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

December 11, 2019

Mrs. Mandy Halter
Acting Vice President, Regulatory
Assurance Licensing
Entergy Services, LLC
M-ECH-61
1340 Echelon Parkway
Jackson, MS 39213

SUBJECT: GRAND GULF NUCLEAR STATION, UNIT 1 AND RIVER BEND STATION, UNIT 1 – ISSUANCE OF AMENDMENTS RE: ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER TSTF-564, REVISION 2, "SAFETY LIMIT MCPR" (EPID L-2019-LLA-0051)

Dear Mrs. Halter:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued amendments consisting of changes to the technical specifications (TSs) in response to your application dated March 7, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19070A227). The following amendments are enclosed:

- Amendment No. 222 to Renewed Facility Operating License No. NPF-29 for Grand Gulf Nuclear Station, Unit 1 (Grand Gulf)

- Amendment No. 200 to Renewed Facility Operating License No. NPF-47 for River Bend Station, Unit 1 (River Bend)

The amendments revise the TSs for each facility based on Technical Specifications Task Force (TSTF) Traveler TSTF-564, Revision 2, "Safety Limit MCPR [Minimum Critical Power Ratio]," dated October 24, 2018 (ADAMS Accession No. ML18297A361). The NRC issued a final safety evaluation approving traveler TSTF-564, Revision 2, on November 16, 2018 (ADAMS Accession No. ML18299A069). The amendments revise the Grand Gulf and River Bend TS Safety Limit 2.1.1.2 and TS 5.6.5, "Core Operating Limits Report (COLR)."

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

Sincerely,

MOBarnes for

Siva P. Lingam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-416 and 50-458

Enclosures:
1. Amendment No. 222 to NPF-29
2. Amendment No. 200 to NPF-47
3. Safety Evaluation

cc: Listserv

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

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

COOPERATIVE ENERGY, A MISSISSIPPI ELECTRIC COOPERATIVE

ENTERGY MISSISSIPPI, LLC

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 222
Renewed License No. NPF-29

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated March 7, 2019, 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, 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 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 Renewed Facility Operating License No. NPF-29 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 222 are hereby incorporated into this renewed license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. NPF-29 and
the Technical Specifications

Date of Issuance: December 11, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 222

RENEWED FACILITY OPERATING LICENSE NO. NPF-29

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

Replace the following pages of the Renewed Facility Operating License No. NPF-29 and the 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
-4-	-4-

Technical Specifications

REMOVE	INSERT
2.0-1	2.0-1
5.0-18	5.0-18

amended, are fully applicable to the lessors and any successors in interest to those lessors, as long as the renewed license of GGNS Unit 1 remains in effect.

(b) SERI is required to notify the NRC in writing prior to any change in (i) the terms or conditions of any new or existing sale or lease agreements executed as part of the above authorized financial transactions, (ii) the GGNS Unit 1 operating agreement, (iii) the existing property insurance coverage for GGNS Unit 1 that would materially alter the representations and conditions set forth in the Staff's Safety Evaluation Report dated December 19, 1988 attached to Amendment No. 54. In addition, SERI is required to notify the NRC of any action by a lessor or other successor in interest to SERI that may have an effect on the operation of the facility.

C. The renewed 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

Entergy Operations, Inc. is authorized to operate the facility at reactor core power levels not in excess of 4408 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 222 are hereby incorporated into this renewed license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

During Cycle 19, GGNS will conduct monitoring of the Oscillation Power Range Monitor (OPRM). During this time, the OPRM Upscale function (Function 2.f of Technical Specification Table 3.3.1.1-1) will be disabled and operated in an "indicate only" mode and technical specification requirements will not apply to this function. During such time, Backup Stability Protection measures will be implemented via GGNS procedures to provide an alternate method to detect and suppress reactor core thermal hydraulic instability oscillations. Once monitoring has been successfully completed, the OPRM Upscale function will be enabled and technical specification requirements will be applied to the function; no further operating with this function in an "indicate only" mode will be conducted.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 21.8% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and process control program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Deleted

5.6.5 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1) LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR),

2) LCO 3.2.2, Minimum Critical Power Ratio (MCPR)(including power and flow dependent limits, and the cycle-specific $MCPR_{99.9\%}$),

3) LCO 3.2.3, Linear Heat Generation Rate (LHGR),

4) Deleted

5) LCO 3.3.1.1, RPS Instrumentation, Table 3.3.1.1-1 APRM Function 2.f

6) The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Flow Biased Simulated Thermal Power - High trip function (Function 2.d) setpoints used in the OPRM Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1.

(continued)

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

ENTERGY LOUISIANA, LLC

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 200
License No. NPF-47

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

   A. The application for amendment by Entergy Operations, Inc., acting as an agent for itself and Entergy Louisiana, LLC, dated March 7, 2019, 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 Renewed 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. 200 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed 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 within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: December 11, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 200

RENEWED FACILITY OPERATING LICENSE NO. NPF-47

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

Replace the following pages of the Renewed 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.

Renewed Facility Operating License

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
2.0-1	2.0-1
5.0-18	5.0-18

(2) EOI, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the above designated location in accordance with the procedures and limitations set forth in this renewed license;

(3) EOI, pursuant to Section 103 of 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 Section 103 of 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 Section 103 of 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 Section 103 of 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 renewed 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 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.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 200 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23.8% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the plant manager and the corporate executive responsible for overall plant nuclear safety.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and process control program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Deleted

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1) LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR),
2) LCO 3.2.2, Minimum Critical Power Ratio (MCPR)(including power and flow dependent limits and the cycle-specific $MCPR_{99.9\%}$).
3) LCO 3.2.3, Linear Heat Generation Rate (LHGR)(including power and flow dependent limits).
4) LCO 3.2.4, Fraction of Core Boiling Boundary (FCBB)
5) LCO 3.3.1.1, RPS Instrumentation (RPS), Function 2.b
6) LCO 3.3.1.3, Periodic Based Detection System (PBDS)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-29 AND AMENDMENT NO. 200 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

RIVER BEND STATION, UNIT 1

DOCKET NOS. 50-416 AND 50-458

1.0 INTRODUCTION

By application dated March 7, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19070A227), Entergy Operations, Inc. (the licensee) submitted a license amendment request for Grand Gulf Nuclear Station, Unit 1 (Grand Gulf) and River Bend Station, Unit 1 (River Bend), which are boiling water reactors (BWRs).

The proposed amendments to the Grand Gulf and River Bend technical specifications (TSs) would revise the reactor core safety limit for the minimum critical power ratio (MCPR), which protects against boiling transition on the fuel rods in the core. The current MCPR safety limit for Grand Gulf and River Bend ensures that 99.9 percent of the fuel rods in the core are not susceptible to boiling transition. The revised MCPR safety limit would ensure that there is a 95-percent probability at a 95-percent confidence level (95/95) that no fuel rods will be susceptible to boiling transition based on a statistical analysis of critical power ratio (CPR) data. The TS requirements for the core operating limits report (COLR) would also be modified.

The proposed changes are based on Technical Specifications Task Force (TSTF) traveler TSTF-564, Revision 2, "Safety Limit MCPR," dated October 24, 2018 (ADAMS Accession No. ML18297A361). The U.S. Nuclear Regulatory Commission (NRC, the Commission) issued a final safety evaluation (SE) approving traveler TSTF-564, Revision 2, on November 16, 2018 (ADAMS Accession No. ML18299A069).

The licensee has proposed several variations from the TS changes described in traveler TSTF-564, Revision 2. The variations are evaluated in Section 3.0 of this SE.

1.1 Background on Boiling Transition

During steady-state operation in a BWR, most of the coolant in the core is in a flow regime known as annular flow. In this flow regime, a thin liquid film is pushed up the surface of the fuel rod cladding by the bulk coolant flow, which is mostly water vapor with some liquid water droplets. This provides effective heat removal from the cladding surface; however, under certain conditions, the annular film may dissipate, which reduces the heat transfer and results in an increase in fuel cladding surface temperature. This phenomenon is known as boiling transition or dryout. The elevated surface temperatures resulting from dryout may cause fuel cladding damage or failure.

1.2 Background on Critical Power Correlations

For a given set of reactor operating conditions (pressure, flow, etc.), dryout will occur on a fuel assembly at a certain power, known as the critical power. Because the phenomena associated with boiling transition are complex and difficult to model purely mechanistically, thermal-hydraulic test campaigns are undertaken using electrically heated prototypical fuel bundles to establish a comprehensive database of critical power measurements for each BWR fuel product. These data are then used to develop a critical power correlation that can be used to predict the critical power for assemblies in operating reactors. This prediction is usually expressed as the CPR, which is the ratio of the critical power predicted using the correlation to the actual assembly power.

One measure of the correlation's predictive capability is based on its validation relative to the test data. For each point *j* in a correlation's test database, the experimental critical power ratio (ECPR) is defined as the ratio of the measured critical power to the calculated critical power:

$$\mathrm{ECPR}_j = \frac{\text{Measured Critical Power}_j}{\text{Calculated Critical Power}_j}$$

For ECPR values less than or equal to 1, the calculated critical power is greater than or equal (≥) to the measured critical power and the prediction is considered to be non-conservative. Because the measured critical power includes random variations due to various uncertainties, evaluating the ECPR for all of the points in the dataset (or, ideally, a subset of points that were not used in the correlation's development, for evaluation of the predictive quality of the model) results in a probability distribution. This ECPR distribution allows the predictive uncertainty of the correlation to be determined. This uncertainty can then be used to establish a limit above which it can be assumed that boiling transition will not occur (with a certain probability and confidence level).

1.3 Background on Thermal-Hydraulic Safety Limits

To protect against boiling transition, BWRs have established MCPR safety limits in their TSs. The current MCPR safety limits at Grand Gulf and River Bend are based on preventing 99.9 percent of the fuel in the core from being susceptible to boiling transition. Such limits are typically developed by considering various cycle-specific power distributions and uncertainties, and they are highly dependent on the cycle-specific radial power distribution in the core. As such, the MCPR safety limits may need to be updated as frequently as every cycle.

The TSs for Grand Gulf and River Bend also include MCPR operating limits as limiting conditions for operation (LCOs), which must be met to ensure that anticipated operational occurrences (AOOs) do not result in fuel damage. Currently, the MCPR operating limits are calculated by combining the largest change in CPR from all analyzed transients with the MCPR safety limit.

2.0 REGULATORY EVALUATION

2.1 Description of TS Sections

The licensee proposed to revise the MCPR safety limit (TS 2.1.1.2) for Grand Gulf and River Bend to make it cycle-independent, consistent with the method described in TSTF-564, Revision 2. The current MCPR safety limit (referred to as the $MCPR_{99.9\%}$) ensures that 99.9 percent of the fuel rods in the core are not susceptible to boiling transition. The revised MCPR safety limit (also referred to as the $MCPR_{95/95}$ safety limit) would ensure that there is a 95-percent probability at a 95-percent confidence level that no fuel rods in the core will be susceptible to boiling transition. Unlike the current MCPR safety limit, the $MCPR_{95/95}$ safety limit is not dependent on the number of recirculation loops in operation.

The proposed TS changes replace the current MCPR safety limit values for single and two recirculation loop operation with a single $MCPR_{95/95}$ value associated with the current fuel loading. These changes are consistent with TSTF-564, Revision 2, and are shown in Table 1 below.

Table 1: Changes to the TS MCPR Safety Limit[1]

Plant	Current TS	Proposed TS
Grand Gulf	With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow: MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.	With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow: MCPR shall be ≥ 1.07.
River Bend	With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow: MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.	With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow: MCPR shall be ≥ 1.07.

The $MCPR_{99.9\%}$ is used to determine the MCPR operating limits in LCO 3.2.2, "Minimum Critical Power Ratio (MCPR)." The licensee did not propose any changes to the MCPR definition or the methods for calculating the $MCPR_{99.9\%}$ value and the MCPR operating limits. Consistent with the method described in TSTF-564, Revision 2, the licensee proposed to change the requirements for the COLR in TS 5.6.5 to require the $MCPR_{99.9\%}$ value used in calculating the MCPR operating limits to be included in the cycle-specific COLR.

[1] The pressure units are pounds per square inch gauge (psig)

2.2 Applicable Regulatory Requirements and Guidance

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications," establishes the regulatory requirements related to the content of TSs. Section 50.36(a)(1) of 10 CFR requires an application for an operating license to include proposed TSs. Section 50.36(a)(1) of 10 CFR states, in part, "A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, must also be included in the application, but shall not become part of the TSs."

In accordance with 10 CFR 50.36, TSs for operating reactors are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; and (5) administrative controls. Section 50.36(c)(1)(i)(A) of 10 CFR states, in part:

> Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission.

In accordance with 10 CFR 50.36(c)(2)(i), "limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operations of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met." Additionally, the regulation in 10 CFR 50.36(c)(5) states that "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."

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants. The general design criteria (GDC) were originally published in the *Federal Register* (36 FR 3255) on February 20, 1971, and became effective on May 21, 1971. GDC 10, "Reactor design," of 10 CFR Part 50, Appendix A, states:

> 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 10 of 10 CFR Part 50 Appendix A is applicable to Grand Gulf and River Bend. The limits placed on the MCPR are specified acceptable fuel design limits to prevent boiling transition and are used to meet GDC 10.

Acceptance criteria for the NRC staff's review of fuel design limits is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 4.4, "Thermal and Hydraulic Design," Revision 2, dated March 2007 (ADAMS Accession No. ML070550060). This guidance provides the

following two examples of acceptable approaches to meeting the NUREG-0800 acceptance criteria for establishing fuel design limits:

A. For departure from nucleate boiling ratio (DNBR), CHFR [critical heat flux ratio], or CPR correlations, there should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB [departure from nucleate boiling] or boiling transition condition during normal operation or AOOs.

B. The limiting (minimum) value of DNBR, CHFR, or CPR correlations is to be established such that at least 99.9 percent of the fuel rods in the core will not experience a DNB or boiling transition during normal operation or AOOs.

3.0 TECHNICAL EVALUATION

The proposed amendments are based on the NRC-approved TSTF-564, Revision 2. The NRC staff's evaluation of the proposed amendments relies upon the NRC staff's previous approval of the methodology and $MCPR_{95/95}$ safety limits for certain fuel types described in TSTF-564, Revision 2. The staff also considered the regulations and guidance discussed in Section 2.2 of this SE in its review.

The licensee identified differences between the TSs for each of the facilities and the BWR standard TSs,[2] upon which TSTF-564 is based. These differences included the TS numbering and the plant-specific steam dome pressure value listed in the safety limit. The NRC staff determined that these differences do not affect the applicability of TSTF-564 for Grand Gulf and River Bend.

3.1 MCPR Safety Limit Methodology

As discussed in Section 1.3 of this SE, the current MCPR safety limits (i.e., the $MCPR_{99.9\%}$ safety limits) for Grand Gulf and River Bend are dependent on the cycle-specific core design, especially including the core power distribution, fuel types in the reactor, and the plant power-to-flow operating domain. As such, it is often necessary to change the MCPR safety limits to accommodate new core designs. Changes to the MCPR safety limits are usually determined late in the design process and necessitate NRC review and approval of a license amendment to support the subsequent fuel cycle.

The licensee proposed to change the methodology for determining the MCPR safety limit for Grand Gulf and River Bend so that it is no longer cycle-dependent. The proposed methodology for determining the MCPR safety limit aligns it with the DNBR safety limit used in pressurized-water reactors, which ensures that no fuel rods will experience DNB with a 95-percent probability at a 95-percent confidence level.

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[2] U.S. Nuclear Regulatory Commission, "Standard Technical Specifications: General Electric Plants BWR/4," NUREG-1433, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4.0, April 2012 (ADAMS Accession Nos. ML12104A192 and ML12104A193, respectively).
U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, General Electric Plants BWR/6," NUREG-1434, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4.0, April 2012 (ADAMS Accession Nos. ML12104A195 and ML12104A196, respectively).

The intent of the proposed methodology for determining the revised MCPR safety limit is acceptable to the NRC staff as it is consistent with the acceptance criteria for establishing fuel design limits in NUREG-0800, Section 4.4 (see Example A in Section 2.2 of this SE).

3.2 Revised MCPR Safety Limit Calculational Method

As discussed in Section 1.2 of this SE, the ECPR distribution is used to quantify the uncertainty associated with the critical power correlation. Traveler TSTF-564, Revision 2, provides the following formula for determining the $MCPR_{95/95}$ for a given fuel type (i):

$$MCPR_{95/95}(i) = \mu_i + \kappa_i \sigma_i$$

where μ_i is the mean ECPR and σ_i is the standard deviation of the ECPR distribution. The statistical parameter (κ_i) is selected, based on the number of samples in the critical power database, to provide a one-sided upper tolerance limit with a 95-percent probability at a 95-percent confidence level. This is a commonly used statistical formula to determine one-sided upper tolerance limits for normal distributions, which is appropriate for the situation under consideration. For reactor cores loaded with a single fuel type, the $MCPR_{95/95}$ safety limit is the $MCPR_{95/95}(i)$ value for that fuel type.

In the SE approving TSTF-564, the NRC staff determined that the formula for determining $MCPR_{95/95}$ appropriately establishes a 95/95 upper tolerance limit on the critical power correlation and that any issues in the underlying correlation can be appropriately addressed through adjustments to the correlation mean and standard deviation, as necessary to ensure appropriate conservatism. Therefore, the NRC staff concluded that the proposed method for determining $MCPR_{95/95}$ can be used to establish acceptable fuel design limits.

3.3 Determination of Revised MCPR Safety Limit for Mixed Cores

Section 3.1 of TSTF-564, Revision 2, states, in part:

> For cores with a mix of fuel products, the corresponding [$MCPR_{95/95}$ safety limit] is based on the largest (i.e., most limiting) of the $MCPR_{95/95}$(i) values for the product lines that are fresh or once-burnt at the start of the cycle. The $MCPR_{95/95}(i)$ values for product lines that are twice-burnt or more at the start of the cycle may be ignored, as these higher exposure bundles operate with considerable MCPR margin relative to the more limiting fresh and once-burnt bundles.

Fuel that is twice-burnt or more has a probability of boiling transition that is very small compared to the limiting bundle and can be neglected in determining the safety limit. In its letter dated May 29, 2018 (ADAMS Accession No. ML18149A320), the TSTF provided results of a study that confirmed this is valid even for fuel operated on short (12-month) reload cycles. Fuel that is twice-burnt or more is included in the cycle-specific evaluation of the $MCPR_{99.9\%}$ value and the MCPR operating limits. If a fuel bundle that is twice-burnt or more is found to be limiting, it would be governed by the MCPR operating limits, which will always be more restrictive than both the $MCPR_{95/95}$ safety limit and the $MCPR_{99.9\%}$ value.

The NRC staff reviewed the information provided by the TSTF and determined that the process for establishing the revised $MCPR_{95/95}$ safety limit for mixed cores is acceptable. Specifically, the NRC staff found it acceptable, based on the information above, to determine the $MCPR_{95/95}$

safety limit for the core based on the most limiting $MCPR_{95/95}$ value for fresh and once-burnt fuel in the core.

3.4 Relationship between MCPR Safety and Operating Limits

As discussed in the TSTF letter dated May 29, 2018, the current $MCPR_{99.9\%}$ safety limits are always greater than the proposed $MCPR_{95/95}$ safety limits because (1) the $MCPR_{99.9\%}$ includes uncertainties not factored into the $MCPR_{95/95}$ and (2) even if these additional uncertainties are neglected, a statistical comparison shows that the $MCPR_{99.9\%}$ is more conservative than the $MCPR_{95/95}$. The level of conservatism in the $MCPR_{95/95}$ safety limit is appropriate because the lead fuel rod in the core (i.e., the limiting fuel rod with respect to MCPR) is used to evaluate whether any fuel rods in the core are susceptible to boiling transition. This is consistent with evaluations performed for pressurized water reactors using a 95/95 upper tolerance limit on the correlation uncertainty as a safety limit.

Consistent with TSTF-564, Revision 2, the licensee will determine the MCPR operating limits for LCO 3.2.2 at Grand Gulf and River Bend using the $MCPR_{99.9\%}$ as an input. The licensee also proposed to revise the COLR TS (i.e. TS 5.6.5 for Grand Gulf and River Bend) to require the cycle-specific value of the $MCPR_{99.9\%}$ to be determined for LCO 3.2.2 and included in the COLR. The licensee did not propose any changes to how it determines the $MCPR_{99.9\%}$. The analytical methods for determining $MCPR_{99.9\%}$ are included in the list of COLR references contained in TS 5.6.5. The proposed changes to TS 5.6.5 will ensure that the uncertainties being removed from the MCPR safety limits are still included as part of the MCPR operating limits and will continue to appropriately inform plant operation. Therefore, the NRC staff concludes that the TSs for each facility will continue to provide appropriate administrative controls in accordance with 10 CFR 50.36(c)(5).

The NRC staff therefore determined that, with the proposed changes, the TS safety limits for the MCPR will retain an adequate level of conservatism and that plant- and cycle-specific uncertainties will be appropriately retained in the MCPR operating limits. The $MCPR_{95/95}$ represents a lower limit on the value of the $MCPR_{99.9\%}$, because the $MCPR_{99.9\%}$ should always be higher since it accounts for numerous uncertainties that are not included in the $MCPR_{95/95}$.

3.5 Implementation of the Revised MCPR Safety Limit in the TSs

The licensee has proposed to eliminate the current MCPR safety limits for single and two loop operation for Grand Gulf and River Bend and replace them with a single MCPR safety limit. The proposed MCPR safety limit is the $MCPR_{95/95}$ value associated with the current fuel loading.

Table 1 of TSTF-564, Revision 2, states that the $MCPR_{95/95}$ safety limit for Global Nuclear Fuel 2 (GNF2) and Global Nuclear Fuel 3 (GNF3) fuel is 1.07. As discussed in TSTF-564, Revision 2, the derivation of these values using the methodology described in the TSTF was provided to the NRC in letter dated July 30, 2018 (ADAMS Package Accession No. ML18212A017).

The licensee stated that Grand Gulf is currently fueled with GNF2 and proposed to revise the TS MCPR safety limit to 1.07, consistent with Table 1 of TSTF-564, Revision 2.

River Bend is currently fueled with GNF2 and GNF3. For mixed cores, the $MCPR_{95/95}$ safety limit is based on the largest $MCPR_{95/95}$ value for the fuel types used. Consistent with Table 1 of TSTF-564, the licensee proposed to revise the TS MCPR safety limit to 1.07.

The NRC staff reviewed the licensee's proposed changes to the TS MCPR safety limits for Grand Gulf and River Bend to reflect the change to $MCPR_{95/95}$ safety limits. The NRC staff found the proposed changes acceptable because they are consistent with TSTF-564, Revision 2, as approved by the NRC staff. Specifically, the licensee appropriately eliminated the separate MCPR safety limits for single and two recirculation loop operation, as the $MCPR_{95/95}$ safety limit does not depend on the number of recirculation loops in operation. Additionally, the proposed $MCPR_{95/95}$ safety limit value for Grand Gulf is consistent with the value in Table 1 of TSTF-564 for reactors fueled with GNF2. For River Bend, the licensee appropriately selected the $MCPR_{95/95}$ safety limit value for the transition from GNF2 to GNF3. The $MCPR_{95/95}$ safety limit value for the final core of GNF3 is equal to the value for GNF2 in Table 1 of TSTF-564, which is acceptable to the NRC staff because it is conservative. Therefore, the NRC staff concludes that the proposed $MCPR_{95/95}$ for Grand Gulf and River Bend is an acceptable fuel design limit, and 10 CFR 50.36(c)(1)(i)(A) will be met since the limit will reasonably protect the integrity of the fuel cladding to guard against the uncontrolled release of radioactivity.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Mississippi and Louisiana officials were notified of the proposed issuance of the amendment on November 19, 2019. The State officials had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted areas as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve 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, which was published in the *Federal Register* on May 21, 2019 (84 FR 23076), that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. 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 amendments.

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) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Joshua M. Wilson, NRR

Date: December 11, 2019

SUBJECT: GRAND GULF NUCLEAR STATION, UNIT 1 AND RIVER BEND STATION, UNIT 1 – ISSUANCE OF AMENDMENTS RE: ADOPTION OF TECHNICAL SPECIFICATIONS TASK FORCE TRAVELER TSTF-564, REVISION 2, "SAFETY LIMIT MCPR" (EPID L-2019-LLA-0051) DATED DECEMBER 11, 2019

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