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

December 21, 2018

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 RE: RELOCATION OF THE REACTOR CORE ISOLATION COOLING INJECTION POINT TO THE 'A' FEEDWATER LINE (EPID L-2018-LLA-0029)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 194 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1 (RBS). The amendment consists of changes to the RBS Updated Safety Analysis Report (USAR) in response to your application dated January 29, 2018, as supplemented by letters dated June 21, August 15, and November 13, 2018.

The amendment revises the USAR to reflect the relocation of the reactor core isolation cooling injection point from the reactor vessel head spray nozzle to the 'A' Feedwater line via the 'A' Residual Heat Removal shutdown cooling return line.

A copy of the related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Lisa M. Regner, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 194 to NPF-47
2. Safety Evaluation

cc w/enclosures: 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 FACILITY OPERATING LICENSE

Amendment No. 194
License No. NPF-47

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

   A. The application for amendment by Entergy Operations, Inc. (EOI), dated January 29, 2018, as supplemented by letters dated June 21, August 15, and November 13, 2018, 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, by Amendment No. 194, Facility Operating License No. NPF-47 is hereby amended to authorize revision to the River Bend Station, Unit 1, Updated Safety Analysis Report (USAR), as set forth in the application dated January 29, 2018, as supplemented by letters dated June 21, August 15, and November 13, 2018, and evaluated in the NRC staff's evaluation enclosed with this amendment.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance. The USAR changes shall be implemented in the next periodic update to the USAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: December 21, 2018

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 194 TO

FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By application dated January 29, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18029A187), as supplemented by letters dated June 21, August 15, and November 13, 2018 (ADAMS Accession Nos. ML18172A142, ML18228A619, and ML18317A392, respectively), Entergy Operations, Inc. (Entergy or the licensee) requested approval of a revision to the Updated Safety Analysis Report (USAR) for Facility Operating License No. NPF-47 for River Bend Station, Unit 1 (RBS).

The proposed changes would revise the RBS USAR to reflect the relocation of the reactor core isolation cooling (RCIC) injection point from the reactor vessel head spray nozzle to the 'A' Feedwater line via the 'A' Residual Heat Removal (RHR) shutdown cooling return line.

The supplemental letters dated June 21, August 15, and November 13, 2018, 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 May 22, 2018 (83 FR 23732).

2.0 REGULATORY EVALUATION

2.1 Background

On July 3, 1999, the licensee implemented a design change to the RCIC at RBS. The portion of the design change relevant to this amendment request is the relocation of the RCIC injection point from the reactor vessel head spray nozzle to the 'A' Feedwater line via the 'A' RHR shutdown cooling return line. The change was originally evaluated by the licensee under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59, "Changes, tests, and experiments."

In October 2015, during the 10 CFR 50.59 and Permanent Plant Modification Inspection by the NRC, the inspector identified a deficiency in the 10 CFR 50.59 evaluation performed in 1998 to support the relocation of the RCIC injection point from the reactor vessel head spray nozzle to the 'A' Feedwater line via the 'A' RHR shutdown cooling return line. Specifically, the inspector found that this plant design change had increased the probability of a loss of feedwater heating (LOFH) accident by more than a minimal amount and was made without requesting prior NRC approval in violation of 10 CFR 50.59 requirements. Therefore, the NRC issued to the licensee a non-cited violation of 10 CFR 50.59 for failure to obtain a license amendment. The NRC inspection report dated June 9, 2017 (ADAMS Accession No. ML17160A401), stated that, because the licensee had failed to restore compliance (i.e., obtain a license amendment pursuant to 10 CFR 50.90) within a reasonable amount of time after the violation was initially identified, the violation had become a cited violation.

Subsequently, the licensee requested an amendment to relocate the RCIC injection point from the reactor vessel head spray nozzle to the 'A' Feedwater line via the 'A' RHR shutdown cooling return line. In its application dated January 29, 2018, the licensee provided that there were two purposes for the reroute:

1. Based on a May 5, 1987, General Electric (GE) (now, General Electric Hitachi) Potentially Reportable Condition (PRC) related to RCIC injection through the reactor head spray, it was reported that the flow injected through the reactor head spray nozzle by the RCIC system can induce water level measurement errors. The cause of the level errors was determined to be water from the RCIC injection being drawn into the line from the vessel to the condensing chamber resulting in an indicated vessel level higher than the actual vessel level. RBS experienced this phenomenon on one of the four narrow range level instruments during the startup test program when a RCIC injection test was performed at a reactor pressure of 150 pounds per square inch gauge. In the evaluation of the GE PRC, RBS Engineering Analysis evaluated several corrective actions for this phenomenon and recommended that the RCIC injection point be rerouted to the feedwater line.

2. To eliminate problems associated with the inboard and outboard containment isolation check valves (E51-AOVF065 and E51-AOVF066). These check valves are containment isolation valves needed for RCIC injection to reactor vessel head spray nozzle and are required to undergo a local leak rate test (LLRT) every refueling outage. The check valves had never passed an as-found LLRT and have been reworked to correct the condition during each refueling outage.

The licensee further stated in the application that an additional benefit of rerouting the RCIC injection point is that the main turbine trip that is initiated when RCIC injects to the vessel would no longer be required. In the previous design, the main turbine was tripped by automatic protective signals on a RCIC initiation to protect the turbine from damage. The trip was to prevent damage to the turbine by RCIC system water that could become entrained in the steam from the reactor to the turbine. With the RCIC system injecting into the feedwater line, carryover from this source was no longer a concern and the main turbine trip was deleted. This eliminated the possibility of a turbine trip and reactor scram should a spurious initiation of RCIC occur.

2.2 Proposed Changes

The licensee requested NRC approval of changes to the RBS USAR and Technical Requirements Manual (TRM) describing the relocation of the RCIC injection point from the reactor vessel head spray nozzle to the 'A' Feedwater line via the 'A' RHR shutdown cooling return line. The licensee also provided changes to the TS Bases, a licensee-controlled document. Pursuant to 10 CFR 50.36(a)(1), TS Bases are not part of the technical specifications.

2.3 Regulatory Requirements

The regulatory requirements and guidance documents that the NRC staff considered in its review of the proposed amendment included the following:

- 10 CFR Part 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," GDC 10, "Reactor design," which establishes requirements that the reactor core and associated coolant, control, and protection system 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 (AOOs). The LOFH and spurious RCIC injection events are analyzed AOOs in the RBS USAR.

- 10 CFR 50.55a, "Codes and standards," which, in part, establishes the standards required for piping systems, pipe supports, and components. In particular, pursuant to 10 CFR 50.55a(c)(1), the reactor coolant pressure boundary components must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code. Additionally, pursuant to 10 CFR 50.55a(d)(1), Quality Group B components must meet the requirements for Class 2 components in Section III of the ASME BPV Code.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP). Relevant sections used for the review of the proposed amendment are as follows:

    - Section 5.4.6, "Reactor Core Isolation Cooling System (BWR [Boiling-Water Reactor])," Revision 4, dated March 2007 (ADAMS Accession No. ML070540102).

    - Section 3.12, "ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports," Revision 1, dated April 2014 (ADAMS Accession No. ML14042A513).

    - Branch Technical Position (BTP) 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Revision 3, dated December 2016 (ADAMS Accession No. ML16085A315).

    - Section 15.0, "Introduction – Transient and Accident Analyses," Revision 3, dated March 2007 (ADAMS Accession No. ML070710376).

3.0 TECHNICAL EVALUATION

The RCIC system function is to respond to transient events by providing makeup coolant to the reactor. The RCIC system is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss-of-coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level. Under these conditions, the high pressure core spray (HPCS) and RCIC systems perform similar functions.

The licensee provided the following description of the RCIC system in its application dated January 29, 2018:

> The RCIC system consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core. Suction piping is provided from the Condensate Storage Tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from main steam line A, upstream of the inboard main steam line isolation valve.

In the application, the licensee stated that three systems were considered that could introduce a cold-water perturbation: RCIC, HPCS, and the feedwater system. The RBS analysis showed that improper startup of HPCS or RCIC events would produce no significant power transients. These events were considered as Chapter 15.1 events in the USAR, but the licensee analysis demonstrated that they were bounded by the LOFH accident and, therefore, they were not evaluated every refueling outage as Chapter 15.1 events due to the limited power transient.

The licensee stated that the inadvertent initiation of a RCIC event is bounded by the inadvertent HPCS startup event, as discussed in Chapter 15.5.1 of the USAR. Additionally, in the application, the licensee stated that changing the injection point of the RCIC does not increase the probability or consequences of an inadvertent RCIC injection because all of the affected piping, fittings, and valve pressure boundaries are qualified to the appropriate fluid transients and operational conditions in accordance with the design and licensing basis. The licensee further noted that no instrument setpoints were changed as a result of this modification and that the RCIC system modes of operation were not changed or affected by this modification. Based on the above, the licensee concluded that there was no change in the probability of an inadvertent initiation of RCIC by this modification, and that there was no impact to the probability or consequences of any previously evaluated accident.

The NRC inspection report dated June 9, 2017, stated, in part:

> the licensee's evaluation for this modification was inadequate because the licensee had failed to correctly evaluate that a spurious [RCIC] actuation injecting through the feedwater line would also result in the same characteristics, (and therefore increase the probability of occurrence) of another accident previously evaluated ([LOFH]) and that this would be more than a minimal increase in frequency.

The NRC staff stated that the spurious RCIC injection to feedwater line has similar thermal-hydraulic characteristics as that of the LOFH transient. Upon reviewing the license amendment request and related documentation, including the licensee's analysis, plant technical specifications, USAR, and TRM, the NRC staff determined, however, that the probability of occurrence of an LOFH event or its consequences will not increase as a result of the RCIC relocation because:

1. While the thermal-hydraulic characteristics for LOFH and spurious RCIC injection into the feedwater line are similar, these are two distinct and isolated events, independent and separate from one another, and that these two transients are caused by different failure mechanisms. As such, initiation of one of these events need not necessarily increase the probability of initiating the other event, or its consequences.

2. The licensee's analysis demonstrated that the acceptable fuel design limits in GDC 10 are not exceeded and that a spurious RCIC event is bounded by the LOFH.

Based on the above, the NRC staff concludes that neither the probability nor the consequences of the LOFH event, as currently analyzed in the RBS USAR, increases due to relocating the RCIC injection point from the reactor vessel head spray nozzle to the 'A' Feedwater line via the 'A' RHR shutdown cooling return line. The staff further concludes that the relocation continues to satisfy the fuel design limits of GDC 10. The staff, therefore, finds that the changes to the RBS USAR are acceptable.

The NRC staff also reviewed the licensee's submittals related to the piping stress analysis model, thermal transient curves and cycles, and the stress result summary. To evaluate the piping stress, fluid transient information, and cumulative usage factor (CUF), the licensee's piping analysis applied the ASME Code Case N-411 damping value to reduce the seismic and dynamic analysis stress results. The licensee determined that the high energy line break locations were not increased based on the lower ASME Code Section III, Piping Stress Equation, in accordance with Section B.1.(ii)(1)(c) of BTP 3-4 of the SRP. The stress calculation result demonstrated that all the stresses were within ASME Code Class 1 and 2 allowable limits. The NRC staff reviewed the summary of these analyses and found them acceptable since they continue to meet the ASME Code acceptance criteria as specified in 10 CFR 50.55a.

The licensee stated in the application that the RCIC piping reroute added a new transient since 40 degree Fahrenheit RCIC water would inject into the feedwater piping during an accident condition with no feedwater flowing. The licensee's feedwater piping fatigue evaluation included this new transient and the resultant maximum CUF is below the ASME Class 1 CUF allowable value of 1.0. Based on the above discussion, the NRC staff finds that the licensee's CUFs are acceptable and that there is no change to the postulated high energy line break locations.

The licensee's application, as supplemented by letters dated June 21, August 15, and November 13, 2018, was evaluated by the NRC staff. The NRC staff determined that the applicable regulatory requirements contained in GDC 10 and 10 CFR 50.55a will continue to be met since the acceptance criteria for GDC 10 and the ASME Code for Class 1 and 2 allowable dynamic stress limits will continue to be met. Thus, the NRC staff concludes that the change is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendment on December 11, 2018. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on May 22, 2018 (83 FR 23732). 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 Contributors: M. Razzaque
K. Hsu

Date: December 21, 2018

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: RELOCATION OF THE REACTOR CORE ISOLATION COOLING INJECTION POINT TO THE 'A' FEEDWATER LINE (EPID L-2018-LLA-0029) DATED DECEMBER 21, 2018

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ADAMS Accession No. ML18345A342 *by memo **by email

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