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

December 31, 2019

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

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT NO. 201 RE: CHANGE TO THE NEUTRON ABSORBING MATERIAL CREDITED IN SPENT FUEL POOL FOR CRITICALITY CONTROL (EPID L-2018-LLA-0298)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 201 to Renewed Facility Operating License No. NPF-47 for the River Bend Station, Unit 1 (River Bend). The amendment consists of changes to the technical specifications (TSs) in response to your application dated October 24, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18297A103), as supplemented by letter dated June 4, 2019 (ADAMS Accession No. ML19155A226).

The amendment revises the TSs to allow use of neutron absorbing inserts in the spent fuel pool (SFP) storage racks for the purpose of criticality control in the SFP. Entergy Operations, Inc. (EOI) proposed these changes due to degradation of the Boraflex neutron absorbing material in the River Bend SFP. Specifically, the amendment revises the nuclear criticality safety analysis for the SFP to allow crediting of NETCO-SNAP-IN® neutron absorbing racks; changes the language in River Bend TS 4.3.1, "Criticality," to reflect the new neutron absorbing inserts; and adds a requirement for a monitoring program for the neutron absorbing rack inserts in River Bend TS 5.5, "Programs and Manuals."

By letter dated June 4, 2019, EOI stated that it had initiated the installation of the new NETCO-SNAP-IN® rack inserts at River Bend in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59 "Changes, tests and experiments," using the existing nuclear criticality safety analysis methodology. The NRC staff did not review EOI's proposed use of the 10 CFR 50.59 process as part of the review of the license amendment request.

Enclosure 2 to this letter contains Proprietary Information. Upon separation from Enclosure 2, this letter is DECONTROLLED.

The NRC staff has determined that the related safety evaluation contains proprietary information pursuant to 10 CFR 2.390. The proprietary version of the safety evaluation is provided in Enclosure 2. Accordingly, the NRC staff has also prepared a non-proprietary version of the safety evaluation, which is provided in Enclosure 3.

Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Margaret W. O'Banion, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 201 to NPF-47
2. Safety Evaluation (Proprietary)
3. Safety Evaluation (Non-Proprietary)

cc w/o Enclosure 2: Listserv

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. 201
Renewed 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 October 24, 2018, as supplemented by letter dated June 4, 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 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. 201 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 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Jennifer 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 31, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 201

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
4.0-1	4.0-1
4.0-2	4.0-2
5.0-16b	5.0-16b

(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. 201 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.

4.0 DESIGN FEATURES

4.1 Site Location

The River Bend Station is located in West Feliciana Parish, Louisiana, on the east bank of the Mississippi River approximately 24 miles north-northwest of Baton Rouge (city center), Louisiana. The site comprises approximately 3342 acres. The exclusion area boundary shall have a radius of 3000 feet from the centerline of the reactor.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 624 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 145 cruciform shaped control rod assemblies. The control material shall be boron carbide or hafnium metal, or both.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum k-infinity of 1.28 in the normal reactor core configuration at cold conditions and a maximum average U-235 enrichment of 4.9 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the USAR;

(continued)

4.0 DESIGN FEATURES

4.3.1.1 (continued)

c. A nominal fuel assembly center to center storage spacing of 7 inches within rows and 12.25 inches between rows in the low density storage racks in the upper containment pool; and

d. A nominal fuel assembly center to center storage spacing of 6.28 inches within a rack and 8.5 inches between cell centers of adjacent racks, with a neutron absorber insert within the storage cells, in the high density storage racks in the spent fuel storage facility in the Fuel Building.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1 of the USAR;

b. A nominal fuel assembly center to center storage spacing of 7 inches within rows and 12.25 inches between rows in the new fuel storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 95 ft.

4.3.3 Capacity

4.3.3.1 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3104 fuel assemblies.

4.3.3.2 No more than 200 fuel assemblies may be stored in the upper containment pool.

5.5.14 Control Room Envelope Habitability Program (continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one subsystem of the CRFA System, operating at the flow rate required by the VFTP, at a Frequency in accordance with the Surveillance Frequency Control Program. The results shall be trended and used as part of the CRE boundary assessment specified in 5.5.14.c (ii).

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5.15 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

This program provides controls for monitoring the condition of the neutron absorber inserts used in the high density storage racks in the spent fuel storage facility in the Fuel Building to verify the Boron-10 areal density is consistent with the assumptions in the spent fuel pool criticality analysis. The program shall be in accordance with NEI 16-03-A, "Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools," Revision 0, May 2017.

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

(NON-PROPRIETARY)

Proprietary information pursuant to Section 2.390 of Title 10 of
the *Code of Federal Regulations* has been redacted from this document.

Redacted information is identified by blank space enclosed within double brackets

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By application dated October 24, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18297A103), as supplemented by letter dated June 4, 2019 (ADAMS Accession No. ML19155A226), Entergy Operations, Inc. (EOI), requested changes to the technical specifications (TSs) for the River Bend Station, Unit 1 (River Bend).

The amendment would revise the TSs to allow use of neutron absorbing inserts in the spent fuel pool (SFP) storage racks for the purpose of criticality control in the River Bend SFP. EOI proposed these changes due to degradation of the Boraflex neutron absorbing material in the River Bend SFP. Specifically, the proposed amendment would (1) revise the nuclear criticality safety (NCS) analysis for the fuel handling building SFP to allow crediting of NETCO-SNAP-IN® neutron absorbing racks, (2) change the language in River Bend TS 4.3.1, "Criticality," to reflect the new neutron absorbing inserts, and (3) add a requirement for a monitoring program for the neutron absorbing rack inserts in River Bend TS 5.5, "Programs and Manuals."

By letter dated June 4, 2019, EOI stated that it had initiated the installation of the new NETCO-SNAP-IN® rack inserts at River Bend in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59 "Changes, tests and experiments," using the existing NCS analysis methodology. The NRC staff did not review EOI's proposed use of the 10 CFR 50.59 process as part of the review of the license amendment request (LAR).

The supplemental letter dated June 4, 2019, 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 or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 5, 2019 (84 FR 1805).

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance documents that the NRC staff used in the review of the LAR are listed below.

2.1 Regulatory Requirements

Per paragraph (a) of 10 CFR 50.68, "Criticality accident requirements," each holder of an operating license shall comply with either 10 CFR 70.24, "Criticality accident requirements," or the requirements in 10 CFR 50.68(b). EOI has elected to meet 10 CFR 50.68(b), and accordingly, must comply with the following requirements:

> (1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.
>
> (4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The regulations in 10 CFR 50.36, "Technical specifications," contain the requirements for the content of TSs. The regulations in 10 CFR 50.36(b) require TSs to be derived from the analyses and evaluation included in the safety analysis report and amendments thereto. As required by 10 CFR 50.36(c)(4), "Design features," the TSs will include design features "of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of [10 CFR 50.36]." The regulations in 10 CFR 50.36(c)(5), "Administrative controls," state that the TSs will include "provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner."

General Design Criteria (GDC) 61, "Fuel storage and handling and radioactivity control," of 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," states in part, that "The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety..."

GDC 62, "Prevention of criticality in fuel storage and handling," of 10 CFR Part 50, Appendix A, states that, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

2.2 Guidance Documents

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 9.1.1, Revision 3, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," dated March 2007; and Section 9.1.2, Revision 4, "New and Spent Fuel Storage," dated March 2007 (ADAMS Accession Nos. ML070570006 and ML070550057, respectively), provide guidance regarding the acceptance criteria and review procedures to ensure that the proposed changes satisfy the requirements in 10 CFR 50.68. The NRC staff notes that while Section 9.1.2 of NUREG-0800 is applicable, it is not concerned directly with fuel storage criticality safety considerations. Therefore, Section 9.1.1 of NUREG-0800 contains the primary standard review plan guidance for reviewing the proposed changes in the LAR as it relates to fuel storage criticality safety considerations.

NRC memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants" (hereafter the "Kopp memo"), dated August 19, 1998, contains guidance for NRC staff for performing the review of SFP NCS analysis (ADAMS Accession No. ML003728001). The Kopp memo provides guidance on the more salient aspects of an NCS analysis, including computer code validation. The guidance is germane to boiling water reactors (BWRs) and pressurized water reactors for both borated and unborated fuel storage pools.

The NRC staff used NRC Interim Staff Guidance (ISG) entitled, "Final Division of Safety Systems Interim Staff Guidance, DSS-ISG-2010-01, Revision 0, 'Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools,'" dated September 2011 (ADAMS Accession No. ML110620086) to review the SFP criticality analyses (notice of availability published in the *Federal Register* on October 13, 2011 (76 FR 63676)). The guidance in DSS-ISG-2010-01 is used by the NRC staff to review NCS analyses for the storage of new and spent nuclear fuel as it applies to: (i) future applications for construction and/or operating licenses, and (ii) future applications for license amendments and requests for exemptions from compliance with applicable requirements that are approved after the date of this interim staff guidance.

NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," dated December 2010 (ADAMS Accession No. ML103490041), provides guidance on what constitutes an acceptable monitoring program for neutron absorbing material (NAM) credited for criticality control in the SFP.

3.0 TECHNICAL EVALUATION

The NRC staff's review of the LAR focused on the following areas: (1) SFP NCS analysis, (2) material characteristics and compatibility, and (3) TS changes.

3.1 SFP NCS Analysis Review

3.1.1 Background

The proposed amendment requested TS revisions to support a new NCS analysis that credits new NETCO-SNAP-IN® neutron absorbing inserts in each cell and no longer credits Boraflex in the River Bend SFP.

EOI's NCS analysis describes the methodology and analytical models used to show that the multiplication factor (k_{eff}) of the spent fuel storage racks, loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water, must not exceed 0.95, at a 95 percent confidence level.

The NETCO-SNAP-IN® are chevron-shaped neutron absorbing inserts. All NETCO-SNAP-IN® inserts will have the same orientation, ensuring one leg of an insert will be between fuel assemblies. The inserts extend over the full-length of the active fuel region of the stored assemblies. The inserts are manufactured from a boron-carbide metal matrix composite with a minimum certified areal density of 0.0129 grams (g) of Boron-10 (B^{10}) per centimeter squared (B^{10}/cm^2). In this analysis, a lower areal density of 0.0115 g B^{10}/cm^2 was used in the base model. However, this does not represent any margin in the analysis because the application states that the monitoring program is to "...verify the Boron-10 areal density is consistent with the assumptions in the spent fuel pool criticality analysis." Since the 0.0115 g B^{10}/cm^2 value was used in the analysis, the monitoring program will verify that value.

3.1.2 Methodology for SFP NCS Analysis

There is no generic or standard NRC-approved methodology for performing NCS analyses for fuel storage and handling. The Kopp memo provides some guidance on several aspects of criticality analysis, and it can be used for BWRs with unborated SFPs.

The methods used for the NCS analysis for fuel in the River Bend SFP are described in the criticality analysis, which was provided in Attachment 3 (non-proprietary) and Attachment 7 (proprietary) of EOI's application dated October 24, 2018. The methodology is specific to this analysis and is not appropriate for other applications.

EOI's new SFP NCS analysis describes the criticality analysis and results for the River Bend SFP racks with credit for NETCO-SNAP-IN® neutron absorbing inserts in each cell. No credit for the Boraflex neutron absorber is taken in the analysis. The analysis covers the current Global Nuclear Fuel (GNF)2 and GNF3 fuel product lines and all legacy fuel stored in the River Bend SFP.

In its application, EOI stated that the SFP racks are analyzed using the MCNP-05P Monte Carlo neutron transport program and ENDF/B-VII.0 cross-section library. The methodology used in the analysis is the peak cold in-core k-infinity (k_∞) eigenvalue criterion methodology. A maximum cold, uncontrolled peak in-core k_∞ of 1.28, as defined by the lattice physics code TGBLA06, is set as the limit for this analysis. This value is incorporated into TS 4.3.1 and is the nominal k_{eff} of the fuel assembly used in the analysis.

In the application, the licensee's analysis states that the as analyzed SFP k_{eff} is **[[ ]]** at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. This included a Δk_{eff} of **[[ ]]** for the NRC to use to address possible issues in the analysis, as referenced in the application. The analysis demonstrates this results in an SFP k_{eff} with significant margin to the regulatory limit of 0.95 k_{eff} for normal and credible abnormal operation with tolerances and uncertainties taken into account.

The maximum cold, uncontrolled peak in-core k_∞ value of 1.28, which is referenced in the TSs, is the nominal k_{eff} of the fuel assembly used in the analysis. Therefore, the licensee's analysis indicates there is significant margin to the regulatory limit, but not to the k_∞ value that is referenced in the TSs. The NRC staff took this into account during its review by using engineering judgment to determine that small changes to the licensee's analysis would not change the NRC staff's overall conclusion.

3.1.2.1 Computational Methods and Validation

EOI used two computational methods in the criticality analysis, TGBLA06 and MCNP-05P. GNF lattice design code TGBLA06 was used to calculate burned fuel compositions and the in-core k_∞ values. The burned fuel compositions were then used in MCNP-05P, the GNF proprietary version of MCNP5, to obtain fuel storage rack k_{eff} values. The NRC staff evaluated the use of the two methods, as discussed below.

<u>TGBLA06</u>

TGBLA06 is a two-dimensional lattice design computer program for BWR fuel bundle analysis. It assumes that a lattice is uniform and infinitely long along the axial direction, and that the lattice geometry and material are reflecting with respect to the lattice boundary along the transverse directions. The NRC staff previously reviewed and accepted the use of TGBLA06 for BWR core depletion calculations, as part of the approval of Amendment 26 of NEDE-24022-P-A, "GESTAR II – Implementing Improved GE [General Electric] Steady-State Methods" (ADAMS Package Accession No. ML993230387), for operating BWRs.

TGBLA06 is considered to be adequately validated by its virtue of being previously approved by the NRC for reactor core design and modeling, and its continued use in core design of operating reactors is benchmarking against actual plant operations. The NRC staff finds the use of TGBLA06A acceptable in support of the River Bend SFP NCS analysis.

<u>MCNP-05P</u>

For the SFP k_{eff} calculations, EOI used the Monte Carlo neutron transport program (MCNP)-05P and ENDF/B-VII.0 cross-section library. The NRC staff has previously accepted this code and the nuclear data library for use in criticality analysis for SFP license amendments. EOI confirmed that all calculations converged using appropriate convergence checks.

MCNP-05 is a commonly used computer code for criticality analyses, provided it is properly validated. However, MCNP-05 is not generically approved by the NRC. The purpose of the criticality code validation is to ensure that appropriate code bias and bias uncertainty are determined for use in the criticality calculation.

The ISG DSS-ISG-2010-01 references NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," dated January 2001 (ADAMS Accession No. ML050250061), for guidance on criticality code validation. NUREG/CR-6698 outlines the basic elements of validation, including identification of operating conditions and parameter ranges to be validated, selection of critical benchmarks, modeling of benchmarks, statistical analysis of results, and determination of the area of applicability.

NUREG/CR-6698 states, in part, that:

> In general, the critical experiments selected for inclusion in the validation must be representative of the types of materials, conditions, and operating parameters found in the actual operations to be modeled using the calculational method. A sufficient number of experiments with varying experimental parameters should be selected for inclusion in the validation to ensure as wide an area of applicability as feasible and statistically significant results.

The NRC staff used NUREG/CR-6698 as guidance for review of the code validation methodology provided in the application. The suite of experiments EOI used to validate MCNP-05P is similar to those used in NRC-approved license amendments for other facilities. The validation was performed in a manner consistent with NUREG/CR-6698. Therefore, this validation of MCNP-5P is acceptable.

3.1.3 SFP and Fuel Storage Racks

3.1.3.1 SFP Water Temperature

The majority of the analyses was done at a constant SFP water temperature. Guidance in the Kopp memo calls for the analysis to be performed at the temperature that results in the maximum k_{eff}. EOI performed a sensitivity analysis for the SFP water temperature over an appropriate range of temperatures, to determine which temperature resulted in the maximum k_{eff} and included a bias for the reactivity effect. This is an acceptable way to meet the guidance; thus, the NRC staff finds that this part of the analyses is acceptable.

3.1.3.2 SFP Storage Rack Models

The River Bend SFP storage racks are constructed of stainless steel boxes with Boraflex positioned on the outside of each face. The NAM is held in place by a thin sheet of stainless steel termed a "wrapper" or "sheathing." The wrapper is open at the top and tack welded along the sides and bottom. These boxes are attached at the corners forming what is called an "egg crate" with "formed" cells created between the manufactured cells. The "egg crate" is completed using filler panels to enclose "formed" cells on the periphery. The resultant cell is attached to a base plate, forming a storage module.

The wrapper and cell wall create a local environment that surrounds the NAM and is open to the SFP water. EOI modeled the volume taken up by the Boraflex as SFP water. While initially a conservative model, eventually the Boraflex will degrade to the point of being ineffective, and this modeling will become the actual condition.

NETCO-SNAP-IN® inserts are chevron shaped with two legs. The NETCO-SNAP-IN® inserts will be installed in the same orientation, ensuring one leg of an insert will be between fuel assemblies. Because of the chevron shape on two peripheral sides, there will be no insert material between the fuel and the surrounding SFP environment. The inserts extend over the full-length of the active fuel region of the stored assemblies.

The actual River Bend SFP rack modules are simulated in the analysis by being modeled as a 10x10 array of storage cells with identical fuel assemblies in each storage location. The 10x10 array has a periodic boundary condition. The periodic boundary condition simulates an infinite number of identical 10x10 arrays with no gap between them.

3.1.3.3 SFP Storage Rack Models Manufacturing Tolerances and Uncertainties

The manufacturing tolerances of the storage racks contribute to SFP reactivity. The ISG DSS-ISG-2010-01 does not explicitly discuss the approach to be used in determining manufacturing tolerances, but the licensees' past practice has been consistent with the Kopp memo. The Kopp memo discusses that determination of the maximum k_{eff} should consider either: (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant tolerance variations in the material and mechanical specifications of the racks. EOI utilized the latter approach with the appropriate as-built parameters of the SFP rack modules and NETCO-SNAP-IN® rack inserts that would result in a significant reactivity impact. EOI combined the uncertainties using the root sum squares (RSS) statistical method. EOI's process is consistent with the guidance in the Kopp memo. Therefore, the NRC staff finds it acceptable.

3.1.3.4 SFP Storage Rack Interfaces

The ISG DSS-ISG-2010-01 provides guidance on consideration for rack interface effects. The River Bend SFP has a single rack design. Because EOI is establishing a peak reactivity limit applicable to every storage cell, and the analysis does not take any credit for gaps between rack modules, there are no intra-rack interfaces to consider. However, because there will be two peripheries of the SFP storage racks without a poison panel, there is a potential interface with the surrounding pool walls or water. EOI considered this possibility and performed a conservative assessment that included a bias for the interface effect of the two peripheral sides without an insert leg and the surrounding environment. This is an acceptable manner to treat the interface between the SFP storage racks and their surroundings that is consistent with the guidance; therefore, the NRC finds that this part of the analysis is acceptable.

3.1.3.5 Missing NETCO-SNAP-IN® Rack Insert

The NETCO-SNAP-IN® monitoring program stipulates that an insert is to be removed periodically for inspection. EOI chose to treat a single missing NETCO-SNAP-IN® rack insert as a normal condition. This allows for the removal of one insert for inspection. EOI included a bias for this condition. Inserts might also be inadvertently removed when withdrawing a fuel assembly from the storage cell. EOI's analysis covers both scenarios, for a single insert. EOI modeled one missing insert in its array of storage cells. While EOI used periodic boundary conditions on its model to simulate an infinite number of arrays and an infinite number of missing inserts, EOI's modeled array is too large for there to be any neutron communication between its modeled arrays. Therefore, EOI only effectively modeled one missing insert. However, modeling one missing insert is acceptable for the periodic removal of an insert for inspection or the inadvertent removal of an insert during fuel handling. The NRC staff finds that this part of the analyses is acceptable.

3.1.4 Fuel Assembly

3.1.4.1 Bounding Fuel Assembly Design

The regulation at 10 CFR 50.68(b)(4) states, in part, "...fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water." To meet this requirement, licensees must determine the most reactive fuel assemblies they have or intend to have. For BWR fuel assemblies, the determination is made by evaluating the different lattice designs, with some fuel assemblies having multiple lattice designs. EOI performed an analysis to determine the most limiting lattice from GNF2 and GNF3 fuel designs. GNF2 and GNF3 fuel designs represent current and potential future River Bend fuel. EOI then used the limiting lattice in the analysis. EOI analyzed legacy fuel assembly designs and demonstrated they were bounded by the limiting lattice. This is an acceptable manner to treat current, potential future fuel, and legacy fuel designs and meet the regulatory requirement, and is therefore acceptable.

3.1.4.2 Fuel Assembly Manufacturing Tolerances and Uncertainties

As discussed in Section 3.1.3.3 of this safety evaluation, the manufacturing tolerances of the fuel assemblies contribute to SFP reactivity. As discussed in the Kopp memo, determination of the maximum k_{eff} should consider either: (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant tolerance variations in the material and mechanical specifications of the racks. EOI utilized the latter approach with the appropriate design parameters and tolerances of the limiting lattice that would result in a significant reactivity impact. EOI combined the uncertainties using the RSS statistical method. EOI's process is consistent with the guidance in the Kopp memo. Therefore, the NRC staff finds it acceptable.

3.1.4.3 Spent Fuel Characterization

Characterization of fresh fuel is based primarily on Uranium-235 (U-235) enrichment, fuel rod gadolinia content and distribution, and various manufacturing tolerances. The manufacturing tolerances are typically manifested as uncertainties, as discussed in Sections 3.1.3.3 and 3.1.4.2 of this safety evaluation, or are bounded by values used in the analysis. These tolerances and bounding values would also carry through to the spent nuclear fuel; common industry practice has been to treat the uncertainties as unaffected by the fuel depletion. The characterization of spent nuclear fuel is more complex. Its characterization is based on the specifics of its initial conditions and its operational history in the reactor. That characterization has three main areas: a burnup uncertainty, the axial and radial apportionment of the burnup, and the core operation that achieved that burnup.

EOI used a peak reactivity method. This method is commonly used for BWR fuel as the initial fuel rod gadolinia content makes a fresh fuel assembly less reactive than at a future burnup where most of the gadolinia has been consumed. The general method involves depleting the limiting lattice design to its peak reactivity.

EOI set a depletion uncertainty. The submittal does not describe how the value was determined. However, the value is consistent with values used in similar analysis indicating that the method of determination is consistent with ISG DSS-ISG-2010-01. When considered in concert with the substantial margin to the regulatory limit, the NRC staff accepts the depletion uncertainty value used by EOI as an approximation.

EOI did not explicitly address the axial and radial apportionment of the burnup. This is acceptable as BWR peak reactivity does not reach a fuel assembly burnup where the axial and radial apportionment become significant.

The regulation at 10 CFR 50.68(b)(4) states, in part, "...fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water." To meet this requirement, applicants must determine the impact of operation in the reactor on the post-irradiation reactivity of the fuel assemblies. EOI performed an evaluation of a range of reactor operating parameters and set a bias to account for the reactivity impact. Using biases in this manner rather than performing the analysis at the limiting parameters should produce similar results. However, performing the analysis at the limiting parameters may reveal synergistic effects the use of biases misses. The NRC staff did not ask EOI to consider whether use of biases would be identical to performing the analysis at the limiting parameters, rather the NRC staff relied upon the substantial margin to the regulatory limit to accept this method for this specific analysis.

3.1.5 Analysis of Abnormal Conditions

Licensees must meet the regulatory requirements for maintaining subcritical margin in the SFP even during abnormal or accident conditions. EOI considered the following abnormal conditions:

- SFP temperature exceeding the normal range
- Dropped fuel assembly
- Mislocated fuel assembly (fuel assembly positioned outside the storage rack)
- Rack movement

This is a suite of abnormal/accident conditions consistent with a BWR peak reactivity analysis. EOI's analysis determined the mislocated fuel assembly is the limiting accident. EOI included a bias should this occur. The NRC staff found this acceptable for this analysis.

3.1.6 Technical Summary

The licensee's analysis states the as analyzed SFP k_{eff} is **[[** **]]** at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. That included a Δk_{eff} value of **[[** **]]** for the NRC to use to address possible issues in the analysis. The maximum cold, uncontrolled peak in-core k_∞ value of 1.28, which is referenced in the licensee's TSs, is the nominal k_{eff} of the fuel assembly used in the analysis. The licensee's analysis indicates there is significant margin to the regulatory limit, but not to the k_∞ that is referenced in the TSs. Therefore, the licensee's analysis indicates there is significant margin to the regulatory limit, but not to the k_∞ value referenced in the TSs. The NRC staff took this into account during its review by using engineering judgment to determine that small changes to the licensee's analysis would not change the NRC staff's overall conclusion. Thus, NRC staff concluded that there is reasonable assurance that the River Bend SFP analysis demonstrates compliance with the applicable regulatory requirements in 10 CFR 50.68.

3.2 Material Characteristics and Compatibility

3.2.1 Background

The credited NAM installed in SFP storage racks ensures that the k_{eff} does not exceed the values and assumptions used in the criticality analysis of record (AOR) and other licensing basis documents. The AOR is the basis, in part, for demonstrating compliance with plant TSs and with applicable NRC regulations. Degradation or deformation of the credited NAM may reduce safety margin and potentially challenge the subcriticality requirement. NAMs utilized in SFP racks exposed to treated water or treated borated water may be susceptible to reduction of neutron absorbing capacity, changes in dimension that increase k_{eff}, and loss of material. A monitoring program is implemented to ensure that degradation of the NAM used in SFPs, which could compromise the ability of the NAM to perform its safety function as assumed in the AOR, will be detected.

3.2.2 Evaluation of the NAM Monitoring Program

River Bend currently credits Boraflex NAM in the NCS analysis for the SFP. Due to degradation of the Boraflex material, EOI's proposed amendment would allow the crediting of NETCO-SNAP-IN® neutron absorbing rack inserts made from the boron-carbide metal matrix material BORALCAN. These inserts have previously been reviewed and approved by the NRC staff for installation in other similar units.

The proposed license amendment also includes the addition of new TS 5.5.15 to TS Section 5.5 to incorporate a program into the TSs to monitor the condition of the neutron absorber inserts used in the SFP storage racks to ensure they will continue to perform their design function. This change is discussed in Section 3.3.2 of this safety evaluation. The new TS 5.5.15 is consistent with Technical Specifications Task Force (TSTF) traveler TSTF-557, Revision 1, "Spent Fuel Storage Rack Neutron Absorber Monitoring Program," which has been reviewed and approved by the NRC staff in a letter dated January 15, 2019 (ADAMS Accession No. ML19007A225). If adopted by a licensee and approved by the NRC, the TSTF-557 traveler revises TS 5.5 by adding a new program titled, "Spent Fuel Storage Rack Neutron Absorber Monitoring Program."

The TS Section 5.5 program imposes a requirement to have a licensee-controlled program that is in accordance with Nuclear Energy Institute (NEI) 16-03-A, "Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools," Revision 0, dated May 2017 (ADAMS Accession No. ML17263A133). In the NRC's Final Safety Evaluation dated March 3, 2017 (ADAMS Accession No. ML16354A486), the NRC approved NEI 16-03-A and accepted the document for referencing in licensing applications for nuclear power plants. The purpose of a NAM monitoring program is to verify that the NAM installed in SFPs continues to perform its safety function (i.e., criticality control) as assumed in the AOR. The guidance provided in NEI 16-03-A for a NAM monitoring program, relies on periodic inspection, testing, monitoring, and analysis of the NAM to ensure that the required subcriticality margin is maintained in accordance with

10 CFR 50.68 requirements. To accomplish this purpose, the guidance document states that a monitoring program must be capable of identifying unanticipated changes in the absorber material and determining whether anticipated changes can be verified. The guidance recommends a combination of coupon testing, in situ measurement, and SFP water chemistry monitoring as a means to monitor potential changes in characteristics of the NAM. The NRC staff reviewed the proposed guidance for what constitutes an acceptable monitoring program and its ability to ensure that potential degradation of SFP NAM will be detected, monitored, and mitigated. The NRC staff determined that an appropriate combination of the three methods listed above (coupon testing, in situ measurement, and SFP water chemistry monitoring) can comprise an effective NAM monitoring program.

3.2.3 Technical Summary

Based on its previous approval of TSTF-557 and NEI 16-03-A, the NRC staff determined that a NAM monitoring program that meets the provisions in NEI 16-03-A will allow EOI to reasonably ensure that the ability of the NAM to perform its safety function, as assumed in the AOR, is maintained; thus, demonstrating compliance with the subcriticality requirements of 10 CFR 50.68. The NRC staff concluded that implementation of such a monitoring program into the TSs, as described in TSTF-557, meets the regulatory requirements and provides reasonable assurance that plants that adopt these TSs will have the requisite requirements and controls to operate safely.

3.3 <u>Evaluation of TS Changes</u>

EOI requested NRC approval to use a new NCS analysis that credits the use of the NETCO-SNAP-IN® rack inserts and does not credit Boraflex. EOI also proposed changes to River Bend TS 4.3.1 and TS 5.5 as described below.

3.3.1 Changes to TS 4.3.1.1

EOI proposed to revise River Bend TS 4.3.1.1 to identify the neutron absorber inserts as design features of the SFP storage racks and to add two fuel-related parameters used in the NCS analysis crediting the NETCO-SNAP-IN® rack inserts. TS 4.3.1.1 will read as follows (added text shown in bold):

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. **Fuel assemblies having a maximum k-infinity of 1.28 in the normal reactor core configuration at cold conditions and a maximum average U-235 enrichment of 4.9 weight percent**;

b. $K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the USAR;

c. A nominal fuel assembly center to center storage spacing of 7 inches within rows and 12.25 inches between rows in the low density storage racks in the upper containment pool; and

d. A nominal fuel assembly center to center storage spacing of 6.28 inches within a rack and 8.5 inches between cell centers of adjacent racks, **with a neutron absorber insert within the storage cells,** in the high density storage racks in the spent fuel storage facility in the Fuel Building.

The NRC finds the proposed changes to TS 4.3.1.1 are consistent with EOI's criticality analysis with the NETCO-SNAP-IN® rack inserts installed. Therefore, the NRC staff finds that the proposed TS changes are acceptable in accordance with 10 CFR 50.36(c)(4).

3.3.2 Change to TS 5.5

In addition, EOI proposed to add Section 5.5.15, "Spent Fuel Storage Rack Neutron Absorber Monitoring Program," to River Bend TS 5.5, which incorporates a program into the TSs to monitor the condition of the neutron absorber inserts used in the SFP storage racks to ensure they will continue to perform their design function. The new section will read as follows:

5.5.15 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

This program provides controls for monitoring the condition of the neutron absorber inserts used in the high density storage racks in the spent fuel storage facility in the Fuel Building to verify the Boron-10 areal density is consistent with the assumptions in the spent fuel pool criticality analysis. The program shall be in accordance with NEI 16-03-A, "Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools," Revision 0, May 2017.

The NRC finds the proposed change to TS 5.5 is consistent with TSTF-557, Revision 1, and EOI's criticality analysis with the NETCO-SNAP-IN® rack inserts installed. In addition, the NRC staff finds that the new TS 5.5.15 provides for monitoring the condition of the neutron absorber inserts. Therefore, the NRC staff finds that the proposed TS change is acceptable in accordance with 10 CFR 50.36(c)(5).

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, published in the *Federal Register* on February 5, 2019 (84 FR 1805), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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: Kent Wood
Matthew Yoder

Date: December 31, 2019

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT NO. 201 RE: CHANGE TO THE NEUTRON ABSORBING MATERIAL CREDITED IN SPENT FUEL POOL FOR CRITICALITY CONTROL (EPID L-2018-LLA-0298) DATED DECEMBER 31, 2019

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