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

December 29, 2021

Mr. Fadi Diya
Senior Vice President and
Chief Nuclear Officer
Ameren Missouri
Callaway Energy Center
8315 County Road 459
Fulton, MO 65077

SUBJECT: CALLAWAY PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 226 REGARDING ADOPTION 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 PLANTS" (EPID L-2020-LLA-0235)

Dear Mr. Diya:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 226 to Renewed Facility Operating License No. NPF-30 for the Callaway Plant, Unit No. 1 (Callaway). The amendment is in response to your application dated October 30, 2020, as supplemented by letters dated July 29, 2021, October 13, 2021, and December 22, 2021.

The amendment revises the Callaway Renewed Facility Operating License No. NPF-30 to add a new license condition to allow for the implementation of Title 10 of the *Code of Federal Regulations* Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors."

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

Sincerely,

/RA/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. Amendment No. 226 to NPF-30
2. Safety Evaluation

cc: Listserv

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

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. 50-483

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 226
License No. NPF-30

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

   A. The application for amendment by Union Electric Company (UE, the licensee), dated October 30, 2020, as supplemented by letters dated July 29, 2021, October 13, 2021, and December 22, 2021, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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-30 is hereby amended to add paragraph 2.C.(19) to read as follows:

   (19) Implementation of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

   Ameren Missouri 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 SSCs using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire, high winds, and seismic risk; 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 SSCs and their associated supports; and the results of non-PRA evaluations that are based on a screening of other external hazards updated using the external hazard screening significance process identified in American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard RA-Sa-2009; as specified in License Amendment No. 226 dated December 29, 2021. Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

3. This amendment is effective as of its date of issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Samson S. Lee

Digitally signed by Samson S. Lee
Date: 2021.12.29 17:04:07 -05'00'

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-30

Date of Issuance: December 29, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 226

RENEWED FACILITY OPERATING LICENSE NO. NPF-30

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. 50-483

Replace the following pages of Renewed Facility Operating License No. NPF-30 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
-9-	-9-
-10-	-10-
-11-	-11-
-12-	-12-
---	-13-

1. In order to ensure that the threads for RPV closure stud hole No. 18 can perform their intended function throughout the period of extended operation, UE shall remove stuck stud No. 18. If repair of stud hole No. 18 is required following removal of the stud, the repair plan shall include inspection of the stud hole prior to and after thecompletion of the repair.

2. In order to ensure that RPV stud holes with damaged threads can continue to perform their intended function throughout the period of extended operation, UE shall perform a laser inspection for the threads of repaired RPV stud hole location Nos. 2, 4, 5, 7, 9, and 53. If inspection of these RPV stud holes reveals that there is additional degradation in any of these stud holes, the condition will be entered in the Corrective Action Program forevaluation and corrective action, and UE shall also inspect the remaining repaired RPV stud hole locations (Nos. 13, 25, 39 and 54).

(18) Implementation Actions for New Technical Specification 3.7.20

The planned plant modifications and emergency operating procedure changes described as commitments in Attachment 5 of Ameren Missouri letter ULNRC-06477, “Supplement to License Amendment Request for Addition of New Technical Specification 3.7.20, ‘Class 1E Electrical Equipment Air Conditioning (A/C) System’ (LDCN 16-0013),” dated January 23, 2019, shall be completed prior to implementation of the license amendment requested per Ameren Missouri letter ULNRC-06401, “License Amendment Request for Addition of New Technical Specification 3.7.20, ‘Class 1E Electrical Equipment Air Conditioning (A/C) System’ (LDCN 16-0013),” dated March 9, 2018, as supplemented by the noted January 23, 2019 letter (ULNRC-06477) and Ameren Missouri letter ULNRC-06491, “Additional Supplement to License Amendment Request for Addition of New Technical Specification 3.7.20, ‘Class 1E Electrical Equipment Air Conditioning (A/C) System’ (LDCN 16-0013),” dated March 7, 2019. Completion of the planned plant modifications means physical completion, including completion of the post-modification testing.

(19) Implementation of 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors”

Ameren Missouri 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 SSCs using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire, high winds, and seismic risk; 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 SSCs and their associated supports; and the results of non-PRA evaluations that are based on a screening of other external hazards updated using the external hazard screening significance process identified in American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard RA-Sa-2009; as specified in License Amendment No. 226 dated December 29, 2021. Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

D. An Exemption from certain requirements of Appendix J to 10 CFR Part 50, are described in the October 9, 1984 staff letter. This exemption is authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, this exemption is hereby granted pursuant to 10 CFR 50.12. With the granting of this exemption 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. UE 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 Requirement 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 combined set of plans, which contain Safeguards Information protected under 10 CFR 10 CFR 73.21, are entitled: “Callaway Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 0” submitted by letter dated October 20, 2004, as supplemented by the letter May 11, 2006.

   UE 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 Callaway Plant Unit 1 CSP was approved by License Amendment No. 203, as supplemented by changes approved per License Amendment No. 214.

F. Deleted per Amendment No. 169.

G. UE 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 October 18, 2044.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

William M. Dean, Director
Office of Nuclear Reactor Regulations

Attachments/Appendices:

1. Attachment 1 (Deleted per Amendment No. 169)
2. Attachment 2 (Deleted per Amendment No. 169)
3. Appendix A - Technical Specifications (NUREG-1058, Revision 1)
4. Appendix B - Environmental Protection Plan
5. Appendix C - Additional Conditions

Date of Issuance: March ___6___, 2015

ATTACHMENT 1

Deleted per Amendment No. 169.

ATTACHMENT 2

Deleted per Amendment No. 169.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 226 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By application dated October 30, 2020 (Reference [1]) as supplemented by letters dated July 29, 2021 (Reference [2]),October 13, 2021 (Reference [3]), and December 22, 2021 [4] Union Electric Company, dba Ameren Missouri (the licensee), requested changes to Renewed Facility Operating License (RFOL) No. NPF-30 for Callaway Plant, Unit No. 1 (Callaway).

The proposed amendment would modify the Callaway licensing basis by the addition of a license condition (i.e., License Condition 2.(C).(19)), to allow for the implementation of the provisions 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 provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance (LSS), alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance (HSS), requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance, resulting in improved plant safety.

The supplemental letters dated July 29, 2021, October 13, 2021, and December 22, 2021, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 26, 2021(86 FR 7117).

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

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

Section 50.69 of 10 CFR contains requirements regarding how a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories.

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

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” (Reference [5])

- RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”; and RG 1.200, Revision 3, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities” (References [6] and [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” (Reference [8])

- NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs [Probabilistic Risk Assessments] in Risk-Informed Decisionmaking” (Reference [9])

- NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition,” Chapter 19, Section 19.2, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance” (Reference [10])

NRC-Endorsed Guidance

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

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

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

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

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

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

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

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

Additionally, Section 11, “Program Documentation and Change Control,” of NEI 00-04 provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(f). Section 12, “Periodic Review,” of NEI 00-04 provides guidance on the periodic review related to the requirements in 10 CFR 50.69(e). 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).

2.3 Licensee's Proposed Changes

The licensee proposed the following License Condition 2.(C).(19) to the Callaway RFOL to allow the implementation of 10 CFR 50.69

(19) Implementation of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

Ameren Missouri 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 SSCs using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire, high winds, and seismic risk; 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 SSCs and their associated supports; and the results of non-PRA evaluations that are based on a screening of other external hazards updated using the external hazard screening significance process identified in American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009 PRA Standard; as specified in License Amendment 226 dated December 29, 2021. Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

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.

In an e-mail dated September 14, 2021, the NRC staff requested additional information from the licensee (Reference [12]). The licensee responded to the requests for additional information (RAIs) in the supplemental letter dated October 13, 2021 (Reference [3]).

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

An acceptable approach for making risk-informed decisions about proposed technical specification changes, including both permanent and temporary changes, is to show that the proposed licensing basis changes meet the five key principles stated in Section C of RG 1.174, Revision 3 (Reference [8]). These key principles are:

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

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

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

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

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

3.2 Traditional Engineering Evaluation

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

3.2.1 Key Principle 1: Licensing Bases Change Meets the Current Regulations

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

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

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

Paragraph 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 approves an application for a license amendment, a licensee 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
(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 (References [11] and [5]), and the acceptability of the licensee's PRA for use in the application of the 10 CFR 50.69 categorization process. The NRC staff's review, as documented in this safety evaluation (SE), used the framework provided in RG 1.174, Revision 3, and NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1.

Section 2 of NEI 00-04, Revision 0, states, in part, that the SSC 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 [Integrated Decisionmaking Panel] 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, "Categorization Process Description (10 CFR 50.69(b)(2)(i))," of the License Amendment Request (LAR) as updated (Reference [2]), 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.

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

Section 7.1, "Engineering Categorization," of NEI 00-04, Revision 0, states, in part, that "[d]ue to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC or part thereof, should be assigned the highest risk significance for any function that the SSC or part thereof supports." Section 4 of NEI 00-04 states that a candidate LSS SSC that supports an interfacing system should remain uncategorized until all interfacing systems are categorized. In the LAR, as supplemented, the licensee explained that it is taking no exception from the guidance cited above as SSCs that support functions for more than one system will remain uncategorized until the applicable systems are categorized, and then the SSCs will be assigned the highest categorization.

The regulatory requirements in 10 CFR 50.69 and 10 CFR Part 50, Appendix B, and the 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 and that any performance deficiencies will be identified and appropriate corrective actions will be taken. The licensee's SSC categorization program includes the appropriate steps and elements prescribed 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 the NEI 00-04 guidance, as endorsed. In light of the above, the NRC staff concludes that the proposed 10 CFR 50.69 program meets the first key principle for risk-informed decisionmaking prescribed in RG 1.174, Revision 3.

3.2.2 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 defense-in-depth 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 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. The criteria provided in paragraph 50.69(b)(1)(x) of 10 CFR are not to determine the proper RISC category for containment isolation valves or penetrations.

In Section 3.1.1, "Overall Categorization Process," of the LAR as updated (Reference [2]), the licensee clarified that it would require an SSC to be categorized as HSS based on the defense-in-depth assessment performed in accordance with NEI 00-04, Revision 0. Based on the above, the NRC staff concludes that the proposed change is consistent with the defense-in-depth philosophy described in Key Principle 2 of RG 1.174, Revision 3 (Reference [8]), and is, therefore, acceptable. The NRC staff finds that the licensee's process is consistent with the NRC-endorsed guidance in NEI 00-04 and would meet the 10 CFR 50.69(c)(1)(iii) criterion that requires defense-in-depth to be maintained.

3.2.3 Key Principle 3: Licensing Basis Change Maintains Sufficient Safety Margins

The regulations in 10 CFR 50.69(c)(1)(iv) require the evaluations to provide reasonable confidence that for SSCs categorized as RISC–3, sufficient safety margins are maintained, and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF), resulting from changes in treatment, are small. 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. By assuring sufficient safety margins, (1) the codes and standards or their alternatives approved for use by the NRC are met, and (2) the safety analysis acceptance criteria in the licensing basis (e.g., Final Safety Analysis Report (FSAR), supporting analyses) are met, or proposed revisions provide sufficient margin to account for uncertainty in the analysis and data. RG 1.174, Revision 3, provides guidelines for making that assessment, including evaluations to ensure the categorization of the SSC does not adversely affect any assumptions or inputs to the safety analysis; or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist.

The SSCs design basis function, as described in the plant's licensing basis, including the Updated FSAR and Technical Specifications bases do not change and should continue to be met. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. On this basis, the NRC staff concludes that the licensee has established a program to ensure sufficient safety margins are maintained in accordance with the third key principle of RG 1.174, Revision 3 (Reference [8]) and would therefore meet the requirements set forth in 10 CFR 50.69(c)(1)(iv).

3.3 Risk-Informed Assessment

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

The risk-informed considerations prescribed in NEI 00-04, Revision 0, endorsed by RG 1.201, Revision 1, addresses the fourth and fifth key principles of the NRC staff's standards for risk-informed decisionmaking, 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:

In Sections 3.2.1, "Internal Events and Internal Flooding"; 3.2.2, "Fire Hazards"; 3.2.3, "Seismic Hazards"; and 3.2.4, "High Winds Hazards" of the LAR as updated (Reference [2]), the licensee described that the Callaway categorization process uses PRA modeled hazards to assess risks for the internal events (including internal floods), internal fires, seismic, and high winds. For the other risk contributors, the licensee's process uses the following non-PRA methods to characterize the risk:

- Other External Hazards: Screening analysis performed for Individual Plant Examination of External Events (IPEEE) (Reference [13]) updated and evaluated using a peer review against criteria from Part 6 of the ASME/ANS RA-Sa–2009, "Addendum A to RA-S–2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference [14]), as endorsed by the NRC
- Shutdown Events: Safe Shutdown Risk Management program consistent with NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (Reference [15])
- Passive Components: ANO-2 passive categorization methodology (Reference [16])

The approaches and methods proposed by the licensee to address internal events, seismic, external events, other hazards, defense-in-depth, and shutdown events are consistent with the approaches and methods included in the guidance in NEI 00-04, Revision 0 (Reference [11]). The non-PRA method for the categorization for passive components is consistent with the ANO-2 methodology for passive components (Reference [16]) 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 3.3.1.2 of this SE.

3.3.1.1 Scope of the PRA

The Callaway PRA is comprised of a full-power, Level 1, internal events PRA (IEPRA), fire PRA (FPRA), seismic PRA (SPRA), and high winds PRA (HW PRA), which evaluate the CDF and LERF risk metrics. The licensee discussed in Section 3.2.1 of the LAR, as updated (Reference [2]), that the IEPRA (including internal floods) model has been assessed against RG 1.200, Revision 2 (Reference [6]). Furthermore, LAR Section 3.3, "PRA Review Process Results (10 CFR 50.69(b)(2)(iii))," states that finding closure reviews were conducted on the IEPRA model in November 2019 and June 2020. Open findings were reviewed and closed using the NRC-accepted process documented in the NEI letter to the NRC "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-out of Facts and Observations," dated February 21, 2017 (Reference [17]).

Section 3.2, "Technical Adequacy Evaluation (10 CFR 50.69(b)(2)(ii))" and Attachment 8, "LAR Supplement to Address Audit Discussion Points and Potential RAIs Summarized in NRC Letter Dated June 9, 2021 (ML21139A022)," of the LAR as updated (Reference [2]) by letter dated July 29, 2021, indicate that all PRA upgrades have been peer reviewed including upgrades associated with resolutions to close findings associated with the most recent PRA model update 9.01. (The transition from PRA model update 8 to PRA update 9.01 occurred after the original LAR was submitted in October 2020.) Moreover, given the significant length of time between the last FPRA peer review in 2009 and the fire facts and observations (F&Os) closure review in 2020, the licensee, provided in the updated LAR Attachment 8, a list of the major updates made to the FPRA and explanation for why they are not considered PRA upgrades. The licensee cited explicit guidance from the ASME/ANS RA-Sa 2009 PRA standard in its determination that the changes were maintenance updates. The NRC staff reviewed this list relative to the criteria presented in the ASME/ANS RA-Sa-2009 PRA standard (Reference [14]) and Regulatory Position C.2.2.2.1, "Peer Review of a PRA Upgrade," in RG 1.200, Revision 3 (Reference [7]). The NRC staff did not find that any of the FPRA model changes met the definition of a PRA upgrade, and therefore the licensee's determination that the changes were maintenance updates was acceptable.

The NRC staff finds that the information provided in the LAR as supplemented (Reference [2]), to support the staff review of the IEPRA (including internal flooding), FPRA, SPRA, and HW PRA for technical acceptability was provided, and therefore, meets the requirements set forth in 10 CFR 50.69(b)(2)(iii).

The NRC staff evaluated the scope of the PRA including: (1) peer review history and results, (2) the Appendix X, Independent Assessment process, (3) credit for diverse and flexible coping strategies (FLEX) in the PRA, and (4) assessment of assumptions and approximations. In an e-mail to the licensee dated September 14, 2021 (Reference [12]), the NRC staff issued RAIs to further assess the acceptability of the Callaway IEPRA (including internal floods), FPRA, SPRA, and HW PRA for consistency with RG 1.200, Revision 2 (Reference [6]), and NEI 00-04, Revision 0 (Reference [11]), as endorsed in RG 1.201, Revision 1 (Reference [5]). The NRC staff's review of these aspects of the PRA and supplemental responses to assess for consistency with the applicable processes as endorsed by the NRC, where necessary, are provided below.

Internal Events PRA (includes internal floods) Peer Review History

In Section 3.3 of the LAR as updated (Reference [2]), the licensee states that the IEPRA (including internal floods) model was subjected to a full-scope peer review in April 2019, consistent with RG 1.200, Revision 2 (Reference [6]). Subsequently, in November 2019, the licensee conducted an Independent Assessment for closure of the finding-level F&Os from the full-scope peer review. All initial F&Os were closed except one regarding implementation of the newly developed method (NDM) described in Pressurized Water Reactor Owners Group (PWROG)-18027, Revision 0, "Loss of Room Cooling in PRA Modeling" (Reference [18]). During the November 2019 Independent Assessment, two F&O resolutions were determined to be PRA upgrades, and therefore, a focused-scope peer review was concurrently performed, which resulted in one new IEPRA F&O. In June 2020, another Independent Assessment closure of F&Os and a concurrent focused-scope peer review were performed, which included review of the new methodology incorporated from PWROG-18027. As a result of the June 2020 F&O closure and concurrent focused-scope peer review, all the IEPRA (including internal floods) F&Os were closed. A detailed NRC staff review of this Independent Assessment is included below in this subsection of this SE.

Concerning the NDM referred to above, Section B of RG 1.200, Revision 3 (Reference [7]) endorses NEI 17-07, Revision 2, "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard," issued August 2019 (Reference [19]). NEI 17-07 establishes a NDM peer review process. Section B of RG 1.200, Revision 3, also endorses certain portions from PWROG-19027-NP, Revision 2, "Newly Developed Method Requirements and Peer Review," issued July 2020 (part of Reference [20]) including "requirements for the peer review of newly developed methods (NDMs) (see Regulatory Positions C.2.2.2 through C.2.2.4)." RG 1.200 Revision 3, Regulatory Position C.2.2.2.2 states that Section 5.1 of the PWROG-19027-NP, Revision 2, provides a set of technical requirements for Section 1-6 of the ASME/ANS Level 1/LERF PRA standard that can be used to peer review a NDM.

The LAR as updated (Reference [2]), explains that, as part of an effort unrelated to the Callaway reviews, a peer review was performed in February and March of 2020 on the PWROG-18027 methodology following the guidance in NEI 17-07, Revision 2, and PWROG-19027-NP, Revision 2, which contains the Supporting Requirements (SRs) for new methods. This NDM pilot review is documented in a letter to the NRC dated August 6, 2020, as PWROG-19020-NP (Reference [21]). Section E.4.2, "Technical Adequacy," of PWROG-19020-NP states that a peer review of the presented method is needed when it is implemented at a plant that includes evaluation certain specified SRs. The licensee explained in the LAR, as supplemented, in Attachment 8, that exclusion of the cited SRs from the scope of that focused-scope peer review of PWROG-18027 is justified because the method was implemented in the PRA and peer reviewed prior to the method being accepted through the NDM peer review process, endorsed by Regulatory Guide 1.200, Revision 3 (Reference [7]). Moreover, the cited SRs were reviewed and assessed as meeting Capability Category II (CC II) and no F&Os were generated. With regard to the fire, seismic and high wind PRAs, the licensee explained that it found no context related to those external hazards that would change how PRA standard SRs are met (e.g., the "back referencing" SRs from fire SR to basic internal events SRs). The licensee clarified that it used the primary approach option in PWROG-18027, in which room cooling is assumed to be required if heatup calculations show temperatures greater than the screening value. For the FPRA, the licensee explained that, in practice, application of the PWROG-18027 method is not impacted by fire modeling considerations, such as room heatup, because the FPRA explicitly addresses the fire contribution to room heatup and any fire impact on cabling and equipment including heating, ventilation, and air conditioning. The NRC staff finds that application of the method to the seismic and high wind PRAs is not impacted because those initiators do not directly impact room heatup, and application of the method on FPRA does not change the FPRA methods. The NRC staff concludes that the licensee's treatment meets the guidance in RG 1.200, Revision 3, because requirements for a peer review of NDMs are met through a peer review relative to PWROG-19027-NP, Revision 2 and no open F&Os remain.

Therefore, the NRC staff concludes that the Callaway IEPRA (including internal floods) has been appropriately peer reviewed, consistent with RG 1.200, Revision 2 (Reference [6]), and the F&O's have been closed using an NRC approved approach. Based on the above, the NRC staff finds that the Callaway 10 CFR 50.69 program uses an IEPRA that is of sufficient quality to meet the requirements set forth in 10 CFR 50.69(c)(1)(i).

<u>Internal Fire PRA Peer Review History</u>

The licensee's FPRA was subject to a full-scope industry peer review in October 2009, consistent with RG 1.200, Revision 2 (Reference [6]). Subsequently, in June 2020, the licensee conducted Independent Assessments for closure of the finding-level F&Os to address ASME/ANS 2009 SRs that were not met at CC II during the 2009 peer review. In conjunction with the F&O closure review, the licensee conducted a focused-scope peer review. The focused-scope peer review generated additional FPRA F&Os that were included in the Independent Assessment team review. As a result of the June 2020 Independent Assessment all finding-level F&Os were closed. However, the resolution of Suggestion F&O FSS-B1-03 was determined to be a PRA upgrade during the F&O closure review. In November 2020, a focused-scope peer review was performed on the resolution of F&O FSS-B1-03. The focused-scope review pertained to the use of a detailed human reliability analysis (HRA) of the main control room abandonment actions rather than using screening values. In February 2021, the F&Os from the November 2020 focused-scope peer review were subsequently closed in an F&O closure review. A detailed NRC staff review of this Independent Assessment is included below in this subsection of this SE. The NRC staff has reviewed the FPRA peer review results and the licensee's resolution of the results concluding that the Callaway FPRA was appropriately peer-reviewed consistent with RG 1.200, Revision 2 (Reference [6]), and that the F&O's have been adequately dispositioned to assess the impact on the risk-informed application.

Regarding FPRA quality but not concerning disposition of F&Os, the guidance in RG 1.200, Revisions 2 and 3, states in part, that "NRC reviewers, [will] focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application." The relatively extensive and detailed review of FPRAs undertaken in support of LARs to transition to National Fire Protection Association (NFPA)-805 determined that implementation of some of the complex FPRA methods often used non-conservative and over-simplified assumptions to apply the method to specific plant configurations. Some of these issues were not always identified in F&Os by the peer review teams but are considered potential key assumptions by the NRC staff because using more defensible and less simplified assumptions could substantively affect the fire risk and fire risk profile of the plant.
Section 3.2.2 of the LAR as updated (Reference [2]), states that the numerous new or revised FPRA guidance documents issued since Callaway was approved to implement NFPA-805 are

being addressed through the PRA maintenance and update process. Attachment 8 of the updated LAR lists the guidance documents on FPRA methods that have been addressed in the FPRA since the full-scope peer review in 2009. The NRC staff reviewed this list and finds that updated NRC guidance, with the most potential to impact the FPRA, has already been implemented in the FPRA or does not apply.

This guidance includes NRC-approved guidance in the form of FPRA Frequently Asked Questions (FAQs). With regard to use of NRC's guidance in the form of FAQ 13-0004, "Clarifications on Treatment of Sensitive Electronics" (Reference [22]), the licensee stated, in part, in Attachment 8 of the updated LAR that "[w]ithout explicitly citing FAQ 13-0004 for the treatment of sensitive electronics, the fire PRA does implement the salient conclusion that a generic screening heat flux damage threshold for thermoset cables, as observed on the outer surface of the cabinet, can be used as a conservative surrogate for assessing the potential for thermal damage to solid-state and sensitive electronics within an electrical panel (cabinet)." The LAR, however, does not address the two caveats cited in FAQ 13-0004 that can invalidate this approach, which are: (1) sensitive electronics mounted on the surface of the cabinet where it can be exposed to the convective or radiant energy of a fire, and (2) the presence of a louver or other typical ventilation means.

Therefore, in RAI 02 (Reference [12]), the NRC staff stated that this source FPRA modeling uncertainty appears to have the potential to impact the application and requested explanation of how sensitive electronics are modeled in the FPRA for the two sensitive electronics cabinet configurations that can invalidate the FAQ 13-0004 approach consistent with the guidance in FAQ 13-0004. In response to RAI 02 by supplemental letter dated October 13, 2021 (Reference [3]), the licensee explained that sensitive electronics for Callaway FPRA were modeled using the thermoset cable heat flux damage threshold, and because the FAQ had not yet been published at the time of the analysis, the caveats about cabinet configurations that could invalidate the approach were not addressed. The licensee stated that in response to the RAI, a review was performed on all fire areas to identify cabinets with configurations that could invalidate the use of the thermoset cable heat flux damage threshold. Only one cabinet that supported a PRA function was found to meet the conditions of the caveat. The licensee explained that it reviewed the fire scenarios in the fire area that contained the sensitive electronics and found just one transient fire scenario that would be affected by changing the zone of influence (ZOI) to account for use of the lower heat flux damage threshold for sensitive electronics (which is provided in Appendix H of NUREG/CR-6850, "Fire PRA Methodology for Nuclear Power Facilities" (Reference [23])). The licensee explained that it performed a sensitivity study in which it increased the ZOI (floor area) for that scenario to account for transient fires further away that could damage the sensitive electronics and reduced the floor area for the adjoining floor area to conserve compartment fire ignition frequency. The licensee presented the change in fire CDF and LERF from using a lower heat flux damage threshold for sensitive electronics showing that the impact to risk is negligible. The NRC staff finds that the licensee's treatment of sensitive electronics for Callaway FPRA is almost consistent with the NRC guidance in FAQ 13-0004, and where it is not consistent, the treatment was found to have no impact on the application.

Section 3.2.2 of the LAR as updated (Reference [2]), states that all NFPA-805 Attachment S items committed to in the Callaway NFPA-805 SE in Amendment No. 206 (Reference [24]), have been implemented, therefore, there is no impact on the application from modifications credited in the FPRA but not yet implemented.

The NRC staff has reviewed the FPRA peer review results and the licensee's resolution of the results concluding that the Callaway FPRA was appropriately peer-reviewed, consistent with RG 1.200, Revision 2 (Reference [6]), and that the F&O's and NRC's RAI about FPRA methods have been adequately dispositioned to assess the impact on the risk-informed application.

External Events PRAs Evaluation

In accordance with Sections 5-1.2 and 7-1.2 of the ASME/ANS RA-Sa 2009 PRA standard (Reference [14]), it is assumed that full-scope, internal events, at-power, Level 1 and Level 2 LERF PRAs exist, and that those PRAs are used as the basis for the SPRA and HW PRA. Therefore, the technical acceptability of the IEPRA model used as the foundation for the SPRA and HW PRA is an important consideration.

In Sections 3.2.3 and 3.2.4 of the LAR as updated (Reference [2]), the licensee stated that the proposed categorization process will use peer-reviewed SPRA and HW PRA models. The NRC staff's review of the technical acceptability of the SPRA and HW PRA models for this application is discussed below.

Seismic PRA Peer Review History and Model Evaluation

The NRC staff's review of the licensee's SPRA is based on the results of the peer review and the associated Independent Assessments for closure of F&Os described in Section 3.3 of the LAR as updated. In the course of the review of this LAR, the NRC staff utilized information from the licensee's submittal in response to the 10 CFR 50.54(f) information request arising from Near Term Task Force (NTTF) Recommendation 2.1 dated March 28, 2014 (Reference [25]), and the corresponding staff response letter dated April 21, 2015 (Reference [26]). The last full-scope peer review of the SPRA was performed in June 2018 against the SPRA requirements in ASME/ANS RA-S Case 1 (Reference [27]), also known as Code Case for the ASME/ANS RA-Sb-2013 PRA standard. NRC accepted the use of the Code Case on an interim basis until it was endorsed in a letter dated March 12, 2018 (Reference [28]), and is now endorsed by the NRC in RG 1.200, Revision 3, dated December 2020 (Reference [7]). The first Independent Assessment was performed in March 2019 on the SRs associated with all but two F&Os from the June 2018 full-scope peer review and included a focused-scope peer review of three SRs that were determined to be the subject of a PRA upgrade. The final F&O closure review was performed June 2020 and closed all remaining F&Os. A detailed NRC staff review of this Independent Assessment is included below in this subsection of this SE.

Regarding SPRA quality, but not concerning disposition of F&Os, the guidance in RG 1.200 states, in part, that "NRC reviewers, [will] focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application." Some of these issues were not always identified in F&Os by the peer review teams but are considered potential key assumptions by the NRC staff. The NRC staff notes that truncation convergence, if not achieved, is a general source of potential modeling uncertainty and use of seismic hazard intervals to represent a seismic hazard curve is a simplification which introduces modeling uncertainty.

Regarding these sources of modeling uncertainty, the licensee states, in Attachment 8 of the updated LAR, that the truncation convergence was reached for update 9.01 of the SPRA when the truncation level was lowered by one decade, and the change in CDF or LERF was found to be less than about 5 percent. This is consistent with the suggestion in SR QU-B3 of the ASME/ANS RA-Sa-2009 PRA standard (Reference [14]). The licensee explained that individual quantifications were performed to separately derive CDF and LERF for each of the seismic hazard intervals. The truncation convergence tests in both cases were conducted on the aggregate quantification results across all seismic hazard intervals. The licensee explained that the seismic hazard intervals were defined to cover the range of possible accelerations up to the point the plant level fragility approaches 1.0. This occurs at 0.7g for seismic CDF and 1.17g for seismic LERF, because the SSCs performing the containment function have a higher seismic capacity than SSCs that prevent core damage. Accordingly, the seismic hazard intervals were defined differently for LERF than they were for CDF, in part, to account for the seismic hazard interval between 0.7g and 1.17g where the containment function is not yet failed. The NRC staff finds the licensee's treatment discussed above meets the ASME/ANS PRA standard SR for truncation convergence, and that the discretization of the hazard curve separately for CDF and LERF quantification provides a more optimal representation of the hazard curve than if the same discretization was used.

In RAI 03 (Reference [12]), the NRC staff noted that the highest seismic hazard bin (i.e., %G10) for CDF covers all accelerations greater than 0.8g. Accordingly, the NRC staff requested a sensitivity study to demonstrate that refining the analysis by subdividing the seismic hazard bin into several sub-bins does not impact the application. In response to RAI 03 (Reference [3]), the licensee provided the results of a sensitivity study in which bin %G10 was subdivided into six new intervals (i.e., 0.8g - 0.9g, 0.9g - 1.0g, 1.0g - 1.2g, 1.2g - 1.5g, 1.5g - 2.0g, and >2.0g). The results of the sensitivity study show that dividing this seismic hazard bin into the six intervals resulted in an increase in seismic CDF of 0.6 percent. The licensee further explained that at these high acceleration levels the conditional core damage probability (CCDP) is overestimated, and if CCDPs greater than 1.0 are not allowed to exceed 1.0, then the increase in CDF is only 0.3 percent. Therefore, the NRC staff concludes that the impact of using a single interval for the seismic hazard acceleration range greater than 0.8g has an insignificant impact on seismic CDF, and therefore, on the 10 CFR 50.69 application.

In RAI 04 (Reference [12]), the NRC staff noted that the high seismic CDF and LERF values relative to the CDF and LERF values from the other hazards, including internal events, suggest that the uncertainty in SPRA modeling involving the level of detail used to model fragility could potentially impact the 10 CFR 50.69 risk-informed categorization. The NRC staff stated that it is not clear from the discussion provided in Attachment 8 of the updated LAR, in response to Audit Question APLC 04, about the level of fragility analyses performed for just the four dominant CDF and LERF importance contributors whether the Callaway seismic CDF and LERF values could be further reduced by further refining the fragility analyses of other SSCs. The NRC staff observed that since the point estimate seismic CDF of 5.59E-05 per year presented in Section 6 of the Callaway SPRA report dated July 10, 2020 (Reference [29]), the seismic point estimate CDF value was reduced to 4.01E-05 based on refinements according to Attachment 1, "List of Categorization Prerequisites," of the updated LAR (Reference [2]). An overly conservative SPRA model could skew the integrated importance measures calculated for the 10 CFR 50.69 categorization. Importance measures provide a relative measure of risk importance and, if the importance of certain SSCs is significantly overestimated, then the importance of other SSCs will be underestimated. Therefore, the NRC staff requested justification that not using a more refined fragility analysis for certain important SSCs will have an inconsequential impact on the 10 CFR 50.69 risk-informed categorization. In response to RAI 04 in the supplemental letter dated October 13, 2021 (Reference [3]), the licensee presented the dominant failures to seismic CDF and LERF along with the associated failure modes, seismic fragilities parameters, and the level of seismic fragility assessment performed. The seismic failures presented were those with a Fussell-Vesley (F-V) importance measure above 0.01 and consisted of four CDF failures and four different LERF failures. The NRC staff reviewed the level of assessment for each seismic failure and observes that all the associated SSC fragility values, with the exception of the fragility for offsite power, are based on plant-specific evaluations that address location-specific considerations. The licensee indicated that further refinement is not expected to result in significant change in the SSC capacities. The licensee stated that the loss of offsite power, which is the most dominant seismic failure, is an aggregate of possible failure modes within the switchyard and transmission grid. The licensee stated that the generic offsite power fragility used in the SPRA comes from a 1995 Electric Power Research Institute (EPRI) report for advanced Light Water Reactors (LWRs) based on an earlier evaluation of industry fragility data, which included data from conventional power plants subjected to seismic events. The "governing" failure mode was found to be the failure of transformer ceramic insulators. The licensee stated that the distribution of fragilities from this data was "small" (narrow) with a median value of 0.29g, which was increased to 0.30g for the Callaway SPRA. The licensee stated that the applicability of this fragility value was confirmed based on factors such as major equipment spatial information, the applicability of failure modes to major equipment, and soil structures. The licensee stated that EPRI 3002000709, "Seismic Probabilistic Risk Assessment Guide" (Reference [30]) recommends using a fragility value of 0.30g and that a more recent EPRI report, EPRI 3002015993, "Loss of Offsite Power Fragility Guidance" dated August 2019 (Reference [31]) states that "[t]o derive a more realistic fragility value, a comprehensive response analysis of the entire offsite power system would be needed, with varying soil profiles, seismic hazards and structural response analysis." The NRC staff finds the fragility analysis performed for the Callaway SPRA to be sufficient to support the 10 CFR 50.69 application, in part, because the fragility analysis is not overly conservative but refined consistent with the state-of-practice.

The NRC staff concludes that the Callaway SPRA was appropriately peer reviewed, consistent with RG 1.200, Revision 2 (Reference [6]), and the F&O's have been closed using an NRC approved approach. In addition, the licensee's responses to RAI 03 and RAI 04 on SPRA methods are acceptable.

<u>High Winds PRA Peer Review History and Model Evaluation</u>

In Section 3.3 of the LAR as updated (Reference [2]), the licensee stated that the Callaway HW PRA model was subjected to a full-scope peer review in April 2019 against the technical elements in Part 7 of the ASME/ANS RA-Sa-2009 PRA Standard, endorsed by RG 1.200, Revision 2 (Reference [6]). RG 1.200, Revision 2, provides the NRC regulatory position on peer review processes described in NEI 00-02, 05-04, and 07-12. Regulatory Position 2.2, "Industry Peer Review Program," of RG 1.200, Revision 2, states, in part, that "[a]n acceptable peer-review approach is one that is performed according to an established process…." Regulatory Position 2.2 in RG 1.200, Revision 2, further states, in part, that "[w]hen the staff's regulatory positions contained in the appendices are taken into account, use of a peer review can be used to demonstrate that the PRA [with regard to an at-power Level 1/LERF PRA for internal events (excluding external hazards)] is adequate to support a risk informed application." Therefore, RG 1.200, Revision 2, does not endorse any peer review guidance for external hazards. The NRC staff issued a letter accepting the use of NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," as modified by the NRC staff's comments and qualifications, in letter dated March 2018 (Reference [32]). The Callaway HW PRA models were subject to an Independent Assessment for closure on all the F&Os generated in the April 2019 full-scope peer review. As a result of the F&O closure review, all F&Os were closed. A detailed NRC staff review of this Independent Assessment is included below in this subsection of this SE.

In response to NRC audit questions APLC-06 and APLC-07 (Reference [2]), the licensee provided a detailed description of high winds and their generated missile hazards development and high winds PRA modeling. The NRC staff finds that the description is very thorough, the hazards development is conservative, and the HW PRA model is adequate and meets Part 7 of the ASME/ANS RA-Sa-2009 PRA standard.

Therefore, the NRC staff concludes that the Callaway HW PRA has been appropriately peer reviewed, consistent with RG 1.200, Revision 2 (Reference [6]), and the F&O's have been closed using an NRC approved approach. In addition, the licensee's responses to the NRC's audit questions on HW PRA modeling are acceptable.

Appendix X, Independent Assessment Process for F&O Closure

Section X.1.3, "Close Out F&Os by Independent Assessment," of Appendix X to NEI 05-04, NEI 07-12 and NEI 12-16 (Reference [17]) provides guidance to perform an Independent Assessment for the closure of F&Os identified from a full-scope or focused-scope peer review. Section 3.3 of the LAR, as updated (Reference [2]) states that all F&O closure reviews were performed in accordance with the processes documented in Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13 as accepted by NRC's guidance letter dated May 3, 2017 (Reference [33]), as well as the requirements published in the ASME/ANS PRA Standard (RA-Sa-2009).

Based on a review of the LAR as updated and the peer review reports during the audit, 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 floods), FPRA, SPRA, and HW PRA in accordance with the ASME/ANS RA-Sa-2009 PRA standard, or Code Case for the ASME/ANS RA-Sb-2013 PRA standard, as endorsed by the NRC. Therefore, the NRC staff finds that the Callaway IEPRA, FPRA, SPRA, and HW PRA have been appropriately peer reviewed consistent with RG 1.200, Revision 2 (Reference [6]) and meet the requirements set forth in 10 CFR 50.69(c)(1)(i).

Credit for FLEX Equipment

The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (Reference [34]), provides the NRC staff's assessment of challenges incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decisionmaking in accordance with the guidance of RG 1.200, Revision 2 (Reference [6]).

In LAR Section 3.2.9.2 and Attachment 8, as updated (Reference [2]), the licensee stated that the FLEX strategies (i.e., portable diesel driven FLEX Steam Generator Makeup Auxiliary Feedwater (AFW) Pump and portable diesel 480 VAC generators to supply power to the battery chargers) have not been credited in the current PRA model update 9.01. The licensee stated that the FLEX strategy involving automatic realignment of the safety-related AFW pumps from the condensate storage tank to the hardened condensate storage tank is credited in all the PRAs but only involves use of permanently installed equipment. Therefore, the NRC staff concludes that the licensee's treatment of FLEX strategies is acceptable for this application because (1) the first two FLEX strategies are not credited in the current PRA model update 9.01, and (2) the third FLEX strategy, credited in all the PRA models, involves use of permanently installed nuclear power plant equipment and actions, which do not require new modeling methods because they are similar to equipment and actions already modeled in the PRAs

Identification of Key Assumptions and Sources of Uncertainty

The licensee stated that NUREG-1855, Revision 1 (Reference [9]) was used 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 NUREG-1855, Revision 1, is a qualitative screening process that involves identifying and validating whether consensus[1] models have been used in the PRA to evaluate identified model uncertainties. In Section 3.2.8, "PRA Uncertainty Evaluations"; Attachment 6: "Key Assumptions and Sources of Uncertainty"; and Attachment 8 of the updated LAR, the licensee explains that a list of plant-specific assumptions and sources of uncertainty identified from the PRA notebooks was compiled along with generic industry sources of assumptions and sources of uncertainty from EPRI 1016737 (Reference [35]) and EPRI 1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty" (Reference [36]), which the NRC staff notes include generic sources of modeling uncertainty for the fire, seismic, and high wind hazards.

The licensee stated that if the PRA models "used a non-conservative treatment or methods that are not commonly accepted, then the underlying assumption or source of uncertainty was reviewed to determine its impact on the 50.69 implementation LAR application." In LAR Attachment 8 as updated (Reference [2]) the licensee stated that in its updated uncertainty analysis (which was reviewed by the NRC staff during the audit) the comprehensive compilation of assumptions and sources of uncertainty were dispositioned using justifications such as: (1) the modeling associated with the source of uncertainty was based on a current industry consensus modeling approach, (2) the treatment associated with the source of uncertainty applied the most recent industry data, (3) the assumption or source of uncertainty has no impact on the PRA results and, therefore, no impact on the 10 CFR 50.69 program, (4) the guidance in NEI 00-04 already requires a sensitivity study for this modeling (e.g., Human Failure Events), and (5) a sensitivity study was performed on the base model showing that this source of uncertainty has no impact on the risk results and, therefore, no impact on the 10 CFR 50.69 program. The NRC staff's review of the licensee's assessment finds that it includes consideration of modeling choices and approximations. Section 3.2.8 of the LAR as updated states, in part, that "Attachment 6, 'Key Assumptions and Sources of Uncertainty,'…documents the conclusion of this review, which found that no additional sensitivity analyses are required to address the Callaway Plant, Unit No. 1 model specific assumptions or sources of uncertainty." The NRC staff reviewed the dispositions to the candidate key assumptions and sources of uncertainty in Attachment 6 of the updated LAR. In the following, the resolution to concerns about assumptions and sources of uncertainty that appeared to the NRC staff to have the potential to impact the 10 CFR 50.69 risk-informed categorization are discussed.

In Attachment 8 of the LAR as updated (Reference [2]), the licensee stated, in part, in response to Audit Question APLC 07 that "[d]epending on the severity of the wind and the likelihood of failure of the offsite electrical grid, high wind events follow either the turbine trip or [Loss of

[1] Per NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking" (ADAMS Accession No. ML17062A466), 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.

Offsite Power] LOOP sequence logic.…" In Table E6-3 of Attachment 6 to the LAR as updated, the licensee stated, in part, that "the electrical grid fragility, is assigned based on wind speed" and identifies this treatment as a source of modeling uncertainty. In Attachment 8 of the LAR, as updated, the licensee stated that the approach is an "enhancement" compared to the assumption made in HW PRA studies in which all high wind is assumed to result in a LOOP. In the LAR as updated, the licensee also stated that wind speeds greater than (>) 112 miles per hour (mph) were assumed to result in a LOOP with certainty. The LAR, however, does not discuss the high wind induced LOOP probabilities or their bases. Table E6-3 of the LAR indicates that these probabilities are an area of ongoing investigation for the nuclear industry. The NRC staff notes, based on sensitivity study results for HW PRA update 9.01, that if the high wind induced LOOP probabilities are considerably increased, then the HW PRA CDF increases by 114 percent to 1.26E-05 per year. This sensitivity study result, discussed in Table E6-3 of the LAR, appears to indicate that the HW PRA risk results are sensitive to the assumed conditional LOOP probabilities. Therefore, a HW PRA CDF of 1.26E-05 per year would be a significant contributor to the total CDF and the assumption discussed above could impact the 10 CFR 50.69 risk-informed categorization. However, in response to Audit Question 05.a the licensee stated that the cited sensitivity study also showed that there are no components with Risk Achievement Worth (RAW) > 2 or F-V > 0.005 in the baseline case that are not already significant in the "pessimistic" electrical grid fragility sensitivity case.

In Attachment 8 of the LAR as updated (Reference [2]), the licensee confirmed that sensitivity studies to address HRA uncertainty for seismic human failure events and combinations will include adjustment to the four seismic HRA bins used in the seismic PRA bins and is consistent with NEI 00-04, Revision 0.

In Attachment 8 of the LAR as updated (Reference [2]), the licensee provided quantification results based on PRA update 9.01 using mean values that account for the state of knowledge correlation (SOKC) and confirmed that the total CDF and LERF meet the RG 1.174 risk acceptance guidelines. The updated LAR also provides the CDF and LERF values for the internal events, internal flooding, fire, seismic and high wind PRAs based on point estimate values compared to calculated mean values. The results show that the differences for the internal events, internal flooding, and fire PRAs are minimal (i.e., between 0.6 percent and 3.6 percent) but are significantly higher for the seismic and high winds PRAs. RG 1.174, Revision 3 (Reference [8]) and Section 6.4 of NUREG-1855, Revision 1 (Reference [9]), for a CC II risk evaluation, indicate that the mean values of the risk metrics (total and incremental values) need to be compared against the risk acceptance guidelines. The mean values referred to are the means of the probability distributions that result from the propagation of the uncertainties on the PRA input parameters and model uncertainties explicitly reelected in the PRA models. In general, the point estimate CDF and LERF obtained by quantification of the cutset probabilities using mean values for each basic event probability does not produce a true mean of the CDF and LERF. Under certain circumstances, a formal propagation of uncertainty may not be required if it can be demonstrated that the SOKC is unimportant (i.e., the risk results are well below the acceptance guidelines).

In RAI 01 (Reference [12]), NRC staff noted that the LAR as updated (Reference [2]) did not identify fire, seismic and high wind PRA parameters that were correlated to account for SOKC and whether fire parameters such as circuit failure probabilities, suppression probabilities, and ignition frequencies for the FPRA were included. Also, the NRC staff cited the licensee in its response stating it presented conservative total mean CDF and LERF values of 19 and 27 percent higher, respectively, than the point estimates values. Therefore, the NRC staff requested (1) identification of the parameters derived from the same data (other than for same type code data) that were correlated for the fire, seismic, and high wind PRAs and justification that these parameters are sufficient to estimate the SOKC, and (2) discussion of how the SOKC will be treated for the 10 CFR 50.69 program consistent with NUREG-1855, Revision 1, which requires mean values and the SOKC be considered. In response to RAI 01, by supplemental letter dated October 13, 2021 (Reference [3]), the licensee explained that for the FPRA the random failure and the hot short events were correlated in the FPRA parameter uncertainty analysis. The licensee indicated that other FPRA parameters such as the non-suppression probabilities and ignition frequencies do not occur in the same cutset, though NRC notes that they could be derived from the same data. The NRC staff agrees that the impact from the correlation of these parameters is less significant because the impact occurs across cutsets opposed to within a cutset. Attachment 8 of the LAR as updated (Reference [2]), shows that the difference in fire CDF and LERF values based on point estimate and calculated mean values is minimal (i.e., between 0 and 3.6 percent). The licensee states that no specific treatment of SOKC is necessary for the 10 CFR 50.69 application, indicating that the application will use FPRA point estimate values for CDF and LERF. The NRC staff finds the investigation of the impact of the SOKC on the FPRA results and the use of FPRA point estimate values sufficient for 10 CFR 50.69 risk-informed categorization because: (1) the difference between the CDF and LERF values based on point estimates and calculated means is minimal, (2) the licensee investigated the dominant contributors to the SOKC for the FPRA, and (3) 10 CFR 50.69 risk categorization is supported by non-PRA elements that have a greater impact on categorization.

For the HW PRA, the licensee explained that only the random failure events were correlated in the HW PRA uncertainty analysis. The licensee indicated that other HW PRA parameters such as SSC failure from high wind fragility do not occur in the same cutset, though NRC notes that they may be derived from the same data. The NRC staff agrees that in most cases the failure probabilities of SSCs from high wind fragility are independently determined from separate data, but notes that fragilities for like-kind SSCs (e.g., emergency diesel generator exhaust stacks) may come from the same data. The licensee’s response to RAI 01 by supplemental letter dated October 13, 2021 (Reference [3]), shows that the difference in high wind CDF and LERF values based on point estimate and calculated mean values using the PRA quantification tool ACUBE is minimal (i.e., between 1 and 7 percent). The licensee stated that no specific treatment of SOKC is necessary for the 10 CFR 50.69 application, indicating that the application will use HW PRA point estimate values for CDF and LERF. The NRC staff finds the investigation of the impact of the SOKC for the HW PRA and the use of HW PRA point estimate values are sufficient for 10 CFR 50.69 risk-informed categorization because: (1) the difference between the high wind CDF and LERF values based on point estimates and calculated means is minimal, (2) the licensee included the impact of the SOKC for random failures and the impact of the contribution from high wind fragilities is judged to be minimal because the failure probabilities of SSCs from high wind fragility are mostly independently determined from separate data, (3) the high wind CDF and LERF are minor contributors to the total CDF and LERF, and (4) 10 CFR 50.69 risk categorization is supported by non-PRA elements that have a greater impact on risk-informed categorization.

For the SPRA, the licensee explained that no failure events were correlated in the SPRA uncertainty analysis. The licensee explained that impact of the SOKC for random failures is only significant at low seismic accelerations before the seismic failures dominant the results, and lower seismic acceleration events are not significant to overall seismic CDF (SCDF) and seismic LERF (SLERF). The licensee indicated that the seismic hazard frequency is discretized into hazard bins, and the NRC staff notes that a single frequency (opposed to a distribution) is used to represent the bin so SOKC does not apply. The licensee stated that impact from the SOKC is possible when multiple component groups utilize the same developed fragility (based on the same data), but this did not occur for the Callaway SPRA. The licensee explained that the seismic failures of like-kind equipment in the same location are based on the same fragility analyses. However, SOKC is addressed because the like-kind equipment is conservatively assumed to be 100 percent correlated. Fragility analyses of similar equipment at different locations were performed independently. The NRC staff agrees that for the reasons stated above, the impact of the SOKC correlation should be minimal. The licensee explained that the difference between the SCDF and SLERF, determined using point estimates and calculated mean values reported in the supplement dated July 29, 2021 (Reference [2]), is not driven by SOKC but by ACUBE and UNCERT software functionality. The licensee showed that when a preprocessing step is not performed, there is no significant difference between the CDF and LERF values calculated using point estimates and mean values. The NRC staff notes that in the seismic fragility analyses, normal distributions are used, which result in significantly less SOKC impact. The licensee stated that the cited preprocessing step involves removing events with the probability of 1.0 from the cutsets, which the NRC notes increases the accuracy of the ACUBE estimates. The licensee stated that the UNCERT program used to propagate parametric uncertainty cannot apply certain ACUBE preprocessing steps. The licensee shows that the difference between the SPRA CDF and LERF presented in Attachment 8 of the LAR as updated appears to be due entirely to the fact that the cited preprocessing step was performed for determining the point estimate SCDF and SLERF and was not performed for determining the calculated mean SCDF and SLERF using the UNCERT. As stated above, the licensee stated that no specific treatment of SOKC is necessary for the 10 CFR 50.69 application, indicating that the application will use SPRA point estimate values for CDF and LERF. The NRC staff finds the investigation of the impact of the SOKC for the SPRA and the use of SPRA point estimate values sufficient for 10 CFR 50.69 risk-informed categorization because: (1) the difference between the CDF and LERF values based on point estimates and calculated mean is negligible, and (2) the licensee demonstrated that though the impact from the SOKC was not addressed in the parametric uncertainty analysis, it is minimal because it is not applicable to the seismic initiating event frequencies and the fragility analyses, and its impact from random failures is minimal.

In Section 3.2.8 of the LAR as updated (Reference [2]), the licensee confirmed that sensitivity studies will be performed consistent with Table 5-2, "Sensitivity Studies for Internal Events PRA," of the NEI 00-04, Revision 0 (Reference [11]), guidance. In accordance with Section 9 of NEI 00-04, as endorsed by RG 1.201, Revision 1 (Reference [5]), the licensee's IDP will use information and risk insights compiled in the initial categorization process, including awareness of the limitations and assumptions of the PRA, and combine that with other information from design bases, defense-in-depth, and safety margins to finalize the categorization of the SSCs. As a result, the NRC staff finds that the licensee will perform a sensitivity study consistent with Table 5-2 of the NEI 00-04 guidance, and the IDP will appropriately consider PRA assumptions and simplifications during the SSC categorization process to address the identified key assumptions and sources of uncertainty.

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

The NRC staff finds that the assessment performed to identify the key assumptions and sources of uncertainty, and to address SOKC, is consistent with the guidance provided in NUREG-1855, Revision 1, and is acceptable for this application.

Integrated Importance Measures

Section 50.69(c)(1)(ii) of 10 CFR requires that the SSC functional importance be determined using an integrated, systematic process. Section 5.6, "Integral Assessment," of NEI 00-04, Revision 0 (Reference [11]), discusses the need for an integrated computation using the available importance measures. The guidance further states, in part, that the, "integrated importance measure essentially weights the importance from each risk contributor (e.g., internal events, fire, seismic and [high wind] PRAs) by the fraction of the total core damage frequency [or large early release frequency] contributed by that contributor." The guidance also provides formulas to compute the integrated F-V, and integrated RAW.

In Attachment 8 of the LAR as updated (Reference [2]), the licensee stated that, to calculate the integrated importance measures, the equations presented in NEI 00-04, Revision 0, Section 5.6 are used, and the resulting integrated importance measures are compared against the screening criteria in NEI 00-04. These equations essentially weight the importance from each risk contributor (i.e., internal events (including flooding), fire, seismic, and high winds) by the fraction of that contributor to the total CDF or total LERF. The license stated that all basic events are mapped to the affected component and, therefore, random and hazard-induced failures of a component are mapped to the appropriate component. The licensee clarified that an integrated assessment is not performed on a component ranked HSS based on the IEPRA because the component will be ranked HSS regardless of the results of the integrated assessment. The licensee also clarified that an integrated assessment will not be performed on a component ranked LSS by all PRA assessments because the results from the integrated assessment cannot be higher than the maximum of the individual contributors.

The scope of modeled hazards for Callaway includes the IEPRA (including internal floods), FPRA, SPRA, and HW PRA. The NRC staff finds that the licensee's use and treatment of importance measures are consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1 (References [11] and [5]).

PRA Acceptability Conclusions

Pursuant to 10 CFR 50.69(c)(1)(i), the categorization process must consider results and insights from a plant-specific PRA. The use of the IEPRA, FPRA, SPRA, and HW PRA to support SSC categorization is endorsed by RG 1.201, Revision 1. The PRAs must be acceptable to support the categorization process and must be subjected to a peer review process assessed against a standard that is endorsed by the NRC. Revision 2 of RG 1.200 (Reference [6]), which was the applicable guidance at the time the initial LAR (Reference [1]) was submitted, provides guidance for determining the acceptability of the PRA by comparing the PRA to the relevant parts of the ASME/ANS RA-Sa 2009 PRA standard (for IEPRA, FPRA, and HWPRA) using a peer review process. The SPRA was reviewed against ASME/ANS RA-A Case 1 (Reference [27]) which was accepted in a letter by the NRC dated March 12, 2018 (Reference [28]), and is now endorsed by the NRC in RG 1.200, Revision 3, dated December 2020 (Reference [7]).

The licensee has subjected the IEPRA, FPRA, SPRA, and HW PRA 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 NRC staff concludes that (1) the licensee's IEPRA, FPRA, SPRA, and HW PRA are acceptable to support the categorization of SSCs using the process endorsed by 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 NUREG-1855 (References [5], [6], and [9], respectively), as applicable, and addressed appropriately for this application.

The NRC staff finds the licensee provided the required information, and the IEPRA (including internal floods), FPRA, SPRA, and HW PRA, are acceptable, and therefore, meet the requirements set forth in Sections 50.69(c)(1)(i) and (ii) of 10 CFR.

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

Other External Hazards

This hazard category includes all external hazards, except for seismic and high winds, such as external floods, transportation, nearby facility accidents, and other hazards.

In the staff evaluation report for the Callaway IPEEE (Reference [37]), the NRC staff states, in part, "floods, transportation and other external events areas were adequately addressed" based on either compliance with the 1975 NRC Standard Review Plan (SRP) criteria or on the basis of a bounding probabilistic assessment resulting in a CDF estimate less than 1E-6 per reactor year (i.e., below the NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," screening criterion) (References [38] and [37]). In Section 3.2.5 of the LAR as updated (Reference [2]), the licensee stated, in part, all other external hazards (i.e., excluding seismic and high winds hazards) were screened for applicability to Callaway per a plant-specific evaluation against the criteria in Part 6 of the ASME/ANS PRA Standard RA-Sa-2009. In Attachment 4 of the LAR as updated (Reference [2]), the licensee provided the updated results of the plant-specific evaluation that assessed the IPEEE results to the endorsed criteria in the ASME/ANS RA-Sa-2009 PRA Standard and current plant hazard information.

In Section 3.2.5 and Attachment 8 of the LAR as updated (Reference [2]), the licensee states that the guidance of NEI 00-04, Revision 0, Figure 5-6, "Other External Hazards," will be applied for SSCs being credited in screening an external hazard at the time of categorizing an SSC. NEI 00-04, Figure 5-6, provides the guidance to use in determining SSC safety significance for other external hazards (excluding in this case internal fires, seismic hazards, and high winds) by addressing whether an SSC "participates" in the screening hazard scenario. The NRC staff finds that Callaway will assess the risk from all other external hazards consistent with Figure 5-6 of NEI 00-04 as endorsed in RG 1.201, Revision 1.

In summary, update of the IPEEE results described by the licensee in the updated LAR (Reference [2]) and the licensee's assessment of the other external hazards (i.e., external flood) is consistent with Section 5 of NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1 (References [11] and [5], respectively). The NRC staff concludes that the licensee's treatment of other external hazards is acceptable and meets the requirements set forth in 10 CFR 50.69(c)(1)(ii).

Component Safety Significance Assessment for Passive Components

In Section 3.1.2 of the LAR as updated (Reference [2]), the licensee proposed using a categorization method for passive components not cited in NEI 00-04, Revision 0, or RG 1.201, Revision 1 (References [11] and [5], respectively), 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 [39]). 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 repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs.

In Section 3.1.2 of the LAR as updated, the licensee stated, in part:

> The passive categorization process is intended to apply the same risk-informed process accepted by the NRC in the… ANO2-R&R-004 document for the passive categorization of Class 2, 3, and non-class components. This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662 as published in Regulatory Guide 1.147, Revision 15. Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned high safety-significant, HSS, for passive categorization, which will result in HSS for its risk-informed safety classification and cannot be changed by the IDP.

The NRC staff finds the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process.

3.3.1.3 Key Principle 4 Conclusions

Based on the above, the NRC staff's review for IEPRA (including internal floods), FPRA, SPRA, and HW PRA acceptability and evaluation of the use of non-PRA methods, concludes that the proposed change satisfies the fourth key principle for risk-informed decisionmaking prescribed in RG 1.174, Revision 3.

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

NEI 00-04, Revision 0 (Reference [11]), 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., including 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 over time are continually incorporated.

Sections 11 and 12 of NEI 00-04, Revision 0, include discussions 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 Callaway as-built-as-operated plant.

The NRC staff finds the risk management process described by the licensee in the LAR as updated (Reference [2]) is consistent with Sections 11 and 12 of NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1, and consistent with the requirements in 10 CFR 50.69(e). Based on the above, the NRC staff has determined that the proposed change satisfies the fifth key principle for risk-informed decisionmaking prescribed in RG 1.174, Revision 3.

4.0 CHANGES TO THE OPERATING LICENSE

Based on the staff's review of the LAR and the licensee's responses to the staff's RAIs, 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 and the guidance in RG 1.201, Revision 1 (Reference [5]), and NEI 00-04, Revision 0 (Reference [11]). Note: Additional actions (e.g., final procedures) need not, and have not been submitted or reviewed by the NRC 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 (Reference 1) and other changes that were described by the licensee, the NRC staff finds to be routine and systematically addressed through the configuration management and control and periodic update processes as described in Sections 3.3.1.1 and 3.3.2 of this SE.

The licensee proposed the following amendment to the RFOLs for the Callaway Plant, Unit No. 1. The proposed license condition states:

> Ameren Missouri 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 SSCs using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire, high winds, and seismic risk; 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 SSCs and their associated supports; and the results of non-PRA evaluations that are based on a screening of other external hazards updated using the external hazard screening significance process identified in American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009 PRA Standard; as specified in License Amendment 226 dated December 29, 2021. Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process

specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

The NRC staff finds that the proposed license condition is acceptable, because: (1) it adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed by the NRC; and (2) the evaluation in SE Section 3.3.1.2, finds the non-PRA method for assessing risk for passive components, which is a deviation from NEI 00-04, to be acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri State official was notified of the proposed issuance of the amendment on November 26, 2021. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, as published in *Federal Register* on January 26, 2021 (85 FR 7117), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

8.0 REFERENCES

[1] Banker, S., Ameren Missouri, letter to U.S. Nuclear Regulatory Commission, Docket Number 50-483, Callaway Plant Unit No. 1, Union Electric Co., Renewed Facility Operating License NPF-30, "Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors,'" dated October 30, 2020 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML20304A454).

[2] Meyer, S. J., Ameren Missouri, letter to U.S. Nuclear Regulatory Commission, Docket Number 50-483, Callaway Plant Unit No. 1, Union Electric Co., Renewed Facility Operating License NPF-30, "Supplemental Information for Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,'" dated July 29, 2021 (ADAMS Package Accession No. ML21210A025).

[3] McLachlin, M., Ameren Missouri, letter to U.S. Nuclear Regulatory Commission, Docket Number 50-483, Callaway Plant Unit No. 1, Union Electric Co., Renewed Facility Operating License NPF-30, "Response to Request for Additional Information Regarding License Amendment Requests to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,'" dated October 13, 2021 (ADAMS Package Accession No. ML21286A680).

[4] Meyer, S. J., Ameren Missouri, letter to U.S. Nuclear Regulatory Commission, Docket Number 50-483, Callaway Plant Unit No. 1, Union Electric Co., Renewed Facility Operating License NPF-30, "Final Supplement to License Amendment Request to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,'" dated December 22, 2021 (ADAMS Package Accession No. ML21356B505).

[5] U.S. Nuclear Regulatory Commission, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," Regulatory Guide 1.201, Revision 1, dated May 2006 (ADAMS Accession No. ML061090627).

[6] U.S. Nuclear Regulatory Commission, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200, Revision 2, dated March 2009 (ADAMS Accession No. ML090410014).

[7] U.S. Nuclear Regulatory Commission, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200, Revision 3, dated December 2020 (ADAMS Accession No. ML20238B871).

[8] U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, Revision 3, dated January 2018 (ADAMS Accession No. ML17317A256).

[9] U.S. Nuclear Regulatory Commission, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," Final Report, NUREG-1855, Revision 1, dated March 2017 (ADAMS Accession No. ML17062A466).

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Date: December 29, 2021

SUBJECT: CALLAWAY PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 226 REGARDING ADOPTION 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 PLANTS" (EPID L-2020-LLA-0235) DATED DECEMBER 29, 2021

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