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

June 22, 2022

Mr. David P. Rhoades
Senior Vice President
Constellation Energy Generation, LLC
President and Chief Nuclear Officer
Constellation Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT – ISSUANCE OF AMENDMENT NO. 151 RE: ADOPT 10 CFR 50.69, “RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS” (EPID L-2021-LLA-0092)

Dear Mr. Rhoades:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 151 to Renewed Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant in response to your application[1] dated May 20, 2021, as supplemented by letters dated October 14, 2021 and May 26, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML21141A009, ML21287A006 and ML22146A104, respectively).

The amendment added a new license condition to the Renewed Facility Operating License to allow the implementation of risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors in accordance with Title 10 of the *Code of Federal Regulations* Section 50.69.

[1] By letter dated February 1, 2022 (ML22032A333), Constellation Energy Generation, LLC (CEG) notified the NRC that CEG completed a license transfer and reorganization, and requested the NRC to continue processing pending NRC actions previously requested by Exelon Generation Company, LLC.

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

Sincerely,

/RA/

V. Sreenivas, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 151 to
   Renewed License No. DPR-18
2. Safety Evaluation

cc: Listserv

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

CONSTELLATION GENERATION COMPANY, LLC

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 151
Renewed License No. DPR-18

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

   The application for amendment by Exelon Generation Company, LLC[2] dated May 20, 2021, as supplemented by letters dated October 14, 2021 and May 26, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   B. 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;

   C. 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

   D. 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] By letter dated February 1, 2022 (ML22032A333), Constellation Energy Generation, LLC (CEG) notified the NRC that CEG completed a license transfer and reorganization, and requested the NRC to continue processing pending NRC actions previously requested by Exelon Generation Company, LLC.

2. Accordingly, the paragraph 2.C.(20) of the Renewed Facility Operating License No. DPR-18 is hereby amended to read as follows:

> Constellation Energy Generation, LLC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Exelon's submittal letter dated May 20, 2021, and all its subsequent associated supplements as specified in License Amendment No. 151 dated June 22, 2022.
>
> Constellation Energy Generation, LLC will complete the implementation items listed in Attachment 7 of Exelon Letter to the NRC dated May 20, 2021, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa -2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to the implementation of 10 CFR 50.69.
>
> 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 license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

Digitally signed by James G. Danna
Date: 2022.06.22 21:46:21 -04'00'

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: June 22, 2022

ATTACHMENT TO LICENSE AMENDMENT NO. 151

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

Replace the following pages of Renewed Facility Operating License No. DPR-18 with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

Remove	Insert
9	9
10	10
- - - - - -	11

(17) Adoption of Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times -RITSTF Initiative 4b"

Constellation Energy Generation, LLC is approved to implement TSTF-505, Revision 2, modifying the Technical Specification requirements related to Completion Times (CT) for Required Actions to provide the option to calculate a longer, risk-informed CT (RICT). The methodology for using the new Risk-Informed Completion Time Program is described in NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0, which was approved by the NRC on May 17, 2007.

Constellation Energy Generation, LLC will complete the implementation items listed in Attachment 6 of Exelon Letter to the NRC dated May 20, 2021, prior to implementation of the RICT Program. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to the implementation of the RICT Program.

(18) Deleted

(19) Constellation Energy Generation, LLC shall provide to the Director of the Office of Nuclear Reactor Regulation or the Director of the Office of Nuclear Material Safety and Safeguards, as applicable, a copy of any application, at the time it is filed, to transfer (excluding grants of security interests or liens) from Constellation Energy Generation, LLC to its direct or indirect parent, or to any other affiliated company, facilities for the production, transmission, or distribution of electric energy having a depreciated book value exceeding ten percent (10%) of Constellation Energy Generation, LLC's consolidated net utility plant, as recorded on Constellation Energy Generation, LLC's books of account.

(20) Constellation Energy Generation, LLC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Exelon's submittal letter dated May 20, 2021, and all its subsequent associated supplements as specified in License Amendment No. 151 dated June 22,2022.

Constellation Energy Generation, LLC will complete the implementation items listed in Attachment 7 of Exelon Letter to the NRC dated May 20, 2021, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa - 2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to the implementation of 10 CFR 50.69.

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. The facility requires an exemption from certain requirements of 10 CFR 50.46(a)(1). This includes an exemption from 50.46(a)(1), that emergency core cooling system (ECCS) performance be calculated in accordance with an acceptable calculational model which conforms to the provisions in Appendix K (SER dated April 18, 1978). The exemption will expire upon receipt and approval of revised ECCS calculations. The aforementioned exemption is authorized by law and will not endanger life property or the common defense and security and is otherwise in the public interest. Therefore, the exemption is hereby granted pursuant to 10 CFR 50.12

E. Constellation Energy Generation, LLC shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27827 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: “R. E. Ginna Nuclear Power Plant Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan,” submitted by letter dated May 15, 2006.

Constellation Energy Generation, LLC shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The licensee’s CSP was approved by License Amendment No. 113 and modified by License Amendment No. 117. The licensee has obtained Commission authorization to use Section 161A preemption authority under 42 U.S.C. 2201a for weapons at its facility.

F. The Updated Final Safety Analysis Report supplement, submitted pursuant to 10 CFR 54.21 (d), describes certain future activities to be completed prior to the period of extended operation. Ginna LLC shall complete these activities no later than September 18, 2009, and shall notify the Commission in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement, as revised, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 1O CFR 50.71 (e)(4) following issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs and activities described in the supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

G. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. Any capsules placed in storage must be maintained for future insertion, unless approved by the NRC.

H. This renewed license is effective as of the date of issuance and shall expire at midnight on September 18, 2029.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachment: Appendix A - Technical Specifications

Date of Issuance: May 19, 2004

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 151 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-18

CONSTELLATION GENERATION COMPANY, LLC

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated May 20, 2021 (Reference [1]), as supplemented by letters dated October 14, 2021 (Reference [2]) and May 26, 2022 (Reference [44]), Exelon Generation Company, LLC[1] submitted a license amendment request (LAR) for the R. E. Ginna Nuclear Power Plant (Ginna). The licensee proposed a license condition to the Renewed Facility Operating License to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors."

The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (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[2].

On August 25, 2021 (Reference [3]), the NRC staff issued a regulatory audit plan that included audit questions. The NRC staff participated in the regulatory audit in September 2021. The NRC staff performed the audit to ascertain the information needed to support its review of the application and develop requests for additional information, as needed. The licensee responded to the audit questions in a supplement dated October 14, 2021. The licensee submitted an updated license condition in a second supplement dated May 26, 2022. The supplemental letters provided additional information that clarified the application and the license condition, did not expand the scope of the application as originally noticed, and did not change

[1] By letter dated February 1, 2022 (ML22032A333), Constellation Energy Generation, LLC (CEG) notified the NRC that CEG completed a license transfer and reorganization, and requested the NRC to continue processing pending NRC actions previously requested by Exelon Generation Company, LLC.
[2] Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," May 2006 (Reference 4), describes the SSC categorization process in its entirety as an overarching approach that includes multiple approaches and methods identified for a PRA hazard and non-PRA methods.

NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 10, 2021 (86 FR 43691).

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

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

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 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 high safety significance.

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 [4])

- RG 1.200, Revision 2 and 3, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference [5] and [20])

- 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 [6])

- NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking" (Reference [7])

- NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (Reference [8])

- *Federal Register* notice (69 FRN 68008, 68028-68029), dated November 22, 2004, related to "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" (Reference [45])

NRC-Endorsed Guidance

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference [9]), as endorsed by RG 1.201 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/elements of the SSC categorization process for meeting the requirements of 10 CFR 50.69:

- Sections 3.2 and 5.1 provide specific guidance corresponding to 10 CFR 50.69(c)(1)(i).
- Sections 3, 4, 5, and 7 provide specific guidance corresponding to 10 CFR 50.69(c)(1)(ii).
- Section 6 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iii).
- Section 8 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iv).
- Section 2 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(v).
- Sections 9 and 10 provide specific guidance corresponding to 10 CFR 50.69(c)(2).

Additionally, Section 11 of NEI 00-04 provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(f). Section 12 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).

The proposed amendment would use methodology from the NRC approved LARs related to 10 CFR 50.69 from the Arkansas Nuclear One, Unit 2 (ANO-2). For the purposes of 10 CFR 50.69 categorization, passive components are those components that have a pressure retaining function. Passive components and the passive function of active components were evaluated using the (ANO-2) Risk-Informed Repair/Replacement Activities (RI-RRA) methodology contained in NRC letter to Use Risk-Informed Safety Classification (RISC) and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems (Reference [12]).

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

An acceptable approach for making risk-informed decisions about proposed technical specification (TS) changes, including both permanent and temporary changes, is to show that the proposed licensing basis changes meet the five key principles stated in Section C of RG 1.174, Revision 3 (Reference [6]). 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 increase should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

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

3.2 Traditional Engineering Evaluation

The traditional engineering evaluation below addresses the first three key principles of RG 1.174, Revision 3 and are pertinent to: (1) compliance with current regulations, (2) evaluation of defense-in-depth, and (3) 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 decision-making process to categorize safety-related and non-safety-related SSCs according to the safety significance of the functions they perform into one of the following four RISC categories, which are defined in 10 CFR 50.69(a), as follows:

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

The SSCs are classified as having either HSS functions (i.e., RISC-1 and RISC-2 categories) or LSS functions (i.e., RISC-3 and RISC-4 categories). For HSS SSCs, 10 CFR 50.69 maintains current regulatory requirements for special treatment (i.e., it does not remove any requirements from these SSCs). In addition, 10 CFR 50.69(d)(1) requires that the licensee or applicant shall

[3] NEI 00-04, Revision 0 (Reference [[9]), uses the term "high-safety-significant" to refer to SSCs that perform safety-significant functions. The NRC understands HSS to have the same meaning as "safety-significant" (i.e., SSCs that are categorized as RISC-1 or RISC-2), as used in 10 CFR 50.69.

ensure that RISC-1 and RISC-2 SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance. For LSS SSCs, licensees may implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2). For RISC-3 SSCs, licensees can replace certain special treatment requirements with an alternative treatment approach that must meet 10 CFR 50.69(d)(2). 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 certain 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) certain requirements for containment leakage testing
- (xi) certain requirements of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100

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 [9] 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, and NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1 (Reference [4]) with specific clarifications.

Section 2 of NEI 00-04, Revision 0, (Reference [9]) in part, states that the categorization process includes eight primary steps:

1. Assembly of Plant-Specific Inputs (Section 3 of NEI 00-04, Revision 0)
2. System Engineering Assessment (Section 4 of NEI 00-04, Revision 0)
3. Component Safety Significance Assessment (Section 5 of NEI 00-04, Revision 0)
4. Defense-In-Depth Assessment (Section 6 of NEI 00-04, Revision 0)
5. Preliminary Engineering Categorization of Functions (Section 7 of NEI 00-04, Revision 0)
6. Risk Sensitivity Study (Section 8 of NEI 00-04, Revision 0)
7. 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 3.1.1 of the LAR (Reference [1]), the licensee stated that it will implement the risk-informed categorization process in accordance with NEI 00-04, Revision 0 (Reference [9]), as endorsed in RG 1.201, Revision 1 (Reference [4]). In Sections 3.1.1 and 3.2.3 of the LAR, the licensee has proposed the use of the alternate seismic approach as an alternative method to assess the seismic hazard contribution. The NRC notes that use of alternative methods is a deviation from the NEI 00-04 guidance, as endorsed. A more detailed staff review of this alternative method is provided in Section 3.3.1.2 of this SE.

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 0 (References [9] and [4], respectively). Categorization process is performed for entire systems and structures, rather than for selected components within a system or structure, as reviewed in this SE, and therefore, the requirements in 10CFR 50.69(c)(1)(v) will be met upon implementation.

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 (Reference [9]) and clarifications in RG 1.201, Revision 1 (Reference [4]), 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 taken. The licensee’s SSC categorization program includes the appropriate steps/elements prescribed in NEI 00-04, Revision 0 to assure that SSCs specified are appropriately categorized consistent with 10 CFR 50.69. The staff performed a more detailed review of specific steps/elements of the licensee’s SSC categorization process where necessary to confirm consistency with the NEI 00-04 guidance, as endorsed. 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 decision making prescribed in RG 1.174, Revision 3 (Reference [6]).

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 (Reference [4]), endorses the guidance in Section 6 of NEI 00-04 (Reference [9]), 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 and B of

Appendix J 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 of the LAR (Reference [1]), 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 (Reference [9]). Based on the above, the 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 [6]), 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 NRC staff reviewed the licensee's established procedure(s) to be implement prior to the use of the categorization process on a plant system. The procedure(s) will ensure that the evaluations 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. With sufficient safety margins, (1) the codes and standards or their alternatives approved for use by the NRC are met and (2) safety analysis acceptance criteria in the licensing basis (e.g., Updated Final Safety Analysis Report (UFSAR), 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 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 [6]) and would therefore meet 10 CFR 50.69(c)(1)(iv). There is no impact to safety analysis acceptance criteria as described in the plant licensing basis. In addition, the SSCs design basis function as described in the plant's licensing basis, including the UFSAR will continue to be met.

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 address the fourth and fifth key principles of the NRC staff's standards for risk-informed decision making, pertaining to the assessment for change in risk and monitoring the impact of the licensing basis change.

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

In Sections 3.2.1 and 3.2.2 of the LAR (Reference [1]), the licensee indicates that the Ginna categorization process uses PRA modeled hazards to assess risks for the internal events (includes internal flood), and internal fires. For the other risk contributors, the licensee's process uses the following non-PRA methods to characterize the risk:

- Seismic Hazard: Alternative seismic treatment using guidance from Electric Power Research Institute (EPRI) Report 3002017583 dated February 29, 2020 (Reference [25]), and qualitative insights about seismic risk at Ginna.
- Other External Hazards: Screening analysis performed for Individual Plant Examination of External Events (IPEEE) (Reference [11]) updated using criteria from Part 6 of the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa–2009, "Addendum A to RA-S–2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," (the PRA Standard, Reference [15]), as endorsed by the NRC.
- Shutdown Events: Safe Shutdown Risk Management program consistent with NUMARC 91-06 (Reference [17]).
- Passive Components: ANO-2 passive categorization methodology (Reference [12]).

The approaches and methods proposed by the licensee to address internal events, including internal flooding, seismic, other external events, defense-in-depth, and shutdown events are consistent with the approaches and methods included in the guidance in NEI 00-04, Revision 0 (Reference [9]). The licensee proposed to use the shutdown safety assessment process based on NUMARC 91-06. The shutdown safety assessment method is consistent with the guidance in NEI 00-04. The non-PRA method for the categorization for passive components is consistent with the ANO-2 methodology for passive components (Reference [12]) approved for risk-informed safety classification and treatment for repair/replacement activities in ASME BPV Code Class 2 and 3 moderate- and high-energy systems. The NRC staff concludes that the internal events PRA with the completion of the proposed implementation items meet the internal events PRA requirement in 10 CFR 50.69(c)(1)(i).The licensee's use of the ANO-2 methodology in the SSC categorization process is addressed in Section 3.3.1.4 of this SE. To address seismic hazard in the SSC categorization process, the licensee proposed to use an alternative method not endorsed by the NRC in NEI 00-04. A detailed NRC staff review of the licensee's proposed alternative seismic approach is provided in Section 3.3.1.2 of this SE.

3.3.1.1 Scope of the PRA

The NRC staff reviewed two aspects of the PRA with regard to the impact of the proposed changes on plant operational risk: (1) scope and acceptability of the PRA models and their application to the proposed changes, and (2) a review of the PRA results and insights described in the licensee's application.

<u>Evaluation of Modeled PRAs</u>

Probabilistic Risk Assessment models to evaluate risk associated with internal events (including internal flooding) PRA (IEPRA) and internal fires, hereinafter fire PRA (FPRA). In Section 3.2 of Enclosure 1 of the LAR, the licensee confirmed that the PRA models had been peer reviewed using the ASME/ANS RA-Sc -2007 PRA Standard for the IEPRA, and ASME/ANS RA-Sa -2009 PRA Standard for the FPRA, as endorsed by RG 1.200, Revision 1 and 2, respectively. For the open Facts and Observations (F&Os) resulting in these peer reviews the licensee stated that closure of the F&Os was performed using an independent assessment process. The NRC staff confirmed that the licensee performed closure of the F&Os consistent with Appendix X to

NEI 05-04, 07-12, and 12-13, as endorsed in RG 1.200, Revision 3 (Reference [20]). The NRC evaluated the two remaining open F&Os, along with their dispositions. In Enclosure 9 of the LAR, the licensee provided a brief discussion and list of the key assumptions and sources of uncertainty, along with treatment for the application of 50.69.

In its supplement dated October 14, 2021 (Reference [2]), Ginna confirmed that the IEPRA and FPRA models credit installed and portable equipment used as part of the FLEX strategy.

The NRC staff concluded that the licensee's credit for FLEX equipment in the 10 CFR 50.69 application is appropriate because the licensee used consensus human reliability analysis methodologies and practices and acceptable failures rates, and performed sensitivity studies to assess the impact on the 50.69 application.

The NRC staff notes that Ginna confirmed for the equipment credited in the adoption of National Fire Protection Association (NFPA) -805 that is also used to mitigate non-fire events. Use of this equipment is directly referenced in the emergency response procedures, thereby ensuring this equipment can be used for all hazards.

The NRC staff reviewed the PRA models peer-review history provided by the licensee in Enclosure 1 of the LAR, as supplemented. The NRC staff further considered the key assumptions and sources of uncertainty identified by the licensee and credit for FLEX. The NRC staff finds the Ginna scope and acceptability of the modeled IEPRA and FPRA to be commensurate with the 50.69 application for use in the integrated decision-making process and consistent with RG 1.174.

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

Alternative Seismic Approach

As part of its proposed process to categorize SSCs according to safety significance, the licensee proposed to use a non-PRA method to consider seismic hazards. The regulations in 10 CFR 50.69(c)(1)(ii) and 50.69(b)(2)(ii) permit the use of systematic evaluation techniques in the risk-informed categorization process. The licensee provided a description of its proposed alternative approach for considering seismic risk in the categorization process in Section 3.2.3 of the LAR (Reference [1]) and its supplement dated October 14, 2021 (Reference [2]). In part, the licensee based its plant-specific evaluation on the case studies performed in EPRI Report 3002017583 (Reference [25]), and stated that the case studies are applicable to Ginna and are used in the alternative seismic approach; and that the measures for assuring the quality and level of detail for the licensee's proposed alternative seismic approach are adequate for the categorization of SSCs. Therefore, based on the above, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(ii) for the proposed alternative seismic approach are met.

The EPRI Report 3002017583 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from seismic PRAs (SPRAs). The licensee states in Footnote 1, Item 2 of Section 3.2.3 of the Ginna LAR that the Ginna LAR incorporates a prior and similar licensee response to request for additional information (RAI) APLC-03 for the Clinton Power Station 10 CFR 50.69 LAR (Reference [37]) using the same alternative seismic approach.

Evaluation of the Information Provided for the Proposed Alternative Seismic Approach

The licensee indicated that aside from updates included in an RAI submittal for the Calvert Cliffs Nuclear Power Plant 50.69 LAR (the first submittal to propose the alternative seismic approach) into the previous version of this report, EPRI 3002012988 (Reference [26]), the technical criteria in EPRI Report 3002017583 is unchanged from its predecessor report EPRI 3002012988. The NRC staff's review confirmed that the case studies in EPRI Report 3002017583 used by the licensee to support its proposed alternative seismic approach, as well as the information in its supplements, provided sufficient plant-specific evaluation of the applicability and differences for Ginna as compared to the amendment approved by the NRC for Calvert Cliffs on February 28, 2020 (Reference [30]). Accordingly, the use of EPRI Report 3002017583 and its cited case studies have been previously approved by the NRC. The information presented in the LAR and its supplement dated October 14, 2021 (Reference [2]) provided a sufficient description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv) for the alternative seismic approach. Therefore, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(iv) are met for the proposed alternative seismic approach.

Evaluation of Technical Acceptability of the PRAs Used for Case Studies Supporting the Proposed Alternative Seismic Approach

In Section 3.2.3 of the LAR, the licensee stated that the plant specific case studies from other licensees in the EPRI Report 3002017583 are incorporated by reference to support its proposed alternative seismic approach. The licensee stated that the case study Plants A, C, and D, pertaining to the technical acceptability of the PRAs used, as well as the technical adequacy of certain technical details of the conduct of the case studies, are applicable to Ginna. The NRC staff reviewed and evaluated the technical acceptability of the PRAs used in case studies for Plants A, C, and D, in EPRI Report 3002017583, and the licensee's assertion of plant-specific applicability to the approach used in the Calvert Cliffs amendment. The NRC staff also evaluated the peer review process and resolution of peer review findings, and key assumptions and sources of uncertainties for Plants A, C, and D. Accordingly, the use of EPRI Report 3002017583 and its cited case studies have been previously approved by the NRC and is also applicable to Ginna, with supporting information.

Based on the above, the NRC staff finds that the acceptability of PRAs used in the Plants A, C, and D, case studies in EPRI Report 3002017583, the mapping approach used in those case studies, and the conclusions on the determination of unique HSS SSCs from the case studies in the Calvert Cliffs amendment are applicable to this licensee's proposed plant-specific alternative seismic approach. Therefore, the NRC staff concludes that the Plants A, C, and D, PRAs are technically acceptable and applicable to use in the corresponding case studies supporting the licensee's proposed alternative seismic approach; the mapping of SSCs between the SPRA, the full-power IEPRA and, as applicable, the FPRA for the Plants A, C, and D, case studies. The licensee's plant-specific evaluation is technically justifiable to support conclusions on the determination of unique HSS SSCs from SPRAs in Plants A, C, and D, case studies in EPRI Report 3002017583; and applicable to Ginna and the licensee's proposed alternative seismic approach.

Evaluation of the Criteria for the Proposed Alternative Seismic Approach

In the LAR the licensee states, in part, that the ground motion response spectrum (GMRS) peak acceleration for Ginna is significantly below the safe-shutdown earthquake (SSE) between 1.0 Hertz (Hz) and 10 Hz. The NRC staff also notes that the GMRS is below 0.2g except for frequencies between 8 to 10 Hz, where it is slightly above 0.2g.

The licensee further states that the GMRS to SSE comparison demonstrates that Ginna qualifies as a Tier 1 plant under the criteria in EPRI Report 3002017583 and that this comparison confirms the expected seismic risk at Ginna would be very low. The NRC staff notes that the licensee's plant-specific evaluation is supported by its NRC 10 CFR 50.54(f) response dated March 31, 2014 (Reference [16]). The NRC staff reviewed the licensee's submittal and supplements and plant-specific evaluation and concluded that the proposed criteria in EPRI Report 3002017583 to determine the applicability and use of the proposed seismic Tier 1 approach is acceptable.

Evaluation of Applicability of Criteria for the Proposed Alternative Seismic Approach

In Section 3.2.3 of the LAR, the licensee compared the Ginna GMRS from the reevaluated seismic hazard developed and submitted by the licensee in response to Near-Term Task Force (NTTF) Recommendation 2.1 against the site's design basis SSE, as shown in Attachment 4 of the LAR to demonstrate that the site meets the criteria for application of the proposed alternative seismic approach as a Tier 1 plant.

In Section 3.2.3 of the LAR, the licensee stated that the NRC staff concluded that the methodology used by the licensee in determining the GMRS was acceptable and that the GMRS determined by the licensee adequately characterized the reevaluated hazard for the Ginna site. The NRC staff's review confirmed the licensee's statements and the comparison of the GMRS from the reevaluated seismic hazard against the SSE. Based on its review, the NRC staff finds that the licensee's seismic hazard meets the criteria for the proposed alternative seismic approach.

In Section 3.2.3 of the LAR, the licensee stated that the small percentage contribution of seismic risk to total plant risk makes it unlikely that an integral importance assessment for a component, as defined in NEI 00-04, would result in an overall HSS determination. In APLC Question 02 in the audit plan (Reference [3]), the NRC staff cited Section 2.2.2 of the EPRI report, which identifies the expectation that low contribution of seismic risk to the total plant risk reduces the likelihood of a unique seismic condition that would cause an SSC to be designated HSS, and, in contrast, Enclosure 5 of the Ginna Technical Specification Task Force (TSTF)-505 LAR (Reference [36]), which indicates that the seismic LERF (SLERF) contribution to the total plant LERF is high (i.e., 68 percent). In response to APLC Question 02 (Reference [2]), the licensee states that the GMRS peak acceleration for Ginna is significantly below the SSE in the range of 1 – 10 Hz, and SLERF estimate in the Ginna RICT LAR is to be used as an SLERF penalty value in RICT calculation and contains conservatism in the calculation.

The NRC staff verified the licensee's estimate of the Ginna seismic CDF (SCDF) by mathematically convolving the plant-level high confidence in low probability of failure (HCLPF) of 0.2g peak ground acceleration (PGA) with composite uncertainty of 0.4 provided in the Ginna TSTF-505 submittal, which is the most recent publicly available information, and the site-specific reevaluated seismic hazard for the mean PGA. The NRC staff noted that SCDF contribution to total plant CDF is low (i.e., 7 percent). The NRC staff reviewed the licensee's description of its SLERF estimate in the Ginna TSTF-505 submittal by using a plant-limiting fragility for containment integrity of 0.2g PGA HCLPF, which is acceptably bounding. The staff noted the licensee's SLERF estimate in the TSTF-505 LAR is expected to be conservative because it was estimated using a fragility value appreciably lower than the fragility of components that are known to be dominant contributors to seismically induced LERF, such as the estimates in Appendix B of NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants" (Reference [38]). The staff also noted that SLERF is about 50 percent of total plant LERF, considering HCLPF of 0.3g for containment integrity based on NUREG/CR-4334 for SLERF estimate. However, overall seismic risk is relatively low compared to total plant risk due to its low SCDF contribution of 7 percent.

Further, as noted in Section 3.6.5 of EPRI Report 3002017583, containment defense-in-depth assessment addresses containment failures and containment bypass situations.

Section 3.6.6 of EPRI Report 3002017583, used for the licensee's proposed alternative seismic approach, recommends that if the licensee chooses to categorize civil structures housing HSS SSCs, the structures are considered as HSS. Therefore, based on its evaluation and review, the NRC staff concludes that the proposed alternative seismic approach, in conjunction with the other elements of the 10 CFR 50.69 categorization program, will appropriately determine the safety significance of any SSCs whose seismic-induced failures led directly to core damage and large early release and that the seismic risk contribution would not solely result in any additional SSC being categorized as HSS.

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

Evaluation of the Implementation of Conclusions from the Case Studies

The categorization conclusions from EPRI Report 3002017583 case studies performed for GMRS to SSE ratios significantly higher than Ginna indicated that seismic-specific failure modes resulted in HSS categorization uniquely from SPRAs. Therefore, such seismic-specific failure modes, such as correlated failures, relay chatter, and passive component structural failure mode, can influence the categorization process. The NRC staff reviewed the proposed alternative seismic approach to evaluate whether the categorization-related conclusions from EPRI Report 3002017583 were appropriately included and implemented.

In Section 3.2.3 of the LAR, the licensee discussed the proposed alternative seismic approach. The licensee stated that the proposed categorization approach for seismic hazards will include qualitative consideration of the mitigation capabilities of SSCs during seismically induced events and seismic failure modes, based on insights obtained from prior seismic evaluations performed for Ginna.

The licensee stated that the qualitative characterization of seismic risk performed for the Integrated Decision-Making Panel will include information from the various post-Fukushima seismic reviews including results of seismic walkdowns, seismic mitigation strategy assessment, and seismic high frequency evaluations. The objective of the alternative seismic approach, as described in Section 3.2.3 of the LAR, is to identify plant-specific seismic insights derived from the components in the system being categorized.

The NRC staff's review of the licensee's proposed alternative seismic approach determined that there are no differences that exist between the Ginna proposed alternative approach and the approach approved by the NRC staff in the Calvert Cliffs 10 CFR 50.69 SE (Reference [30]) and that the plant-specific implementation of the alternative seismic approach is acceptable. The

NRC staff's review of the proposed alternative seismic approach, in conjunction with the requirements in 10 CFR 50.69 and the corresponding statement of consideration, finds that the proposed alternative seismic approach includes the evaluations required by 10 CFR 50.69(c)(1)(ii), as well as 10 CFR 50.69(c)(1)(iv) because:

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

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

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

4. The proposed alternative seismic approach presents the Integrated Decision-Making Panel with the basis for the proposed alternative seismic approach, including the low seismic hazard for the plant and the criteria for use of the proposed alternative seismic approach.

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

Consideration of Changes to Seismic Hazard

An important input to the NRC staff's evaluation of the proposed alternative seismic approach is the current knowledge of the seismic hazard at the plant. The possibility exists for the seismic hazard at the site to increase such that the criteria for use of the proposed alternative seismic approach are challenged. In such a situation, the categorization process may be impacted from a seismic risk perspective either solely due to the seismic risk or by the integrated importance measure determination.

In Section 3.2.3 of the enclosure to its letter dated May 20, 2021 (Reference [1]), the licensee stated that "U.S. nuclear power plants that utilize the 10 CFR 50.69 Seismic Alternative (EPRI Report 3002017583) will continue to compare GMRS to SSE." Since the alternative seismic approach explicitly cites and is based on EPRI Report 3002017583, the continued comparison of GMRS to SSE applies to the Ginna site. The licensee also stated that the seismic hazard at the plant is subject to periodic reconsideration as new information became available through industry evaluations.

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

Monitoring Inputs to and Outcome of Proposed Alternative Seismic Approach

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

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

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

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

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

3.3.1.3 Method for Assessing Other Non-Seismic External Hazards

This hazard category includes all non-seismic external hazards such as high winds, external floods, transportation, nearby facility accidents, and other hazards. The licensee discussed its consideration of other external hazards in Section 3.2.4 of the LAR. The licensee stated, in part, that all external hazards, except for seismic, were screened from applicability to Ginna per a plant-specific evaluation in accordance with Generic Letter 88-20 and the criteria in ASME/ANS RA-Sa-2009 PRA Standard (Reference [15]).

In Attachment 4 of the LAR (Reference [1]), the licensee provided the 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.

Except for the external flooding hazard addressed in the Attachment 4 table of the LAR, the guidance in NEI 00-04, Figure 5-6 regarding SSCs that play a role in screening a hazard is not discussed in the LAR. In APLC Question 03.a in the NRC's audit plan (Reference [3]), the NRC staff stated that it appears that SSC categorization may not be evaluated using the guidance in

NEI 00-04, Figure 5-6 to confirm that the SSC is not credited in screening an external hazard (except the external flooding hazard) because that evaluation has already been made. The NRC staff noted that plant changes, operational experience, and identified errors or limitations in the PRA models could potentially impact the conclusion that an SSC is not needed to screen an external hazard. In response to audit question APLC Question 03.a (Reference [2]), the licensee clarified that, during categorization of SSCs consistent with the NEI 00-04, guidance in Figure 5-6 of NEI 00-04 will be followed. Based on this response, the NRC staff finds that the licensee's SSC categorization process will evaluate the safety significance of SSCs for all other external hazards consistent with the guidance provided in NEI 00-04, as endorsed by the NRC, and is therefore acceptable.

In APLC Question 03.c presented by NRC in the audit plan (Reference [3]), the NRC staff stated, with regard to the extreme wind or tornado hazard, the LAR appears to indicate that screening of this hazard was determined on the success of tornado missile barriers "after upgrades to several of the barriers are made." Attachment 7 of the LAR identifies the upgrades and modifications needed to protect against 3-inch pipe missiles generated by tornadoes as commitments but does not propose a mechanism to ensure they are completed prior to implementation of the 10 CR 50.69 program. In response to audit question APLC Question 03.c (Reference [2]), the licensee explained that the tornado missile plant upgrades and modifications discussed in Attachment 7 of the LAR to protect against 3-inch pipe missiles generated by tornadoes are needed to support the screening of the extreme wind and tornado hazards. In Attachment 3 of the LAR Supplement letter dated October 14, 2021 (Reference [2]), the licensee proposes a condition to the operating license stating that it will complete the implementation items listed in Attachment 7 of the LAR prior to implementation of the 10 CFR 50.69 program. Additionally, Attachment 3 of the LAR Supplement states that a focused-scope peer review will be performed on any corresponding PRA upgrades, but the licensee does not commit to closing F&Os from the focused-scope peer review prior to implementation of the 10 CFR 50.69 program. The NRC staff reviewed the list of committed plant modifications and finds that upgrades to IEPRA and FPRA models will not be needed to reflect the as-built plant because the plant changes consist of re-enforcing and replacing barriers for exhaust pipes, air intakes, and air vents. The proposed approach to upgrade these barriers should provide protection against wind hazards and maintain the proposed safety categorization.

In audit question APLC Question 03.e (Reference [3]), the NRC staff cited the Attachment 4 table of the LAR that indicates the ice cover hazard (i.e., the accumulation of frozen water on bodies of water such as lakes, rivers or on SSCs) is screened based on the criteria defined in Attachment 5 table of the LAR criteria as "C1" (Event damage potential is less than events for which the plant was designed) and "C4" (Event is included in the definition of another event). The NRC staff also cited Section 5.0 of the SE for the Ginna IPEEE (Reference [11]) that states: "The licensee reported that in an earlier plant modification, the power for the heaters on the cooling water intake screens on Lake Ontario had been increased to protect against ice formation (slush)." The NRC staff stated based on these observations that it appears there is a potential for ice to form on the cooling water intake screens in the winter that could potentially fail the cooling water supply for such systems as the ultimate heat sink, particularly if the heaters (or power to the heaters) are unavailable. In response to audit question APLC Question 03.e (Reference [2]), the licensee stated that there are no SSCs credited in screening the ice cover hazard. The licensee explained that although heater racks were originally installed in the screenhouse to mitigate the effects of frazil ice on the cooling water intake screens, these racks were shown to actually restrict water flow. Removal of the heater resulted in water flow velocity that ensured floating ice could not plug the submerged water intake screens. Per the

Ginna UFSAR, all heater racks were removed. Therefore, there is no impact from the ice cover hazard to the SSC being categorized.

In summary, the use of the Ginna IPEEE results described by the licensee in the LAR, and the licensee's assessment of the other external hazards (i.e., high winds, tornadoes, and external flood) is consistent with Section 5 of NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1 (References [9] and [4], respectively). The NRC staff concludes that the licensee's treatment of other external hazards is acceptable and meets 10 CFR 50.69(c)(1)(ii).

3.3.1.4 Component Safety Significance Assessment for Passive Components

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

In Section 3.1.2 of the LAR (Reference [1]), the licensee proposed using a categorization method for passive components that is not cited in NEI 00-04, Revision 0, or RG 1.201, Revision 1 (References [9] and [4], respectively). The licensee proposed to categorize passive components and the passive function of active components using the Arkansas Nuclear One (ANO) Risk-Informed Repair/Replacement Activities (RI-RRA) methodology that was approved by the NRC for Arkansas Nuclear One, Unit 2 (ANO-2) (Reference [12]). 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 [14]). The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measures the safety significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment. Therefore, the NRC staff finds that the use of the repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs. The ANO-2 methodology includes specific treatment requirements for LSS SSCs.

In Section 3.1.2 of the LAR, the licensee stated, “[t]he passive categorization process is intended to apply the same risk-informed process accepted in the ANO2-R&R-004 for the passive categorization of Class 2, 3, and non-class components.” The licensee stated that “[a]ll ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned high safety-significance, HSS, for passive categorization which will result in HSS for its risk-informed safety classification and cannot be changed by the [Integrated Decision-Making Panel].” The NRC staff finds the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process.

3.3.1.5 Key Principle 4: Conclusions

Based on the above, the NRC staff review for IEPRA (includes internal floods) and FPRA acceptability and evaluation of the use of non-PRA methods, concludes that the proposed change satisfies the fourth key principle for risk-informed decision making 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 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, includes discussion on periodic review, and program documentation and change control. Maintaining change control and periodic review will also maintain confidence that all aspects of the 10 CFR 50.69 program and risk categorization for SSCs continually reflect the Ginna as-built, as-operated plant.

The NRC staff finds the risk management process described by the licensee in Section 3.5 of the LAR 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 decision-making prescribed in RG 1.174, Revision 3.

4.0 PROPOSED REVISION TO THE OPERATING LICENSE

The licensee proposed the following amendment to the RFOL for Ginna. The proposed license condition would state:

> Constellation Energy Generation, LLC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in Exelon's submittal letter dated May 20, 2021 and all its subsequent associated supplements as specified in License Amendment No. 151 dated June 22, 2022.
>
> Constellation Energy Generation, LLC will complete the implementation items listed in Attachment 7 of Exelon Letter to the NRC dated May 20, 2021, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa -

2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to the implementation of 10 CFR 50.69.

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 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 Section 3.3 above finds the non-PRA methods for assessing risk for seismic, and passive components, which are deviations from NEI 00-04, to be acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the staff notified the New York State official on April 11, 2022, of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 or change inspections or surveillance requirements. The NRC staff has determined that the amendments involve 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, which was published in the *Federal Register* on August 10, 2021 (86 FR 43691), that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. The supplemental letters provided additional information that clarified the application and the license condition, did not expand the scope of the application as originally noticed, and did not change Accordingly, the amendments meet 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 amendments.

7.0 CONCLUSION

Based on the aforementioned considerations, the NRC staff has concluded 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: June 22, 2022

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