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

May 16, 2024

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. 214 RE: ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2023-LLA-0038)

Dear Vice President, Operations:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 214 to Renewed Facility Operating License (RFOL) No. NPF-47 for River Bend Station, Unit 1 (River Bend). The amendment consists of changes to the RFOL in response to your application dated February 27, 2023, as supplemented by letter dated January 12, 2024.

The amendment revises the River Bend RFOL No. NPF-47 to add a new license condition to allow for the implementation of Title 10 of the *Code of Federal Regulations* Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors."

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

Sincerely,

/RA/

Jason J. Drake, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 214 to NPF-47
2. Safety Evaluation

cc: 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. 214
Renewed License No. NPF-47

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

   A. The application for amendment by Entergy Operations, Inc. (EOI, the licensee), dated February 27, 2023, as supplemented by letter dated January 12, 2024, 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 as indicated in the attachment to this license amendment, and Renewed Facility Operating License No. NPF-47 is hereby amended to add paragraph 2.H to read as follows:

    H. Entergy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Entergy's submittal letter dated February 27, 2023, and all its subsequent associated supplements; as specified in License Amendment No. 214 dated May 16, 2024.

       Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jennivine K. Rankin

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Date: 2024.05.16 15:16:32 -04'00'

Jennivine K. Rankin, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. NPF-47

Date of Issuance: May 16, 2024

ATTACHMENT TO LICENSE AMENDMENT NO. 214

RENEWED FACILITY OPERATING LICENSE NO. NPF-47

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

Replace the following page of Renewed Facility Operating License No. NPF-47 with the attached revised page. The revised page is identified by Amendment number and contains marginal lines indicating the areas of change.

Renewed Facility Operating License

Remove	Insert
-8-	-8-

G. Entergy Louisiana LLC. and EOI shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. Entergy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Entergy's submittal letter dated February 27, 2023, and all its subsequent associated supplements; as specified in License Amendment No. 214 dated May 16, 2024.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

I. This renewed license is effective as of the date of issuance and shall expire at midnight on August 29, 2045.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Ho K. Nieh, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Attachments 1-5
2. Appendix A - Technical Specifications (NUREG–1172)
3. Appendix B - Environmental Protection Plan
4. Appendix C - Antitrust Conditions

Date of Issuance: December 20, 2018

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 214 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By letter dated February 27, 2023 (Reference 1), as supplemented by letter dated January 12, 2024 (Reference 2), Entergy Operations, Inc. (Entergy, the licensee) submitted a license amendment request (LAR) for changes to Renewed Facility Operating License (RFOL) No. NPF-47 for River Bend Station, Unit 1 (River Bend).

The proposed amendment would modify the River Bend licensing basis by adding a license condition to allow for the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The proposed license condition would state:

> Entergy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE [Individual Plant Examination of External Events] Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS [American Society of Mechanical Engineers/American Nuclear Society] PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Entergy's submittal letter dated, February 27, 2023, and all its subsequent associated supplements as specified in License Amendment No. 214 dated May 16, 2024.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on an integrated and systematic risk-informed process that includes several approaches and methods for categorizing SSCs according to their safety significance.[1]

The proposed amendment would also use the methodology from the U.S. Nuclear Regulatory Commission (NRC, the Commission)-approved LAR related to 10 CFR 50.69 from the Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs) seismic approach (Reference 3)

The NRC staff participated in a regulatory audit in October 2023 (Reference 4) to ascertain the information needed to support its review of the application and to develop requests for additional information, as needed. Following the regulatory audit, the licensee submitted a supplemental letter dated January 12, 2024, which included additional information resulting from the audit. On May 6, 2024, the staff issued an audit summary (Reference 5).

The licensee's supplemental letter dated January 12, 2024, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 11, 2023 (88 FR 44166).

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs perform their design basis functions. For SSCs categorized as low safety significance (LSS), alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety significance (HSS), requirements may not be changed.

In order for a licensee to adopt alternate treatments, Section 50.69 of 10 CFR contains the requirements describing how to categorize SSCs using a risk-informed process; how to adjust treatment requirements consistent with the relative significance of the SSC; and how to manage the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four RISC categories.

SSC categorization does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, 10 CFR 50.69 enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. For SSCs that are categorized as HSS, existing treatment requirements are maintained or potentially enhanced. Conversely, for SSCs categorized as LSS that do not significantly contribute to plant safety on an individual basis, the regulation allows an alternative risk-informed approach to treatment that provides a reasonable level of confidence

[1] Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," dated May 2006 (Reference 4), describes the SSC categorization process in its entirety as an overarching approach that includes multiple approaches and methods identified for a PRA hazard and non-PRA methods.

that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve focus on equipment that has HSS.

2.2 Regulatory Guidance

The NRC staff considered the following regulatory guidance during its review of the proposed changes:

- Regulatory Guide (RG) 1.201, Revision 1 "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," dated May 2006 (Reference 6).

- RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009 (Reference 7).

- RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated January 2018 (Reference 8).

- NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," dated March 2017 (Reference 9).

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition [SRP]," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," dated June 2007 (Reference 10).

NRC-Endorsed Guidance

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," dated July 2005 (Reference 11), as endorsed by RG 1.201 for trial use with clarifications, which describes a process that the NRC staff considers acceptable for complying with 10 CFR 50.69. This process determines the safety significance of SSCs and categorizes them into one of four RISC categories defined in 10 CFR 50.69.

Sections 2 through 10 of NEI 00-04 describe the following steps/elements of the SSC categorization process for meeting the requirements of 10 CFR 50.69:

- Sections 3.2, "Use of Risk Information"; and 5.1, "Internal Events Assessment," provide specific guidance corresponding to 10 CFR 50.69(c)(1)(i).

- Sections 3, "Assembly of Plant-Specific Inputs"; 4, "System Engineering Assessment"; 5, "Component Safety Significance Assessment"; and 7, "Preliminary Engineering Categorization of Functions," provide specific guidance corresponding to 10 CFR 50.69(c)(1)(ii).

- Section 6, "Defense-in-Depth Assessment," provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iii).

- Section 8, “Risk Sensitivity Study,” provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iv).

- Section 2, “Overview of Categorization Process,” provides specific guidance corresponding to 10 CFR 50.69(c)(1)(v).

- Sections 9, “IDP [Integrated Decisionmaking Panel] Review and Approval”; and 10, “SSC Categorization,” provide specific guidance corresponding to 10 CFR 50.69(c)(2).

Additionally, section 11, “Program Documentation and Change Control,” of NEI 00-04 provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(f). Section 12, “Periodic Review,” of NEI 00-04 provides guidance on the periodic review related to the requirements in 10 CFR 50.69(e), “Feedback and process adjustment.” Maintaining change control and periodic review provides confidence that all aspects of the program reasonably reflect the current as-built, as-operated plant configuration and applicable plant and industry operational experience as required by 10 CFR 50.69(c)(1)(ii).

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

An acceptable approach for making risk-informed decisions about proposed technical specification changes, including both permanent and temporary changes, is to show that the proposed licensing basis changes meet the five key principles stated in section C of RG 1.174, Revision 3. These key principles are:

Principle 1: The proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption….

Principle 2: The proposed licensing basis change is consistent with the defense-in-depth philosophy.

Principle 3: The proposed licensing basis change maintains sufficient safety margins.

Principle 4: When the proposed licensing basis changes result in an increase in risk, the increase should be small and consistent with the intent of the Commission’s policy statement on safety goals for the operations of nuclear power plants.

Principle 5: The impact of the proposed licensing basis change should be monitored by using performance measures strategies.

3.2 Traditional Engineering Evaluation

The traditional engineering evaluation below addresses the first three key principles of RG 1.174, Revision 3 and are pertinent to: (1) compliance with current regulations, (2) evaluation of defense-in-depth (DID), and (3) evaluation of safety margins.

Key Principle 1: Licensing Bases Change Meets the Current Regulations

Section 50.69(c) of 10 CFR requires licensees to use an integrated decision-making process to categorize safety-related and non-safety-related SSCs according to the safety significance of the functions they perform into one of the following four RISC categories, which are defined in 10 CFR 50.69(a), as follows:

RISC–1: Safety-related SSCs that perform safety significant functions[2]
RISC–2: Non-safety-related SSCs that perform safety significant functions
RISC–3: Safety-related SSCs that perform low safety significant functions
RISC–4: Non-safety-related SSCs that perform low safety significant functions

The SSCs are classified as having either HSS functions (i.e., RISC-1 and RISC-2 categories) or LSS functions (i.e., RISC-3 and RISC-4 categories). For HSS SSCs, 10 CFR 50.69 maintains current regulatory requirements for special treatment (i.e., it does not remove any requirements from these SSCs). In addition, 10 CFR 50.69(d)(1) requires that "[t]he licensee or applicant shall ensure that RISC-1 and RISC-2 SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance." For LSS SSCs, licensees may implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2). For RISC-3 SSCs, licensees may replace certain special treatment requirement with an alternative treatment approach that meet 10 CFR 50.69(d)(2). For RISC-4 SSCs, 10 CFR 50.69 does not impose new treatment requirements.

Section 50.69(b)(3) of 10 CFR states that the Commission will approve a licensee's implementation of this section by issuance of a license amendment if the Commission determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). As stated in 10 CFR 50.69(b), after the NRC issues a license amendment to implement 10 CFR 50.69, a licensee or applicant specified under 10 CFR 50.69(b)(1) may voluntarily comply with 10 CFR 50.69 as an alternative to compliance with the following requirements for LSS SSCs:

(i) 10 CFR Part 21
(ii) a certain portion of 10 CFR 50.46a(b)
(iii) 10 CFR 50.49
(iv) 10 CFR 50.55(e)
(v) certain requirements of 10 CFR 50.55a
(vi) 10 CFR 50.65, except for paragraph (a)(4)
(vii) 10 CFR 50.72
(viii) 10 CFR 50.73
(ix) Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50
(x) certain requirements for containment leakage testing
(xi) certain requirements of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100

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[2] NEI 00-04, Revision 0, uses the term "high-safety-significant" to refer to SSCs that perform safety-significant functions. The NRC understands HSS to have the same meaning as "safety-significant" (i.e., SSCs that are categorized as RISC-1 or RISC-2), as used in 10 CFR 50.69.

The NRC staff reviewed the licensee's SSC categorization process against the categorization process described in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, and the acceptability of the licensee's PRA for use in the application of the 10 CFR 50.69 categorization process. The NRC staff's review, as documented in this safety evaluation (SE), used the framework provided in RG 1.174, Revision 3, and NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1 including clarifications described in Section C.9 of RG 1.201. Section 2 of NEI 00-04, Revision 0, states, in part, that the categorization process includes eight primary steps:

1. Assembly of Plant-Specific Inputs (Section 3 of NEI 00-04, Revision 0)
2. System Engineering Assessment (Section 4 of NEI 00-04, Revision 0)
3. Component Safety Significance Assessment (Section 5 of NEI 00-04, Revision 0)
4. Defense-In-Depth Assessment (Section 6 of NEI 00-04, Revision 0)
5. Preliminary Engineering Categorization of Functions (Section 7 of NEI 00-04, Revision 0)
6. Risk Sensitivity Study (Section 8 of NEI 00-04, Revision 0)
7. IDP Review and Approval (Section 9 of NEI 00-04, Revision 0)
8. SSC Categorization (Section 10 of NEI 00-04, Revision 0)

In section 3.1.1, "Overall Categorization Process," of the LAR enclosure, the licensee stated that it will implement the risk-informed categorization process in accordance with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. In sections 3.2.3, "Seismic Hazards," and 3.2.4, "Other External Hazards," of the enclosure to the LAR, the licensee has proposed the use of the Electric Power Research Institute (EPRI) Tier 1 alternate seismic approach as alternative methods to assess the applicable hazard contributions. The NRC notes that use of this alternative method is a deviation from the NEI 00-04 guidance as endorsed. A more detailed staff review of the alternative methods is provided in section 3.3.2 of this SE.

The licensee provided further discussion of specific elements within the 10 CFR 50.69 categorization process that are delineated in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 0.

The regulatory requirements in 10 CFR 50.69 and 10 CFR Part 50, Appendix B; implemented in accordance with the guidance for monitoring outlined in NEI 00-04, Revision 0; and clarifications in RG 1.201, Revision 1, ensure that the SSC categorization process is sufficient to assure that the SSC functions continue to be met, that any performance deficiencies will be identified, and appropriate corrective actions taken. The licensee's SSC categorization program includes the appropriate steps/elements described in NEI 00-04, Revision 0, to assure that SSCs specified are appropriately categorized consistent with 10 CFR 50.69. The NRC staff performed a more detailed review of specific steps/elements of the licensee's SSC categorization process, where necessary, to confirm consistency with the NEI 00-04 guidance, as endorsed. Based on the above, the NRC staff concludes that the proposed program to implement risk-informed categorization and treatment of SSCs meets the first key principle of RG 1.174 for risk-informed decision-making.

Key Principle 2: Licensing Basis Change is Consistent With the Defense-In-Depth Philosophy

In RG 1.174, Revision 3, the NRC identified the following considerations used for evaluating how the licensing basis change is maintained for the DID philosophy:

- Preserve a reasonable balance among the layers of defense.

- Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.
- Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.
- Preserve adequate defense against potential CCFs [common-cause failures].
- Maintain multiple fission product barriers.
- Preserve sufficient defense against human errors.
- Continue to meet the intent of the plant's design criteria.

RG 1.201, Revision 1, endorses the guidance in section 6 of NEI 00-04, Revision 0, but notes that the containment isolation criteria in this section of the guidance, are separate and distinct from those set forth in 10 CFR 50.69(b)(1)(x). The criteria in 10 CFR 50.69(b)(1)(x) are to be used in determining which containment penetrations and valves may be exempted from the Type B and Type C leakage testing requirements in both Options A, "Prescriptive Requirements," and B, "Performance-Based Requirements," of Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The criteria provided in 10 CFR 50.69(b)(1)(x) are not to determine the proper RISC category for containment isolation valves or penetrations.

In section 3.1.1 of the enclosure to the LAR, the licensee clarified that if the DID assessment determines that the SSC must be categorized as HSS, the SSC will be categorized as HSS in accordance with NEI 00-04, Revision 0. The NRC staff finds that the licensee's process is consistent with DID philosophy and the NRC-endorsed guidance in NEI 00-04; therefore, Key Principle 2 of risk-informed decision-making in RG 1.174, Revision 3 is met and fulfills the 10 CFR 50.69(c)(1)(iii) criterion that requires DID to be maintained.

<u>Key Principle 3: Licensing Basis Change Maintains Sufficient Safety Margins</u>

The engineering evaluation that will be conducted by the licensee under 10 CFR 50.69 for SSC categorization will assess the design function(s) and risk significance of the SSC to assure that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring the categorization of the SSC does not adversely affect any assumptions or inputs to the safety analysis; or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist.

The SSCs design-basis function as described in the plant's licensing basis, including the Updated Final Safety Analysis Report and Technical Specification Bases, do not change and therefore will continue to be met. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. On this basis, the NRC staff concludes that safety margins are maintained by the proposed methodology, and therefore satisfies the third key safety principle of RG 1.174, Revision 3.

System Engineering Assessment (NEI 00-04, Revision 0, Section 4)

Section 4 NEI 00-04, Revision 0 describes a process for system selection and boundary definition, identification of system functions, and mapping of components to functions. In section 2.2 of the enclosure to the LAR, "Reason for Proposed Change," the licensee identifies the selected systems by stating, in part, that "[t]he safety functions [in the categorization process] include the design basis functions, as well as functions credited for severe accidents (including external events)." Figure 3-1 of the enclosure the LAR comprises a flow chart in which the licensee will define system boundaries and define system functions. Section 3.1.1 of the enclosure to the LAR summarizes the different hazards and plant states for which functional and risk significant information will be collected. In section 3.1.1 of the enclosure to the LAR, the licensee also confirmed that "the mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components."

The NRC staff finds that the process described in the LAR is consistent with NEI 00-04, Revision 0. RG 1.201, Revision 1, states, in part, "this trial regulatory guides provides interim guidance for complying with the NRC's requirements in 10 CFR 50.69, by using the process described in Revision 0 of NEI 00-04." Because the process described in LAR meets the requirements of NEI 00-04, the NRC staff finds that the process meets the requirements set forth in 10 CFR 50.69(c)(1)(ii) and 10 CFR 50.69(c)(1)(iv).

3.3 Risk-Informed Assessment

Key Principle 4: Change in Risk is Consistent with the Safety Goals

The risk-informed considerations described in NEI 00-04, Revision 0, endorsed by RG 1.201, Revision 1 addresses the fourth and fifth key principles of the NRC staff's guidance for risk-informed decision-making, pertaining to the assessment for change in risk and monitoring the impact of the licensing basis change.

A summary of how the licensee's SSC categorization process is consistent with the guidance and methodology prescribed in NEI 00-04, Revision 0, and RG 1.201, Revision 1 is provided in the sections below:

Assembly of Plant-Specific Inputs (NEI 00-04, Revision 0, Section 3)

The NRC staff acknowledges that elements of the categorization process are not always performed in chronological order and may be performed in parallel, such as the systematic process (e.g., system selection, system boundary definition, identification of system functions, and mapping of components to functions) that is further discussed in section 3.2 of this SE. The licensee's risk categorization process uses PRAs to assess risks from the internal events PRA (IEPRA) (including internal flooding), and fire PRA (FPRA). For non-PRA methods that depart from the methodology prescribed in NEI 00-04, additional staff review is discussed in section 3.3.2 of this SE.

Section 50.69(c)(1)(v) of 10 CFR requires that SSC categorization be performed for entire systems and structures, not for selected components within a system or structure. Section 3.1.1 of the enclosure the LAR states, in part, that "Entergy will implement the risk categorization process in accordance with NEI 00-04, as endorsed by Regulatory Guide 1.201." Attachment 2 of the LAR supplement also describes an overall method for selecting systems and system boundaries consistent with NEI 00-04, Revision 0. Because NEI 00-04, Revision 0, was endorsed as an acceptable means to comply with the requirements of 10 CFR 50.69, the NRC staff finds the process described in the LAR, as supplemented, meets the requirements set forth in 10 CFR 50.69(c)(1)(v).

Component Safety Significance Assessment (NEI 00-04, Section 5)

This step in the licensee's categorization process assesses the safety significance of components using quantitative or qualitative risk information from a modeled PRA hazard, other hazards that can be screened, and non-PRA method(s). In the NEI 00-04 guidance, component risk significance is assessed separately for the following hazard groups:

- internal events (including internal floods)
- internal fire events
- seismic events
- external hazards (e.g., high winds, external floods)
- other hazards
- shutdown events
- passive categorization

In sections 3.2.1, "Internal Events and Internal Flooding," and 3.2.2, "Fire Hazards," of the enclosure to the LAR, the licensee described that the River Bend categorization process uses PRA to assess risks for the internal events (includes internal flood) and internal fires. For the other risk contributors, the licensee's process uses non-PRA methods to characterize the risk.

- External Hazards: Screening analysis performed for IPEEEs (Reference 12) updated using criteria from Part 6 of the ASME/ANS RA-Sa–2009, "Addendum A to RA-S–2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (the ASME/ANS 2009 PRA standard) (Reference 13), as endorsed by the NRC in Appendix A of RG 1.200 (Reference 7)

- Other Hazards: Screening analysis performed for the IPEEE updated using criteria from Part 6 of the ASME/ANS 2009 PRA standard, as endorsed by the NRC in Appendix A of RG 1.200 (Reference 7).

- Shutdown Events: Safe Shutdown Risk Management program consistent with Nuclear Management and Resources Council (NUMARC) 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (Reference 14).

- Passive Components: ANO-2 passive categorization methodology (Reference 15).

- Seismic: EPRI Technical Update 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Tier 1 (Reference 16), as reviewed and approved by the NRC in the Calvert Cliffs 10 CFR 50.69 SE (Reference 3).

The approaches and methods proposed by the licensee to address internal events, non-seismic external events, DID, and shutdown events are consistent with the approaches and methods included in the guidance in NEI 00-04, Revision 0. The non-PRA method for the categorization for passive components is consistent with the ANO-2 methodology for passive components approved for risk-informed safety classification and treatment for repair/replacement activities in class 2 and 3 moderate- and high-energy systems. The use of the ANO-2 methodology in the SSC categorization process is provided in section 3.3.2 of this SE. To address the seismic events, the licensee proposed to use an alternative method. A detailed NRC staff review of the licensee's proposed alternate seismic approach is provided in section 3.3.2 of this SE.

3.3.1 Scope of the PRA

The River Bend PRA is comprised of a full-power, Level 1, IEPRA, internal flooding PRA (IFPRA), and FPRA, which evaluate the core damage frequency (CDF) and large early release frequency (LERF) risk metrics. The licensee discussed in section 3.3, "PRA Review Process Results (10 CFR 50.69(b)(2)(iii))" of the enclosure to the LAR that the IEPRA (includes floods) and FPRA models have been peer reviewed against RG 1.200, Revision 2. Furthermore, section 3.3 states that finding closure reviews and focused-scope peer reviews were conducted on both the IEPRA and FPRA models. Open findings were reviewed and closed using the NRC-accepted process documented in the NEI letter to the NRC "Final Revision of Appendix X to NEI 05-04/07-12/12-16, 'Close-out of Facts and Observations,'" dated February 21, 2017 (Reference 17), hereafter referred to as the Appendix X Process.

The NRC staff finds that the information regarding the PRA review process provided in the LAR is of sufficient detail to support the staff's review of the technical acceptability of the IEPRA (including internal flooding) and FPRA and therefore meets the requirements set forth in 10 CFR 50.69(b)(2)(iii).

Aspects considered by the NRC staff to evaluate the scope of the PRA include: (1) peer-review history, process, and results, (2) credit for FLEX in the PRA, and (3) assessment of assumptions and approximations.

<u>IEPRA (Includes Internal Floods) Peer-Review History</u>

In section 3.3 of the enclosure to the LAR, the licensee states that the IEPRA model was subjected to a full-scope peer review in April 2011 consistent with RG 1.200 Revision 2. Subsequently, in September 2017 and June 2019 the licensee conducted focused scope peer reviews regarding LERF, internal flooding, and pre-initiator human error events. In August 2020 and December 2021, the licensee performed an independent assessment for closure of the finding-level facts and observations (F&Os) using the Appendix X process. These reviews concluded that all the IEPRA (includes internal floods) F&Os have been closed.

In section 3.2, "Technical Adequacy Evaluation (10 CFR 50.69(b)(2)(ii))," of the enclosure to the LAR, for the IEPRA (including internal floods), states in part, "there are no PRA upgrades that have not been peer reviewed." In review of the licensee's reports, the NRC staff concluded that all F&Os were appropriately assessed by the independent assessment team to assure that no new methods or upgrades were inadvertently incorporated into the IEPRA without a peer review in accordance with the ASME/ANS 2009 PRA standard as endorsed by the NRC.

Therefore, the NRC staff concludes that the River Bend IEPRA (including internal floods) was appropriately peer reviewed, consistent with RG 1.200, Revision 2, and the F&O's have been adequately closed.

Internal FPRA Peer-Review History

The licensee's FPRA was subject to a full-scope industry peer review in June 2019, consistent with RG 1.200, Revision 2. The finding-level F&Os from the 2019 full-scope were considered fully resolved and closed by the independent assessment review team based on a closure review conducted in June 2020 using the Appendix X Process.

The NRC staff has reviewed the FPRA peer review results and the licensee's resolution of the results and concludes that the River Bend FPRA was appropriately peer-reviewed, consistent with RG 1.200, Revision 2, and the F&O's have been adequately closed. The LAR further clarifies that the independent assessment team concluded that the dispositions for the finding-level F&Os were PRA maintenance activities, and none constituted a PRA model upgrade that would necessitate a focused-scope peer review.

Credit for Diverse and Flexible Mitigation Capability (FLEX) Equipment

The NRC memorandum dated May 6, 2022, "Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Risk Assessments" (Reference 18), provides the NRC staff's assessment of challenges to incorporating diverse and flexible mitigation capability (FLEX) equipment and strategies into a PRA model in support of risk-informed decision-making in accordance with the guidance of RG 1.200, Revision 2.

In its LAR, as supplemented, the licensee confirmed that the internal events (includes internal flooding) and fire PRA models credit permanently installed and portable equipment used as part of the FLEX strategy and described how the modeling is consistent with the NRC memorandum dated May 6, 2022, regarding the NRC's staff updated assessment of identified challenges and strategies for incorporating FLEX equipment into a PRA model in support of risk-informed decision-making in accordance with the guidance of RG 1.200. The licensee provided the results of a sensitivity study in which credit for FLEX equipment was completely removed. The licensee reported an overall increase in CDF of 12 percent. In the LAR supplement, the licensee stated that the 12 percent increase in CDF for FLEX credit results from the complete removal of FLEX credit, which is conservative compared to a more realistic model. The licensee stated the current modeling of FLEX credit was developed using failure probability values listed in Pressurized Water Reactor Owners Group (PWROG)-18042, which is consistent with the NRC memorandum dated May 6, 2022, with the exception of a motor-operated centrifugal pump. For this pump, the licensee modeled impact to CDF using data from NUREG/CR-6928 (Reference 19) multiplied by a conservative multiplier. For modeling of human reliability analysis (HRA), the licensee followed the guidance in EPRI Technical Report 300201018, "Human Reliability Analysis (HRA) for Diverse and Flexible Mitigation Strategies (FLEX) and Use of Portable Equipment" (Reference 20), as approved in the NRC memorandum dated May 6, 2022. For each of the three HRA exceptions to methods in EPRI TR-300201018, (i.e., trailer connections, refueling, and load shedding recovery), the licensee provided details for accounting for the uncertainties in the PRA in the supplement to the LAR. Therefore, the NRC staff finds that the licensee's credit for FLEX equipment is acceptable for the application because the licensee used consensus HRA methodologies, practices, and failures rates consistent with the NRC guidance.

Assessment of PRA Model Assumptions and Approximations

Identification of Key Assumptions and Sources of Uncertainty

In the LAR, the licensee stated that sources of model uncertainty and related assumptions in the base PRA were identified, screened, and characterized using processes consistent with NUREG-1855, Revision 1, EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty" (Reference 21), and EPRI Technical Report 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments" (Reference 22). In the LAR, as supplemented, the licensee provided an overview of the criteria used to identify sources of uncertainty, such as impact on baseline CDF and LERF, impact on risk achievement worth, impact on Fussell-Vessely importance values, and whether the issue had been previous addressed by a consensus approach or model. The NRC staff finds that the assessment performed to identify the key assumptions and sources of uncertainty is consistent with the NRC guidance provided in NUREG-1855, Revision 1.

Treatment of the Key Assumptions and Sources of Uncertainty

NUREG-1855, Revision 1 and NEI 00-04 provides guidance regarding how to address PRA uncertainties and provides guidance on how to treat uncertainties associated with PRA in risk-informed decision-making. The guidance fosters an understanding of the uncertainties associated with PRA and their impact on the results of the PRA and provides an approach to addressing these uncertainties in the context of the decision-making. The licensee identified 29 total candidate sources of uncertainty in all PRAs, as listed in attachment 6, "Disposition of Key Assumptions/Sources of Uncertainty," of the LAR. After applying the screening methods, one source of uncertainty was identified to impact the categorization according to 10 CFR 50.69. In the supplement to the LAR, the licensee confirmed that for any key sources of uncertainty, sensitivity studies will be performed consistent with the NEI 00-04 guidance. In accordance with NEI 00-04, the results of the sensitivity study are given to the IDP for consideration in the final risk characterization for components initially classified as LSS that may be reclassified to HSS. The NRC staff finds that the licensee's approach to treatment of PRA key assumptions and sources of uncertainty is consistent with the NEI 00-04 guidance.

In addition, the NRC staff recognizes that the licensee will perform routine PRA changes and updates to assure the PRA continually reflects the as-built, as-operated plant, in addition to changes made to the PRA to support the context of the analysis being performed (i.e., sensitivities), as required by sections 50.69(e) and (f) of 10 CFR. These sections stipulate the process for feedback and adjustment to assure configuration control is maintained for these routine changes and updates to the PRA(s).

PRA Acceptability Conclusions

Pursuant to 10 CFR 50.69(c)(1)(i), the categorization process must consider results and insights from a plant-specific PRA. The use of the IEPRA, IFPRA, and FPRA to support SSC categorization is endorsed by RG 1.201, Revision 1. Furthermore, pursuant to 10 CFR 50.69(c)(1)(i) the PRAs must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer-review process assessed against a standard that is endorsed by the NRC. Revision 2 of RG 1.200 provides guidance for determining the acceptability of the PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 PRA standard using a peer-review process.

The licensee has subjected the IEPRA, IFPRA, and FPRA to the peer-review processes and submitted the results of the peer review. The NRC staff reviewed the peer-review history (which included the results and findings), the licensee's resolution of peer-review findings, and the identification and disposition of key assumptions and sources of uncertainty. The staff concludes that (1) the River Bend IEPRA (includes internal floods) and FPRA were appropriately peer reviewed consistent with RG 1.200, Revision 2 and meets the requirements set forth in 10 CFR 50.69(c)(1)(i), and that all F&Os have been closed, (2) the licensee's IEPRA and FPRA are acceptable to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201, Revision 1, and (3) the key assumptions for the PRAs have been identified consistent with the guidance in RG 1.200, Revision 2 and NUREG-1855, as applicable, and addressed appropriately for this application.

The NRC staff finds the licensee provided the required information, and the IEPRA (includes internal floods) and FPRA are acceptable and therefore meets the requirements set forth in 10 CFR 50.69(c)(1)(i).

3.3.2 Evaluation of the Use of Non-PRA Methods in SSC Categorization

The licensee's categorization process uses the following non-PRA method(s), respectively:

- Seismic Hazard: alternative seismic treatment using guidance from EPRI Technical Update 3002017583 and qualitative insights about seismic risk at River Bend.

- Other External Hazards: Screening analysis performed for IPEEE and updated using criteria from Part 6 of the ASME/ANS 2009 PRA standard, as endorsed by the NRC.

- Shutdown Events: Safe Shutdown Risk Management program consistent with NUMARC 91-06.

- Passive Components: ANO-2 passive categorization methodology.

The NRC staff's review of these methods is discussed below.

<u>Seismic Risk</u>

As part of its proposed integrated decision-making process to categorize SSCs according to their safety significance, the licensee proposed to use a non-PRA method to consider seismic hazards. The regulation in 10 CFR 50.69(b)(2)(ii) requires a description of the measures taken to assure that the quality and level of detail of the systematic evaluation process for risk-informed categorization of SSCs to be included in the application. In section 3.2.3 of the LAR, as supplemented, the licensee described its proposed alternative seismic approach for considering seismic risk in the categorization process. The licensee also described how its proposed alternative seismic approach would be used in the categorization process and the measures taken to assure that the quality and level of detail for the licensee's proposed alternative seismic approach are adequate for the categorization of SSCs. Based on the above, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(ii) for the proposed alternative seismic approach are met.

The licensee based its plant-specific evaluation, in part, on the case studies performed in EPRI Report 3002017583 (Reference 16) and stated that the case studies are applicable to River Bend and are used in its proposed alternative seismic approach. EPRI Report 3002017583 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from seismic PRAs (SPRAs). The NRC staff's review confirmed that the case studies in EPRI Report 3002017583 used by the licensee as well as the information in its supplement provided sufficient plant-specific basis for applicability of its proposed alternative seismic approach to River Bend. Accordingly, EPRI Report 3002017583 and its cited case studies previously approved by the NRC staff (Reference 3) and information presented in the LAR, as supplemented provided a sufficient description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv). Therefore, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(iv) are met for the licensee's proposed alternative seismic approach.

Evaluation of the Criteria for the Proposed Alternative Seismic Approach

In section 3.2.3 of the LAR, the licensee states, in part, that the ground motion response spectrum (GMRS) peak spectral acceleration for River Bend is below the safe shutdown earthquake (SSE) between 1 hertz (Hz) and 10 Hz, which demonstrates that River Bend qualifies as a Tier 1 plant under the criteria in EPRI Report 3002017583. The NRC staff notes that the licensee's plant-specific evaluation is supported by its response to the NRC 10 CFR 50.54(f) request dated March 26, 2014 (Reference 23) in that the plant SSE exceeded the GMRS in the 1 to 10 Hz range of the response spectrum. The NRC staff reviewed the LAR, as supplemented, and the licensee's plant-specific evaluation and concluded that the use of the criteria in EPRI Report 3002017583 to determine the applicability and use of the proposed seismic Tier 1 approach is acceptable.

Evaluation of the Applicability of the Proposed Alternative Seismic Approach to 10 CFR 50.69 Categorization at River Bend

In section 3.2.3 of the LAR enclosure, the licensee states that "[t]he overall seismic risk is relatively low compared to total plant risk," and that "[t]he small seismic risk contribution at [River Bend] makes it unlikely that an integral importance assessment for a component, as defined in NEI 00-04, would result in an overall HSS determination." The NRC staff noted that section 2.2.2 of the EPRI Report 302017583, which identifies the expectation that low contribution of seismic risk to total plant risk reduces the likelihood of a unique seismic condition that would cause an SSC to be designated HSS.

The NRC staff's evaluation of seismic risk to total plant risk was based on information in the River Bend Technical Specifications Task Force (TSTF) Traveler TSTF-505 LAR (Reference 24). The purpose of the seismic risk estimate in the River Bend TSTF-505 LAR is to provide a conservative estimate for use in calculating risk-informed completion times for technical specifications. The staff determined that the licensee used conservatively biased median fragility values in calculating the seismic risk estimate for seismic CDF and LERF for the River Bend TSTF-505 LAR. Further, as noted in section 3.6.5, "Defense-in-Depth Assessment," of EPRI Report 3002017583, containment DID assessment addresses containment failures and containment bypass situations. Section 3.6.6, "Civil Structures," of EPRI Report 3002017583, used for the licensee's proposed alternative seismic approach, recommends that if the licensee chooses to categorize civil structures housing HSS SSCs, the structures are considered as HSS. Therefore, based on its evaluation and review, the staff concludes that the proposed alternative seismic approach, in conjunction with the other elements of the 10 CFR 50.69

categorization program, will appropriately determine the safety significance of any SSCs whose seismic-induced failures would lead directly to core damage and large early release. Furthermore, staff finds that contribution from seismic risk alone to the 10 CFR 50.69 categorization program would not result in any additional SSCs being categorized as HSS.

The NRC staff finds that the licensee's basis for applying the proposed alternative seismic approach to its site is applicable for use in the licensee's 10 CFR 50.69 program because (1) the reevaluated hazard meets the criteria for use of the proposed alternative seismic approach, (2) in conjunction with other elements of the 10 CFR 50.69 categorization program, the approach will appropriately determine the safety significance of any SSCs whose seismic-induced failures would lead directly to core damage and large early release, and (3) the seismic risk contribution would not solely result in any additional SSCs being categorized as HSS.

<u>Evaluation of the EPRI 3002017583 Case Studies</u>

In its supplement to the LAR, the licensee stated that the plant specific case studies from other licensees included in EPRI Report 3002017583 are incorporated by reference to support its proposed alternative seismic approach. The licensee also stated in the supplement that there are no differences between the proposed River Bend alternative seismic approach and that, reviewed and approved in the NRC staff's SE on Calvert Cliffs 10 CFR 50.69 LAR (Reference 3). The NRC staff finds that the acceptability of PRAs used in Plants A, C, and D case studies in EPRI Report 3002017583, the mapping approach used in those case studies, and the conclusions on the determination of unique HSS SSCs from the case studies, which were reviewed and approved by the NRC staff and used as technical basis for the Calvert Cliffs 10 CFR 50.69 license amendment, are applicable to the proposed alternative seismic approach for River Bend.

<u>Evaluation of the Implementation of Conclusions from the Case Studies</u>

The licensee stated that the proposed categorization approach for seismic hazards will include qualitative consideration of the mitigation capabilities of SSCs during seismically-induced events and seismic failure modes, based on insights obtained from prior seismic evaluations performed for River Bend. The licensee explained that the qualitative characterization of seismic risk performed for the IDP will include information from the various post-Fukushima seismic reviews including results of seismic walkdowns, seismic mitigation strategy assessment, and seismic high-frequency evaluations. The objective of the alternative seismic approach is to identify plant-specific seismic insights derived from the components in the system being categorized.

The NRC staff's review of the licensee's proposed alternative seismic approach determined that the approach used in the Calvert Cliff's amendments are applicable to this licensee's proposed alternative seismic approach and that the plant-specific evaluation of the implementation of the alternative seismic approach is acceptable. There are no differences that exist between the River Bend proposed alternative approach and the approach used in the NRC staff-approved Calvert Cliffs 10 CFR 50.69 SE. The NRC staff's review of the proposed alternative seismic approach, in conjunction with the requirements in 10 CFR 50.69 and the corresponding statement of consideration, finds that the proposed alternative seismic approach includes the evaluations required by 10 CFR 50.69(c)(1)(ii), as well as 10 CFR 50.69(c)(1)(iv) because:

1. The proposed alternative seismic approach includes qualitative consideration of seismic events at several steps of the categorization process, including documentation of the

information for presentation to the IDP as part of the integrated, systematic process for categorization.

2. The proposed alternative seismic approach presents system-specific seismic insights to the IDP for consideration as part of the IDP review process as each system is categorized, thereby providing the IDP a means to consider potential impacts of seismic events in the categorization process.

3. The insights presented to the IDP include potentially important seismically-induced failure modes, as well as mitigation capabilities of SSCs during seismically-induced design basis and severe accident events consistent with the conclusions on the determination of unique HSS SSCs from SPRAs in EPRI 3002017583. The insights will use prior plant-specific seismic evaluations and, therefore, in conjunction with performance monitoring for the proposed alternative seismic approach, reasonably reflect the current plant configuration. Further, the recommendation for categorizing civil structures in the alternative seismic approach provides appropriate consideration of such failures from a seismic event.

4. The proposed alternative seismic approach presents the IDP with the basis for the proposed alternative seismic approach, including the low seismic hazard for the plant and the criteria for the use of the proposed alternative seismic approach.

5. The proposed alternative seismic approach includes qualitative consideration and insights related to the impact of a seismic event on SSCs for each SSC that is categorized and does not limit the scope to SSCs from the case studies supporting this application.

<u>Consideration of Changes to Seismic Hazard</u>

The possibility exists for the seismic hazard at the site to increase such that the criteria for the use of the proposed alternative seismic approach may no longer be appropriate. The licensee stated that the continued comparison of GMRS to SSE applies to the River Bend site. The licensee also stated that the seismic hazard at the plant is subject to periodic reconsideration as new information becomes available through industry evaluations.

The NRC staff finds that the consideration of changes to the seismic hazard in the licensee's plant-specific proposed alternative seismic approach is the same as that approved in the Calvert Cliffs amendments. Consequently, the NRC staff finds that the consideration of changes to the seismic hazard at River Bend that exceeds the criteria for use of the proposed alternative seismic approach is acceptable for the proposed approach because (1) the criteria for use of the proposed alternative seismic approach is clear and traceable, (2) the proposed alternative seismic approach includes periodic reconsideration of the seismic hazard as new information becomes available, (3) the proposed alternative seismic approach satisfies the requirements in 10 CFR 50.69 discussed above, and (4) the licensee has included a proposed license condition in the LAR to require NRC approval for a change to the specified seismic categorization approach.

<u>Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach</u>

In section 3.5, "Feedback and Adjustment Process," of the enclosure to the LAR, the licensee stated that its configuration control process ensures that changes to the plant, including a physical change and changes to documents, are evaluated to ensure that the qualitative determinations for the seismic hazard continue to remain in compliance with the requirements of 10 CFR 50.69.

Based on its review, the NRC staff found that consideration of the feedback and adjustment process in the licensee's proposed alternative seismic approach is acceptable. The NRC staff finds that:

1. The licensee's programs provide reasonable assurance that the existing seismic capacity of LSS components would not be significantly impacted, and

2. The monitoring and configuration control program ensures that potential degradation of the seismic capacity would be detected and addressed before significantly impacting the plant risk profile.

Therefore, the NRC staff finds that the potential impact of the seismic hazard on the categorization is maintained acceptably low and the requirements in 10 CFR 50.69(c)(1)(iv) are met for the proposed alternative seismic approach.

External Hazards and Other Hazards (Non-Seismic)

This hazard category includes all non-seismic external hazards such as high winds, external floods, transportation, nearby facility accidents, and other hazards. In Section 3.2.4 of the enclosure to the LAR, the licensee stated, in part, that "[a]ll other external hazards, except for seismic, were screened from applicability to RBS per a plant-specific evaluation in accordance with Generic Letter 88-20 and updated to use the criteria in ASME/ANS PRA Standard RA-Sa-2009."

In the safety evaluation report for the River Bend IPEEE (Reference 25), the NRC staff confirmed that the high winds, floods, and other external events (HFO) were eliminated based on conformance with the criteria in the 1975 NRC Standard Review Plan (SRP) using the progressive screening approach described in NUREG-1407 (Reference 26), a procedural and submittal guidance for Generic Letter 88-20, Supplement 4.

The licensee confirmed that River Bend will subject the external hazards (except for seismic) to the process described by the flow chart in NEI 00-04, figure 5-6, which provides guidance to be used to determine SSC safety significance for these external hazards. The NRC staff finds that River Bend will assess the risk from all other external hazards consistent with figure 5-6 of NEI 00-04 as endorsed in RG 1.201, Revision 1.

In summary, the NRC staff finds that use of the updated River Bend IPEEE results described by the licensee in the LAR, and the licensee's assessment of other external hazards (i.e., high winds, tornadoes, and external flood) in the LAR are consistent with section 5 of NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. The NRC staff concludes that the licensee's treatment of other external hazards is acceptable and meets 10 CFR 50.69(c)(1)(ii).

Shutdown Risk

Consistent with the guidance in NEI 00-04, Revision 0, the licensee proposed using the shutdown safety assessment based on NUMARC 91-06. NUMARC 91-06 provides considerations for maintaining DID for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment-primary/secondary. NUMARC 91-06 also specifies that a DID approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and an alternative system/train to accomplish the given key safety function.

The use of NUMARC 91-06 described by the licensee in the submittal is consistent with the guidance in NEI 00-04, Revision 0, as endorsed in the NRC in RG 1.201, Revision 1. The approach uses an integrated and systematic process to identify HSS components, consistent with the shutdown evaluation process. Therefore, the NRC staff finds that the licensee's use of NUMARC 91-06 to assess shutdown safety is acceptable, and meets the requirements set forth in 10 CFR 50.69(c)(1)(ii).

Component Safety Significance Assessment for Passive Components

Passive components are not modeled in the PRA; therefore, a different assessment method is necessary to assess the safety significance of these components. Passive components are those components having only a pressure retaining function. This process also addresses the passive function of active components such as the pressure/liquid retention of the body of a motor-operated valve.

In section 3.1.2, "Passive Categorization Process," of the enclosure to the LAR, the licensee proposed using a categorization method for passive components not cited in either NEI 00-04, Revision 0, or RG 1.201, Revision 1, for passive component categorization, but was approved by the NRC for ANO-2. The ANO-2 methodology is a risk-informed safety classification and treatment program for repair/replacement activities for Class 2 and 3 pressure retaining items and their associated supports (exclusive of Class CC and MC items), using a modification of the ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1" (Reference 27). The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measures the safety significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment. Therefore, the NRC staff finds that the use of the ANO-2 repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs at River Bend.

In section 3.1.2 of the LAR, the licensee stated, "[t]he passive categorization process is intended to apply the same risk-informed process accepted by the NRC in the ANO 2-R&R-004 for the passive categorization of Class 2, 3, and non-class components." Consistent with ANO-2-R&R-004, Class 1 pressure retaining SSCs in the scope of the system being categorized will be assigned HSS and cannot be changed by the IDP. That is, the ANO-2 repair/replacement methodology does not allow a Class 1 pressure retaining SSC to be recategorized from HSS to LSS. Therefore, the NRC staff finds the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process for Class 1, Class 2, and Class 3 pressure retaining SSCs.

3.3.3 Risk Sensitivity Study (NEI 00-04, Section 8)

In section 3.1.1 of the enclosure to the LAR, the licensee states, in part that "[a]n unreliability factor of three will be used for the sensitivity studies described in Section 8 of NEI 00-04." In section 3.2.7, "PRA Maintenance and Updates," of the enclosure to the LAR, the licensee further confirms that a cumulative sensitivity study will be performed where the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in PRAs for all systems that have been categorized are increased by a factor of three. The NRC staff finds the application of a factor of three for the sensitivities is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

In section 3.1.1, the licensee specifically noted that RG 1.201 states, in part, that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 (Reference [9]) must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv)." In Section 3.4 of the LAR, the licensee states, "[s]ensitivity studies described in NEI 00-04, Section 8, will be used to confirm that the categorization process results in acceptably small increases to CDF and LERF." This sensitivity study together with the periodic review process discussed in section 3.4 of this SE, assure that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. The NRC staff finds that the performance of the risk sensitivity study consistent with the guidance in section 8 of NEI 00-04, Revision 0, and, therefore, will assure that the potential cumulative risk increase from the categorization is maintained acceptably low, as required by 10 CFR 50.69(c)(1)(iv).

3.3.4 Integrated Decision-Making

Appendix B of SRP Section 19.2 provides guidance and the NRC staff expectations for the licensee's integrated decision-making process. The appendix states, in part, that "[r]isk-informed applications are expected to require a process to integrate traditional engineering and probabilistic considerations to form the basis for acceptance." NEI 00-04 guidance identifies two steps in the categorization process: (1) Preliminary Engineering Categorization of Function and (2) IDP Review and Approval that are responsible for the integrated assessment of the traditional engineering analyses and the risk results from the PRA and non-PRA assessments that are performed to determine the approval of the safety significance of the SSC for categorization. The staff review of the two steps to ensure the processes is well-defined, systematic, repeatable, and scrutable are provided as follows:

<u>Preliminary Engineering Categorization of Function (NEI 00-04, Section 7)</u>

All the information collected and evaluated in the licensee's engineering evaluations is provided to the IDP as described in section 7 of NEI 00-04, Revision 0. The IDP will make the final decision about the safety significance of SSCs based on guidelines in NEI 00-04, Revision 0, the information they receive, and their expertise.

In section 3.1.1 of the enclosure to the LAR, the licensee stated, in part, "... if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS." The licensee also stated that, "[o]nce a system function is identified as HSS, then all the components that support that function are preliminary HSS."

The NRC staff finds that the above description provided by the licensee for the preliminary categorization of functions is consistent with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, and therefore meets the requirements of 10 CFR 50.69.

IDP Review and Approval (NEI 00-04, Sections 9 and 10)

In section 3.1.1 of the enclosure to the LAR, the licensee states, in part, that "[t]he IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and PRA." Based on this information, the staff finds that the IDP will comprise the required expertise consistent with Section 9.1 of NEI 00-04, Revision 0.

The guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, provides confidence that the IDP expertise is sufficient to perform the categorization and that the results of the different evaluations (PRA and non-PRA) are used in an integrated, systematic process as required by 10 CFR 50.69(c)(1)(ii). In section 3.1.1, the licensee states that "[a]t least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in the modeling and updating of the plant-specific PRA." The licensee further states that "[t]he IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address, at a minimum, the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant-specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the defense-in-depth philosophy and requirements to maintain this philosophy." The NRC staff finds that the licensee's IDP areas of expertise meet the requirements in 10 CFR 50.69 (c)(2) and the additional descriptions of the IDP characteristics, training, processes, and decision guidelines are consistent with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1.

As discussed in NEI 00-04, Revision 0, the only LSS SSC requirements that are relaxed for RISC-3 (LSS) SSCs are those related to treatment, not design or capability, and 10 CFR 50.69(d)(2)(i) requires that the licensee ensures, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions. Therefore, the NRC staff finds that the IDP for the River Bend categorization process, is consistent with the endorsed guidance in NEI 00-04, Revision 0, and, therefore, fulfills 10 CFR 50.69(c)(1)(iv).

Based on the above NRC staff review for: (1) IEPRA, IFPRA, and FPRA acceptability, (2) PRA importance measures and integrated importance measure, (3) evaluation of the use of non-PRA methods, (4) risk sensitivity study, and (5) integrated decision making, the staff has determined that the proposed change satisfies the fourth key principle for risk-informed decision making described in RG 1.174, Revision 3.

3.4 Key Principle 5: Monitor the Impact of the Proposed Change

NEI 00-04, Revision 0 provides guidance that includes programmatic configuration control and a periodic review to ensure that the all aspects of the 10 CFR 50.69 program (i.e., includes traditional engineering analyses) and PRA models used to perform the risk assessment continue to reflect the as-built,-as-operated plant and that plant modifications and updates to the PRA overtime are continually incorporated.

Programmatic Configuration Control (NEI 00-04, Sections 11 and 12)

Sections 11 and 12 of NEI 00-04, Revision 0, includes discussion on periodic review; and program documentation and change control. Maintaining change control and periodic review will also maintain confidence that all aspects of the 10 CFR 50.69 program and risk categorization for SSCs, continually reflect the River Bend as-built, as-operated plant. A more detailed staff review is provided as follows:

Periodic Review (NEI 00-04, Section 12)

Section 50.69(e) of 10 CFR requires that periodic updates to the licensee's PRA and SSC categorization must be performed. Changes over time to the PRA and to the SSC reliabilities are inevitable and such changes are recognized by the 10 CFR 50.69(e) requirement for periodic updates.

In section 3.2.7 of the enclosure to the LAR, the licensee described the process for maintaining and updating the River Bend PRA models used for the 10 CFR 50.69 categorization process. Consistent with NEI 00-04, the licensee confirmed that the River Bend risk management process ensures that the applicable PRA model(s) used in this application continue to reflect the as-built, as-operated plant. The licensee's process includes provisions for: monitoring issues affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience); assessing the risk impact of unincorporated changes; and controlling the model and associated computer files. The process also includes reevaluating previously categorized systems to ensure the continued validity of the categorization.

Section 12.1 of NEI 00-04, Revision 0, states, in part, "[s]cheduled periodic reviews (e.g. once per two fuel cycles in a unit) should evaluate new insight resulting from available risk information." In Section 3.5 of the enclosure to the LAR, the licensee states, in part, "[s]cheduled periodic reviews at least once every other refueling outage will evaluate new insights resulting from available risk information." Therefore, NRC staff finds the risk management process described by the licensee in the LAR is consistent with section 12 of NEI 00-04, Revision 0 guidance as endorsed by the NRC and therefore satisfies the requirement of 10 CFR 50.69(e)(1). Furthermore, based on the above, the staff has determined that the proposed change satisfies the fifth key principle for risk-informed decision making described in RG 1.174, Revision 3.

Program Documentation and Change Control (NEI 00-04, Section 11)

Section 50.69(f) of 10 CFR requires, in part, program documentation, change control, and records. In section 3.2.7 of the enclosure to the LAR, the licensee stated that it will implement a process that addresses the requirements in section 11 of NEI 00-04, Revision 0, pertaining to program documentation and change control records. In section 3.1.1 of the enclosure to the LAR, the licensee states that the RISC categorization process documentation will include the following ten elements:

- Program procedures used in the categorization
- System functions, identified and categorized with the associated bases

- Mapping of components to support function(s)
- PRA model results, including sensitivity studies
- Hazards analyses, as applicable
- Passive categorization results and bases
- Categorization results including all associated bases and RISC classifications
- Component critical attributes for HSS SSCs
- Results of periodic reviews and SSC performance evaluations
- IDP meeting minutes and qualification/training records for the IDP members

The NRC staff also recognizes that for facilities licensed under 10 CFR Part 50, Appendix B, Criterion VI, for document control procedures, which are considered formal plant documents that require, in part, that "[m]easures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality." The NRC staff finds that the elements provided in section 3.1.1, in addition to the list of implementation items provided in attachment 1 of the enclosure to the LAR, as supplemented, for the River Bend 10 CFR 50.69 categorization process will be documented in formal licensee procedures consistent with section 11 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, and therefore sufficient for meeting the 10 CFR 50.69(f) requirement for program documentation, change control and records.

4.0 CHANGES TO THE OPERATING LICENSE

Based on the NRC staff's review of the LAR, as supplemented, the staff identified specific actions, as described below that are identified as being necessary to support the NRC staff's conclusion that the proposed program meets the requirements in 10 CFR 50.69, the guidance in RG 1.201, Revision 1, and NEI 00-04, Revision 0. Note: Additional actions (e.g., final procedures and proposed alternative treatment) need not, and have not been developed, submitted, or reviewed by the staff for issuance of the SE, but will be completed before implementation of the program as specified in the 10 CFR 50.69 rule.

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 process is conditioned upon the License Condition provided below. For the clarifications to the NEI 00-04, Revision 0 guidance and other changes that were described by the licensee, the NRC staff finds those to be routine and systematically addressed through the configuration management and control and periodic update processes as described in section 3.3 of this SE.

The licensee proposed the following license condition to the RFOL for River Bend. The proposed license condition states:

> Entergy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance

process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Entergy's submittal letter dated February 27, 2023, and all its subsequent associated supplements; as specified in License Amendment No. 214 dated May 16, 2024

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

The NRC staff finds that the proposed license condition is acceptable, because: (1) it adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed by the NRC; and (2) the evaluation in SE section 3.3.2, finds the non-PRA method for assessing risk for passive components, which is a deviation from NEI 00-04, to be acceptable.

The NRC staff notes that the guidance for implementing 10 CFR 50.69 provided by the Commission in the *Federal Register* notice published on November 22, 2004 (69 FR 68008, 68028-68029),[3] Section III.4.10.2, "Section 50.36 Technical Specifications," stated that the 10 CFR 50.69 rule does not include 10 CFR 50.36 in the list of special treatment requirements that may be replaced by the alternative 10 CFR 50.69 requirements for RISC-3 and RISC-4 SSCs when implementing a 10 CFR 50.69 license amendment. As a result, the NRC staff does not consider the TSs (including Improved Technical Specifications and the associated Technical Requirements Manual) to be part of the 10 CFR 50.69 rule. Therefore, the licensee needs to address proposed changes to its TS separately.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendment on April 19, 2024. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of 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, as published in *Federal Register* on July 11, 2023 (88 FR 44166), 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.

[3] Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" Final Rule, published in the *Federal Register* on November 22, 2004 (69 FR 68008, 68028-68029).

7.0 CONCLUSION

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

8.0 REFERENCES

1. Couture, P., Entergy, letter to NRC, "Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors,'" dated February 27, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23058A217).

2. Couture, P., Entergy, letter to NRC, "Supplement to License Amendment Request to Adopt Risk-Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b,' and Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors,'" dated January 12, 2024 (ML24016A248).

3. Marshall, Jr., M. L., NRC, letter to B. Hanson, Exelon Generation Company, LLC "Calvert Cliffs Nuclear Power Plant, Units 1 and 2 – Issuance of Amendment Nos. 332 and 310 RE: Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors (EPID L-2018-LLA-0482)," dated February 28, 2020 (ML19330D909).

4. Drake, J. J., NRC, letter to Entergy, "River Bend Station, Unit 1 – Regulatory Audit Plan in Support of License Amendment Requests to Revise Technical Specifications to Adopt Risk-Informed Competion Times and Implement the Provisions of 10 CFR 50.69 (EPID L-2023-LLA-0037 and EPID L-2023-LLA-0038)," dated October 6, 2023 (ML23278A240).

5. Drake, J. J., NRC, letter to Entergy, "River Bend Station, Unit 1 – Summary of Regulatory Audit in Support of License Amendment Requests to Revise Technical Specifications to Adopt Risk-Informed Completion Times and Implement the Provisions of 10 CFR 50.69 (EPID L-2023-LLA-0037 and EPID L-2023-LLA-0038)," dated May 6, 2024 (ML24102A284).

6. NRC, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," RG 1.201, Revision 1, dated May 2006 (ML061090627).

7. NRC, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," RG 1.200, Revision 2, dated March 2009 (ML090410014).

8. NRC, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," RG 1.174, Revision 3, dated January 2018 (ML 17317A256).

9. NRC, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking,” NUREG 1855, Revision 1, Final Report, dated March 2017 (ML17062A466).

10. NRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800 Section 19.2, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance,” dated June 2007 (ML071700658).

11. NEI, “10 CFR 50.69 SSC Categorization Guideline,” NEI 00-04, Revision 0, dated July 2005 (ML052910035).

12. NRC, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f),” Generic Letter 88-20, Supplement 4, dated June 28, 1991 (ML031150485).

13. ASME/ANS, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” Addendum A to RA-S-2008, PRA Standard ASME/ANS RA-Sa-2009, dated February 2009.

14. NUMARC, “Guidelines for Industry Actions to Assess Shutdown Management,” NUMARC 91-06, dated December 1991 (ML14365A203).

15. Markley, M. T., NRC, letter to Entergy, “Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3, Moderate and High Energy Safety Systems (TAC No. MD5250),” dated April 22, 2009 (ML090930246).

16. EPRI, “Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization,” EPRI Technical Update Report 3002017583, dated February 2020 (ML21082A170).

17. Anderson, V.K., Nuclear Energy Institute, letter to S. Rosenberg, NRC, “Final Revision of Appendix X to NEI 05-04/07-12-12-16, Close-Out of Facts and Observations,” February 21, 2017 (ML17086A431).

18. Zoulis, A.M., NRC, memorandum to M. Franovich, NRC, “Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Risk Assessments,” dated May 6, 2022 (ML22014A084).

19. Idaho National Laboratory/NRC, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” NUREG/CR 6928, dated February 2007 (ML070650650).

20. EPRI, “Human Reliability Analysis (HRA) for Diverse and Flexible Mitigation Strategies (FLEX) and Use of Portable Equipment,” Technical Report 3002013018, dated November 2018.

21. EPRI, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," EPRI Technical Report 1026511, dated December 2012.

22. EPRI, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," EPRI TR-1016737, dated December 2008.

23. Mashburn, W. F., Entergy, letter to NRC, "Entergy Operations Inc. Seismic Hazard and Screening Report (CEUS Sites), Response NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 26, 2014 (ML14091A426).

24. Couture, P., Entergy, letter to NRC, "License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b,'" dated February 27, 2023 (ML23058A215).

25. NRC Staff Review of the River Bend Station Individual Plant Examination of External Events (IPEEE) Submittal, dated June 13, 2001 (ML011640678).

26. NRC, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, Final Report, dated June 1991 (ML063550238).

27. American Society of Mechanical Engineers, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, ASME Code Case, N-660", July 2002.

Principal Contributors: April Pulvirenti, NRR
Angelo Stubbs, NRR
Derek Scully, NRR
Fred Forsaty, NRR
Karen Sida, NRR
Mihaela Biro, NRR
Ming Li, NRR
Sunwoo Park, NRR
Varoujan Kalikian, NRR
Vijay K Goel, NRR

Date: May 16, 2024

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT NO. 214 RE: ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID: L-2023-LLA-0038) DATED MAY 16, 2024

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