommodate single loop operation (SLO) in the MELLL region. Use of MELLL has been previously approved for other BWR/6 designs and is acceptable for use at RBS.

By letter dated June 26, 1997, the staff requested additional information on MELLL transient analysis including the Rod Withdrawal Error (RWE) analysis and the revised Average Power Range Monitor (APRM) flow-biased scram and rod block setpoints. The licensee provided a response by letter dated July 7, 1997.

3.0 EVALUATION

Anticipated Operational Occurrences for MELLL Operation

The limiting anticipated operational occurrences (AOOs) (pressurization and non-pressurization events) for the MELLL region were examined to ensure that operating limit minimum critical power ratio (OLMCPR) requirements are satisfied. Maintaining the OLMCPR provides sufficient margin so that the safety limit MCPR (SLMCPR) will not be exceeded. Maintaining this limit assures that 99.9 percent of the fuel rods are expected to avoid boiling transition. The analysis show that the OLMCPR for operation in the MELLL region remains bounded by the OLMCPR established for current rated conditions of 100% power and 107% flow. The following core-wide events were considered to bound OLMCPR requirements, and were evaluated for MELLL operation:

Generator Load Rejection with No Bypass (LRNBP)
Feedwater Controller Failure (FWCF) to Maximum Demand
Fuel Loading Error (FLE)
Pressure Regulator Failure (PRFDS) Downscale

The transient and accident analysis methodologies used for RBS cycle 7 are described in GESTAR-II[1]. These events were examined for OLMCPR impact when operating in the MELLL region. The analysis results for RBS showed that for the pressurization transients at the MELLL condition of 100% power and 75% flow, the LRNBP event yields the most limiting fuel thermal responses for both GE11 and GE8x8EB fuel, and are bounded by the previously analyzed 100% power, 107% flow and 100% power, 100% flow conditions. The power and flow dependent thermal limits (MCPR-p, MCPR-f) for the current operating cycle (Cycle 7) were used in the analysis. These limits will be verified and adjusted as necessary for future cycles and will be reported as required in the Core Operating Limits Report (COLR).

Loss of Coolant Accident (LOCA)

An ECCS/LOCA analysis was conducted to determine the impact of MELLL operation on the RBS LOCA peak cladding temperature (PCT). The result was an estimated increase of less than 5°F from the value for operation in the currently approved power/flow regime. The evaluation was conducted using the current RBS licensing basis methodology SAFE/REFLOOD. The licensee has stated that prior to implementation of MELLL in Cycle 8, adequate PCT margin will exist to absorb the additional PCT impact. The licensee's preliminary SAFER/GESTR analysis was conducted with inputs which bound both Cycle 7 and 8, and have resulted in a PCT of about 1300°F. This is acceptable to the staff.

---

[1] NEDE-24011-P-A-11, "General Electric Standard Application for Reactor Fuel, GESTAR II," and NEDE-24011-P-A-11-US, "GESTAR II U.S. Supplement," November 1995.

Rod Withdrawal Error

The rod withdrawal error (RWE) event is a localized transient event not significantly affected by MELLL operation. The characteristics of the RWE event are such that the most important parameters affecting the transient response are the initial control rod pattern and the error rod position. These parameters are not affected by the low flow operation such as in the MELLL domain. The severity of the RWE event for a BWR/6 is mitigated by the Rod Withdrawal Limiter (RWL) function which is not affected by the MELLL operation. In addition, the APRM flow-biased rod block setpoint is revised to initiate the clamp at 75% core flow condition instead of at 100% core flow to bound the MELLL domain. Therefore, the previously-analyzed RWE remains valid for MELLL operation.

Thermal - Hydraulic Stability

The licensee has stated that RBS has implemented the GE Service Information Letter (SIL) 380, Revision 1, "BWR Core Thermal Hydraulic Stability," recommendations, and that RBS is also in compliance with the interim measures of Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)." In addition, core/channel decay ratio and the stability exclusion region are reanalyzed for new fuel designs to provide assurance of stability performance. RBS is scheduled to implement option I-A included in the BWROG topical report NEDO-32339-A, "Reactor Stability Long-Term Solutions: Enhanced Option I-A." This is acceptable to the staff.

Vessel Overpressure Protection

The Main Steam Isolation Valve (MSIV) closure event was re-analyzed and demonstrated conformance to the ASME code in the MELLL region. The results of this analysis confirmed that the peak pressure of 1295 psig for MELLL operation is bounded by the current cycle 7 result of 1311 psig and is within ASME Section III limit of 1375 psig.

Containment Pressurization Response/Dynamic Loading

Bounding short-term containment response analyses of the design basis LOCA event were performed to demonstrate that operation in the MELLL domain will not result in exceeding containment design limit. The RBS final safety analysis report (FSAR) steamline break analysis is applicable to the MELLL conditions. A main steam line break, although more limiting for peak temperature response, was not re-analyzed for MELLL operation because the vessel dome pressure is unchanged. Therefore, the analysis results in SAR section 6.2 for steamline break remain valid in the MELLL region.

The recirculation line break in the MELLL region was analyzed to show that the drywell-wetwell pressure differential response remains below the 25 psid design value. The drywell airspace temperature is within the current recirculation line break analysis results in the FSAR. The licensee submittal also stated that MELLL operation will not significantly affect containment dynamic loads. The LOCA containment dynamic loads analysis is based upon the short-term LOCA analysis. The LOCA dynamic loads considered for MELLL operation include pool swell, condensation oscillation and chugging loads. These loads are bounded by previously-specified design values.

Control Rod Drop Accident

Banked position withdrawal sequence (BPWS) and rod patterns are used for RBS. For plants using BPWS the control rod drop accident (CRDA) has been statistically analyzed generically and it was found that with a high degree of confidence the peak fuel enthalpy would not approach the 280 cal/gram acceptance criteria for this event. In addition, the CRDA is a startup event which would not be affected by MELLL operation.

Anticipated Transient Without Scram (ATWS)

The basis for ATWS requirements is 10 CFR 50.62. For BWRs, the rule includes requirements for an ATWS Recirculation Pump Trip (RPT), Alternate Rod Insertion (ARI) system and an equivalent Standby Liquid Control System (SLCS) injection rate. A plant-specific ATWS analysis was performed to support RBS operation in the MELLL domain. Results of the re-evaluation of a number of ATWS limiting transients demonstrated continued conformance to FSAR ATWS acceptance criteria. This is acceptable to the staff.

Single Loop Operation

Single Loop Operation (SLO) analyses were reviewed to ensure their applicability in the MELLL region. The maximum power/flow state point achievable with one recirculation pump operation is 83%power/54%flow. The proposed changes revise TS 3.4.1 to reflect the new power limit of 83% rated thermal power and SLO limits will be reviewed for adequacy for each future operating cycle.

Reactor Vessel Internals Integrity

An evaluation was performed to identify potential increases in reactor internal pressure differences following a recirculation pump runout along the MELLL rod line. This evaluation shows that the pressure differences for MELLL operation resulting from recirculation pump runout are bounded by the current design basis results for the upset condition. The licensee has also stated that reactor internal vibrations during MELLL operation will be bounded by previously established acceptance criteria. Therefore, it has been concluded that RBS can operate in the MELLL region without any detrimental effects on the reactor internals due to flow induced vibration or reactor internal pressure differences.

APRM Setpoint Changes

The flow-biased APRM thermal power scram line and the APRM flow-biased rod block line are not credited in any RBS safety licensing analysis. For the current licensed power/flow map, the flow-biased APRM thermal power scram line was defined as 0.66W+ 0.51, and the flow biased APRM rod block line was set at 0.66W+0.457 where W is the recirculation drive flow in percent of rated. These values are reflected in TS Table 3.3.1.1-1 which is being revised. With the current power/flow map expansion to include the MELLL domain, the upper boundary of the licensed operating domain is now extended to approximately the 121% rod line. To accommodate this expanded operating domain above the rated rod line, the setpoints for the flow-biased scram line and the flow-biased rod block line are now redefined as follows: nominal 0.66W+0.67 and 0.66W+0.61 (in note (b) of Table 3.3.1-1). The licensee has stated that the establishment of the revised setpoints is consistent with the previously approved GE setpoint methodology. These proposed TS changes are acceptable.

4.0 SUMMARY

EOI has performed the analyses for implementation of MELLL using current cycle limits to determine the impact of operation in the MELLL region. The analyses have shown that the SLMCPR will not be exceeded for limiting AOOs, and have also show that the impact on LOCA PCT is minimal for Cycle 7 and 8. The licensee has stated that the impact on PCT will be addressed prior to implementation to ensure that the PCT limits in 10 CFR 50.46 are addressed. The report shows that RBS can be operated within the MELLL region while continuing to support required safety margins.

The licensee also provided changes to the technical specifications to reflect the changes for MELLL implementation (see EVALUATION section "Single Loop Operation" and "APRM Setpoint Changes" above). These changes are approved as they are consistent with the safety analysis. Page changes from the Bases were also provided consistent with these changes and are acceptable.

5.0 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.

6.0 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 (62 FR 8799). 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.

7.0 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.

Principal Contributor: G. Golub

Date: August 26, 1997