July 29, 2024

Vice President, Operations
Entergy Operations, Inc.
Grand Gulf Nuclear Station
P.O. Box 756
Port Gibson, MS 39150

SUBJECT: GRAND GULF NUCLEAR STATION, UNIT 1 – ISSUANCE OF AMENDMENT NO. 233 RE: ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2023-LLA-0080)

Dear Vice President, Operations:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 233 to Renewed Facility Operating License (RFOL) No. NPF-29 for Grand Gulf Nuclear Station, Unit 1 (Grand Gulf). The amendment consists of changes to the RFOL in response to your application dated June 6, 2023, as supplemented by letters dated May 1, 2024 and June 13, 2024.

The amendment revises the Grand Gulf RFOL No. NPF-29 to add a new license condition to allow for implementation of Title 10 of the *Code of Federal Regulations* Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors."

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

Sincerely,

/RA/

Zachary M. Turner, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures:
1. Amendment No. 233 to NPF-29
2. Safety Evaluation

cc: Listserv

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

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

COOPERATIVE ENERGY, A MISSISSIPPI ELECTRIC COOPERATIVE

ENTERGY MISSISSIPPI, LLC

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 233
Renewed License No. NPF-29

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated June 6, 2023, as supplements by letters dated May 1, 2024, and June 13, 2024, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes as indicated in the attachment to this license amendment, and Renewed Facility Operating License No. NPF-29 is hereby amended to add paragraph 2.C.(51) to read as follows:

    (51) 10.CFR.50.69 License Condition

    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 June 6, 2023, and all its subsequent associated supplements; as specified in License Amendment No. 233 dated July 29, 2024.

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

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jennivine K. Rankin

Digitally signed by Jennivine K. Rankin
Date: 2024.07.29 15:19:11 -04'00'

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

Attachment:
Changes to Renewed Facility
Operating License No. NPF-29 and
the Technical Specifications

Date of Issuance: July 29, 2024

ATTACHMENT TO LICENSE AMENDMENT NO. 233

RENEWED FACILITY OPERATING LICENSE NO. NPF-29

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

Replace the following pages of Renewed Facility Operating License No. NPF-29 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating License

Remove	Insert
-22-	-22-
-23-	-23-
-24-	-24-

The results of the three MELLLA+ time-critical operator actions training will be reported to the NRC Project Manager within 60 days of completion of the training.

The reported results will include the full range of response times for each time-critical action and the average times for each crew.

Any MELLLA+ time-critical operator training failures during evaluated scenarios will be reported to the NRC within 60 days of any failures with a plan for resolution.

(50) License Renewal Conditions

(a) The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and licensee commitments as listed in Appendix A of the "Safety Evaluation Report Related to the License Renewal of Grand Gulf Nuclear Station, Unit 1," are collectively the "License Renewal UFSAR Supplement." This Supplement is henceforth part of the UFSAR, which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs, activities, and commitments described in this Supplement, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59, "Changes, Tests and Experiments," and otherwise complies with the requirements in that section.

(b) The License Renewal UFSAR Supplement, as defined in license condition 50(a) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation (PEO).

1. The licensee shall implement those new programs and enhancements to existing programs no later than 6 months prior to the PEO

2. The licensee shall complete those activities by the 6-month date prior to the PEO operation or the last refueling outage prior to the PEO, whichever occurs later

3. The licensee shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

(51) 50.69 License Condition

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 June 6, 2023, and all its subsequent associated supplements; as specified in License Amendment No. 233 dated July 29, 2024.

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

D. The facility required exemptions from certain requirements of Appendices A and J to 10 CFR Part 50 and from certain requirements of 10 CFR Part 100. These include: (a) exemption from General Design Criterion 17 of Appendix A until startup following the first refueling outage, for (1) the emergency override of the test mode for the Division 3 diesel engine, (2) the second level undervoltage protection for the Division 3 diesel engine, and (3) the generator ground over current trip function for the Division 1 and 2 diesel generators (Section 8.3.1 of SSER #7) and (b) exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J for the containment airlock testing following normal door opening when containment integrity is not required (Section 6.2.6 of SSER #7). These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. In addition, by exemption dated December 20, 1986, the Commission exempted licensees from 10 CFR 100.11(a)(1), insofar as it incorporates the definition of exclusion area in 10 CFR 100.3(a), until April 30, 1987 regarding demonstration of authority to control all activities within the exclusion area (safety evaluation accompanying Amendment No. 27 to Renewed License (NPF-29). This exemption is authorized by law, and will not present an undue risk to the public health and safety, and is consistent with the common defense and security. In addition, special circumstances have been found justifying the exemption. Therefore, these exemptions are hereby granted pursuant to 10 CFR 50.12 with the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Physical Security, Safeguards Contingency and Training and Qualification Plan," and were submitted to the NRC on May 18, 2006.

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The licensee's CSP was approved by License Amendment No. 186 as supplemented by a change approved by License Amendment Nos. 192, 200 and 210.

F. EOI shall report any violations of the requirements contained in Section 2, Items C. (1), C.(4) through C.(38) of this renewed license within twenty-four (24) hours. Initial notification shall be made in accordance with the provisions of 10 CFR 50.72 with written follow-up in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

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

H. This renewed license is effective as of the date of issuance and shall expire at midnight on November 1, 2044.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY:

/RA/

William M. Dean, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Attachments 1 and 2
2. Appendix A – Technical Specifications (NUREG-0934)
3. Appendix B – Environmental Protection Plan
4. Appendix C - Antitrust Conditions

Date of Issuance: December 1, 2016

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 233 TO RENEWED

FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated June 6, 2023 (Reference 1), as supplemented by letters dated May 1, 2024 (Reference 2), and June 13, 2024 (Reference 3), Entergy Operations, Inc. (Entergy, the licensee) submitted a license amendment request (LAR) for changes to Renewed Facility Operating License (RFOL) No. NPF-29 for Grand Gulf Nuclear Station, Unit 1 (Grand Gulf).

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

> Entergy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE [Individual Plant Examination of External Events] Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS [American Society of Mechanical Engineers / American Nuclear Society] PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Entergy’s submittal letter dated June 6, 2023, and all its subsequent associated supplements; as specified in License Amendment No. 233 dated July 29, 2024.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

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

The NRC staff participated in a regulatory audit from December 2023 to June 2024 to ascertain the information needed to support its review of the application and develop requests for additional information, as needed. By letters dated May 1, 2024 and June 13, 2024, the licensee responded to the audit providing additional information associated with the NRC questions discussed in the audit. The supplemental letters dated May 1, 2024 and June 13, 2024, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 8, 2023 (88 FR 53541). On June 24, 2024, the NRC staff issued an audit summary (Reference 4).

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

A licensee voluntarily choosing to implement 10 CFR 50.69 shall submit an application for license amendment under 10 CFR 50.90 that contains the information specified in 10 CFR 50.69(b)(2)(i)-(iv).

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

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

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

[1] Regulatory Guide 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006 (Reference 4), describes the SSC categorization process in its entirety. It identifies a variety of acceptable approaches and methods, including PRA as well as some that do not use PRA.

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

2.2 Regulatory Guidance

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

- Regulatory Guide (RG) 1.201, Revision 1, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance,” May 2006 (Reference 5).

- RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities” March 2009 (Reference 6).

- RG 1.200, Revision 3, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities,” December 2020 (Reference 7).

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

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

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

2.3 Applicable NRC Endorsed Guidance

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, “10 CFR 50.69 SSC Categorization Guideline,” dated July 2005 (Reference 11), as endorsed by RG 1.201, Revision 1, for trial use with clarifications, which describes a process that the NRC staff considers acceptable for complying with 10 CFR 50.69.[2] This process determines the safety significance of SSCs and categorizes them into one of four RISC categories defined in 10 CFR 50.69.

Sections 2 through 12 of NEI 00-04 describe the following steps/elements of the SSC categorization process for meeting the requirements of 10 CFR 50.69:

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

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[2] All citations of NEI 00-04 refer to Revision 0. Unless otherwise noted, this means the industry guidance as clarified in RG 1.201, Revision 1, and endorsed by the NRC for trial use.

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

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

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

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

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

- Section 11, “Program Documentation and Change Control,” provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(f).

- Section 12, “Periodic Review,” provides guidance on the periodic review related to the requirements in 10 CFR 50.69(e), “Feedback and process adjustment.”

  Maintaining change control and periodic review provides confidence that all aspects of the program reasonably reflect the current as-built, as-operated plant configuration and applicable plant and industry operational experience as required by 10 CFR 50.69 (c)(1)(ii).

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

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

Principle 1: The proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption (i.e., a specific exemption under 10 CFR 50.12).

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

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

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

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

3.2 Traditional Engineering Evaluation

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

Key Principle 1: Licensing Bases Change Meets the Current Regulations

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

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

The SSCs are classified as having either HSS functions (i.e., RISC-1 and RISC-2 categories) or LSS functions (i.e., RISC-3 and RISC-4 categories). For SSCs that are HSS, 10 CFR 50.69 maintains current regulatory requirements for special treatment (i.e., it does not remove any requirements from these SSCs). For SSCs that are LSS, licensees can implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d). For SSCs in RISC-3, licensees can replace special treatment with an alternative treatment. For SSCs in RISC-4, 10 CFR 50.69 does not impose new treatment requirements.

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

(i) 10 CFR Part 21
(ii) a portion of 10 CFR 50.46a(b)
(iii) 10 CFR 50.49
(iv) 10 CFR 50.55(e)
(v) specified requirements of 10 CFR 50.55a

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

(vi) 10 CFR 50.65, except for paragraph (a)(4)
(vii) 10 CFR 50.72
(viii) 10 CFR 50.73
(ix) Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50
(x) specified requirements for containment leakage testing
(xi) specified requirements of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100

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

Section 2 of NEI 00-04, Revision 0, states in part, that the categorization process includes eight primary steps:

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

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

The licensee provided further discussion of specific elements within the 10 CFR 50.69 categorization process that are delineated in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

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

Based on the above, the NRC staff concludes that the proposed 10 CFR 50.69 program to implement risk-informed categorization and treatment of SSCs meets the first key principle prescribed in RG 1.174, Revision 3, for risk-informed decision-making.

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

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

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

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

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

Key Principle 3: Licensing Basis Change Maintains Sufficient Safety Margins

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

The licensee did not request approval of changes to the SSCs design basis functions described in the plant's licensing basis, including the Updated Final Safety Analysis Report. Similarly, the licensee did not request approval of changes to safety analyses acceptance criteria described in the plant licensing basis. On this basis, the NRC staff concludes that safety margins are maintained by the proposed methodology, and therefore, satisfies the third key safety principle of RG 1.174, Revision 3.

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

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

The NRC staff finds that the process described in the LAR is consistent with NEI 00-04, Revision 0. RG 1.201, Revision 1, states, in part, "[t]his trial regulatory guide provides interim guidance for complying with the NRC's requirements in [10 CFR] 50.69, by using the process described in Revision 0 of NEI 00-04 to determine the safety significance of SSCs and place them into the appropriate RISC categories." Because the process described in the LAR meets the requirements of NEI 00-04, the NRC staff finds that the process meets the requirements set forth in 10 CFR 50.69(c)(1)(ii) and 10 CFR 50.69(c)(1)(iv).

3.3 Risk-Informed Assessment

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

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

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

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

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

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

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

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

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

In sections 3.2.1, "Internal Events and Internal Flooding," and 3.2.2, "Fire Hazards," of the enclosure to its LAR, the licensee described that the Grand Gulf categorization process uses a peer reviewed plant-specific PRA model to assess risks for internal events (including internal flooding) and internal fires. These are consistent with the approaches and methods included in NEI 00-04, Revision 0. For the other risk contributors, the licensee's process uses non-PRA methods to characterize the risk:

- Seismic Hazard: As described in section 3.2.3 of the enclosure to the LAR, an alternative seismic approach presented in EPRI Technical Update 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization" (Reference 12, "the EPRI Report" hereafter) is used.

  The EPRI alternative approach is not included in NEI 00-04, Revision 0, and was not endorsed by the NRC as a way to address the seismic hazard in the SSC categorization process. A detailed NRC staff review of the licensee's proposed alternative seismic approach is provided in section 3.3.2 of this SE.

- Other External Hazards: As described in section 3.2.4 of the enclosure to the LAR, a screening analysis performed for the IPEEE (Reference 13) was updated using criteria from ASME/ANS RA-Sa–2009, “Addendum A to RA-S–2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications” (Reference 14, the ASME/ANS PRA Standard), as endorsed by the NRC. More specifically, the update used criteria in part 6, “Requirements for Screening and Conservative Analysis of Other External Hazards At-Power.”

- Low Power and Shutdown Hazard**:** As described in section 3.2.5 of the enclosure to the LAR, a safe shutdown risk management program is used consistent with Nuclear Management and Resources Council (NUMARC) 91-06, “Guidelines for Industry Actions to Assess Shutdown Management” (Reference 15).

- Passive Components**:** A non-PRA method for the categorization of passive components is used consistent with the ANO-2 methodology for passive components. This was approved for risk-informed safety classification and treatment for repair/replacement activities in ASME Boiler and Pressure Vessel Code (ASME Code) class 2 and 3 moderate- and high-energy systems (Reference 16).

  The NRC staff evaluation of the ANO-2 methodology in the SSC categorization process for Grand Gulf is provided in section 3.3.2 of this SE.

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

3.3.1 Scope of the PRA

The Grand Gulf PRA comprises a full-power, IEPRA (including internal flooding), and FPRA. Each one of these assessments evaluates risk metrics of core damage frequency (CDF) and large early release frequency (LERF).

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

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

IEPRA (Including Internal Flooding) Peer-Review History

In section 3.3, "PRA review Process Results (10 CFR 50.69(b)(2)(iii))" of the enclosure to its LAR, the licensee reported that the IEPRA model (including internal flooding) had been peer reviewed in September 2015 against the ASME/ANS PRA Standard; RG 1.200, Revision 2; and NEI 05-04 (Reference 17). All finding-level facts and observations (F&Os) from the peer-review were closed in August 2017 according to the process documented in the NEI letter to the NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-13, 'Close-out of Facts and Observations,'" (Reference 18), and as accepted with conditions by NRC letter dated May 3, 2017 (Reference 19). In section 3.2, "Technical Adequacy Evaluation (10 CFR 50.69(b)(2)(ii))," of the enclosure to its LAR, for both the IEPRA (including internal flooding) and FPRA, the licensee stated that "there are no PRA upgrades that have not been peer reviewed."

In review of the licensee's reports, the NRC staff concluded that all finding-level F&Os were appropriately assessed by the independent assessment team to assure that no new methods or upgrades were inadvertently incorporated into the IEPRA without a peer-review in accordance with the ASME/ANS PRA Standard, as endorsed by the NRC.

Therefore, the NRC staff concludes that the Grand Gulf IEPRA (including internal flooding) were appropriately peer reviewed, consistent with RG 1.200, Revision 2, and the F&Os have been adequately closed.

Internal FPRA Peer-Review History

The licensee's FPRA was subject to a peer-review in July 2022. It was reviewed against the ASME/ANS PRA Standard; RG 1.200, Revision 3; and NEI 17-07, Revision 2 "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard" (Reference 20). All finding-level F&Os from the peer-review were closed in April 2023 in accordance with appendix E of NEI 17-07, as endorsed by NRC in RG 1.200, Revision 3. In the enclosure to its LAR, the licensee stated that "[t]he independent assessment team […] concluded that the dispositions for the finding-level F&Os were PRA maintenance activities, and none constituted a PRA model upgrade." In review of the licensee's reports, the NRC staff concluded that all finding-level F&Os were appropriately assessed by the independent assessment team to assure that no new methods or upgrades were inadvertently incorporated into FPRA without a peer-review in accordance with the ASME/ANS PRA standard, as endorsed by the NRC.

In the LAR, as supplemented, the licensee states that the Grand Gulf plant-specific FPRA model was developed consistent with NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities" (Reference 21), and only utilizes methods accepted by the NRC (e.g., frequently asked questions, NUREGs, or interim guidance documents). More specifically,

- Reduced NUREG/CR-6850 transient heat release rates were not credited for any fire areas.

- To ensure consistency with NUREG-2178, "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE)," (Reference 22), the fire modeling logic used for Grand Gulf prohibits the use of the obstructed plume model when the base of the fire is located at an elevation that is less than half of the electrical enclosure height.

- The FPRA contains a set of basic events for which the component cable routing is not known, developed, or assumed, and therefore, these basic events are assumed to be failed for every fire scenario, except for main control room abandonment scenarios where, if credited, the component circuit is known to be isolated from fire damage. The treatment of these components in the FPRA is not a source of model uncertainty because the treatment is consistent with the accepted methods and industry consensus for developing the FPRA. Additionally, a sensitivity analysis was performed for the change in the Fussell-Vesely (FV) and risk achievement worth (RAW) risk importance measures assuming the associated cables are not failed for every fire scenario while retaining the associated random failures. The resulting changes in FVs and RAWs were small and expected to have an inconsequential impact on SSC categorization under the 10 CFR 50.69 program.

- Bin 15 electrical cabinets that are “well-sealed, robustly-secured,” with circuits limited to less than 440 volts, were screened from the ignition source count during the ignition source walkdowns in accordance with the guidance in NUREG/CR-6850.

Based on the discussion above, the NRC staff concludes that the Grand Gulf FPRA was appropriately peer reviewed consistent with RG 1.200, Revision 3, and the F&Os have been adequately closed utilizing methods acceptable to the NRC.

Appendix X, Independent Assessment Process for F&O Closure

Appendix X of NEI 05-04/07-12/12-13, as accepted with conditions by NRC letter dated May 3, 2017, and Appendix E of NEI 17-07, as endorsed by NRC in RG 1.200, Revision 3, provide guidance to perform an independent assessment for the closure of F&Os identified from a full-scope peer-review.

In review of the LAR, the NRC staff concluded that all F&Os were appropriately assessed by the independent assessment team to assure that no new methods or upgrades were inadvertently incorporated into the IEPRA (including internal flooding) or FPRA without a peer-review in accordance with the ASME/ANS PRA standard as endorsed by the NRC. Therefore, the NRC staff finds that the Grand Gulf IEPRA (including internal flooding) and FPRA were appropriately peer reviewed consistent with RG 1.200, Revisions 2 and 3, respectively, and meet the requirements set forth in 10 CFR 50.69(c)(1)(i).

Assessment of PRA Model Assumptions and Approximations

Identification of Key Assumptions and Sources of Uncertainty

In the LAR, as supplemented, the licensee stated that NUREG-1855, Revision 1, was performed to identify, screen, and characterize those sources of model uncertainty and related assumptions in the base PRA that are relevant to this application. Substep E-1.4 of the guidance in NUREG-1855, is a qualitative screening process that involves identifying and validating whether consensus[4] models have been used in the PRA to evaluate identified model uncertainties. The licensee confirmed that for the Grand Gulf uncertainty analysis, some uncertainties and assumptions were screened based on the use of a consensus method. The

[4] Per NUREG-1855, Revision 1, a consensus model is a model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group.

NRC staff finds that the assessment performed to identify the key assumptions/sources of uncertainty is consistent with the guidance provided in NUREG-1855, Revision 1.

Treatment of the Key Assumptions and Sources of Uncertainty

NUREG-1855, Revision 1, provides guidance regarding how to address PRA uncertainties to assure the risk-informed decision is in the context of the application for the decision under consideration. The licensee confirmed in the LAR, as supplemented, that sensitivity studies will be performed consistent with NEI 0004, Revision 0 guidance. In accordance with NEI 0004, the results of the sensitivity studies are given to the IDP for consideration in the final risk characterization for components initially classified as LSS that may be reclassified to HSS. The NRC staff finds that the licensee's approach to treatment of PRA key assumptions and sources of uncertainty is consistent with the NEI 0004 guidance.

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

PRA Importance Measures and Integrated Importance Measures

The scope of modeled hazards for Grand Gulf includes the IEPRA (including internal flooding) and FPRA. The NRC staff finds that the licensee's use and treatment of importance measures is consistent with the guidance in NEI 00-04, Revision 0**.** A more detailed staff review of the non-PRA methods for assessing the risk for seismic, other hazards, shutdown, and passive components is provided in section 3.3.2 of this SE.

Credit for Diverse and Flexible Mitigation Capability (FLEX) Equipment

Per 10 CFR 50.69(c)(1)(i) and NEI 00-04, Revision 0, guidance, the results of sensitivity studies that were identified in the characterization of PRA technical acceptability should be given to the IDP for consideration in the final risk characterization for components initially classified as LSS that may be reclassified to HSS. The NRC's memorandum dated May 6, 2022, "Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Risk Assessments" (Reference 23), on industry guidance for incorporating diverse and flexible coping strategies (FLEX) into PRAs provides the NRC staff's assessment of challenges to incorporating FLEX into a PRA model in support of risk-informed decision-making.

In its LAR, as supplemented, the licensee indicated that FLEX is not credited in the IEPRA (including internal flooding), but is credited in the FPRA for Extended Loss of AC Power scenarios, supporting actions such as (1) reactor core isolation cooling injection from the upper containment pool, (2) low pressure injection, (3) powering the hydrogen igniters, and (4) powering the safety relief valves to manually depressurize the reactor. Specific portable FLEX equipment and human actions modeled in the FPRA are detailed in the LAR, as supplemented.

The NRC's May 6, 2022, memorandum endorsed portable FLEX equipment failure rates from Pressurized Water Reactor Owners Group (PWROG)-18042, "FLEX Equipment Data Collection and Analysis" (Reference 24), yet the licensee used rates from NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants" (Reference 25), for permanently installed equipment, introducing uncertainty in SSC categorization. In order to address this key source of uncertainty consistent with NEI 00-04, Revision 1, guidance, the licensee proposed in the LAR, as supplemented, to provide the IDP with results of a sensitivity study utilizing PWROG-18042 portable equipment failure rates for each categorized system. This uncertainty would be considered resolved, and the associated sensitivity study would not be required once the FPRA model is updated to include the PWROG-18042 failure rates. The licensee stated that the potential impact from uncertainties associated with FLEX operator actions are inherently addressed with the standard human reliability analyses sensitivities specified in NEI 00-04. In its LAR supplement dated May 1, 2024, the licensee stated that the modeling of FLEX was peer reviewed as part of the peer-review of the FPRA (refer to section 3.3.1 of this SE) in support of risk-informed decision-making in accordance with the guidance of RG 1.200.

Based on the above discussion, the NRC staff finds that the uncertainties associated with the PRA modeling of FLEX equipment and operator actions have been adequately addressed consistent with NEI 00-04, Revision 0, guidance, as endorsed by RG 1.201, Revision 1, to support the Grand Gulf 10 CFR 50.69 categorization process.

PRA Acceptability Conclusions

Pursuant to 10 CFR 50.69(c)(1)(i), the categorization process must consider results and insights from a plant-specific PRA. The use of PRA for IEPRA, (to include internal flooding) and FPRA to support SSC categorization is endorsed by RG 1.201, Revision 1. Furthermore, pursuant to 10 CFR 50.69(c)(1)(i), the PRAs must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer-review process assessed against a standard that is endorsed by the NRC. RG 1.200, Revision 2 and Revision 3, provides guidance for determining the acceptability of the PRA by comparing the PRA to the relevant parts of the ASME/ANS PRA Standard using a peer-review process.

The licensee has subjected the IEPRA (including internal flooding) and FPRA to the peer-review processes and submitted the results of the peer-review. The NRC staff reviewed the peer-review history (which included the results and findings), the licensee's resolution of peer-review findings, and the identification and disposition of key assumptions and sources of uncertainty. The staff concludes that (1) the Grand Gulf IEPRA (including internal flooding) and FPRA are acceptable to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201, Revision 1, and (2) the key assumptions for the PRAs have been identified consistent with the guidance in RG 1.200, Revision 2 and Revision 3, and NUREG-1855 and addressed appropriately for this application.

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

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

The licensee's categorization process uses the following non-PRA methods, respectively:

- Seismic Hazard: Alternative seismic approach described in the EPRI Report 3002017583.

- Other External Hazards: Screening analysis performed for the IPEEE and updated using the external hazard screening significance process identified in the ASME/ANS PRA Standard.

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

- Passive Components: ANO-2 passive categorization methodology.

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

Seismic Hazard

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

The licensee based its plant-specific evaluation, in part, on the case studies performed in EPRI Report 3002017583 and stated that the case studies are applicable to Grand Gulf and are used in its proposed alternative seismic approach. EPRI Report 3002017583 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from seismic PRAs (SPRAs). The NRC staff's review confirmed that the case studies in EPRI 3002017583 used by the licensee as well as the information in its supplements provided sufficient plant-specific basis for applicability of its proposed alternative seismic approach to Grand Gulf. Accordingly, EPRI Report 3002017583 and its cited case studies previously approved by the NRC staff (Reference 26) and information presented in the LAR, as supplemented, provide a sufficient description of and basis for acceptability of the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv). Therefore, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(iv) are met for the licensee's proposed alternative seismic approach.

Evaluation of the Criteria for the Proposed Alternative Seismic Approach

In section 3.2.3 of its LAR, the licensee states, in part, that the ground motion response spectrum (GMRS) peak spectral acceleration for Grand Gulf is below the safe shutdown earthquake (SSE) between 1 hertz (Hz) and 10 Hz, which demonstrates that Grand Gulf qualifies as a Tier 1 plant under the criteria in EPRI Report 3002017583. The NRC staff notes that the licensee's plant-specific evaluation is supported by its response to the NRC 10 CFR 50.54(f) request dated March 31, 2014 (Reference 27) in that the plant SSE exceeded the GMRS in the 1 to 10 Hz range of the response spectrum. The NRC staff reviewed the LAR, as supplemented, and the licensee's plant-specific evaluation and concluded that the use of the criteria in EPRI Report 3002017583 to determine the applicability and use of the proposed seismic Tier 1 approach is acceptable.

Evaluation of the Applicability of the Proposed Alternative Seismic Approach to 10 CFR 50.69 Categorization at Grand Gulf

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

The NRC staff's evaluation of seismic risk to total plant risk was based on information in the Grand Gulf Technical Specifications Task Force (TSTF) Traveler TSTF-505 LAR (Reference 28). The purpose of the seismic risk estimate in the Grand Gulf TSTF-505 LAR is to provide a conservative estimate for use in calculating risk-informed completion times for technical specifications. The NRC staff determined that the licensee used conservative seismic CDF and LERF for the Grand Gulf TSTF-505 LAR. Further, as noted in section 3.6.5, "Defense-in-Depth Assessment," of EPRI Report 3002017583, containment DID assessment addresses containment failures and containment bypass situations. Section 3.6.6, "Civil Structures," of EPRI Report 3002017583, used for the licensee's proposed alternative seismic approach, recommends that if the licensee chooses to categorize civil structures housing HSS SSCs, the structures are considered as HSS. Therefore, based on its evaluation and review, the NRC staff concludes that the proposed alternative seismic approach, in conjunction with the other elements of the 10 CFR 50.69 categorization program, will appropriately determine the safety significance of any SSCs whose seismic-induced failures would lead directly to core damage and large early release. Furthermore, the NRC staff finds that contribution from seismic risk alone to the 10 CFR 50.69 categorization program would not result in any additional SSCs being categorized as HSS.

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

Evaluation of the EPRI 3002017583 Case Studies

In its supplement to its LAR dated May 1, 2024, the licensee stated that the plant-specific case studies from other licensees included in EPRI Report 3002017583 are incorporated by reference to support its proposed alternative seismic approach. The licensee also stated in the supplement that there are no differences between the proposed Grand Gulf alternative seismic approach and that, reviewed and approved in the NRC staff's SE on Calvert Cliffs 10 CFR 50.69 LAR (Reference 26). The NRC staff finds that the acceptability of PRAs used in Plants A, C, and D case studies in EPRI Report 3002017583, the mapping approach used in those case studies, and the conclusions on the determination of unique HSS SSCs from the case studies, which were reviewed and approved by the NRC staff and used as technical basis for the Calvert Cliffs 10 CFR 50.69 license amendment, are applicable to the proposed alternative seismic approach for Grand Gulf.

Evaluation of the Implementation of Conclusions from the Case Studies

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

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

1. The proposed alternative seismic approach includes qualitative consideration of seismic events at several steps of the categorization process, including documentation of the information for presentation to the IDP as part of the integrated, systematic process for categorization.

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

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

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

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

Consideration of Changes to Seismic Hazard

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

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

Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach

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

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

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

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

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

External Hazards and Other Hazards (Non-Seismic)

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

In the safety evaluation report for the Grand Gulf IPEEE (Reference 29), the NRC staff confirmed that the high winds, floods, and other external events (HFO) were eliminated based on conformance with the criteria in the 1975 NRC Standard Review Plan using the progressive screening approach described in NUREG-1407 (Reference 30), a procedural and submittal guidance for Generic Letter 88-20, Supplement 4.

The licensee confirmed that Grand Gulf will subject the external hazards (except for seismic) to the process described by the flow chart in NEI 00-04, Revision 0, figure 5-6, which provides guidance to be used to determine SSC safety significance for these external hazards. The NRC staff finds that Grand Gulf will assess the risk from all other external hazards consistent with figure 5-6 of NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1.

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

Shutdown Risk

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

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

Component Safety Significance Assessment for Passive Components

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

In section 3.1.2, “Passive Categorization Process,” of the enclosure to its LAR, the licensee proposed using a categorization method, for passive components not cited in either NEI 00-04, Revision 0, or RG 1.201, Revision 1, for passive component categorization, but was approved by the NRC for ANO-2 (Reference 16). The ANO-2 methodology is a risk-informed safety classification and treatment program for repair/replacement activities for Class 2 and 3 pressure retaining items and their associated supports (exclusive of Class CC and MC items), using a modification of the ASME Code Case N-660, “Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1” (Reference 31). The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing passive components solely based on consequences, which measures the safety significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment. Therefore, the NRC staff finds that the use of the ANO-2 repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs at Grand Gulf.

In section 3.1.2, of the enclosure to its LAR, the licensee stated, in part, “[t]he 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.” Consistent with ANO2-R&R-004, Class 1 pressure retaining SSCs in the scope of the system being categorized will be assigned HSS and cannot be changed by the IDP. That is, the ANO-2 repair/replacement methodology does not allow a Class 1 pressure retaining SSC to be recategorized from HSS to LSS. Therefore, the NRC staff finds the licensee’s proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process for Class 1, Class 2, and Class 3 pressure retaining SSCs.

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

In section 3.1.1 of the enclosure to its LAR, the licensee states, in part, that “[a]n unreliability factor of 3 will be used for the sensitivity studies described in Section 8 of NEI 00-04.” In section 3.2.8, “PRA Uncertainty Evaluations,” of the LAR enclosure, the licensee further confirms that a cumulative sensitivity study will be performed where the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in PRAs for all systems that have been categorized are increased by a factor of 3. The NRC staff finds the application of a factor of 3 for the sensitivities is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

In section 3.1.1 of the enclosure to its LAR, the licensee specifically cited NEI 00-04 and noted that RG 1.201, states, in part, that “the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence” and that “all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by § 50.69(c)(1)(iv).” In section 3.4, “Risk Evaluations (10 CFR 50.69(b)(2)(iv))” of the LAR enclosure, the licensee states, “[s]ensitivity studies described in NEI 00-04 Section 8 will be used to confirm that the categorization process results in acceptably small increases to CDF and LERF.” These sensitivity studies, together with the periodic review process discussed in section 3.4 of this SE, 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 NRC staff finds that the performance of the risk sensitivity study is consistent with the guidance in section 8 of NEI 00-04, Revision 0, and therefore, will assure that the potential cumulative risk increase from the categorization is maintained acceptably low, as required by 10 CFR 50.69(c)(1)(iv).

3.3.4 Integrated Decision-Making

Appendix B of SRP Section 19.2 provides guidance and the NRC staff expectations for the licensee’s integrated decision-making process. The guidance in appendix B states, in part, “[r]isk-informed applications are expected to require a process to integrate traditional engineering and probabilistic considerations to form the basis for acceptance.” NEI 00-04, Revision 0, guidance identifies two steps in the categorization process: (1) Preliminary Engineering Categorization of Function and (2) IDP Review and Approval that are responsible for the integrated assessment of the traditional engineering analyses and the risk results from the PRA and non-PRA assessments that are performed to determine the approval of the safety significance of the SSC for categorization. The NRC staff review of the two steps to ensure the processes is well-defined, systematic, repeatable, and scrutable are provided as follows:

Preliminary Engineering Categorization of Functions (NEI 00-04, Section 7)

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

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

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

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

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

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

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

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

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

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

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

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

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

Section 50.69(f) of 10 CFR requires, in part, program documentation, change control, and records. In section 3.2.7 of the enclosure to its LAR, the licensee stated that "[Grand Gulf] will implement a process that addresses the requirements in NEI 00-04, Section 11, 'Program Documentation and Change Control.'" In section 3.1.1 of the enclosure to its LAR, the licensee states that the RISC categorization process documentation will include the following ten elements:

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

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

<u>Periodic Review (NEI 00-04, Section 12)</u>

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

In section 3.2.7, "PRA Maintenance and Updates," of the enclosure to its LAR, the licensee described the process for maintaining and updating the Grand Gulf PRA models used for the 10 CFR 50.69 categorization process. Consistent with NEI 00-04, Revision 0, the licensee confirmed that the "[Grand Gulf] risk management process ensures that the applicable PRA model(s) used in this application continue to reflect the as-built and as-operated plant for [Grand Gulf]." Additionally, the licensee stated that "[t]he 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 also includes reevaluating previously categorized systems to ensure the continued validity of the categorization.

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

3.5 Editorial Change to Correct a Typographical Error

During the review of this LAR, the NRC staff found that section 2(C)(50), "License Renewal Conditions," on page 22 of Renewed Facility Operating License NPF-29 for Grand Gulf, paragraph (b)(2), misspells the word 'activities.' The NRC staff has confirmed that this typographical error was inadvertently introduced when issuing Renewed Facility Operating License No. NPF-29 (Reference 32) for Grand Gulf and was included in the licensee's original submittal dated October 28, 2011 (Reference 33). The typographical error is addressed by replacing the current page 22 in conjunction with the issuance of Amendment No. 233 of RFOL NPF-29 for Grand Gulf. Correcting the misspelling does not alter the authority granted in the license, and does not impact the NRC's proposed determination of no significant consideration.

4.0 CHANGES TO THE OPERATING LICENSE

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

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

The licensee proposed the following license condition to the RFOL for Grand Gulf. The proposed license condition states:

> Entergy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Entergy's submittal letter dated June 6, 2023, and all its subsequent associated supplements; as specified in License Amendment No. 233 dated July 29, 2024.

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

The NRC staff finds that the proposed license condition is acceptable because it adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed by the NRC.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Mississippi State official was notified of the proposed issuance of the amendment on June 20, 2024. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, as published in *Federal Register* on August 8, 2023 (88 FR 53541), and there has been no public comment related to such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

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2. Couture, P., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Supplement to License Amendment Request to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b' and Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors,' Grand Gulf Nuclear Station, Unit 1, NRC Docket No. 50-416, Renewed Facility Operating License No. NPF-29," dated May 1, 2024 (ML24122C611).

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Date: July 29, 2024

SUBJECT: GRAND GULF NUCLEAR STATION, UNIT 1 – ISSUANCE OF AMENDMENT NO. 233 RE: ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2023-LLA-0080) DATED JULY 29, 2024

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GBedi, NRR
DWidrevitz, NRR
FForsaty, NRR

ADAMS Accession No.: ML24172A250 ***concurrence via email**

OFFICE	NRR/DORL/LPL4/PM*	NRR/DORL/LPL4/LA*	NRR/DRA/APLA/BC*	NRR/DRA/APLC/BC*
NAME	ZTurner	PBlechman	RPascarelli	SVasavada
DATE	6/18/2024	6/26/2024	6/28/2024	7/1/2024
OFFICE	NRR/DEX/EEEB/BC*	NRR/DEX/EMIB/BC*	NRR/DSS/SNSB/BC*	NRR/DNRL/NVIB/BC*
NAME	WMorton	SBailey	PSahd	ABuford
DATE	6/18/2024	6/27/2024	6/26/2024	6/28/2024
OFFICE	OGC (NLO)*	NRR/DORL/LPL4/BC*	NRR/DORL/LPL4/PM*	
NAME	DRoth	JRankin	ZTurner	
DATE	7/17/2024	7/26/2024	7/29/2024	

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