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

September 29, 2009

Vice President, Operations
Entergy Operations, Inc.
River Bend Station
5485 U.S. Highway 61N
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: CHANGES TO TECHNICAL SPECIFICATION 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)" (TAC NO. ME0157)

Dear Sir or Madam:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 166 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 November 20, 2008, as supplemented by letter dated August 12, 2009.

The amendment revises TS 5.6.5, "Core Operating Limits Report (COLR)," to add a reference to an analytical method that will be used to determine the core operating limits. The change is needed to support the use of GE14 fuel during refueling outage 15 scheduled for the fall of 2009.

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,

Alan Wang

Alan B. Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 166 to NPF-47
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

ENTERGY GULF STATES LOUISIANA, LLC

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 166
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 November 20, 2008, as supplemented by letter dated August 12, 2009, 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

   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. 166 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 and shall be implemented before Cycle 16 operation.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License No. NPF-47 and
Technical Specifications

Date of Issuance: September 29, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 166

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following pages of the Facility Operating License No. NPF-47 and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change.

Facility Operating License

Remove	Insert
-3-	-3-

Technical Specifications

Remove	Insert
5.0-19	5.0-19

(3) EOI, pursuant to the Act and 10 CFR Part 70, to receive, possess and to use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

EOI is authorized to operate the facility at reactor core power levels not in excess of 3091 megawatts thermal (100% rated power) in accordance with the conditions specified herein. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 166 and the Environmental Protection Plan contained in Appendix 8, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

24) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR-II)".

25) NEDC-33383P, "GEXL97 Correlation to ATRIUM-10 Fuel," Global Nuclear Fuel.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 166 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 November 20, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML083570192), as supplemented by letter dated August 12, 2009 (ADAMS Accession No. ML092300229), Entergy Operations, Inc. (Entergy, the licensee), requested changes to the Technical Specifications (TSs) for the River Bend Station, Unit 1 (RBS). The supplement dated August 12, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 23, 2009 (74 FR 4249).

The proposed changes would revise TS 5.6.5, "Core Operating Limits Report (COLR)." Specifically, the change would add a new reference (NEDC-33383P, "GEXL97 Correlation Applicable to ATRIUM-10 Fuel," Global Nuclear Fuel) as an analytical method that will be used to determine the core operating limits. The change will allow the licensee to use a Global Nuclear Fuel (GNF) method to determine the critical power of AREVA (formerly Framatome ANP) ATRIUM-10 fuel. RBS currently operates with a full core of ATRIUM-10 fuel. The licensee intends to load GE14 fuel in place of a portion of the ATRIUM-10 fuel during its upcoming fall 2009 refueling outage. The proposed change supports the licensee's transition from ATRIUM-10 to GE14 fuel and associated analytical methods, and is needed to support the use of GE14 fuel during refueling outage 15 scheduled for the fall of 2009. Entergy plans to use the GEXL97 correlation to determine the core operating limits for the RBS operating Cycle 16, which begins in fall 2009.

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The TSs ensure the operational capability of structures, systems, and components that are required to protect the health and safety of the public. The NRC's regulatory requirements related to the content of the TSs are contained in Title 10 of the *Code of Federal Regulations*, Section 50.36 (10 CFR 50.36), which requires that the TSs include items in the following specific categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) limiting conditions for operation (LCO); (3) surveillance requirements; (4) design features; and (5) administrative controls. In accordance with 10 CFR 50.36(c)(5), administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

The NRC staff reviewed the proposed changes for compliance with 10 CFR 50.36 and agreement with the precedent as established in NUREG-1434 "Standard Technical Specifications –General Electric Plants, [Boiling-Water Reactor] BWR/6." In general, licensees cannot justify TS changes solely on the basis of adopting the Standard Technical Specification (STS) model. Licensees may revise the TSs to adopt the improved STS format and content, provided that a plant-specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative, or provides clarification (i.e., no requirements are materially altered); (2) the change is more restrictive than the licensee's current requirement; or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards.

RBS TS 5.6.5.b. requires, in part, that "[t]he analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC ... ."

Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Section 50.34(b)(4), "Final safety analysis report," requires, in part, that "[a] final analysis and evaluation of the design and performance of structures, systems, and components with the objective [of assessing ... the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.]"

As part of the reload design process, the licensee (or its vendor) performs reload safety analyses with approved methodologies to ensure that the design cycle will continue to meet the applicable regulatory criteria. To confirm that the analyses remain acceptable, the licensee confirms that key results of the safety analyses, such as the critical power ratio (CPR), are conservative with respect to the current design cycle. If key safety analysis results are not acceptable, a reanalysis or reevaluation of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

10 CFR Part 50, Appendix A, General Design Criterion (GDC)-10, "Reactor design," requires that:

> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC-12, "Suppression of reactor power oscillations," requires that:

> The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

GDC-15, "Reactor coolant system design," requires that:

> The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

GDC-29, "Protection against anticipated operational occurrences," requires that:

> The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

The NRC staff also used the regulations in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR 50, Appendix K, in conjunction with GDC-35 of Appendix A. The regulations in 10 CFR 50.46 incorporate GDC-35 by reference, which makes GDC-35 directly applicable for this review.

GDC-35, "Emergency core cooling," requires that:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

3.0 TECHNICAL EVALUATION

Core operating limits and parameters are established prior to each reload cycle, or prior to any remaining portion of a reload, and are documented in the core operating limits report (COLR). To establish these limits and parameters, the methodologies referenced in TS 5.6.5.b are used. The references to the methodologies being added to TS 5.6.5.b under this amendment are in addition to those currently included in the TSs and have been previously approved for use by the NRC for RBS.

The additional references to the approved methodologies (Topical Reports) in TS 5.6.5.b provide only the Topical Report (TR) number and title and do not include further reference to revision numbers, date of issuance, and supplements. This will allow the licensee to use the current TRs (referenced in TS 5.6.5.b) to determine a core operating limit in the COLR without having to obtain a license amendment to use a revised NRC-approved TR, provided that the revised approved methodology is applicable to RBS. Because TS 5.6.5.b requires that the methodology must be previously reviewed and approved by the NRC staff, the licensee would not be able to use a revised TR unless the NRC has approved its use.

To support the license amendment request (LAR), the licensee referenced a licensing topical report (TR), NEDC-33383P, Revision 1, "GEXL97 Correlation Applicable to ATRIUM-10 Fuel." This report contains the methodology, correlation, and associated uncertainties developed for modeling the AREVA ATRIUM-10 fuel design. The licensee will apply the correlation to the ATRIUM-10 fuel that will remain in the RBS BWR core.

The licensee submitted information to demonstrate that GE14 fuel and the legacy fuel are thermal-hydraulically compatible and that the GNF GESTAR II analytical methods are applicable to RBS for use in licensing calculations. The licensee analyzed the affected licensing basis events based on the GNF analytical methods and showed compliance with the applicable regulations.

3.1 GEXL97

The data for the GEXL97 development was generated using the NRC-approved SPCB correlation developed by AREVA (Reference 3). The database consisted of ATRIUM-10 sub-bundle and full-bundle critical power data generated by the sub-channel code XCOBRA, incorporating the NRC-approved SPCB correlation (SPCB is the AREVA critical power correlation for ATRIUM-10 fuel). The objective of this data collection was to obtain ATRIUM-10 quality data appropriate for GEXL analysis.

The span of the data collection encompasses cosine, top-peaked, bottom-peaked, and double-humped axial power shapes. This data was generated to cover the complete range of expected operation of the ATRIUM-10 fuel in the RBS BWR core. The data was used to develop a new General Electric (GE) critical quality-boiling length correlation (GEXL) for the ATRIUM-10 design, designated as GEXL97. The GEXL97 correlation uses the same functional form as previous GEXL correlations with different constants for the GEXL correlation coefficient parameters.

The GEXL correlation was developed to accurately predict the onset of boiling transition in BWR fuel assemblies during both steady-state and reactor transient conditions. The GEXL correlation is necessary for determining the minimum critical power ratio (MCPR) operating limits resulting from transient analysis, the MCPR safety limit analysis, and the core operating performance and design. The GEXL correlation is an integral part of the transient analysis methodology. It is used to confirm the adequacy of the operating limit MCPR (OLMCPR), and it can be used to determine the time of onset of boiling transition in the analysis of other events.

The NRC staff's review considered the following: 1) adequacy of the database generated with the sub-channel code XCOBRA, 2) proper determination of the uncertainty in GEXL97 correlation predictions for the ATRIUM-10 fuel design, and 3) applicability of the proposed operating range of the GEXL97 correlation to the ATRIUM-10 fuel.

3.1.1 Validity of the Database and Associated Uncertainties

ATRIUM-10 fuel is a 10x10 fuel bundle with a water channel design that displaces nine fuel rods. It contains a total of 83 full-length fuel rods and 8 part-length rods. It has 27 unique fuel rod locations within the 10x10 lattice for which dryout data was collected.

The SPCB correlation for the ATRIUM-10 fuel, as encoded in the sub-channel computer code XCOBRA, is used to generate a database of predicted critical power values for a range of operating conditions corresponding to the range of the ATRIUM-10 correlation. This database was then treated in the same way as an experimental database, using the approved methodology for GEXL correlation development. Utilizing this approach, GNF produced a new form of the GEXL correlation, namely GEXL97, applicable only to the ATRIUM-10 fuel design.

The data for the GEXL97 development specific to ATRIUM-10 fuel was generated using NRC-approved AREVA SPCB correlation encoded in the above stated sub-channel code. Specified rod-to-rod peaking factors, axial power shapes, pressure, mass flux, and sub-cooling were used with the AREVA SPCB correlation to determine critical power at dryout.

Because the experimental database and critical power correlation for the previous vendor's fuel is not available to the new vendor for generating the hypothetical databases, the licensee used the AREVA SPCB correlation encoded in the subchannel code XCOBRA to evaluate the mixed core fuel. The AREVA SPCB correlation has been previously approved by the NRC staff and therefore, the NRC staff concludes that this is a reasonable engineering approach for evaluating mixed core fuel.

3.1.2 Determination of Uncertainties

The database used in the development of the GEXL97 correlation for ATRIUM-10 fuel was provided in Table 2-1 of NEDC-33383P, Revision 1. This table shows the number of calculated critical power data points obtained using the AREVA critical power correlation for cosine, inlet, outlet, and double-humped axial power distributions. It also shows the fuel pin dryout location that formed the basis of the 28 different sets of AREVA calculated critical power data. Table 2-2 of the same document provides additional information by further dividing the data collected into subgroups of pressure, mass flux, and inlet sub-cooling.

The GEXL97 database generated in this manner is an artificial construct, created with a computer code that has implemented the SPCB correlation. It is only an approximation of the actual critical power raw data behavior of the ATRIUM-10 fuel. However, with reasonable engineering practices, and proper statistical accountability, the database can reliably predict the critical power behavior with acceptable uncertainties. Testing the hypothetical databases as if it were real data in the regression analysis, therefore, introduces unavoidable error into the correlation being derived from it.

As noted earlier, the database for the GEXL97 development specific to ATRIUM-10 fuel was generated using the NRC-approved SPCB correlation. The database consisted of ATRIUM-10 sub-bundle and full-bundle critical power data generated by the sub-channel code XCOBRA, incorporating the NRC-approved SPCB correlation.

The local critical power values predicted with the approved SPCB correlation can be expected to vary over the range of the database. Since the GEXL97 correlation is fitted to this "hypothetical" database, the error in the critical power prediction of the GEXL97 correlation for a given set of conditions will have some additional error relative to the real critical power value for those conditions, over and above the uncertainty of the correlation's fit to the hypothetical database. Therefore, the approach of the correlation procedure can be valid only if overall uncertainty in the new GEXL97 correlation is appropriately characterized in terms of the uncertainty in its fit to the hypothetical database and the uncertainty of the critical power values in the hypothetical database itself.

The treatment of the overall uncertainty of the GEXL97 correlation for ATRIUM-10 fuel, as originally presented in the submittal, is complete. GNF appropriately combined the uncertainty of the fit of the GEXL97 correlation to the hypothetical database and the uncertainty of the database itself, which is a function of the uncertainty of SPCB correlation.

Based on above, the NRC staff concludes that the total uncertainty in the correlation's critical power predictions appropriately accounts for the uncertainty in the new correlation's fit to the hypothetical database. In addition, the uncertainty in the hypothetical database with respect to the underlying experimental data are also appropriately treated. Therefore, the NRC staff concludes that the treatment of uncertainties is acceptable.

3.1.3 Generation of the GEXL97 Correlation and the Range of Applicability

In developing the GEXL97, GNF took steps to optimize GEXL97 critical power predictions for the ATRIUM-10 fuel design, and to minimize the prediction uncertainty. This process is identical to that used by GNF when developing GEXL correlation coefficients for GNF fuel designs using raw test data, and has been used in past development of GEXL correlations applicable to other legacy fuel.

The procedure used for development of the GEXL97 correlation is summarized below:

a) First, a range of generated data covering all parameter(s) variations is selected to form a correlation development database. This database consists of the majority of the generated data. A separate dataset is set aside to form a correlation verification database.

b) The GEXL97 correlation coefficients are then chosen to minimize the bias and standard deviation in correlating the development database, and to minimize any trend errors in reference to flow, pressure, sub-cooling, and R-factor (the R-factor is an input to the correlation that accounts for the effects of the fuel rod distributions and the fuel assembly and channel geometry on the fuel assembly critical power).

c) Once the optimum coefficients are determined, the apparent R-factors are calculated for each assembly. The apparent R-factor is defined as that R-factor which yields an overall ECPR of 1.0 for a given assembly. ECPR is defined as the ratio of the GEXL97 calculated critical power to the SPCB calculated critical power.

d) A final set of additive constants (Table 4-2 of NEDC-33383P, Revision 1) are determined by adjusting the preliminary additive constants subject to minimizing the difference between the R-Factors.

The range of application for the GEXL97 correlation, as stated in Section 4.2 of NEDC-33383P, Revision 1, is the same as the range of the hypothetical database over which the correlation is derived, and within the AREVA SPCB development database. The application range covers the complete range of expected operation of the ATRIUM-10 fuel during normal steady state and transient conditions in the RBS BWR core. Therefore, the licensee's use of the new GEXL97 correlation within the limits of the hypothetical database, bounded by the experimental limits of the ATRIUM-10 database, is acceptable.

3.2 Thermal-Hydraulic Compatibility of the GE14 fuel with the ATRIUM-10 fuel

The licensee's submittal included an analysis for the mixed core (Reference 1, Attachment 4) confirming that the GE14 and ATRIUM-10 fuel designs are thermal-hydraulically compatible.

The next three cycles at RBS will be designated as mixed cores with the core comprised of ATRIUM-10 fuel and GE14 fuel. Cycle 16 will approximately consist of 2/3 core of ATRIUM-10 fuel and 1/3 core of GE14 fuel; Cycle 17 will approximately consist of 1/2 core of ATRIUM-10 fuel and 1/2 core of GE14 fuel; and Cycle 18 will approximately consist of 1/3 ATRIUM-10 fuel and 2/3 GE14 fuel. Therefore, GNF performed calculations to verify the mixed core calculations results regarding the similarity in thermal-hydraulic performance of the GE14 and ATRIUM-10 fuel designs. Data provided by AREVA was used by GNF to develop computer code models to perform the evaluations.

GNF investigated the thermal-hydraulic compatibility between GE14 and ATRIUM-10 through a series of mixed cores, progressing from the full core of ATRIUM-10 fuel to a full core of GE14 fuel. The mixed core analyses projected the performance of both fuel types during transition cores, from a full core of ATRIUM-10 fuel to a full core of GE14 fuel.

GNF also performed evaluations to demonstrate compliance with safety and performance criteria, including core nuclear design and the thermal hydraulic critical power correlations for the ATRIUM-10 fuel. The GNF calculations provided confirmation that the thermal-hydraulic performance characteristics applied in the calculations met specific acceptance criterion associated with the thermal-hydraulic compatibility of GE14 fuel and the legacy fuel.

3.3 Use of Approved Analytical Methods

Analytical methods (e.g., computer codes, correlations, etc.) used to support licensing calculations are generally documented in TRs, which may be reviewed by the NRC staff on a generic basis. In the NRC staff safety evaluation (SE) approving the TR, the NRC staff defines the basis for acceptance in conjunction with any limitations and conditions on use of the TR, as appropriate. The NRC staff requested the licensee to document that the use of proposed analytical methods for RBS, as summarized below in Table 1, is consistent with the limitations and conditions on use of the TR in the NRC staff approval. In Reference 2, the licensee listed the TRs and analytical methods used for each affected analysis and stated that such use was consistent with the corresponding NRC staff approval. The NRC staff finds that the topical reports and analytical methods used by the licensee are acceptable.

3.4 Licensing Basis Analyses

In Reference 2, the licensee provided plant-specific information to support the methodology change in conjunction with the introduction of GE14 fuel. Specifically, the licensee performed analyses of the limiting updated final safety analysis report events with the GNF methods to demonstrate that the results of the analysis meet the applicable acceptance criteria. The information provided by the licensee is summarized in Section 3.4.4, Table 1 of this safety evaluation.

The events analyzed include:

- Limiting Anticipated Operational Occurrences (AOOs; Turbine Trip with no Bypass, Load Rejection with no Bypass, Feedwater Controller Failure)
- American Society of Mechanical Engineers (ASME) Overpressure (Main Steam Isolation Valve Closure with Flux Scram)
- Stability
- Emergency Core Cooling System (ECCS) – Loss-of-Coolant Accident (LOCA)
- Anticipated Transient without Scram (ATWS; Main Steam Isolation Valve Closure, Pressure Regulator Failure Open)

3.4.1 AOO and ASME Overpressure

The plant responses to the limiting AOOs are analyzed for each reload cycle to establish the OLMCPR. The ASME overpressure analysis is also performed every cycle to ensure that vessel pressurization following a limiting transient is within the acceptable limit. The licensee performed the reload transient analysis to cover the projected operating conditions within the licensed power-to-flow map, the expected core exposures, and equipment availability conditions. In Reference 2, the licensee provided the OLMCPR limits for Cycle 16 core containing GE14 and co-resident ATRIUM-10 fuel. In addition, the licensee showed that ASME overpressure results are acceptable. The results of the analysis satisfy the requirements of GDC-10 and GDC-15 regarding the fuel integrity and RCS pressure boundary integrity. Therefore, the NRC staff concludes that these analyses are acceptable.

3.4.2 Stability

RBS is currently operating under the requirements of the reactor stability Long-Term Solution Enhanced Option I-A (E1A) approved by the NRC staff in GE Licensing Topical Report NEDO 32339-A and its supplements (References 10 through 14). Option E1A is a solution based exclusively on prevention of instabilities. A conservative exclusion region where instabilities are unlikely is defined or validated by cycle-specific calculations based on an approved procedure. This exclusion region is enforced automatically by the reactor protection system. The licensee validated the exclusion region by showing that core and channel decay ratios are within acceptable levels based on the approved ODYSY methodology (Reference 9).

The results of the analysis satisfy the requirements of GDC-10 and GDC-12 regarding the fuel integrity and stability protection. Therefore, the NRC staff concludes that these analyses are acceptable.

3.4.3 ECCS-LOCA

The ECCS is designed to mitigate postulated LOCAs due to ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The analysis methodology used for RBS LOCA analysis is the SAFER/GESTR-LOCA evaluation model (References 12 through 16), which has been approved by the NRC.

The licensee performed the LOCA analysis to support the transition. The licensee examined break sizes ranging from double-ended guillotine break of recirculation suction line to small breaks for which no core heatup is predicted. For the transition cores, the limiting large break was identified as the double-ended guillotine break of the recirculation suction line and the limiting small break was identified as a 0.05 ft^2 break in the recirculation suction line. In addition, the 0.05 ft^2 break was identified as the overall most limiting break assuming the worst single failure of high pressure core spray diesel generator. For ATRIUM-10, the large break was identified as the overall most limiting break.

The licensee evaluated both mid-peaked and top-peaked axial power shapes. The limiting axial power shape was identified as mid-peaked for large breaks and top-peaked for small breaks.

The licensee evaluated potentially limiting power and flow conditions including the Maximum Extended Loadline Limit Analysis (MELLLA) and rated power/flow conditions. For this analysis, the licensee identified the rated power/flow point as the limiting power and flow condition for GE14 and the MELLLA power/flow point as the limiting power and flow condition for ATRIUM-10. The licensee also concluded that the single-loop operation is bounded by two-loop operation, and increased core flow is bounded by rated core flow.

Based on above, the NRC staff concludes that the licensee provided sufficient information to demonstrate compliance with the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

3.4.4 ATWS

ATWS is defined as an AOO followed by the failure of the reactor protection system as required by GDC-20. To demonstrate acceptability, the ATWS analysis must show that 1) the peak vessel bottom pressure is less than the ASME service level C limit of 1500 pounds per square inch gauge (psig); 2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200 degrees Fahrenheit (°F); 3) the peak suppression pool temperature is less than the design limit (185 °F for RBS); and 4) the peak containment pressure is less than the containment design pressure (15 psig for RBS). The licensee performed the ATWS analysis and the results meet the acceptance criteria, as shown in Table 1.

Table 1 - Limiting Analysis Results and Computer Codes and Methodology Used

Analysis	Code(s) Used	Staff Approval	Key Parameter(s)	Result vs. Acceptance Criteria
Anticipated Operational Occurrence (AOO)	ISCOR09 PANAC11 ODYN09 TASC03	References 4, 5, and 6	MCPR	All equipment in service condition OLMCPR range of 1.27 to 1.31 (See Note 1)
ASME Overpressure	PANAC11 ODYN09	References 4 and 5	Peak Dome Pressure (psig) Peak Vessel Pressure (psig)	1313 (≤1325) 1348 (≤1375)
Stability	ISCOR09 PANAC11 ODYSY05	References 4, 7, 8, 9, 10, 11, and 12	Max Channel Decay Ratio Max Core Decay Ratio	0.319 (<0.8 ODYSY Criteria) 0.789 (<0.8 ODYSY Criteria)
ECCS-LOCA	ISCOR09 LAMB08 SAFER04/GESTR08 TASC03	References 13, 14, 15, 16, and 17	ATRIUM-10 PCT (°F) ATRIUM-10 Max local oxidation (%) ATRIUM-10 Core wide Metal-water reaction (%) GE14 PCT (°F) GE14 Max local oxidation (%) GE14 Core wide Metal-water reaction (%)	1810 (≤2200) 3 (≤17) 0.1 (≤1.0) 1660 (≤2200) 1 (≤17) 0.1 (≤1.0)
ATWS	ISCOR09 PANAC11 ODYN09V STEMP04 TASC03	References 4, 5, and 6	Peak Vessel Pressure (psig) Peak Suppression Pool Temperature (°F) Peak Containment Pressure (psig) Peak Cladding Temperature (°F) Peak Local Cladding Oxidation (%)	1497 (≤1500) 182.4 (≤185) 9.2 (≤15) 1405 (≤2200) Insignificant (≤17)

Note 1: The AOO analysis determines the OLMCPR such that the limiting transient would not violate the safety limit MCPR. In Reference 2, the licensee provided a base case with all equipment in service and an OLMCPR of 1.28 for beginning to middle of cycle and 1.31 for middle to end of cycle.

3.5 <u>Revision to TS 5.6.5, "Core Operating Limits Report (COLR)"</u>

TS 5.6.5.b provides a list of TRs documenting the NRC-approved methodologies used to determine the values of cycle-specific parameters included in the COLR. The licensee proposes to add the following TR to the reference list:

25. NEDC-33383P, "GEXL97 Correlation Applicable to ATRIUM-10 Fuel," Global Nuclear Fuel

The NRC staff concludes that the GEXL97 method documented in the referenced TR is acceptable for use in support of RBS licensing applications. Therefore, the proposed change to TS 5.6.5 is acceptable.

3.6 NRC Staff Technical Conclusions

In consideration of information discussed above, the NRC staff concludes that the proposed fuel and methodology transition is acceptable. The NRC staff concludes that the use of GEXL97 is acceptable for the following reasons:

- The total uncertainty in the correlation's critical power predictions appropriately accounts for the uncertainty in the new correlation's fit to the hypothetical database and the uncertainty in the hypothetical database with respect to the underlying experimental data are appropriately treated;

- Generating the hypothetical databases using the SPCB correlation encoded in the sub-channel code XCOBRA is a reasonable engineering approach to dealing with mixed core fuel, where the experimental database and critical power correlation for the previous vendor's fuel is not available to the new vendor;

- The licensee intends to utilize the new GEXL97 correlation within the limits of the hypothetical database, bounded by the experimental limits of the ATRIUM-10 database; and

- The licensee confirmed that the CPR analyses remain bounding, and that key inputs to the safety analyses (such as the CPR) are conservative with respect to the current design cycle.

In addition, the NRC staff concludes that the introduction of the GE14 fuel will not adversely affect the performance of the ATRIUM-10 fuel, and that the two distinct fuel designs are thermal-hydraulically compatible. The NRC staff concludes that the use of the proposed analytical methods for RBS is consistent with the corresponding NRC staff approval and that the results of the analyses meet the applicable regulatory requirements. Based on the above, the NRC staff concludes that the proposed license amendment is acceptable.

4.0 REGULATORY COMMITMENT

In support of the proposed application, the licensee provided in its November 20, 2008, submittal the following regulatory commitment regarding 10 CFR Part 21 Report No. SC05-03 from GE:

> Entergy is following the industry effort to resolve the TS issues associated with the GE Part 21 and will request appropriate TS changes after a generic resolution is reached by the industry and the NRC staff.

The licensee committed to complete the above commitment after the industry resolution of GE Part 21 Report SC05-03. The NRC concludes this commitment is 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 published in the *Federal Register* on January 23, 2009 (74 FR 4249). Accordingly, the amendments meet 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.

8.0 REFERENCES

1. J. C. Roberts, Entergy Operations, Inc., Letter to U.S. Nuclear Regulatory Commission, "License Amendment Request (LAR) Changes to Technical Specifications 5.6.5, "Core Operating Limits Report (COLR)," dated November 20, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No ML083570192).

2. J. C. Roberts, Entergy Operations, Inc., Letter to U.S. Nuclear Regulatory Commission, "Supplement to License Amendment Request (LAR) Changes to Technical Specifications 5.6.5, "Core Operating Limits Report (COLR)," dated August 12, 2009 (ADAMS Accession No ML092300229).

3. EMF-2209(P)(A), Revision 2, "SPCB Critical Power Correlation," Framatome ANP, September 2003.

4. NEDE-30130-P-A, "Steady-State Nuclear Methods," April 1985, and for PANACEA11, Letter from S.A. Richards (NRC) to G.A. Watford (GE), Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady-State Methods," (TAC NO. MA6481), November 10, 1999.

5. NEDC-24154P-A, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors, Supplement 4, Volume 1," January 1998.

6. NEDC-32084P-A, "TASC-03A, A Computer Program for Transient Analysis of a Single Channel, Revision 2," July 2002.

7. NEDE-33213P-A, "ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions," April 2009.

8. NEDO-32339-A, "Reactor Stability Long-Term Solution: Enhanced Option I-A, Revision 1," April 1998.

9. NEDO-32339-A Supplement 3, "Reactor Stability Long-Term Solution: Enhanced Option I-A, Flow Mapping Methodology, Revision 1," April 1998.

10. NEDO-32339-A Supplement 4, "Reactor Stability Long-Term Solution: Enhanced Option I-A, Generic Technical Specifications, Revision 1," April 1998.

11. NEDC-32339P-A Supplement 1, "Reactor Stability Long-Term Solution: Enhanced Option I-A, ODYSY Application to E1A," December 1996.

12. NEDC-32339P-A Supplement 2, "Reactor Stability Long-Term Solution: Enhanced Option I-A, Solution Design, Revision 1," April 1998.

13. NEDE-20566P-A, "General Electric Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K," September 1986.

14. NEDE-23785-1-P-A, Revision 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Vol. 1, GESTR-LOCA - A Model for the Prediction of Fuel Rod Thermal Performance," October 1984.

15. NEDE-23785-1-P-A, Revision 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Vol. 2, SAFER - Long Term Inventory Model for BWR Loss-of-Coolant Analysis," October 1984.

16. NEDE-23785-1-P-A, Revision 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Vol. 3, SAFER/GESTR Application Methodology," October 1984.

17. NEDE-23785P-A, Revision 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Vol. 3 Supplement 1, Additional Information for Upper Bound PCT Calculation," March 2002.

Principal Contributors: A. Attard
T. Nakanishi

Date: September 29, 2009

September 29, 2009

Vice President, Operations
Entergy Operations, Inc.
River Bend Station
5485 U.S. Highway 61N
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: CHANGES TO TECHNICAL SPECIFICATION 5.6.5, "CORE OPERATING LIMITS REPORT (COLR) (TAC NO. ME0157)

Dear Sir or Madam:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 166 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 November 20, 2008, as supplemented by letter dated August 12, 2009.

The amendment revises TS 5.6.5, "Core Operating Limits Report (COLR)," to add a reference to an analytical method that will be used to determine the core operating limits. The change is needed to support the use of GE14 fuel during refueling outage 15 scheduled for the fall of 2009.

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/

Alan B. Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 166 to NPF-47
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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T. Nakanishi, NRR/DSS/SRXB
A. Attard, NRR/DSS/SNPB

ADAMS Accession No. ML092430145 *SE memo dated

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	DIRS/ITSB/BC	DSS/SNPB/BC	DSS/SXRB/BC	OGC	NRR/LPL4/BC	NRR/LPL4/PM
NAME	ABWang	JBurkhardt	RElliott	AMendiola*	GCranston*	BHarris	MMarkley	ABWang
DATE	9/14/09	9/14/09	9/14/09	8/27/09	8/27/09	9/21/09	9/28/09	9/29/09

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