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

June 2, 2015

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

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: HEAVY LOAD MOVEMENT OVER FUEL ASSEMBLIES (TAC NO. MF2495)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 186 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1 (RBS). The amendment consists of changes to the Technical Requirements Manual (TRM) and the Updated Safety Analysis Report (USAR) in response to your application dated July 29, 2013, as supplemented by letters dated September 23, 2014, January 12, and March 30, 2015.

The amendment revises the RBS TRM, Section 3.9.14, "Crane Travel – Spent and New Fuel Storage, Transfer, and Upper Containment Fuel Pools," and the USAR Sections 9.1.2.2.2, "Fuel Building Fuel Storage," and 9.1.2.3.3, "Protection Features of Spent Fuel Storage Facilities," by adding a permanent exception to the current prohibition for travel of loads in excess of 1200 pounds over fuel assemblies in the spent fuel storage pool. This exception will allow the licensee to move the spent fuel pool (SFP) watertight gates, which separate the SFP from the cask and lower inclined fuel transfer pools, to perform maintenance and repairs on the gates and watertight seals.

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

Sincerely,

Alan Wang

Alan B. Wang, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 186 to NPF-47
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

ENTERGY GULF STATES LOUISIANA, LLC

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 186
License No. NPF-47

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated July 29, 2013, as supplemented by letters dated September 23, 2014, January 12, 2015 and March 30, 2015, 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.

Enclosure 1

2. Accordingly, by Amendment No. 186, the license is amended to authorize revision of the Updated Safety Analysis Report (USAR) and Technical Requirements Manual (TRM) as set forth in the application for amendment by Entergy Operations, Inc., dated July 29, 2013, as supplemented by letters dated September 23, 2014, January 12, 2015 and March 30, 2015. Entergy Operations, Inc., shall update the USAR and TRM to reflect the revised licensing basis authorized by this amendment in accordance with 10 CFR 50.71(e).

3. The license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance. The licensee will, prior to any rigging of the spent fuel pool gates, verify that no fuel assembly in the spent fuel pool, cask pool or fuel transfer pool has been part of a critical core within the preceding 14 days as described in the licensee's letter dated January 12, 2015, and the NRC staff's safety evaluation for this amendment. In addition, the licensee shall include the revised information in the River Bend Station, Unit 1 USAR in the next periodic update in accordance with 10 CFR 50.71(e), as described in the licensee's application dated July 29, 2013, as supplemented by letters dated September 23, 2014, January 12, 2015, and March 30, 2015 and the NRC staff's safety evaluation for this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Meena K.

Meena K. Khanna, Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: June 2, 2015

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 186 TO

FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By letter dated July 29, 2013, as supplemented by letters dated September 23, 2014, January 12, 2015, and March 30, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13214A334, ML14272A185, ML15027A422, and ML15096A500), Entergy Operations, Inc. (Entergy or the licensee), requested changes to the licensing basis documents for Facility Operating License Number NPF-47 for the River Bend Station, Unit 1 (RBS). The proposed changes will revise the RBS Technical Requirements Manual (TRM), Section 3.9.14, "Crane Travel – Spent and New Fuel Storage, Transfer, and Upper Containment Fuel Pools," by adding a permanent exception to the current prohibition for travel of loads in excess of 1200 pounds over fuel assemblies in the spent fuel storage pool. This exception will allow the licensee to move the spent fuel pool (SFP) watertight gates, which separate the SFP from the cask and lower inclined fuel transfer pools, to perform maintenance and repairs on the gates and watertight seals. The Updated Safety Analysis Report (USAR) Sections 9.1.2.2.2, "Fuel Building Fuel Storage," and 9.1.2.3.3, "Protection Features of Spent Fuel Storage Facilities," will also be changed to reflect the proposed exception. The licensee has determined that the load of the gate and rigging exceeds the load analyzed over spent fuel (1200 pounds) and further concludes that the requested change requires prior U.S. Nuclear Regulatory Commission (NRC) approval in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.59, "Changes, tests, and experiments."

The supplements dated September 23, 2014, January 12, 2015, and March 30, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 10, 2013 (78 FR 74181).

Enclosure 2

2.0 REGULATORY EVALUATION

The NRC requested licensees to address control of heavy load movements in 1980. The NRC staff provided regulatory guidelines to support this action in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," published July 1980 (ADAMS Accession No. ML070250180). Implementation of these guidelines assure safe handling of heavy loads in areas where a load drop could have an impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. Chapter 5, "Guidelines for Control of Heavy Loads," Section 5.1.1, "General," of NUREG-0612 provides: guidelines for reducing the likelihood of dropping heavy loads and criteria for establishing safe load paths; procedures for load handling operations; training of crane operators; design, testing, inspection, and maintenance of cranes and lifting devices; and analyses of the impact of heavy load drops.

The guidelines in Sections 5.1.2 through 5.1.5 of NUREG-0612 address alternatives to either further reduce the probability of a load-handling accident or mitigate the consequences of heavy load drops in specific areas of the facility. These alternatives include: using a single-failure-proof crane for increased handling system reliability, employing electrical interlocks and mechanical stops to restrict crane travel to safe areas, or performing load drop consequence analyses to assess the effect of dropped loads on plant safety and operations. Section 5.1.2, "Spent Fuel Pool Area – PWR" [pressurized-water reactor], addresses alternatives appropriate for the SFP area. These alternatives include analyses of heavy load drops to demonstrate that the evaluation criteria of Section 5.1 of NUREG-0612 are satisfied. These criteria specify that for any releases resulting from damage to spent fuel, doses produced are well within the 10 CFR Part 100, "Reactor Site Criteria," limits; damage to the fuel does not increase the effective neutron multiplication factor (k_{eff}) above 0.95; damage to the SFP will not result in leakage that could uncover the fuel; and damage does not result in a loss of essential safe shutdown functions. Appendix A to NUREG-0612 provides guidance for analyses of postulated load drops.

The regulatory requirements and guidance, which the NRC staff considered in its review of the license amendment request (LAR), with regard to the assessment of radiological impacts, are as follows:

10 CFR 50.67, "Accident source term" states, in part, that:

> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sievert] (25 rem) total effective dose equivalent (TEDE), (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE), and (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

Appendix A to 10 CFR Part 50, "General Design Criteria [GDC] for Nuclear Power Plants," Criterion 19, "Control room," states, in part, that:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem [0.05 Sv] whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, July 2000 (ADAMS Accession No. ML003716792) provides the methodology for analyzing the radiological consequences of several design basis accidents to show compliance with 10 CFR 50.67. RG 1.183 provides guidance to licensees on acceptable application of alternate source term (AST) (also known as the accident source term) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. ML003734190), provides guidance to the NRC staff for the review of alternative source term amendment requests. SRP 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in RG 1.183. The dose acceptance criteria for the fuel-handling accident (FHA) are a TEDE of 6.3 rem at the exclusion area boundary (EAB) for the maximum 2-hour period, and 6.3 rem at the outer boundary of the low population zone (LPZ) during the entire period of the postulated radioactive cloud passage. The NRC staff also considered relevant information in the RBS USAR.

License Amendment No. 132, "River Bend Station, Unit 1 - Issuance of Amendment Re: Full-Scope Application of Alternative Source Term Insights (TAC No. MB5021)," dated March 14, 2003 (ADAMS Accession No. ML030760746), used an AST methodology for analyzing the radiological consequences of four design-basis accidents using RG 1.183. The FHA was one of the design-basis-accidents analyzed.

The regulatory requirements from which the NRC staff based its acceptance are the reference values in 10 CFR 50.67, the accident specific guideline values in Regulatory Position 4.4 of RG 1.183, and Table 1 of SRP Section 15.0.1.

3.0 TECHNICAL EVALUATION

3.1 Background

The fuel building fuel storage area consists of three interconnected pools constructed of reinforced concrete lined with stainless steel. The spent fuel storage pool contains the fuel storage racks. Two adjacent pools, the cask pool and the inclined fuel transfer system pool, interface with the SFP through fuel transfer slots that may be separated from the SFP by use of watertight gates equipped with pneumatic seals. The RBS preventive maintenance program specifies pneumatic seal replacement on a 14-year frequency. Seal replacement requires removal of the gates from the pools. Each fuel pool gate with the necessary rigging for removal from the pool weighs approximately 2000 pounds.

The protection at RBS against accidental damage to spent fuel resulting from potential load drops consists of a combination of design features and administrative controls. The RBS USAR, Section 9.1.2.3.3, describes that the configuration of the spent fuel cask pool and cask handling crane prevents movement of heavy loads over stored spent fuel. The spent fuel cask handling crane travels in a north-south direction along the eastern end of the fuel building. This allows the crane to travel over the cask pit adjacent to the east wall of the SFP, while preventing movement of the crane over the SFP. The spent fuel cask handling crane main hook is fixed in position laterally, but the 15-ton spent fuel cask handling crane auxiliary hook operates from a trolley that allows movement perpendicular to the cask crane main hook between the runway rails. The RBS USAR, Section 9.1.2.2.2, describes a 15-ton SFP bridge crane that has been used for installation of the fuel storage racks in the SFP and remains available for movement of light loads over spent fuel under administrative control. The administrative control of loads was documented in RBS TRM 3.9.14, which prohibits travel of loads in excess of 1200 pounds from travel over fuel assemblies. The proposed amendment would modify the described administrative controls to permit handling of the SFP gates for periodic replacement of seals as a specific exception.

3.2 Description of the Proposed Changes

In the original LAR by letter dated July 29, 2013, the licensee proposed changes, which add an exception to the movement of loads in excess of 1200 pounds from travel over fuel assemblies in the spent fuel pool. The exception will allow for movement of the SFP gates for gate repair or seal replacement, which weigh more than 1200 pounds. The current TRM 3.9.14 states:

> TLCO 3.9.14: Loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent or new fuel storage, transfer or upper containment fuel pool racks and all loads shall be prohibited from travel over irradiated fuel when water level is < 23' over the irradiated fuel.

In the original LAR, dated July 29, 2013, the licensee proposed TRM 3.9.14 as follows:

> TLCO 3.9.14: Except for movement of spent fuel pool gates, loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent or new fuel storage, transfer or upper containment

> fuel pool racks, and all loads shall be prohibited from travel over irradiated fuel when water level is < 23' over the irradiated fuel.

The current TRM 3.9.14 Surveillance 3.9.14.1 states, "The fuel building crane loads shall be verified to weigh less than or equal to 1200 pounds." In its letter dated July 29, 2013, the licensee originally proposed that TRM 3.9.14 Surveillance 3.9.14.1 would be changed to state, "The fuel building crane loads shall be verified to weigh less than or equal to 1200 pounds, except for movement of spent fuel pool gates."

The licensee originally proposed in its LAR by letter dated July 29, 2013, to change RBS USAR Section 9.1.2.2.2 to state, in part, the following:

> ...Subsequent to the installation of the [high density spent fuel storage] racks, the crane will be utilized for transporting only light loads over stored spent fuel, except for movement of spent fuel pool gates. Administrative controls exist to prevent the transport of heavy loads other than fuel pool gates over stored spent fuel.

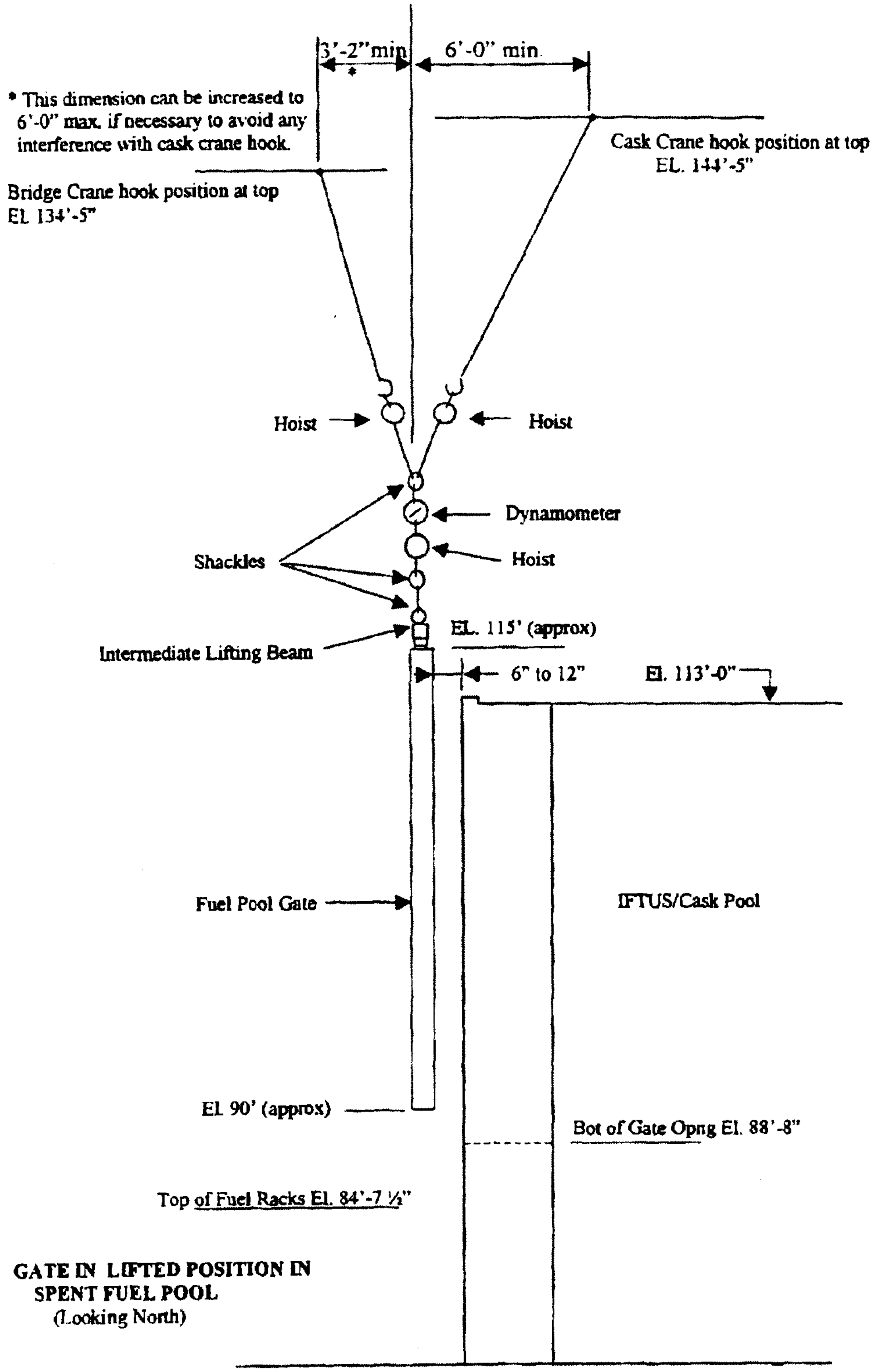
In addition, the licensee originally proposed, in its LAR dated July 29, 2013, to change RBS USAR Section 9.1.2.3.3 to state, in part, the following:

> ...The only heavy objects that will be moved in the vicinity of stored spent fuel are spent fuel pool gates, as required for repair or seal replacement. The load handling of the gates will be performed in accordance with the intent of NUREG-0612, NUREG-0554, RIS-2005-25, and RIS-2005-25 Supplement 1 guidelines for reducing the potential for an accidental load drop. The load associated with movement of spent fuel pool gates is approximately 2500 pounds, which accounts for the weight of a gate plus the rigging.

Note: These original proposed modifications to the licensing basis documents were later revised by the licensee after NRC review and submittal of Requests for Additional Information (RAIs) (see Section 3.4.2.2 of the SE below).

3.3 NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Evaluation

In Section 5.1 of Attachment 1 to the LAR by letter dated July 29, 2013, Entergy described the proposed gate movement process. The design of the fuel storage area precludes either the auxiliary spent fuel cask crane or the SFP bridge crane from positioning either crane hook directly over either SFP gate. The licensee determined from dry-run operation that the SFP bridge crane hook can be positioned as close as 3 feet, 2 inches to the west of the gates, which are located in the east wall of the SFP. The licensee similarly determined that the spent fuel cask handling crane auxiliary hook can be positioned as close as 6 feet to the east of the SFP gates. The hooks for each crane can be aligned with the gate positions in the north-south direction. The licensee provided the below rigging diagram, which shows the planned arrangement of the cranes and intermediate lifting components during the initial lift of the gate, as Attachment 4 to the LAR.



GATE IN LIFTED POSITION IN SPENT FUEL POOL
(Looking North)

The licensee also considered the potential use of an alternate rigging configuration. This configuration would include an additional lifting beam to span the distance between the crane hooks to allow a vertical lift by the crane hoists. This additional beam would be placed between the crane hooks and the hoists connected directly to the crane hooks in the above diagram. This additional lifting beam is designated, as the alternate lifting beam in the load drop analyses evaluated in Section 3.3.2 of this SE.

3.3.1 Load Handling System Design and Operation

The licensee described the proposed operation of the rigging components to transfer the gates from their installed location to the laydown area in Section 5 of Attachment 1 to the LAR. The licensee stated that all rigging equipment (slings, chains, hoists, shackles, etc.) would be selected to have a minimum rated capacity of 5 tons. With the crane hooks aligned with the centerline of the gate, the rigging personnel would establish the rigging configuration depicted in Attachment 4 to the LAR. In this rigging configuration, the lower hoist would primarily be used to lift the gate off its hinges, and the upper pair of hoists would be used to move the gate in the east-west direction between the two cranes, including movement of the gate from the SFP to the cask pool through the cask pool gate opening. Simultaneous motion of the two cranes in the north-south direction would be used to move the gate in that direction. As the licensee described in Attachment 1 to the LAR, this rigging configuration would be used while maneuvering the gate to the upper shelf in the cask pool, where the gate would be rested on the shelf while changing the rigging configuration. The second rigging configuration would have the auxiliary hoist of the spent fuel cask handling crane directly connected to the gate lifting beam without the intermediate hoists, which would provide the hoist headroom necessary to lift the gate out of the cask pool and place it in the laydown area. The process would be reversed to return the gate to its normal position in the SFP.

The licensee addressed the NUREG-0612 guidelines that are intended to reduce the probability of a load drop with significant consequences. In Attachment 5 to the LAR, the licensee provided the following commitments related to movement of the SFP gates:

- Work instructions for gate maintenance will include the proposed load lift rigging plan and the safe load path for movement.

- Crane operator training will be conducted prior to performing the move.

- Lifting and rigging equipment will be inspected prior to the movement.

- Corresponding sections of the RBS USAR will be revised to be consistent with the exception and to state that the provisions [of the following documents will be met:]

    - NUREG-0612;

    - NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," published May 1979 (ADAMS Accession No. ML110450636);

- o Regulatory Issue Summary (RIS) 2005-025, "Clarification of NRC Guidelines for Control of Heavy Loads," dated October 31, 2005 (ADAMS Accession No. ML052340485); and

- o RIS 2005-025, Supplement 1, "Clarification of NRC Guidelines for Control of Heavy Loads," dated May 29, 2007 (ADAMS Accession No. ML071210434).

The guidelines of NUREG-0612 include guidance for handling of heavy loads near spent fuel in the SFP. Section 5.1.4, "Reactor Building – BWR [Boiling-Water Reactor]," of NUREG-0612, is applicable to load handling over SFPs at BWRs such as RBS, and specifies that load handling practices for movement over a BWR SFP should satisfy the general guidelines of Section 5.1.1 of NUREG-0612 and either the load handling system should satisfy the single-failure-proof guidelines of Section 5.1.6 of NUREG-0612 or the effects of a postulated load drop should be evaluated against the criteria in Section 5.1 of NUREG-0612. The above commitments relating to work instructions, safe load paths, crane operator training, and inspection of lifting and rigging equipment satisfy aspects of the NUREG-0612 general guidelines related specifically to the gate movement. Those commitments, combined with the proposed revision of the RBS USAR to indicate the provisions of NUREG-0612, will be satisfied, provide reasonable assurance that the general guidelines of NUREG-0612 related to minimizing the potential of a damaging load drop will be satisfied.

The NRC staff evaluated the conformance of the proposed handling system design with the single-failure-proof handling system guidelines provided in Section 5.1.6 of NUREG-0612, which includes use of a single-failure-proof crane, conforming to the guidelines of NUREG-0554. The licensee proposed to classify the rigging configuration as compliant with the guidance of NUREG-0554. However, the NRC staff questioned the licensee's ability to comply with the NUREG-0554 guidelines in the proposed rigging configuration, particularly with respect to the provision of a dual balanced reeving system specified in Section 4.1, "Reeving System," of NUREG-0554. In response to the NRC staff's RAI, the licensee supplemented its LAR by letter dated September 23, 2014. This response provided an evaluation of the consequences of a postulated drop of a gate in the SFP against the criteria provided in Section 5.1 of NUREG-0612 and supporting calculations. Accordingly, the NRC staff's review focused on the evaluation of the consequences of a load drop because the proposed load handling configuration did not conform to the guidelines of Section 5.1.6 of NUREG-0612 for a single-failure-proof handling system.

3.3.2 Load Drop Analysis

In the supplement provided by letter dated September 23, 2014, the licensee described the results of the load drop analysis. Consistent with the criteria in Section 5.1 of NUREG-0612, the licensee evaluated the consequences of a postulated gate drop against the following criteria:

- Releases of radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem [roentgen equivalent man] thyroid, 25 rem whole body (analyses should show that doses are equal to or less than 1/4 of Part 100 limits).

- Damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of the fuel such that k_{eff} is larger than 0.95.

- Damage to the reactor vessel or the spent fuel pool based on calculations of damage following accidental dropping of a postulated heavy load is limited so as not to result in water leakage that could uncover the fuel, (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost is borated).

- Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in loss of required safe shutdown functions.

In Attachment 2 to the letter dated September 23, 2014, the licensee provided Calculation G.13.1B.2. 7 -116, Revision 0, Load Drop Calculation for SFP Gates (FNS-GATE1 and FNS-GATE2), which provided analyses addressing the above criteria. The NRC staff's evaluation of the assumptions, methodologies, and results used in these analyses is provided below.

3.3.2.1 Analysis of Potential Radioactive Material Release

Through its analyses, the licensee determined the number of damaged fuel pins resulting from an accidental drop of one SFP gate and associated rigging equipment. The analysis considered impacts of the SFP gate, an intermediate lifting beam, and an alternate lifting beam. For each object, the licensee determined the impact energy assuming a drop from the maximum expected height of the equipment above the stored fuel while the gate is in the SFP, which were 6 feet for the gate, 31 feet for the intermediate lifting beam, and 39 feet for the alternate lifting beam. The licensee evaluated the expected fuel damage states for several cases that considered different configurations for the dropped equipment and impacted fuel. All evaluated cases neglected fluid drag by assuming all potential energy was absorbed by the impacted fuel assemblies and the dropped equipment, with the distribution of energy based on a mass ratio. From this evaluation, the licensee determined that 209 fuel pins could be damaged by a drop of the gate and associated rigging. To provide margin to account for potential future changes in the rigging configuration, the licensee assumed up to 266 that fuel rods could be damaged by an accidental drop of the gate and associated rigging. This damage state was used to determine the radiological consequences of this postulated event.

The NRC staff evaluated the assumed accident configuration for the fuel handling accident. The staff finds the assumed scope of equipment, the equipment drop heights, and the evaluated equipment impact configurations to be conservative for handling operations over the SFP and consistent with the guidance provided in Appendix A to NUREG-0612. Therefore, the NRC staff concludes that the assumed accident configuration is acceptable for the SFP gate handling accident.

The NRC staff also evaluated the postulated number of damaged pins for the above accident configuration. The licensee used a fuel damage methodology consistent with the methodology described in the approved GESTARII licensing topical report (General Electric Standard Application for Reactor Fuel, June 2000). Therefore, the NRC staff concludes that the number of postulated damaged pins (266) represents a conservative estimate of fuel damage for the postulated gate handling accident in the SFP. Evaluation of the calculated release from the damaged fuel and the associated consequences are provided in Section 3.4 of this SE.

3.3.2.2 Analysis of Potential Change in Fuel Configuration

In Attachment 2 to the supplement by letter dated September 23, 2014, the licensee evaluated the ability of the rack structure to withstand the impact of the SFP gate and lifting beams. The first part of this evaluation considered the strain energy necessary to deform the plates forming the sides of rack cells to the elevation of the neutron absorbers, which are located more than 16 inches below the top of the rack. The licensee evaluated the energy of the individual components based on a postulated drop from the maximum expected height, considering the drop height through air and the remaining drop through water to the top of the racks. The NRC staff has reviewed the evaluation and concluded that the SFP gate and the rigging components would not have sufficient energy to deform the rack structure to the extent that rack neutron poisons would be affected. The licensee also evaluated the impact energy of the SFP gate and rigging components relative to the design rated impact energy for the racks. The licensee stated and the NRC confirmed, that based on geometry and weight considerations, the impact energy that could result from a drop during SFP gate handling activities is bounded by the impact energy evaluated in the existing analysis for the fuel bundle drop accident. The licensee's fuel bundle drop accident analysis determined that k_{eff} remains acceptable following this event. As the impact energy of the SFP gate handling accident is bounded by the fuel bundle drop accident, k_{eff} is also bounded and remains acceptable for the SFP gate handling activities. Therefore, the NRC staff concludes that since the deformation of the storage racks and the impact energies are within acceptable ranges, the required margin to criticality would be maintained following an accidental drop during SFP gate handling activities.

3.3.2.3 Analysis of Potential Spent Fuel Pool Damage

The licensee also provided an analysis of the effects of a SFP gate handling event on the integrity of the SFP liner in Attachment 2 to the supplement by letter dated September 23, 2014. The stainless steel liner serves as a leak-tight membrane resistant to damage. The stainless steel liner is a Seismic Category I and Safety Class 3 structure formed by welding 3/16th inch thick plates around the periphery of thicker stainless steel plates embedded in the reinforced concrete structure of the SFP. Each weld is provided with a leak collection channel that routes leakage to normally isolated leak detection lines. The details of the SFP liner construction are shown in Figures 9.1-33 through 9.1-35 of the RBS USAR.

Similar to the rack damage analysis, the licensee determined the drop height of the components through air and through water from the maximum expected handling height. However, the licensee considered the greater lift height necessary to lift the gate out of the cask pool, which contains no fuel storage racks. The drop heights were used to determine the final impact velocity considering the drag of the water. The NRC staff reviewed the calculation methodology and the results and concludes they are acceptable.

Section 9.1.4.3.2, "Storage Cask Component Drop Analyses," of the RBS USAR describes analyses of the consequence of fuel storage cask component drop scenarios evaluated to support handling the 125 ton storage cask system components with the cask handling crane. These analyses demonstrate that the integrity of the fuel building reinforced concrete structure would be maintained following heavy load drops. Therefore, the licensee focused the load drop analysis on the integrity of the steel liner.

Attachment 2 to the letter dated September 23, 2014, includes a description of the methodology used to determine the penetration potential of the gate and associated handling equipment given the strike velocity. The methodology used was from Topical Report BC-TOP-9A, "Topical Report, Design of Structures for Missile Impact," Revision 2, September 1974 (ADAMS Accession No. ML14093A217). The methodology was based on a comparison of the missile energy against the available strain energy that could be absorbed without rupture by a steel plate anchored far from the impact location. The licensee determined that the liner had adequate strain energy to absorb the energy of a dropped gate and associated rigging. However, while this methodology was appropriate for its intended application to missile barriers, the NRC staff found the methodology incomplete for evaluation of liner penetration given the potential location of supports under the liner plate. The NRC Topical Report Evaluation, which was included within the approved topical report, states that the evaluation of punching shear effects due to impact was not included as part of the structural response considerations in the report, but should be addressed in the safety analysis. Accordingly, the NRC staff requested that the licensee describe the configuration of the SFP floor liner and address the susceptibility liner to punching shear effects. Subsequently, on March 4, 2015, the NRC staff participated in a teleconference with Entergy to clarify the requested information and the acceptance criteria.

By letter dated March 30, 2015, the licensee responded to the RAI by e-mail dated December 16, 2015 (ADAMS Accession No. ML14352A011). In Attachment 1 to this letter, the licensee provided a supplemental analysis focused on the construction of the pool liner and potential for leakage. The licensee determined from the construction details that the SFP liner plates are in direct contact with the SFP concrete floor. Because the liner plates are directly backed by the reinforced concrete floor, the floor would not be penetrated by the impact from the gate and associated rigging. Only the leakage monitoring channel covers, which form a small portion of the total area of the liner system, are not in direct contact with the reinforced concrete floor. In the event of a load drop impacting the pool floor, a breach of the leakage monitoring channel covers would not result in external liner leakage because the weld between the liner plate and the embedded mounting plate would be protected by its location under the channel cover. The licensee stated that the leakage collection system is normally isolated, so leakage into the leakage monitoring channels would be contained. Nevertheless, the licensee evaluated the potential maximum rate of leakage into the leakage collection system by assuming the gate drop creates a 1/32nd inch crack in the leakage collection system cover with a length equal to the width of the SFP gate. The analysis assumed free flow though the leakage collection system to a collection volume at atmospheric pressure and no resistance to flow other than the size of the crack. Based on these assumptions, the licensee determined a conservative estimate of leakage equal to 178 gallons per minute.

Section 9.1.3, "Fuel Pool Cooling and Cleanup System," of the RBS USAR includes descriptions of several sources of makeup water for the SFP. The Seismic Category I, Safety Class 3 reactor plant component cooling water and standby service water systems provide redundant, safety-related supplies of standby makeup water to the SFP, in addition to the normal source of makeup water from the condensate storage tank. The RBS USAR states that these connections ensure the availability of makeup water to the SFP in all emergency or accident conditions. Although the safety-related inventory of makeup water from the standby service water system is limited, the movement of the gates is so infrequent (i.e., a few times every 15 years) that consideration of an accident (e.g., a seismic event) in combination with gate movement is not credible with respect to defining makeup requirements. Furthermore, the design of the SFP liner system and leakage collection channels, as described above, would not allow a significant loss of water from the SFP as a result of a SFP gate drop because the collection system is normally isolated and the integrity of the reinforced concrete structure would not be affected by the drop. Therefore, the NRC staff concludes that available makeup would be adequate to compensate for any leakage that may result from a SFP gate handling accident.

3.3.2.4 Analysis of Potential Damage to Safe Shutdown Systems

The licensee classified SFP forced cooling as a safe shutdown function and provided an analysis of the effects of a SFP gate handling accident on the integrity of the SFP cooling system piping in Attachment 2 to the supplement by letter dated September 23, 2014. Two cooling system return lines enter the pool approximately 5 feet apart through the north wall, turn downward, and connect to a common header to distribute the return flow along the north side of the pool. Similarly, two cooling system suction lines draw SFP water from a common header and penetrate the south wall of the SFP. The vertical sections of these pipes travel from an elevation about 1 foot below the top of the fuel storage racks to an elevation about 25 feet above the top of the racks and are within a horizontal distance of about 11 to 18 feet from the nearest gate opening. Since the gates are over 24 feet tall, the licensee determined that the gates could strike this piping assuming the end of the gate first struck the top of the racks near the gate opening and the top of the gate tipped toward the suction or return piping. Based on analysis of this impact scenario, the licensee determined that the SFP cooling system piping would not be significantly affected by the effects of a SFP gate handling accident.

The NRC staff evaluated the licensee's analysis of potential damage to safe shutdown systems. The staff concluded that the configuration of the SFP cooling system piping would preclude a complete loss of the cooling function as a result of a SFP gate handling accident. A gate could not credibly damage both suction or return lines in a manner that would interrupt flow. No other equipment essential to performance of safe shutdown functions is present in the vicinity of the SFP gate handling activities. Therefore, the staff concludes that a SFP gate handling accident would not cause a loss of any safe shutdown function.

3.4 Accident Dose Assessment

3.4.1 Accident Source Term

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, "Accident source term," that provided a mechanism for licensed power reactors to voluntarily replace the traditional accident source term used in its design-basis-accident (DBA) analyses with ASTs. Regulatory guidance for the implementation of these ASTs is provided in RG 1.183. Under 10 CFR 50.67, a licensee seeking to use the AST is required to apply for a license amendment and the application is required to contain an evaluation of the consequences of DBAs.

RBS has previously requested and the NRC has approved use of an AST at RBS by License Amendment 132. In License Amendment 132, the FHA was postulated to occur as a consequence of a failure of the fuel assembly lifting mechanism, resulting in a drop of a raised fuel assembly onto stored fuel assemblies in either the spent fuel pool or the reactor core. The analysis was based on assumed damage to 150 General Electric 9 X 9 fuel rods that had experienced a power peaking factor of 2.0 and a fission product decay period of 24 hours. The analysis assumed that an instantaneous release of all noble gases and iodines from the fuel rod gaps occurred and bubbled up through the water above the fuel assemblies. The water provided an overall decontamination factor of 200 for iodines, consistent with the guidelines provided in RG 1.183. All of the noble gases and the remaining iodine reached the fuel-building atmosphere, and were released to the environment in a 2-hour time period.

In the proposed license amendment, RBS provides a similar load drop analysis to support the proposed changes to the TRM and USAR. The RBS load drop analysis assumes a SFP gate and its associated rigging (2375 pounds) are dropped from a maximum allowable height of 6 feet onto stored fuel assemblies in the SFP.

3.4.2 Fuel Handling Accident Radiological Consequences

3.4.2.1 RBS Assessment

The licensee's SFP gate load drop analysis uses the same meteorology data, power level, and control room specifications that were used in the current licensing basis for the FHA. The SFP gates and the associated rigging are assumed to independently strike spent fuel bundles in order to maximize the postulated number of bundles struck by the objects, thereby, increasing potential fuel damage. The number of damaged fuel rods was then increased by 25 percent for conservatism, resulting in a postulated 266 damaged fuel rods. In addition, since the accident occurs in the spent fuel pool, the analysis assumes a fission product decay period of 14 days (336 hours) with no credit taken for the fuel building or control room charcoal adsorber filtration. In accordance with RG 1.183, the licensee assumed the following:

- The gap activity fractions in the fuel rods are: 0.08 for I-131, 0.10 for Kr-85, 0.05 for other noble gases, and 0.05 for other halogens.

- All of the gap activity in the damaged rods is assumed to be instantaneously released.

- Radionuclides considered include xenons, kryptons, halogens, cesiums, and rubidiums.

- Particulate radionuclides are assumed to be retained by the water in the fuel pool.

- Radioiodine released from the damaged fuel rods is assumed to be 99.85% in the form of elemental iodine and 0.15% in the form of organic iodine.

- The overall decontamination factor for iodine species is 200. This results in an iodine release to the environment composed of 57% elemental and 43% organic species.

- All radionuclide releases to the environment occur over a 2-hour period.

The RBS SFP gate load drop analysis determined a TEDE of 1.21 rem at the EAB, 0.160 rem at the LPZ, and 0.873 rem in the control room. RBS concluded that the doses from an accidental drop of a SFP gate are less than 25 percent of the 10 CFR 50.67 TEDE limits for an FHA.

3.4.2.2 NRC Staff's Assessment

The NRC staff reviewed the assumptions, inputs, and methods used by RBS to assess the radiological impacts of dropping a SFP gate and its associated rigging onto stored fuel assemblies in the SFP. The NRC staff's review finds that the RBS current licensing basis for the FHA analysis assumed 150 spent fuel rods were damaged from a drop of a channeled spent fuel bundle onto unchanneled spent fuel in the spent fuel racks and includes a decay time of 24 hours. The licensee's analysis for a spent fuel pool gate drop accident assumed 266 fuel rods were damaged after a minimum of a 14-day restriction prior to movement of the spent fuel gates.

The RBS licensing basis documents (TRM and USAR) prohibit moving heavy loads over stored spent fuel, but do not include a restriction on the minimum amount of time prior to heavy load movement. Therefore, the NRC through an RAI, staff asked the licensee to either provide a revised FHA analysis that accounts for a decay time of 24 hours and 266 damaged fuel rods, or to provide a proposed 14-day time restriction before movement of spent fuel gates over spent fuel assemblies.

In the response to RAI-2 by letter dated January 12, 2015, the licensee stated that the 14-day time restriction was an assumption in its load drop analysis and that it is appropriate that such a condition be made a prerequisite to the movement of spent fuel gates. Therefore, the licensee has made a regulatory commitment that prior to any rigging of the spent fuel pool gates, it will be verified that no fuel assembly in the pools has been part of a critical core within the preceding 14 days. In addition, the licensee has proposed a revised USAR Section 9.1.2.3.3 to reflect this regulatory commitment, as shown below.

In addition, the NRC staff asked the licensee to explain whether Entergy is requesting to move the spent fuel pool gates at any time or only for repair and seal replacement. If the spent fuel pool gates are going to be moved only for repair and seal replacement, then the licensee was asked to provide the updated TRM and USAR sections that reflect this change (similar to those in License Amendment No. 108, "River Bend Station, Unit 1 – Issuance of Amendment Re: Heavy Load Exception to Allow Movement of Spent Fuel Pool Gates to Replace Gate Seal (TAC No. MA7365)," dated January 13, 2000 (ADAMS Accession No. ML003674610) to RBS Facility Operating License Number NPF-47).

In the response to RAI-1 by letter dated January 12, 2015, the licensee stated that the only reasonably foreseeable causes for moving a gate are emergent repairs or seal replacement. As part of the licensee's response to RAI-1, the licensee provided a revised proposed TRM and USAR pages as follows:

The revised proposed TRM 3.9.14 states:

> Except for movement of the spent fuel pool gates for repair or seal replacement, loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent or new fuel storage, transfer or upper containment fuel pool racks and all loads shall be prohibited from travel over irradiated fuel when water level is < 23' over the irradiated fuel.

The revised proposed TRM 3.9.14 Surveillance 3.9.14.1 states, "The fuel building crane loads shall be verified to weigh less than or equal to 1200 pounds, except for movement of the spent fuel pool gates for repair or seal replacement."

The revised proposed RBS USAR Section 9.1.2.2.2 states, in part:

> ...Subsequent to the installation of the [high density spent fuel storage] racks, the crane will be utilized for transporting only light loads over stored spent fuel, except for the movement of the spent fuel pool gates for repair or seal replacement. Administrative controls exist to prevent the transport of heavy loads other than fuel pool gates over stored spent fuel.

The revised proposed RBS USAR Section 9.1.2.3.3 states, in part:

> ...The only heavy objects that will be moved in the vicinity of stored spent fuel are spent fuel pool gates, as required for repair or seal replacement. The load handling of the pool gates will be performed in accordance with the intent of NUREG-0612, NUREG-0554, RIS-2005-25, and RIS-2005-25 Supplement 1, for reducing the potential for an accidental load drop. The load associated with movement of pool gates is approximately 2500 pounds, which accounts for the weight of a gate plus the rigging. Prior to movement of the fuel pool gates, it will be verified that no fuel assembly in the pools has been part of a critical core within the previous 14 days.

The NRC staff evaluated the licensee's revised proposed changes to the TRM and USAR. The NRC staff concludes that the licensee's revised analysis methods and assumptions are consistent with the guidance of RG 1.183, including Regulatory Position 3.1 that radioactive decay from the time of shutdown may be modeled. The NRC staff performed a confirmatory analysis assuming 14 days of decay time before moving the SFP gates over irradiated fuel in the SFP. The NRC analyses conclude that the EAB, LPZ and control room doses are less than (1) the current licensing basis for an FHA, (2) RG 1.183 dose acceptance criteria, and (3) SRP 15.0.1 dose acceptance criteria.

3.5 Technical Conclusion

The licensee has provided adequate information and completed suitable analyses to demonstrate that the infrequent handling of the SFP gates will be completed safely and the potential consequences of a SFP gate handling accident will be within acceptable limits. The licensee has implemented appropriate administrative controls to use existing overhead handling equipment to safely handle the SFP gates for infrequent replacement of the gate seals, consistent with the general guidelines of Section 5.1.1 of NUREG-0612. In the unlikely event of an SFP gate handling accident, the licensee has determined that the consequences of such an event would fall within acceptable bounds specified in Section 5.1 of NUREG-0612 with respect to potential radiological consequences, changes to the configuration of stored fuel, damage to the SFP structure, and damage affecting essential safe shutdown functions. The load drop analyses appropriately conform to the intent of the load drop analysis guidelines of Appendix A to NUREG-0612 with respect to determination of drop height, evaluation of missile energy, and evaluation of potential damage to structures and components. In addition, the NRC staff performed confirmatory analyses and concludes that the EAB, LPZ and control room doses are less than: (1) the current licensing basis for a FHA, (2) RG 1.183 dose acceptance criteria, and (3) SRP 15.0.1 dose acceptance criteria. As discussed above, the NRC staff concludes with the implementation requirement discussed below that the proposed modification to use administrative controls to permit handling of the SFP gates for periodic replacement of seals as a specific exception to existing administrative controls is acceptable.

3.6 Regulatory Commitment and License Condition

In its letter dated January 12, 2015, the licensee proposed the following Regulatory Commitment:

> Prior to any rigging of the spent fuel pool gates, it will be verified that no fuel assembly in the pools has been part of a critical core within the preceding 14 days.

The licensee has proposed this as a "continuing compliance."

The NRC staff used RG 1.183 methods, assuming a 14-day hold period, to evaluate the licensee's dose assessment of the SFP gate load drop analysis. The NRC staff has determined that this proposed Regulatory Commitment is required for the approval of the USAR change and is part of the basis for NRC staff approval of this license amendment. As such, the NRC staff has determined that the 14-day decay time prior to movement of spent fuel gates must be incorporated into the licensing basis documents, and any future changes to this action must be evaluated under the criteria of 10 CFR 50.59. The NRC staff has elevated this action to implementation requirement as described in the amendment issuance pages. Per the implementation requirement, this action will be incorporated into the licensee's USAR upon implementation of this amendment. Therefore, the action, originally proposed as regulatory commitment, has been elevated by the staff to a license condition and cannot be modified or deleted by the licensee under its commitment management program. This action will be incorporated into the licensee's USAR and any future changes to this action must be evaluated under the criteria of 10 CFR 50.59.

4.0 STATE CONSULTATION

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

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 amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on December 10, 2013 (78 FR 74181). 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.

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A. Wang

Date: June 2, 2015

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

Sincerely,

/RA/

Alan B. Wang, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 186 to NPF-47
2. Safety Evaluation

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DATE	5/14/15	5/14/15	2/25/15	4/16/15
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