Phil Couture
Senior Manager
Regulatory Assurance
601-368-5102

RBG-48143

10 CFR 50.90
10 CFR 50.69

February 27, 2023

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

River Bend Station, Unit 1
NRC Docket No. 50-458
Renewed Facility Operating License No. NPF-47

In accordance with the provisions of 10 CFR 50.69 and 10 CFR 50.90, Entergy Operations, Inc. (Entergy) is requesting an amendment to the license of River Bend Station Unit 1 (RBS). The proposed amendment would modify the RBS licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The enclosure to this letter provides the basis for the proposed change to the RBS Operating License. The categorization process being implemented through this change is consistent with NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, dated July 2005, which was endorsed by the Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, dated May 2006. Attachment 1 of the enclosure provides a list of categorization prerequisites. Use of the categorization process on a plant system will only occur after these prerequisites are met.

Entergy intends to submit a separate request on February 27, 2023 titled "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'" using the same probabilistic risk assessment (PRA) models described in this license amendment request (LAR). Entergy requests that the NRC coordinate their review of the PRA technical adequacy descriptions in Section 3.2 and 3.3 of the enclosure to this letter for both applications. This would reduce the number of Entergy and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action, as the details of the PRA models in each LAR are complete which will allow the NRC to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

A proposed License Condition is described in Section 2.3 of the enclosure to this letter.

Entergy requests approval of the proposed license amendment by April 1, 2024 with the NRC-approved amendment being implemented within 60 days of issuance of the amendment.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), a copy of this license amendment request, with enclosure, is being provided to the designated State Officials.

This letter and its enclosure do not contain any new commitments.

Should you have any questions or require additional information, please contact Mr. Tim Schenk at (225) 381-4177 or tschenk@entergy.com.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on February 27, 2023

Respectfully,

Philip Couture

Digitally signed by Philip Couture
Date: 2023.02.27 11:29:15 -06'00'

Phil Couture
PC/bj

Enclosure: Evaluation of the Proposed Change

Attachments to the Enclosure:

1. List of Categorization Prerequisites
2. Description of PRA Models Used in Categorization
3. Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items
4. External Hazards Screening
5. Progressive Screening Approach for Addressing External Hazards
6. Disposition of Key Assumptions/Sources of Uncertainty

cc: NRC Regional Administrator – Region IV
NRC Project Manager – River Bend Station
NRC Senior Resident Inspector – River Bend Station
Louisiana Department of Environmental Quality

Enclosure

RBG-48143

Evaluation of the Proposed Change

TABLE OF CONTENTS

EVALUATION OF THE PROPOSED CHANGE

1.0 SUMMARY DESCRIPTION

The proposed amendment modifies the licensing basis to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

2.0 DETAILED DESCRIPTION

2.1 CURRENT REGULATORY REQUIREMENTS

The Nuclear Regulatory Commission (NRC) has established a set of regulatory requirements for commercial nuclear reactors to ensure that a reactor facility does not impose an undue risk to the health and safety of the public, thereby providing reasonable assurance of adequate protection to public health and safety. The current body of NRC regulations and their implementation are largely based on a "deterministic" approach.

This deterministic approach establishes requirements for engineering margin and quality assurance in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design basis events (DBEs). The deterministic approach then requires that the facility include safety systems capable of preventing or mitigating the consequences of those DBEs to protect public health and safety. The Structures, Systems and Components (SSCs) necessary to defend against the DBEs are defined as "safety-related," and these SSCs are the subject of many regulatory requirements, herein referred to as "special treatments," designed to ensure that they are of high quality and high reliability and have the capability to perform during postulated design basis conditions. Treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. The distinction between "treatment" and "special treatment" is the degree of NRC specification as to what must be implemented for particular SSCs or for particular conditions. Typically, the regulations establish the scope of SSCs that receive special treatment using one of three different terms: "safety-related," "important to safety," or "basic component." The terms "safety-related" and "basic component" are defined in the regulations, while "important to safety," used principally in the general design criteria (GDC) of Appendix A to 10 CFR Part 50, is not explicitly defined.

2.2 REASON FOR PROPOSED CHANGE

A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety significance, and allowing

consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, Probabilistic Risk Assessments (PRAs) address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures. The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner.

To take advantage of the safety enhancements available through the use of PRA, in 2004 the NRC published a new regulation, 10 CFR 50.69. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The rule contains requirements on how a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages 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 risk-informed safety class (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, as described by NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline" (Reference 1), which uses both risk insights and traditional engineering insights. The safety functions include the design basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability and is a function of the SSC categorization results and associated bases. Finally, periodic assessment activities are conducted to make adjustments to the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable requirements.

The rule 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, the rule enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. For SSCs that are categorized as high safety significant, existing treatment requirements are maintained or enhanced. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, the rule allows an alternative risk-informed approach to treatment that provides a reasonable, though reduced, level of confidence that these SSCs will satisfy functional requirements.

Implementation of 10 CFR 50.69 will allow Entergy to improve focus on equipment that has safety significance resulting in improved plant safety.

2.3 DESCRIPTION OF THE PROPOSED CHANGE

Entergy proposes the addition of the following condition to the renewed operating license of RBS to document the NRC's approval of the use 10 CFR 50.69.

> 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. [XXX] dated [DATE].

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

3.0 TECHNICAL EVALUATION

10 CFR 50.69 specifies the information to be provided by a licensee requesting adoption of the regulation. This request conforms to the requirements of 10 CFR 50.69(b)(2), which states:

> A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:
>
> (i) A description of the process for categorization of RISC–1, RISC–2, RISC–3 and RISC–4 SSCs.
>
> (ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.
>
> (iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i).
>
> (iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).

Each of these submittal requirements are addressed in the following sections.

The PRA models described within this license amendment request (LAR) are the same as those described within the Entergy LAR dated February 27, 2023, "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.'"

Though routine maintenance updates have since been applied, the NRC has previously reviewed the technical adequacy of the RBS internal events PRA model (which includes

internal flooding) identified in this application. The NRC concluded that the model was of sufficient technical adequacy to support the Surveillance Frequency Control Program (Reference 2).

Entergy requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the LAR to adopt TSTF-505, Revision 2. This would reduce the number of Entergy and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action as the details of the PRA models in each LAR are complete which will allow the NRC to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

3.1 CATEGORIZATION PROCESS DESCRIPTION (10 CFR 50.69(b)(2)(i))

3.1.1 Overall Categorization Process

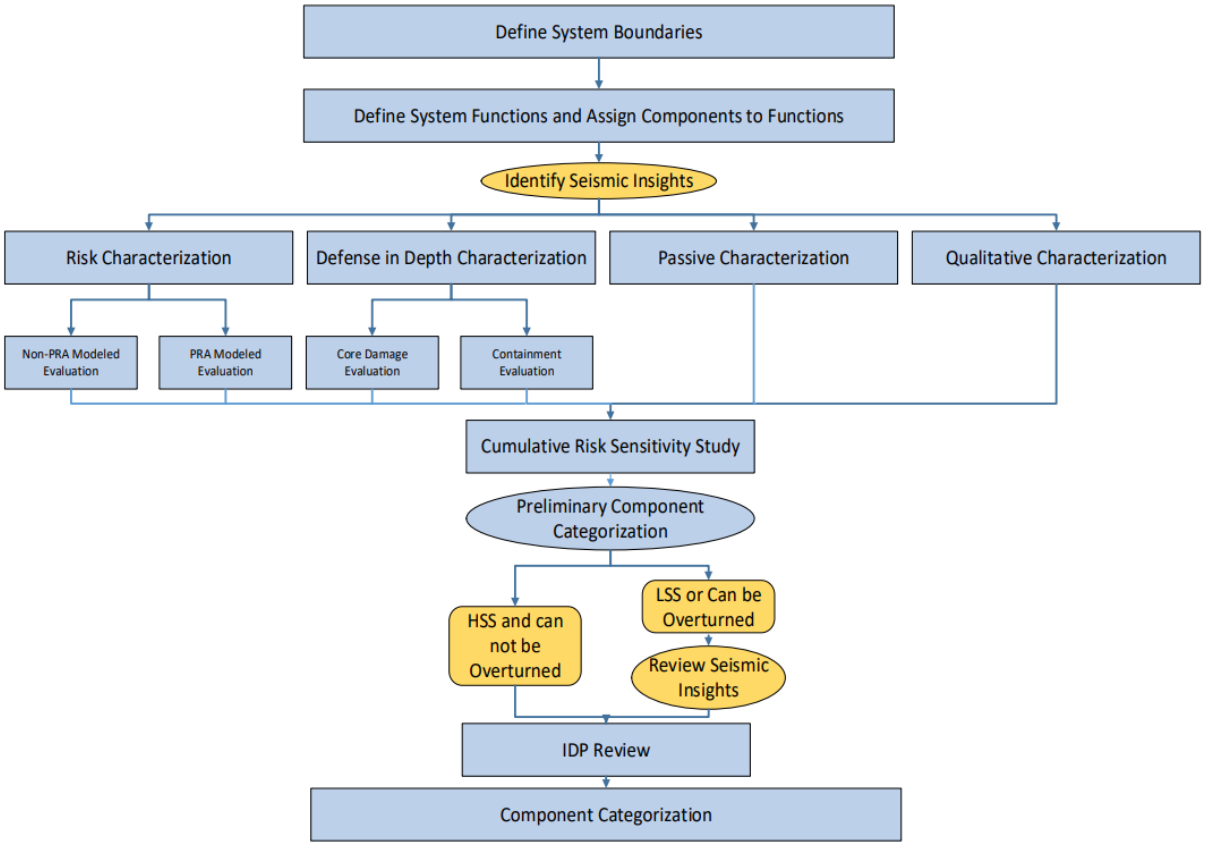
Entergy will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Reference 3). NEI 00-04 Section 1.5 states "Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is assessed separately from each of five risk perspectives and used to identify SSCs that are potentially safety-significant." A separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The process to categorize each system will be consistent with the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201, with the exception of the evaluation of impact of the seismic hazard, which will use the Electric Power Research Institute (EPRI) 3002017583 (Reference 4) approach for seismic Tier 1 sites, which includes RBS, to assess seismic hazard risk for 10 CFR 50.69. Inclusion of additional process steps discussed below to address seismic considerations will ensure that reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv) is achieved. RG 1.201 states 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 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. Therefore, the order in which each of the elements of the categorization process (listed below) is completed is flexible and as long as they are all completed they may even be performed in parallel. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as Low Safety Significant (LSS) by all other elements. Similarly, NEI 00-04 only requires Item 4 to be completed for safety-related active components/functions categorized as LSS by all other elements.

1. PRA-based evaluations (e.g., the internal events, internal flooding, and fire PRAs)
2. Non-PRA approaches (e.g., seismic safe shutdown equipment list, other external events screening, and shutdown assessment)
3. Seven qualitative criteria in Section 9.2 of NEI 00-04
4. The defense-in-depth assessment
5. The passive categorization methodology

Figure 3-1 is an example of the major steps of the categorization process described in NEI 00-04; two steps (represented by four blocks on the figure) have been included to highlight the review of seismic insights as it pertains to this application, as explained further in Section 3.2.3:

Figure 3-1: Categorization Process Overview



Categorization of SSCs will be completed per the NEI 00-04 process, as endorsed by RG 1.201, which includes the determination of safety significance through the various elements identified above. The results of these elements are used as inputs to arrive at a preliminary component categorization (i.e., High Safety Significant (HSS) or LSS) that is presented to the Integrated Decision-Making Panel (IDP). Note: the term "preliminary HSS or LSS" is synonymous with the NEI 00-04 term "candidate HSS or LSS." A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 3-1 below. The safety significance determination of each element identified above is independent of the others, and therefore, the sequence of the elements does not impact the resulting preliminary categorization of each component or function. Consistent with NEI 00-04, the categorization of a component or function will only be "preliminary" until it has been confirmed by the IDP. Once the IDP confirms that the categorization process was followed appropriately, the final Risk Informed Safety Class (RISC) category can be assigned.

The IDP may direct and approve detailed categorization of components in accordance with NEI 00-04, Section 10.2. The IDP may always elect to change a preliminary LSS component or

function to HSS, however, the ability to change component categorization from preliminary HSS to LSS is limited. This ability is only available to the IDP for select process steps as described in NEI 00-04 and endorsed by RG 1.201. Table 3-1 summarizes these IDP limitations in NEI 00-04. The steps of the process are performed at either the function level, component level, or both. This is also summarized in Table 3-1. A component is assigned its final RISC category upon approval by the IDP.

Table 3-1: Categorization Evaluation Summary

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
Risk (PRA Modeled)	Internal Events Base Case – Section 5.1	Component	Not Allowed	Yes
	Fire, Seismic and Other External Events Base Case		Allowable	No
	PRA Sensitivity Studies		Allowable	No
	Integral PRA Assessment – Section 5.6		Not Allowed	Yes
Risk (Non-modeled)	Fire and Other External Hazards	Component	Not Allowed	No
	Seismic	Function/Component	Allowed[2]	No
	Shutdown – Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage – Section 6.1	Function/Component	Not Allowed	Yes
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable[1]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

Notes:
[1] The assessments of the qualitative considerations are agreed upon by the IDP in accordance with NEI 00-04, Section 9.2. In some cases, a 50.69 categorization team may provide preliminary assessments of the seven considerations for the IDP's consideration, however, the final assessments of the seven considerations are the direct responsibility of the IDP.

The seven considerations are addressed preliminarily by the 50.69 categorization team for at least the system functions that are not found to be HSS due to any other categorization step. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that consideration. If the 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS. Conversely, if all the seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing

the preliminary assessment to the same level of detail as the 50.69 team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.

[2] IDP Consideration of seismic insights can also result in an LSS to HSS determination.

The mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components. Specifically, functions with mapped components that are determined to be HSS by the PRA-based assessment (i.e., internal events PRA or integral PRA assessment) or defense-in-depth evaluation will be initially treated as HSS. However, NEI 00-04 Section 10.2 allows detailed categorization which can result in some components mapped to HSS functions being treated as LSS; and Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with a HSS function but which do not support the critical attributes of that HSS function. Note that certain steps of the categorization process are performed at a component level (e.g., passive, non-PRA-modeled hazards – see Table 3-1). Except for seismic, these components from the component level assessments will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped. Components having seismic functions may be HSS or LSS based on the IDP's consideration of the seismic insights applicable to the system being categorized. Therefore, if a HSS component is mapped to an LSS function, that component will remain HSS. If an LSS component is mapped to an HSS function, that component may be driven HSS based on Table 3-1 above or may remain LSS. For the seismic hazard, given that RBS is a seismic Tier 1 (low seismic hazard) plant as defined in Reference 4, seismic considerations are not required to drive an HSS determination at the component level, but the IDP will consider available seismic information pertinent to the components being categorized and can, at its discretion, determine that a component should be HSS based on that information.

If an SSC supports two functions, one being LSS and the other being HSS, the SSC will receive a categorization of HSS. If an SSC supports only an LSS function but is considered HSS from a component level evaluation, the SSC will be considered HSS. The only exception applies to those components that support an HSS function but do not have a credible means to fail the HSS function. These components may be considered LSS in accordance with the guidance in Section 10.2 of NEI 00-04.

NEI 00-04, Sections 4 and 7.1, will be followed for SSCs that support an interfacing system. Those SSCs will typically remain uncategorized until all interfacing systems are categorized. In some cases, impacts that an interfacing component could have on an interfacing system can be fully determined and the interface component can be categorized (and alternative treatment implemented) without categorizing the entire interfacing system. In this event, an assessment of interface component risk associated with uncategorized systems will be limited to cases where the following two conditions are met: 1) the interface SSC failure cannot prevent performance of interface system functions, and 2) the risk is limited to passive failures assessed as LSS, following the passive categorization process for the applicable pressure boundary segments.

The following are clarifications to be applied to the NEI 00-04 categorization process:

- The 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. At 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 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 decision criteria for the IDP for categorizing SSCs as safety significant or low safety significant pursuant to § 50.69(f)(1) will be documented in Entergy procedures. Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. However, a simple majority of the panel is sufficient for final decisions regarding HSS and LSS.

- Passive characterization will be performed using the processes described in Section 3.1.2 of this enclosure. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

- An unreliability factor of 3 will be used for the sensitivity studies described in Section 8 of NEI 00-04. The factor of 3 was chosen as it is representative of the typical error factor of basic events used in the PRA model.

- NEI 00-04, Section 7, requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5, but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. This requirement is further clarified in the Vogtle SER (Reference 5) which states "…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."

- Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The IDP must intervene to assign any of these HSS Function components to LSS.

- With regard to the criteria that considers whether the active function is called out or relied upon in the plant Emergency/Abnormal Operating Procedures, Entergy will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

- RBS proposes to apply an alternative seismic approach to those listed in NEI 00-04 Sections 1.5 and 5.3. This approach is specified in EPRI 3002017583 (Reference 4) for Tier 1 plants and is discussed in Section 3.2.3.

The risk analysis to be implemented for each hazard is described below:

- Internal Event Risks: Internal events and internal flooding PRA models Revision 7, dated December 2022. Revision 6 of this model was accepted by the NRC to support the evaluation of changes to Technical Specification surveillance frequencies that were relocated to the Surveillance Frequency Control Program (Reference 2). The model has since undergone routine maintenance and those changes have been reviewed as described in Section 3.3.

- Fire Risks: Fire PRA model, EC88663 Interim Update, dated January 2021.

- Seismic Risks: Alternative Approach in EPRI 3002017583 (Reference 4) for Tier 1 plants with the additional considerations discussed in Section 3.2.3 of this enclosure.

- Other External Risks (e.g., tornados, external floods): The other external hazards were determined to be insignificant contributors to plant risk. Under the IPEEE (Individual Plant Examination of External Events) program, a systematic reevaluation of selected external hazards was performed (Reference 6). The external hazards from the IPEEE were re-examined in 2022 to ensure the IPEEE conclusions remained bounding and to account for updated information. The results of the 2022 re-examination were that the external hazards (other than seismic, internal fire, and internal flooding) can be screened out; therefore, there is no need for further detailed PRA of these hazards.

- Low Power and Shutdown Risks: Qualitative defense-in-depth (DID) shutdown model for shutdown configuration risk management (CRM) based on the framework for DID provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management" (Reference 7), which provides guidance for assessing and enhancing safety during shutdown operations.

A change to the categorization process that is outside the bounds specified above (e.g., change from the Tier 1 alternate seismic approach described in EPRI 3002017583 to a seismic PRA approach) will not be used without prior NRC approval. The SSC categorization process documentation will include the following elements:

1. Program procedures used in the categorization

2. System functions, identified and categorized with the associated bases

3. Mapping of components to support function(s)

4. PRA model results, including sensitivity studies

5. Hazards analyses, as applicable

6. Passive categorization results and bases

7. Categorization results including all associated bases and RISC classifications

8. Component critical attributes for HSS SSCs

9. Results of periodic reviews and SSC performance evaluations

10. IDP meeting minutes and qualification/training records for the IDP members

3.1.2 Passive Categorization Process

For the purposes of 10 CFR 50.69 categorization, passive components are those components that have a pressure retaining function. Passive components and the passive function of active components will be evaluated using the Arkansas Nuclear One (ANO) Risk-Informed Repair/Replacement Activities (RI-RRA) methodology consistent with the related Safety Evaluation (SE) issued by the Office of Nuclear Reactor Regulation (Reference 8).

The RI-RRA methodology is a risk-informed safety classification and treatment program for repair/replacement activities for pressure retaining items and their associated supports. In this method, the component failure is assumed with a probability of 1.0 and only the consequence evaluation is performed. It additionally applies deterministic considerations (e.g., DID, safety margins) in determining safety significance. Component supports, if categorized, are assigned the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

The use of this method was previously approved by the NRC in the final SE for Vogtle dated December 17, 2014 (Reference 5). The RI-RRA method as approved for use at Vogtle for 10 CFR 50.69 does not have any plant specific aspects and is generic. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, this RI-RRA process categorizes components solely based on consequence, which measures the safety significance of the passive component given that it ruptures. This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment. The passive categorization process is intended to apply the same risk-informed process accepted by the NRC in the ANO2-R&R-004 for the passive categorization of Class 2, 3, and non-class components. This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662 as published in RG 1.147, Revision 15. Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned HSS for passive categorization which will result in HSS for its risk-informed safety classification and cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at RBS for 10 CFR 50.69 SSC categorization.

3.2 TECHNICAL ADEQUACY EVALUATION (10 CFR 50.69(b)(2)(ii))

The following sections demonstrate that the quality and level of detail of the processes used in categorization of SSCs are adequate. The PRA models described below have been peer

reviewed and there are no PRA upgrades that have not been peer reviewed. The PRA models described within this LAR are the same as those described within the Entergy LAR "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.'"

3.2.1 Internal Events and Internal Flooding

The RBS categorization process for the internal events and internal flooding hazard will use the peer reviewed plant-specific PRA model. The Entergy risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant. Attachment 2 of this enclosure identifies the applicable internal events and internal flooding PRA models.

3.2.2 Fire Hazards

The RBS categorization process for fire hazards will use a peer reviewed plant-specific fire PRA model. The internal fire PRA model was developed consistent with NUREG/CR-6850 and only utilizes methods previously accepted by the NRC. The Entergy risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant. Attachment 2 of this enclosure identifies the applicable fire PRA model.

3.2.3 Seismic Hazards

10 CFR 50.69(c)(1) requires the use of PRA to assess risk from internal events. For other risk hazards such as seismic, 10 CFR 50.69(b)(2) allows, and NEI 00-04 summarizes, the use of other methods for determining SSC functional importance in the absence of a quantifiable PRA (such as Seismic Margin Analysis or IPEEE Screening) as part of an integrated, systematic process. For the RBS seismic hazard assessment, Entergy proposes to use a risk-informed graded approach that meets the requirements of 10 CFR 50.69(b)(2) as an alternative to those listed in NEI 00-04 sections 1.5 and 5.3. This approach is specified in EPRI Technical Update 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization" (Reference 4) and includes additional qualitative considerations that are discussed in this section. RBS meets the EPRI 3002017583 Tier 1 criteria for a "Low Seismic Hazard/High Seismic Margin" site. The Tier 1 criteria are as follows:

> Tier 1: Plants where the GMRS [Ground Motion Response Spectrum] peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE [Safe Shutdown Earthquake] between 1.0 Hz and 10 Hz. Examples are shown in Figures 2-1 and 2-2 [in Reference 4]. At these sites, the GMRS is either very low or within the range of the SSE such that unique seismic categorization insights are not expected.

Note: Reference 4 applies to the Tier 1 sites in its entirety except for the sections 2.3 (Tier 2 sites), 2.4 (Tier 3 sites), Appendix A (seismic correlation), and Appendix B (criteria for capacity-based screening).

The Tier 1 criterion (i.e., basis) in EPRI 3002017583 is a comparison of the ground motion response spectrum (GMRS, derived from the seismic hazard) to the safe shutdown earthquake

(SSE, i.e., seismic design basis capability). U.S. nuclear power plants that utilize this approach will continue to compare GMRS to SSE.

The trial studies in EPRI 3002017583 show that seismic categorization insights are overlaid by other risk insights even at plants where the GMRS is far beyond the seismic design basis. Therefore, the basis for the Tier 1 classification and resulting criteria is not that the design basis insights are adequate. Instead, it is that consideration of the full range of the seismic hazard produces limited unique insights to the categorization process. That is the basis for the following statements in Table 4-1 of the EPRI report.

> At Tier 1 sites, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low. Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the FPIE [full power internal events] PRA and other risk evaluations along with the required Defense-in-Depth and Integrated Decision-making Panel (IDP) qualitative considerations are expected to adequately identify the safety-significant functions and SSCs required for those functions and no additional seismic reviews are necessary for 50.69 categorization.

The proposed categorization approach for RBS is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic PRA. For Tier 1 plants, this approach relies on the insights gained from the seismic PRAs examined in Reference 4 along with confirmation that the site GMRS is low. Reference 4 demonstrates that seismic risk is adequately addressed for Tier 1 sites by the results of additional qualitative assessments discussed in this section and existing elements of the 10 CFR 50.69 categorization process specified in NEI 00-04.

For example, the 10 CFR 50.69 categorization process as defined in NEI 00-04 includes an Integral Assessment that weighs the hazard-specific relative importance of a component (e.g., internal events, internal fire, seismic) by the fraction of the total Core Damage Frequency (CDF) contributed by that hazard. For Tier 1 sites, the seismic risk (CDF/Large Early Release Frequency (LERF)) will be low such that seismic hazard risk is unlikely to influence an HSS decision. In applying the EPRI 3002017583 process for Tier 1 sites to the RBS 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the EPRI guidance and informed of plant SSC-specific seismic insights for their consideration in the HSS/LSS deliberations.

EPRI 3002017583 recommends a risk-informed graded approach for addressing the seismic hazard in the 10 CFR 50.69 categorization process. There are a number of seismic fragility fundamental concepts that support a graded approach and there are important characteristics about the comparison of the seismic design basis (represented by the SSE) to the site-specific seismic hazard (represented by the GMRS) that support the selected thresholds between the three evaluation Tiers in the EPRI report. The coupling of these concepts with the categorization process in NEI 00-04 are the key elements of the approach defined in EPRI 3002017583 for identifying unique seismic insights.

The seismic fragility of an SSC is a function of the margin between an SSC's seismic capacity and the site-specific seismic demand. References such as EPRI NP-6041-SL (Reference 9) provide inherent seismic capacities for most SSCs that are not directly related to the site-specific seismic demand. This inherent seismic capacity is based on the non-seismic design loads (pressure, thermal, dead weight, etc.) and the required functions for the SSC. For

example, a pump has a relatively high inherent seismic capacity based on its design and that same seismic capacity applies at a site with a very low demand and at a site with a very high demand. At sites with lower seismic demands such as RBS, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference 9. Low seismic demand sites have lower likelihood of seismically-induced failures and lesser challenges to plant systems. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazards at RBS.

There are some plant features such as equipment anchorage that have seismic capacities more closely associated with the site-specific seismic demand since those specific features are specifically designed to meet that demand. However, even for these features, the design basis criteria have intended conservatisms that result in significant seismic margins within SSCs. These conservatisms are reflected in key aspects of the seismic design process. The SSCs used in nuclear power plants are intentionally designed using conservative methods and criteria to ensure that they have margins well above the required design bases. Experience has shown that design practices result in margins to realistic seismic capacities of 1.5 or more.

The following provides the basis for establishing Tier 1 criteria in EPRI 3002017583:

a. SSCs for which the inherent seismic capacities are applicable, or which are designed to the plant SSE, will have low probabilities of failure at sites where the peak spectral acceleration of the GMRS < 0.2g or where the GMRS < SSE between 1 and 10 Hz.

b. The low probabilities of failure of individual components would also apply to components considered to have correlated seismic failures.

c. These low probabilities of failure lead to low seismic CDF and LERF estimates, from an absolute risk perspective.

d. The low seismic CDF and LERF estimates lead to reasonable confidence that seismic risk contributions would allow reducing an HSS to LSS due to the 10 CFR 50.69 Integral Assessment if the equipment is HSS only due to seismic considerations.

Test cases described in Section 3 of Reference 4 showed that it would be unusual even for moderate hazard plants to exhibit any unique seismic insights, including due to correlated failures. Hence, while it is prudent to perform additional evaluations to identify conditions where correlated failures may occur for Tier 2 sites, for Tier 1 sites such as RBS, correlation studies would not lead to new seismic insights or affect the site risk results in any significant way.

The Tier 1 to Tier 2 threshold as defined in Reference 4 provides a clear and traceable boundary that can be consistently applied plant site to plant site. Additionally, because the boundary is well defined, if new information is obtained on the site hazard, a site's location within a particular Tier can be readily confirmed. In the unlikely event that the RBS seismic hazard changes to medium risk (i.e., Tier 2) at some future time, Entergy will follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e). In accordance with the proposed license condition, Entergy will seek prior NRC approval if it is

determined that a process different from the proposed alternative seismic Tier 1 approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69 (e.g., a change from the Tier 1 approach to a seismic PRA approach).

The following provides the basis for concluding that RBS meets the Tier 1 site criteria.

In response to the NRC 50.54(f) letter associated with post-Fukushima recommendations (Reference 10), RBS submitted a seismic hazard screening report (Reference 11) which concluded that the plant SSE exceeded the GMRS in the 1 to 10 Hz portion of the response spectrum, which is consistent with the Tier 1 criteria in Reference 4.

The NRC's staff assessment of the RBS seismic hazard and screening response is documented in Reference 12. Figure 3.4-1 of that assessment showed that the RBS GMRS shape was very similar to that calculated by the NRC, both of which indicated that the plant SSE exceeded the GMRS in the 1 to 10 Hz portion of the response spectrum. In section 3.4 of Reference 12, the NRC concluded that the methodology used by Entergy in determining the GMRS was acceptable and that the GMRS determined by Entergy adequately characterized the reevaluated hazard for the RBS.

Section 1.1.3 of Reference 4 cites various post-Fukushima seismic reviews performed for the U.S. fleet of nuclear power plants. For RBS, the specific seismic reviews prepared by the licensee and the NRC's staff assessments are provided here.

1. NTTF Recommendation 2.1 seismic hazard screening (References 11,12)
2. NTTF Recommendation 2.3 seismic walkdowns (References 13,14,15,16)
3. NTTF Recommendation 4.2 seismic mitigation strategy assessment (References 17,18)

In addition to the NTTF related seismic studies, a calculation was performed to provide a conservative estimate of the seismic risk to support the Risk-Informed Completion Time (RICT) program LAR. The estimation of the seismic CDF (SCDF) and the seismic LERF (SLERF) is based on the current RBS seismic hazard curve and an estimate of the seismic capacity of a controlling component whose seismic failure would lead directly to core damage. The estimation of SCDF uses the plant level high confidence of low probability of failure (HCLPF) of 0.3g and convolves the corresponding failure probabilities as a function of seismic hazard level with the seismic hazard curve. This is a commonly used approach to conservatively estimate SCDF when a seismic PRA is not available (per section 10-B.9 of PRA Standard ASME/ANS RA-Sa-2009). Using this approach, the SLERF was conservatively estimated by including the containment fragility in the convolution calculation. The conclusion of the calculation was to apply a 9.81E-07/year penalty to CDF and a 4.66E-07/year penalty to LERF to conservatively account for seismic risk when calculating RICTs. This approach is consistent with approaches used in other regulatory applications. The details of the seismic penalty calculation methodology and results are available for NRC audit.

The overall seismic risk is relatively low compared to total plant risk. The seismic risk contributions are consistent with those approved by the NRC for the R.E. Ginna Nuclear Power Plant's adoption of 10 CFR 50.69 (Reference 19). The small seismic risk contribution at RBS makes it unlikely that an integral importance assessment for a component, as defined in NEI 00-04, would result in an overall HSS determination. Further, the low hazard relative to plant

seismic capability makes it unlikely that any unique seismic condition would exist that would cause an SSC to be designated HSS for a Tier 1 site such as RBS.

As an enhancement to the EPRI study results as they pertain to RBS, the proposed RBS 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 RBS. For example, as part of the categorization team's preparation of the System Categorization Document (SCD) that is presented to the IDP, a section will be included in the SCD that summarizes the identified plant seismic insights pertinent to the system being categorized and will also state the basis for applicability of the EPRI 3002017583 study and the bases for RBS being a Tier 1 plant. The discussion of the Tier 1 bases will include such factors as:

- The low seismic hazard for the plant, which is subject to periodic reconsideration as new information becomes available through industry evaluations; and
- The definition of Tier 1 in the EPRI study

At several steps of the categorization process (e.g., as noted in Figure 3-1 and Table 3-1) the categorization team will consider the available seismic insights relative to the system being categorized and document their conclusions in the SCD. Integrated importance measures over all modeled hazards (i.e., internal events, internal flooding, and internal fire) for RBS are calculated per Section 5.6 of NEI 00-04, and components for which these measures exceed the specified criteria are preliminary HSS which cannot be changed to LSS.

For HSS SSCs uniquely identified by the RBS PRA models but having design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events, these will be addressed using non-PRA based qualitative assessments in conjunction with any seismic insights provided by the PRA.

For components that are HSS due to fire PRA but not HSS due to internal events PRA, the categorization team will review design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events and characterize these for presentation to the IDP as additional qualitative inputs, which will also be described in the SCD.

The categorization team will review available RBS plant-specific seismic reviews and other resources such as those identified above. The objective is to identify plant-specific seismic insights derived from the above sources, relevant to the components in the system being categorized, that might include potentially important impacts such as:

- Impact of relay chatter
- Implications related to potential seismic interactions such as with block walls
- Seismic failures of passive SSCs such as tanks and heat exchangers
- Any known structural or anchorage issues with a particular SSC
- Components that are implicitly part of PRA-modeled functions (including relays)
- Components that may be subject to correlated failures

Such impacts would be compiled on an SSC basis. As each system is categorized, the system specific seismic insights will be provided to the IDP for consideration as part of the IDP review process, as noted in Figure 3-1. As such, the IDP can challenge, from a seismic perspective, any candidate LSS recommendation for any SSC if they believe there is basis for doing so. Any decision by the IDP to lower preliminary HSS components to LSS will also consider the applicable seismic insights in that decision. These insights will provide the IDP a means to consider potential impacts of seismic events in the categorization process.

Use of the EPRI approach outlined in Reference 4 to assess seismic hazard risk for 10 CFR 50.69 with the additional reviews discussed above will provide a process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs that satisfies the requirements of 10 CFR 50.69(c).

Based on the above, Section 2.2.3 (Summary/Conclusion/Recommendation) of the EPRI report (Reference 4) applies to RBS, i.e., RBS is a Tier 1 plant for which the GMRS is very low such that unique seismic categorization insights are expected to be minimal. As discussed in Reference 4, the likelihood of identifying a unique seismic insight that would cause an SSC to be designated HSS is very low. Therefore, with little to no anticipated unique seismic insights, the 10 CFR 50.69 categorization process using the full power internal events (FPIE) PRA and other risk evaluations along with the defense-in-depth and qualitative assessment by the IDP adequately identify the safety-significant functions and SSCs.

3.2.4 Other External Hazards

All other external hazards, except for seismic, were screened from applicability to RBS per a plant-specific evaluation in accordance with Generic Letter 88-20 (Reference 20) and updated to use the criteria in ASME/ANS PRA Standard RA-Sa-2009 (Reference 21). The external hazards from the IPEEE were re-examined in 2022 to ensure the IPEEE conclusions remained bounding and to account for updated information. Attachment 4 to this enclosure provides a summary of the external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

As part of the categorization assessment of other external hazard risk, an evaluation will be performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered HSS.

3.2.5 Low Power & Shutdown

Consistent with NEI 00-04, the RBS categorization process will use the shutdown safety management plan described in NUMARC 91-06 (Reference 7) for evaluation of safety significance related to low power and shutdown conditions. The overall process for addressing shutdown risk is illustrated in Figure 5-7 of NEI 00-04.
NUMARC 91-06 specifies that a defense-in-depth approach should be used with respect to each defined shutdown key safety function. The key safety functions defined in NUMARC 91-06 are evaluated for categorization of SSCs.

SSCs that meet either of the two criteria (i.e., considered part of a "primary shutdown safety system" or a failure would initiate an event during shutdown conditions) described in Section 5.5 of NEI 00-04 will be considered preliminary HSS.

3.2.6 Integral Assessment

The internal events and fire PRA models will remain as separate models when calculating importance measures for the 10 CFR 50.69 categorization process. As such, integration of importance measures across all hazards will be performed manually using the methodology specified in NEI 00-04, Section 5.6.

3.2.7 PRA Maintenance and Updates

The Entergy risk management process ensures that the applicable PRA model(s) used in this application continues to reflect the as-built and as-operated plant for RBS (Reference 22). The process delineates the responsibilities and guidelines for updating the PRA models and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, and industry operational experience) for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process will assess the impact of these changes on the plant PRA model in a timely manner. If there is a significant impact on the PRA model, the SSC categorization will be re-evaluated. A significant impact on the PRA model is defined as a greater than 25% increase in CDF or LERF, or significant impacts to basic event importance measures (a factor of three increase in the corrected Birnbaum value of a monitored Mitigating System Performance Index train or component).

In addition, Entergy will implement a process that addresses the requirements in NEI 00-04, Section 11, "Program Documentation and Change Control." The process will review the results of periodic and interim updates of the plant PRA that may affect the results of the categorization process. If the results are affected, adjustments will be made as necessary to the categorization or treatment processes to maintain the validity of the processes. In addition, any PRA model upgrades will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.2.8 PRA Uncertainty Evaluations

Uncertainty evaluations associated with any applicable baseline PRA model(s) used in this application were evaluated during the assessment of PRA technical adequacy and confirmed through the self-assessment and peer review processes as discussed in Section 3.3 of this enclosure.

Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in Section 8 of NEI 00-04 and in the prescribed sensitivity studies discussed in Section 5 of NEI 00-04.

In the overall risk sensitivity studies, Entergy will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference 5. Consistent with the NEI 00-04 guidance, Entergy will perform both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS (i.e., RISC-3) components modeled in all identified PRA models for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures 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 RBS internal events and fire PRA models and documentation were reviewed for plant-specific and generic modeling assumptions and related sources of uncertainty. The process to evaluate uncertainties is defined in NUREG-1855 Rev. 1 (Reference 23), EPRI TR-1026511 (Reference 24), and EPRI TR-1016737 (Reference 25). Each PRA model includes an evaluation of the potential sources of uncertainty for the base case models using the approach that is consistent with the ASME/ANS RA-Sa-2009 requirements for identification and characterization of uncertainties and assumptions.

The uncertainty notebooks for each PRA model were reviewed to identify key assumptions and sources of uncertainty that potentially impact the 50.69 categorization process. A total of 29 items were identified as potentially impacting the 50.69 categorization process. Each of the candidate sources of uncertainty was evaluated for its potential to significantly impact the importance evaluations used for performing SSC categorization. The SSC categorization for 10 CFR 50.69 involves the use of Fussell-Vesely (F-V) and Risk Achievement Worth (RAW) importance measures based on CDF and LERF. In reviewing each of the candidate sources of uncertainty for the internal events/internal flooding and fire PRA models, the following considerations were applied to determine if an impact to 50.69 could exist.

- Criterion #1: Candidate uncertainties that are associated with topics addressed by sensitivity as part of the 10 CFR 50.69 process in accordance with NEI 00-04, such as human error probabilities and common cause factors. The process acknowledges these candidate uncertainties as potentially impacting categorization and accounts for those impacts.
- Criterion #2: Candidate uncertainties that are qualitatively shown to have a very small impact on total risk and would be expected to have a negligible impact on RAW and F-V (particularly uncertainties that pertain to parts of the model that would not impact components that are in the 10 CFR 50.69 program, such as changes to non-support system initiating event frequencies, human error probabilities not related to 50.69-eligible equipment, etc.).
- Criterion #3: Candidate uncertainties that were identified, but for which current industry-accepted approaches and data were used, are not considered key sources of uncertainty. This is consistent with the ASME/ANS PRA Standard definition of a "source of modeling uncertainty" which states: "a source is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model."
- Criterion #4: Candidate uncertainties that were examined via sensitivity studies to confirm that the impact on baseline CDF and LERF and/or the change in importance measures is negligibly small are not considered as key sources of uncertainty for the 50.69 program.

There were 11 potential key assumptions and uncertainties identified from the internal events/internal flooding PRA model. After review of each uncertainty was completed, there was

one key assumption or uncertainty identified that could impact the 10 CFR 50.69 categorization process and will be addressed with a sensitivity study.

For the fire PRA model, there were 18 potential key assumptions and uncertainties identified. After review of each assumption or uncertainty was completed, there were no key assumptions or uncertainties identified that could impact the 10 CFR 50.69 categorization process.
The supplementary list for other potential sources of model uncertainty (items from Table A-3 of EPRI TR-1016737) were assessed and the evaluation determined that none of these items would be a significant source of uncertainty for the RBS 10 CFR 50.69 application.

The key assumptions and uncertainties for the 10 CFR 50.69 application are discussed in more detail in Attachment 6. Attachment 6 also provides discussion on the uncertainties associated with human error probabilities and FLEX, although both of these areas were determined not to be key assumptions or uncertainties with respect to the 10 CFR 50.69 application.

The sensitivity studies described in Attachment 6 will be run as part of the system categorization process and the results will be provided to the IDP as an additional input to their categorization decision.

3.2.9 Modeling of FLEX

Entergy intends to credit the PRA modeling of FLEX equipment during the 10 CFR 50.69 categorization process. Therefore, this section provides background information, a summary of FLEX strategies that are modeled in the PRA, the modeling methodology, the results of an independent review of the modeling, and the results of sensitivity analyses.

Background

The Diverse and Flexible Coping Strategies (FLEX) system is a combination of pre-planned approaches to establish an indefinite coping capability during which key safety functions (core cooling, containment integrity, and spent fuel pool cooling) are maintained during an extended loss of AC power (ELAP) concurrent with a loss of normal access to the ultimate heat sink, as required by NRC Order EA-12-049 (Reference 26). The RBS FLEX program is based upon NEI 12-06 (Reference 27) and is implemented through Entergy programs and procedures. The station's ability to successfully execute FLEX strategies, such as deployment of portable equipment from its storage location to its designated staging point, have been demonstrated and documented. RBS is a "N+1" FLEX strategy plant, meaning there are provisions for on-site spare capability to support the safety functional requirements beyond the minimum necessary to support the single unit.

Modeling of FLEX Equipment and Actions

FLEX equipment is credited in the RBS internal events, internal flooding, and internal fire PRA models. The FLEX equipment modeled in the RBS PRA includes:

- FLX-EG1 and EG2 portable diesel generators, which provide electrical support for powering of the battery chargers for DC power
- FLX-EG3 and EG4 portable diesel generators, which provide electrical support for powering alternate decay heat removal (ADHR) suppression pool cleanup and cooling (SPC) pumps

- FLX-EG5 portable diesel generator, which provides electrical support to power FLEX pump FLX-P1
- FLX-EG7 portable diesel generator, which provides electrical support to power hydrogen igniters
- FLX-P1 FLEX pump and FLX-P2 FLEX pump, which provide service water to the ADHR heat exchanger
- FLX-P3 FLEX pump and FPW-P4 FLEX pump, which provide injection to the reactor pressure vessel

For a declared ELAP condition, abnormal operating procedures direct operators to perform a DC load shed within two hours of the ELAP. The operators would extend DC power during an ELAP condition by cross-connecting the Division II DC battery to Division I within four hours of the ELAP. Additionally, the operators would align FLX-EG1 or FLX-EG2 to power the DC buses within seven hours of the ELAP.

An alternate core cooling method for an ELAP is available should the SPC core cooling path be unavailable following depletion of the upper containment pool and following the loss of a suction source for the reactor core isolation cooling system. This method uses FLEX pump 3 (FLX-P3) or FLEX pump 4 (FPW-P4) to pump water from the standby cooling tower basin, at 200 gpm, through fire hoses to the low-pressure core spray flushing connection and into the vessel. Additionally, a FLEX pump (FLX-P1 or FLX-P2) provides service water to the ADHR heat exchanger.

In addition to the ELAP modeling, the RBS PRA models also credit the use of FLEX equipment in specific loss of decay heat scenarios to prevent a very late (i.e., greater than 16 hours after start of accident) failure of containment. This strategy calls for providing service water to the ADHR heat exchanger via FLX-P1 powered by FLX-EG5 or via FLX-P2. Simultaneously, FLX-EG3 or FLX-EG4 is staged to power the ADHR system to cool the suppression pool. The deployment of this strategy is supported by operator interviews and the latest RBS emergency operating procedures, which provide the operations staff with greater latitude to use FLEX equipment without an ELAP needing to be declared.

Methodology

The FLEX strategies discussed above were incorporated into the PRA models as part of the PRA maintenance process. The incorporation of FLEX strategies into the PRA models does not constitute a PRA model upgrade because modeling of FLEX has been performed in a manner that:

- is consistent with other modeling aspects used in the PRA model and no new methodology was introduced;
- does not result in a change in scope or capability that impacts the significant accident sequences or the significant accident progression.

While the integrated human event analysis system (IDHEAS) method used in the development of the ELAP declaration is new to the PRA model, application of this method is not considered a PRA upgrade, which is consistent with Conclusion 12 contained in a recent NRC memorandum (Reference 28). Therefore, a focused-scope peer review was not required for incorporation of the RBS FLEX strategies into the PRA models.

Following the peer reviewed pre-initiating human reliability analysis (HRA) event screening methodology employed for non-FLEX SSCs in the internal events model, no pre-initiating HRA events unique to the FLEX equipment were developed in the FLEX fault tree model.

FLEX operator actions modeled for implementation of the above strategies are evaluated using technical approaches consistent with the endorsed ASME/ANS RA-Sa-2009 PRA Standard.

The failure to start and failure to run data for the FLEX equipment was developed using the generic values in PWROG-18042-P, Rev. 1 (Reference 29) and NUREG/CR-6928 (Reference 30) for diesel generators and diesel pumps.

Independent Review

In June 2022, an independent review of the RBS FLEX modeling was completed. The purpose of the review was to assess the RBS PRA against the NRC memorandum dated May 6, 2022, "Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Probabilistic Risk Assessments" (Reference 28). The independent review provided four recommendations to gain consistency with the NRC's memorandum. Three of the four recommendations were associated with documentation and one recommendation was related to the modeling of FLEX Pump 1.

FLEX Pump 1 (FLX-P1) is a motor-driven centrifugal pump. Since PWROG-18042-P does not include failure data for motor-driven centrifugal pumps, a conservative multiplier of five was applied to the motor-driven pump failure data from NUREG/CR-6928. The recommendation from the independent review was that a justification should be provided for use of the factor of five multiplier. Entergy will update the appropriate basic events associated with FLX-P1 to increase the multiplier to a factor of ten, which is based upon updated data (i.e., change the multiplier applied to the NUREG/CR-6928 motor-driven pump failure data from five to ten). Entergy will also address the three recommendations related to documentation.

Sensitivity Analysis

A sensitivity analysis was performed to measure the overall risk impact of FLEX by removing credit for FLEX strategies from the internal events and internal flooding PRA models. The result of the sensitivity analysis was that removing credit for FLEX strategies from the RBS Revision 7 model increased CDF by less than 12%.

3.3 PRA REVIEW PROCESS RESULTS (10 CFR 50.69(b)(2)(iii))

The PRA models described in Section 3.2 have been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (Reference 31), consistent with NRC RIS 2007-06 (Reference 32).

Internal Events and Internal Flooding PRA Model

The internal events PRA model was subject to a self-assessment and a full-scope peer review conducted in April 2011 in accordance with NEI 05-04 (Reference 33), PRA Standard ASME/ANS RA-Sa-2009 (Reference 21), and RG 1.200, Revision 2. The April 2011 review was

a full-scope peer review of the technical elements of the internal events (including internal flooding), at-power PRA model.

A focused-scope peer review was conducted in September 2017 covering the LERF and internal flooding related technical elements. The review was conducted in accordance with NEI 05-04, PRA Standard ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2.

In June 2019, a focused-scope peer review was performed of the at-power internal events HRA High Level Requirements HLR-HR-D and HLR-HR-I for specific pre-initiator HEPs modified for the fire PRA.

A finding closure review was conducted on the internal events (including internal flooding) PRA model in August 2020. The closure review team evaluated how RBS addressed the finding-level F&Os from the April 2011 full-scope peer review of the internal events PRA model and the September 2017 focused-scope peer review of the internal flooding and LERF models. Findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13, "Close-out of Facts and Observations (F&Os)" (Reference 34). The independent closure team evaluated changes made to the models in response to the F&Os for determining whether those changes constituted a PRA upgrade or introduced new PRA methods. No F&Os remained open due to changes being characterized as a PRA upgrade. The results of the independent closure review were that two F&Os, of the 40 selected for review, remained open (2017 focused scope peer review F&O 3-3 and 3-4). Note that one open internal flooding F&O (4-5) was not selected for review during this closure review.

In December 2021, a finding closure review was conducted on the three remaining F&Os, which were related to the internal flooding PRA model and were identified during the September 2017 focused-scope peer review. This closure review was performed in accordance with PRA Standard ASME/ANS RA-Sa-2009 and met the documentation requirements of Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13. In addition to assessing the closure status, changes made to the PRA model to address the F&Os were also evaluated. It was determined that no F&Os remained open and there were no changes characterized as a PRA upgrade. The result of the closure review was that all three F&Os were considered closed.

Fire PRA Model

The fire PRA model was subject to a self-assessment and a full-scope peer review conducted in June 2019 in accordance with NEI 07-12 (Reference 35), PRA Standard ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2. The review was conducted against all technical elements in Part 4 of the ASME/ANS PRA Standard.

In March of 2020, a self-assessment was conducted and determined that all finding-level F&Os were successfully resolved, and all associated supporting requirements were met.

A finding closure review was conducted on the RBS fire PRA model in June 2020. The purpose was to perform an independent assessment in accordance with Appendix X of NEI 05-04, NEI 07-12, and NEI 12-13, "Closeout of Facts and Observations (F&Os)." The finding-level F&O dispositions were reviewed by the independent assessment team; the team determined that all F&Os were adequately addressed and were considered closed. The independent assessment team concurred with the RBS self-assessment, which concluded that

the dispositions for the finding-level F&Os were PRA maintenance activities, and none constituted a PRA model upgrade.

Summary

The self-assessments and peer reviews demonstrate that the internal events, internal flooding, and fire PRA models are of sufficient quality and level of detail to support the categorization process and have been subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC as required by 10 CFR 50.69(c)(1)(i).

There are no open finding-level F&Os from the RBS peer reviews of the PRA models. The results of the reviews have been documented and are available for NRC audit.

3.4 RISK EVALUATIONS (10 CFR 50.69(b)(2)(iv))

The RBS 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04. The overall risk evaluation process described in the NEI guidance addresses both known degradation mechanisms and common cause interactions and meets the requirements of §50.69(b)(2)(iv). Sensitivity 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. The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, and human errors). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data and provide timely insights into the need to account for any important new degradation mechanisms.

3.5 FEEDBACK AND ADJUSTMENT PROCESS

If significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can (or actually did) prevent a safety significant function from being satisfied, an immediate evaluation and review will be performed prior to the normally scheduled periodic review. Otherwise, the assessment of potential equipment performance changes and new technical information will be performed during the normally scheduled periodic review cycle.

To more specifically address the feedback and adjustment (i.e., performance monitoring) process as it pertains to the proposed RBS Tier 1 approach discussed in section 3.2.3, implementation of the Entergy design control and corrective action programs will ensure the inputs for the qualitative determinations for seismic continue to remain valid to maintain compliance with the requirements of 10 CFR 50.69(e).

The performance monitoring process will be described in the RBS 10 CFR 50.69 program documents. The program requires that the periodic review assess changes that could impact the categorization results and provides the IDP with an opportunity to recommend categorization and treatment adjustments. Station personnel from engineering, operations, risk management, regulatory affairs, and others have responsibilities for preparing and conducting various performance monitoring tasks that feed into this process. The intent of the performance monitoring reviews is to discover trends in component reliability, to identify and reverse negative performance trends, and take corrective action if necessary.

The Entergy configuration control process ensures that changes to the plant, including a physical change to the plant and changes to documents, are evaluated to determine the impact to drawings, design bases, licensing documents, programs, procedures, and training. The configuration control program will include a checklist of configuration activities to recognize those systems that have been categorized in accordance with 10 CFR 50.69 to ensure that any physical change to the plant or change to plant documents is evaluated prior to implementing those changes.

The checklist includes:

- A review of the impact on the SCD for configuration changes that may impact a categorized system under 10 CFR 50.69.
- Steps to be performed if redundancy, diversity, or separation requirements are identified or affected. These steps include identifying any potential seismic interaction between added or modified components and new or existing safety related or safe shutdown components or structures.
- Review of impact to seismic loading and SSE seismic requirements, as well as the method of combining seismic components.
- Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters Seismic Category I mechanical or electrical components.

Entergy has a comprehensive problem identification and corrective action program that ensures that issues are identified and resolved. Any issue that may impact the 10 CFR 50.69 categorization process will be identified and addressed through the problem identification and corrective action program, including seismic-related issues.

The Entergy 10 CFR 50.69 program requires that SCDs cannot be approved by the IDP until the panel's comments have been resolved to the satisfaction of the IDP. This includes issues related to system-specific seismic insights considered by the IDP during categorization.

Scheduled periodic reviews at least once every other refueling outage will evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process will be updated. This scheduled review will include:

- A review of plant modifications since the last review that could impact the SSC categorization
- A review of plant specific operating experience that could impact the SSC categorization
- A review of the impact of the updated risk information on the categorization process results
- A review of the importance measures used for screening in the categorization process
- An update of the risk sensitivity study performed for the categorization

In addition to the normally scheduled periodic reviews, if a PRA model or other risk information is upgraded, a review of the SSC categorization will be performed.

The periodic monitoring requirements of the 10 CFR 50.69 process will ensure that these issues are captured and addressed at a frequency commensurate with the issue severity. The 10 CFR 50.69 periodic monitoring program includes immediate and periodic reviews, which include the requirements of the regulation, to ensure that all issues that could affect 10 CFR 50.69 categorization are addressed. The periodic monitoring process also monitors the performance and condition of categorized SSCs to ensure that the assumptions for reliability in the categorization process are maintained.

4.0 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The following NRC requirements and guidance documents are applicable to the proposed change.

- The regulations in Title 10 of the Code of Federal Regulations (10 CFR) Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

- NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006

- Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018

- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009

The proposed change is consistent with the applicable regulations and regulatory guidance.

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

Entergy proposes to modify the licensing basis for River Bend Station to allow for the voluntary implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of Structures, Systems and Components (SSCs) subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The process used to evaluate SSCs for changes to NRC special treatment requirements and the use of alternative requirements ensures the ability of the SSCs to perform their design function. The potential change to special treatment requirements does not change the design and operation of the SSCs. As a result, the proposed change does not significantly affect any initiators to accidents previously evaluated or the ability to mitigate any accidents previously evaluated. The consequences of the accidents previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition following an accident will continue to perform their design functions.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not change the functional requirements, configuration, or method of operation of any SSC. Under the proposed change, no additional plant equipment will be installed.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not affect any Safety Limits or operating parameters used to establish the safety margin. The safety margins included in analyses of accidents are not affected by the proposed change. The regulation requires that there be no significant effect on plant risk due to any change to the special treatment requirements for SSCs and that the SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 CONCLUSIONS

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

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Nuclear Energy Institute (NEI), NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, dated July 2005
2. NRC letter to Entergy, "River Bend Station, Unit 1 – Issuance of Amendment Re: Adoption of Technical Specifications Task Force Traveler TSTF-425, Revision 3 (EPID L-2018-LLA-0056)," (ADAMS Accession No. ML19066A008), dated April 29, 2019
3. NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, dated May 2006
4. Electric Power Research Institute (EPRI) Technical Update 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," dated February 2020
5. NRC letter to Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Re: Use of 10 CFR 50.69 (TAC Nos. ME9472 and ME9473)," (ADAMS Accession No. ML14237A034), dated December 17, 2014
6. River Bend Station Engineering Report SEA-95-001, "Individual Plant Examination of External Plants (IPEEE)," dated June 1995
7. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," dated December 1991
8. NRC letter to Entergy, "Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative AN02-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 Systems (TAC No. MD5250)," (ADAMS Accession No. ML090930246), dated April 22, 2009
9. EPRI Technical Report NP-6041-SL, "A Methodology for Assessment of Nuclear Plant Seismic Margin (Revision 1)," dated August 1, 1991
10. NRC letter to All Power Reactor Licensees and Holders of Construction Permits in Active or Deferred Status, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," (ADAMS Accession No. ML12053A340), dated March 12, 2012
11. Entergy letter to NRC, RBG-47453, "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," (ADAMS Accession No. ML14091A426) dated March 26, 2014
12. River Bend Station, Unit 1 – Staff Assessment of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), "Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident (TAC No. MF3706)," (ADAMS Accession No. ML15295A186), dated November 3, 2015

13. Entergy Letter to NRC RBG-47307, "Seismic Walkdown Report – Entergy's Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," (ADAMS Accession No. ML12342A094), dated November 27, 2012

14. Entergy Letter to NRC RBG-47366, "Seismic Walkdown Report Revision 1 – Planned Update to Entergy's Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," (ADAMS Accession No. ML13198A075), dated June 18, 2013

15. Entergy Letter to NRC RBG-47409, "Supplemental Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Hazard Walkdowns Conducted to Verify Current Plant Compliance With the Current Licensing Basis for Seismic Requirements," (ADAMS Accession No. ML13330A999), dated November 21, 2013

16. NRC Letter to Entergy, "River Bend Station, Unit 1 – Staff Assessment of the Seismic Walkdown Report Supporting Implementation of Near-Term Task Force Recommendation 2.3 Related to the Fukushima Dai-ichi Nuclear Power Plant Accident (TAC No. MF0167)," (ADAMS Accession No. ML14073A548), dated May 1, 2014

17. Entergy Letter to NRC, "Seismic Mitigating Strategies Assessment Report River Bend Station – Unit 1," (ADAMS Accession No. ML16251A069), dated August 31, 2016

18. NRC Letter to Entergy, "River Bend Station, Unit 1 – Staff Review of Mitigation Strategies Assessment Report of the Impact of the Reevaluated Seismic Hazard Developed in Response to the March 12, 2012, 50.54(f) Letter," (ADAMS Accession No. ML16259A362), dated September 29, 2016

19. NRC Letter to Constellation Energy, "R.E. Ginna Nuclear Power Plant – Issuance of Amendment No. 151 Re: Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors' (EPID L-2021-LLA-0092)," (ADAMS Accession No. ML22094A107), June 22, 2022

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

21. The American Society of Mechanical Engineers, ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated February 2009

22. Entergy Nuclear Management Manual EN-DC-151, "PRA Maintenance and Update," Revision 9

23. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision making," Revision 1, dated March 2017

24. Electric Power Research Institute (EPRI), Technical Report (TR) 1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," dated December 2012

25. Electric Power Research Institute (EPRI), Technical Report (TR) 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," dated December 2008

26. NRC Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," (ADAMS Accession No. ML12054A735), dated March 12, 2012

27. Nuclear Energy Institute (NEI), NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 0, (ADAMS Accession No. ML12242A378), dated August 2012

28. NRC Memorandum, "Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Probabilistic Risk Assessments," (ADAMS Accession No. ML22014A084) dated May 6, 2022

29. Pressurized Water Reactor Owner's Group (PWROG) report, PWROG-18042-P, "FLEX Equipment Data Collection and Analysis," Revision 1, PA-RMSC-1651 R1, dated August 2021

30. NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants – 2015 Update," dated December 2016, https://nrcoe.inl.gov/publicdocs/AvgPerf/ComponentUR2015.pdf

31. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009

32. NRC Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007

33. Nuclear Energy Institute (NEI), NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2, dated November 2008

34. NEI letter to NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-16 [sic], Close-Out of Facts and Observations (F&Os)," (ADAMS Accession No. ML17086A431), dated February 21, 2017

35. Nuclear Energy Institute (NEI), NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Revision 1, dated June 2010

36. Nuclear Energy Institute (NEI), NEI 17-02, "Tornado Missile Risk Evaluator (TMRE) Industry Guidance Document," Revision 1, dated September 2017

37. NUREG/CR-2300, "PRA Procedures Guide," dated January 1983

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

7.0 ATTACHMENTS

1. List of Categorization Prerequisites

2. Description of PRA Models Used in Categorization

3. Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

4. External Hazards Screening

5. Progressive Screening Approach for Addressing External Hazards

6. Disposition of Key Assumptions/Sources of Uncertainty

Enclosure, Attachment 1

RBG-48143

List of Categorization Prerequisites

List of Categorization Prerequisites

Entergy will implement the four recommendations that were provided from the June 2022 independent review of River Bend FLEX PRA modeling.

Entergy will develop fleet level procedures to outline the process for categorization of plant systems. The Entergy fleet procedures will contain the elements/steps listed below for categorizing systems at River Bend Station.

- Integrated Decision-Making Panel (IDP) member qualification requirements.
- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary High Safety Significant (HSS) or Low Safety Significant (LSS) based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.2 of the enclosure). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting an LSS function are categorized as preliminary LSS.
- Component safety significance assessment. Safety significance of active components is assessed through a combination of Probabilistic Risk Assessment (PRA) and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.
- Assessment of defense-in-depth (DID) and safety margin. Safety-related components that are categorized as preliminary LSS are evaluated for their role in providing DID and safety margin and, if appropriate, upgraded to HSS.
- Review by the IDP. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.
- Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of Regulatory Guide 1.174.
- Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized.
- Documentation requirements per Section 3.1.1 of the enclosure

Enclosure, Attachment 2

RBG-48143

Description of PRA Models Used in Categorization

Description of PRA Models Used in Categorization

Model	Baseline CDF (per year)	Baseline LERF (per year)	Comments
Internal events and internal flooding PRA models Revision 7 dated **December 2022** Peer reviewed against RG 1.200, Rev. 2 (see LAR Section 3.3)	3.03E-06	3.06E-08	NRC reviewed PRA model technical adequacy during the review of River Bend's application to adopt the Surveillance Frequency Control Program (ML19066A008)
Fire PRA model, EC88663 interim update, dated January 2021 Peer reviewed against RG 1.200, Rev. 2 (see LAR Section 3.3)	1.47E-05	7.98E-07	

Enclosure, Attachment 3

RBG-48143

Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

This attachment is not applicable. There are no open peer review findings or open self-assessment items.

Enclosure, Attachment 4

RBG-48143

External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Aircraft Impact	Y	PS4	Bounding analysis demonstrates that the frequency of aircraft-induced radiological consequences is less than 1E-7/yr. Conservatively assuming the Large Early Release Frequency (LERF) is a surrogate for the radiological consequences and that Core Damage Frequency (CDF) is typically an order of magnitude greater than LERF for BWRs, this implies that CDF is less than 1E-6/yr. Therefore, the aircraft impact hazard can be screened out from an external events PRA for River Bend Station (RBS).
Avalanche	Y	C3	Topography is such that no avalanche is possible.
Biological Event	Y	C1	The hypochlorite system inhibits growth and is controlled and monitored. There would be adequate warning for these events. Also note that the Mississippi River is not the Ultimate Heat Sink (UHS) for RBS.
Coastal Erosion	Y	C3	The RBS site is inland just east of the Mississippi River.
Drought	Y	C1 C5	There have been low levels in the Mississippi River; however, the UHS consists of a 200-percent cooling tower located atop a 100-percent capacity water storage facility with sufficient storage capacity to accommodate evaporative and drift losses over a 30-day period. In addition, the plant can receive water from on-site water sources and/or the Mississippi River.
External Flooding and Intense Precipitation	Y	C1 PS1	The external flooding hazard at the site was updated as a result of the post-Fukushima 50.54(f) request for information and the flood hazard

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			reevaluation report (FHRR) was submitted to the NRC for review. The FHRR determined that flood hazard reevaluation water elevations were bounded by the existing design basis, however Local Intense Precipitation (LIP) and Probable Maximum Flood (PMF) for streams and rivers exceeded the current design basis flood protection elevation at the plant site. The NRC's staff assessment of the FHRR agreed with that conclusion and stated that RBS should submit an integrated assessment or a focused evaluation for LIP and PMF. RBS submitted a focused evaluation that determined that there is effective flood protection for maintaining key safety functions during both mechanisms through the demonstration of adequate Available Physical Margin and reliability of flood protection features. Entergy also developed and implemented the mitigating strategies flood hazard assessment (MSA), which was reviewed by the NRC. The updated examination of external flood risk, including the updated plant data, flood history and new measures for risk management validate the current flood mitigation strategy of the current design basis. External flooding events will cause no flooding damage to RBS safety-related structures, systems and components.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Extreme Wind or Tornado	Y	PS4	All Seismic Category I structures are designed for the 100-year wind speed of 100 mph sustained at 30 ft above grade. Tornado loadings are based on a 290-mph rotational wind speed and a 70-mph translational wind speed, with a simultaneous maximum atmospheric pressure drop of 3 psi at a rate of 2 psi/second. Non-Category I structures have been designed to not collapse on or impact Seismic Category I structures. According to the original analysis performed to verify compliance with the licensing basis, there are no openings in the walls or roofs of Seismic Category I structures which could allow a tornado missile to pass through and hit any safety-related targets. Subsequent to the above analysis, some non-conformances were found and evaluated. To assess the threat of missiles penetrating the unprotected areas of RBS, drawings depicting the exterior faces of buildings were reviewed, and walkdowns were performed to identify potential penetration points. The review identified deviations from the design basis requirements. A Tornado Missile Risk Evaluator (TMRE) analysis was performed using NEI 17-02 (Reference 36) methodology to determine the missile strike probability for each of the identified openings. The results of the TMRE analysis were that not providing missile protection to preclude damage related to nonconformances represented a risk increase of 2.15E-10/yr ΔCDF and 4.23E-12/yr ΔLERF, which was

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			sufficiently low to conclude that additional tornado missile protection need not be provided. In summary, RBS has been designed for extreme winds and tornado loadings that are higher than the current regulatory guidance. A TMRE analysis was completed for nonconforming SSCs, and the total risk increase was very small. Therefore, it is concluded that the hazard event of extreme winds and tornadoes can be screened out for RBS.
Fog	Y	C4	Fog affects the frequency of occurrence of other hazards, e.g., highway accidents, aircraft landing and take-off accidents. Fog and heavy fog (<400m visibility) occurred approximately 10 and 1 percent of the time with average fog and heavy fog episodes lasting 4.4 and 2.8 hours, respectively.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Forest or Range Fire	Y	C1 C3 C4	The terrain is heavily wooded with several unnamed intermittent streams crossing and draining to either Grants Bayou on the east or Alligator Bayou on the west of the site boundary. However, the chance of a fire traveling onto the site is minimal due to the areas around the plant being under cultivation. The potential impacts of such a fire would include loss of offsite power (LOOP), which is modeled in the PRA, and smoke and gases entering the control room. If such an event were to occur, operators would have sufficient time to take action, such as donning protective air masks within the control room if the concentration of smoke begins to increase. Also, the main control room air intake header can be manually isolated. Therefore, the release of toxic combustion products from forest fire does not pose a hazard to control room personnel, nor will it cause thermal damage to the RBS safety-related structures.
Frost	Y	C4	Snow and ice govern this hazard.
Hail	Y	C1 C4	Hail may occur, but there are no openings in the walls or roofs of safety related buildings through which hail may enter and damage essential equipment. Tornado missile protection features, structural walls and roofs are adequate to withstand the impact of hail for most components. The UHS is designed to withstand the effects of externally generated and internally generated missiles.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
High Summer Temperature	Y	C1 C5	The highest recorded temperature near Baton Rouge was 109.9°F on August 19, 1909. The maximum allowable cold-water temperature is nominally 95°F, which corresponds to the value assumed for evaluation of the containment heat removal systems. The highest recorded water temperature in Baton Rouge does not approach that value given that water heats and cools more slowly than air. Given the thermal inertia of the concrete structures where safety-related equipment is located, high summer temperatures should not have an impact. The longest period in Baton Rouge with temperatures of 90°F or higher on successive days was 56 days, July 2 through August 24, 1962, and the temperature exceeded 96°F for 16 of those days.
High Tide, Lake Level, or River Stage	Y	C3 C4	Included in external flooding analysis.
Hurricane	Y	C1	The plant is near the Gulf Coast; however, wind speeds generated during hurricanes and other storms are less intense and lower in magnitude than those generated by tornadoes. Thus, plant structures that are designed to satisfy the design criteria for tornadoes will also satisfy the design criteria for those events categorized as "high winds." In addition, a hurricane undergoes significant weakening by the time it reaches the RBS, which is about 70 miles inland.
Ice Cover	Y	C1 C4	Due to the relatively infrequent occurrence of ice cover at RBS, this

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			event screens out from further consideration.
Industrial or Military Facility Accident	Y	C3 PS1	<u>Military Facility Accident</u> There are no military facilities within five miles of RBS. Therefore, this potential contributor to risk was eliminated. <u>Industrial Facility Accident</u> The IPEEE concluded that based on the size and nature of the products, industrial facilities pose no hazard to RBS as a result of potential explosion or fire. The only significant nearby industrial facilities were the Crown-Zellerbach (now owned by Hood Container) papermill located approximately three miles south-southeast of the RBS site and Big Cajun No. 2 Power Plant located across the Mississippi River approximately 2.9 miles southwest of the RBS site. Chlorine was the main concern regarding offsite hazardous materials. It was determined that chlorine detectors were not required at RBS because Crown-Zellerbach agreed to notify RBS within 30 minutes following a spill, which is ample time since it was calculated that it would take two hours for the chlorine concentration in the RBS reactor control room to reach toxic levels. The chlorine containers shipped to Big Cajun No. 2 are of smaller size and therefore not of concern. <u>Flammable Stationary Sources</u> With the exception of several retail gasoline stations located along Hwy 61, no new stationary sources of hazardous materials stored within a

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			five-mile radius of RBS were identified. The IPEEE remains bounding.
Internal Flooding	N	None	The RBS internal events PRA model includes evaluation of risk from internal flooding events.
Internal Fire	N	None	The RBS fire PRA model includes evaluation of risk from internal fires.
Landslide	Y	C3	There are no steep hills in the immediate vicinity of the plant.
Lightning	Y	C1 C4	Lightning is considered in the plant design. It is also considered in the LOOP analysis for the site PRA.
Low Lake Level or River Stage	Y	C1 C5	RBS does not rely on the Mississippi River as the UHS; there are 30 days of water supply in the water storage facility.
Low Winter Temperature	Y	C1	The lowest recorded temperature in Baton Rouge was 2°F on February 13, 1899. January and February average 41.6°F and 45.3°F for lows and 62.3°F and 66.6°F for highs, respectively. The Standby Cooling Tower is relatively insensitive to extreme temperatures. Potential failures due to cold weather are generally limited to full or partial LOOP, which is sufficiently analyzed in the internal events PRA. Severe temperature transients are, therefore, eliminated from any further analysis.
Meteorite or Satellite Impact	Y	C2	This event has a very low frequency of occurrence for any site in the United States.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Pipeline Accident	Y	C3	Several natural gas and hazardous liquid pipelines are present south and east of RBS. They are owned or operated by Transcontinental Pipeline Company, Texas Eastern Transmission Corporation, Colonial Pipeline Company, Mission Natural Gas Company, and Mid Louisiana Gas Transmission Company. All of the pipelines are located beyond the 2-mile search area criterion.
Release of Chemicals in Onsite Storage	Y	C4 PS1	Chemicals on the RBS site were analyzed in accordance with Regulatory Guide 1.78. The RBS Chemical Control Program regulates the use, storage and disposal of chemicals present on site. New chemicals are required to be evaluated for toxic characteristics under this program, and to have a control room habitability evaluation performed, if applicable. The updated survey of onsite chemical source listings was performed most recently in 2018 and no new threats that would challenge control room habitability in event of a postulated accident were found. Therefore, onsite sources of toxic chemicals do not pose a threat to control room habitability and the IPEEE remains bounding.
River Diversion	Y	C3	The Mississippi River is not used as the UHS; therefore, this event will not impact RBS.
Sand or Dust Storm	Y	C3	This is not a relevant hazard for this region.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Seiche	Y	C3	There is no large body of water capable of producing a seiche within 50 miles of the site; therefore, a seiche cannot affect the plant.
Seismic Activity	N	None	Entergy will use a graded approach as discussed in EPRI 3002017583. RBS is a Tier 1 plant, which means the ground motion response spectrum (GMRS) peak acceleration is at or below approximately 0.2g or the GMRS is below or approximately equal to the Safe Shutdown Earthquake between 1.0 Hz and 10 Hz.
Snow	Y	C3	Snow or freezing precipitation is not a serious concern for a power plant in the Baton Rouge area. The maximum 2-day snowfall of 12.5 inches (total) was recorded in February 1895 and there have only been a total of 20 measurable snow days, with measurable being defined as 0.1 inch or more. There have been 97 trace events (less than 0.1 inch) from 1895 to present day. From previous plant design analyses, the weight of the 100-year return snowpack was conservatively assumed to be 5 lb/sq ft for the RBS site. The 48-hr probable maximum winter precipitation (PMWP) for the site was conservatively estimated to be 35 inches water equivalent. Based on the record of historical snow and ice storms for this area, it is unlikely that a significant percentage of the PMWP would fall in a frozen form which would result in roof load exceeding the design live loads.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Soil Shrink-Swell Consolidation	Y	C3	The soil at the site is founded on granular fill, placed directly on the dense sands and gravels of the Citronelle Formation which overlie hard clays and are compacted to an average relative density of 93.8 percent. These are strong, statically and dynamically stable materials. Due to the permeable nature of the granular soils at the site, the soil is resistant to shrink-swell.
Storm Surge	Y	C4	Included in external flooding analysis.
Toxic Gas	Y	C4	Toxic gas is covered under release of chemicals in onsite storage, industrial or military facility accident, and transportation accident.
Transportation Accident	Y	C3 PS1	<u>Shipments by River</u> The Mississippi River is the nearest waterway to the plant, with its eastern bank lying approximately two miles from safety-related structures. Because of this distance, there should be no damage resulting from potential accidents since the criterion for a 5,000-ton river vessel explosion given in Regulatory Guide 1.91 is met. No significant changes in hazardous materials transported by waterway within a five-mile radius of RBS was identified; therefore, the IPEEE remains bounding. <u>Shipments by Rail</u> Two railway companies transport freight within five miles of RBS. The Kansas City Southern company operates a rail spur into the Big Cajun No. 2 power plant across the river in Point Coupee Parish. Canadian National reopened the east-west rail

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			line into the Hood Container facility south of RBS in 2018. There is no new information available concerning hazardous materials shipped by rail. However, the distance of approximately four miles precludes explosive overpressures or toxic gas releases from endangering the plant site. The criteria in Regulatory Guide 1.91 are met. Per the updated hazardous material survey, no new railways were identified near RBS, although several rail lines noted in UFSAR Section 2.2 have since been decommissioned and removed from service. The IPEEE remains bounding. <u>Shipments by Truck</u> The nearest highway on which explosive materials may be transported is US Highway 61, which is a minimum distance of 5,000 feet from the center of the reactor and meets the separation criteria in Regulatory Guide 1.91. Therefore, truck explosions would pose no danger to the site. An updated hazardous material survey was performed in 2018 and it was determined that while road traffic had nearly doubled since the UFSAR Section 2.2 assessment had been done, no new mobile source pathways or significant changes in hazardous materials transported by highway within a five-mile radius of RBS were identified. Available U.S. DOT records show only one accidental release of hazardous materials on highways near RBS between 2015 and 2019 in which a minimal amount of diesel fuel was released approximately one mile east

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			of 2105 Highway 964 (the Hood Container facility) in St. Francisville in February 2015. The IPEEE remains bounding. <u>Toxic Gas from Transient Sources</u> Transient chemicals transported by truck or rail in the RBS vicinity were analyzed and bounded by the discussion of hazardous materials release presented in UFSAR Section 2.2.3.1.3. None of the sources were found to pose a hazard to control room operation. The 2018 hazardous materials survey states that no new mobile source pathways or significant changes in hazardous materials transported by highway, railway, or waterway within a five-mile radius of RBS were identified. Therefore, transient sources of toxic chemicals do not pose a threat to control room habitability and the IPEEE remains bounding.
Tsunami	Y	C3	RBS is located about 70 miles inland and about two miles from the Mississippi River; the site is not exposed to the tsunami threat.
Turbine-Generated Missiles	Y	C1	Analysis determined that for the current design this is not a credible hazard at RBS (UFSAR Section 3.5.1.3).
Volcanic Activity	Y	C3	The site is not close to any active volcanoes.
Waves	Y	C4	Included in external flooding analysis.
Note a – See Attachment 5 for descriptions of the screening criteria.			

Enclosure, Attachment 5

RBG-48143

Progressive Screening Approach for Addressing External Hazards

Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is less than events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
	PS3. Design basis event mean frequency is < 1E-05 per year and the mean conditional core damage probability is < 0.1.	NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009	
	PS4. Bounding mean CDF is < 1E-06 per year.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Enclosure, Attachment 6

RBG-48143

Disposition of Key Assumptions/Sources of Uncertainty

Disposition of Key Assumptions/Sources of Uncertainty

Assumption/ Uncertainty	Discussion	Disposition
The Internal Events PRA model assumes that that Fire Protection Water cannot be aligned for any station blackout sequences due to timing to complete the actions. No credit is taken for the type and timing of the DG failures which could result in the service water valves to be in the desired position to make the action feasible.	The assumption that Fire Protection Water cannot be aligned within the required time criterion for any station blackout sequences is a potentially conservative modeling assumption.	A sensitivity study will be performed as part of the categorization process to determine if any changes in HSS/LSS determination would occur because of the uncertainty of credit for Fire Protection Water in station blackout sequences.
The RBS PRA does not credit potential recovery actions to dewater the Auxiliary Building 70' elevation after flooding events, which would be required to allow use of RBS-FSG-012 actions for FLEX suppression pool cooling. Significant time is available for such actions even though the action has not been proceduralized.	Not crediting non-proceduralized actions is a potentially conservative modeling assumption.	A sensitivity study will be performed as part of the categorization process to determine if any changes in HSS/LSS determination would occur if credit is taken for non-proceduralized flooding recovery actions.
Human error probabilities (HEPs) represent a potentially large uncertainty for the internal events and fire PRA models given the importance of human actions.	Human Reliability Analysis (HRA) uncertainties could have some impact on the risk. Accepted industry methods (e.g., EPRI HRA Calculator) were used to perform the HRA.	This item does not represent a key source of uncertainty in the internal events and fire PRA models but sensitivity to HEP variations will be assessed as part of the 50.69 process per Tables 5-2 and 5-3 of NEI 00-04.

Assumption/ Uncertainty	Discussion	Disposition
Conservative FLEX credit for aligning RCIC to the Upper Containment Pool (UCP)	The PRA assumes that that all scenarios involving credit for the FLEX action to align RCIC suction to the UCP will require manually opening the SFC-MOV139 valve within containment. This MOV is only required to be manually opened if AC power remained available to close it at the time of a RPV Low Low Level (level 2) or Drywell High Differential Pressure isolation signal. Not having to manually open this valve within the degraded containment atmosphere greatly simplifies this action and would accordingly reduce its HEP.	The sensitivity study required by NEI 00-04 to reduce all HEPs to their 5th percentile value can assess the impact of this conservative assumption. Therefore, no additional sensitivity study is needed for the 50.69 process.
Credit for non-standard success paths (e.g., use of alternate injection systems)	SW/RHR Cross-tie and FPW are non-standard injection systems that have been reviewed to determine that sufficient flow and inventory is available. Both SW/RHR Cross-tie and FPW injection are proceduralized in EOPs and Operators are trained on these actions. FPW injection to the RPV to prevent core damage is credited if other injection sources are initially successful and subsequently fail later in the scenario.	As these actions are proceduralized in EOPs and have been evaluated, this is not a key source of uncertainty for 50.69.