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

May 19, 2021

Mr. David P. Rhoades
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 237 REGARDING ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS OF NUCLEAR POWER REACTORS" (EPID L-2019-LLA-0098)

Dear Mr. Rhoades:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 237 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. The amendment is in response to your application dated April 30, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20121A241) as supplemented by letter dated November 24, 2020 (ADAMS Accession No. ML20329A433).

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

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

Sincerely,

/RA/

Joel S. Wiebe, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosures:
1. Amendment No. 237 to NPF-62
2. Safety Evaluation

cc: Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 237
License No. NPF-62

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

    A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated April 30, 2020, as supplemented by letter dated November 24, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, by Amendment No. 237, Facility Operating License No. NPF-62 is hereby amended to add license condition 2.C(26) to specify the conditions for use of a risk-informed process for the categorization and treatment of structures, systems, and components as set forth in the licensee's application dated April 30, 2020, as supplemented by letter dated November 24, 2020, and evaluated in the NRC staff's safety evaluation enclosed with this amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2021.05.19 09:00:03 -04'00'

Nancy L. Salgado, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License

Date of Issuance: May 19, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 237

FACILITY OPERATING LICENSE NO. NPF-62

CLINTON POWER STATION, UNIT NO. 1

DOCKET NO. 50-461

Replace the following pages of the Facility Operating License No. NPF-62, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Page 7	Page 7
Page 8	Page 8 (repagination)
	Page 9 (repagination)

(25) Irradiated GE14i fuel bundles shall be stored at least four feet from the wall of the Spent Fuel Pool.

(26) Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

Exelon Generation Company, LLC (EGC) 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 EGC's submittal letter dated April 30, 2020, and all its subsequent associated supplements as specified in License Amendment No. 237 dated May 19, 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. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include: (a) an exemption from the requirements of 10 CFR 70.24 for the criticality alarm monitors around the fuel storage area; (b) an exemption from the requirement of 10 CFR Part 50, Appendix J – Option B, paragraph III.B, exempting the measured leakage rates from the main steam isolation valves from inclusion in the combined leak rate for local leak rate tests (Section 6.2.6 of SSER 6); and (c) an exemption from the requirements of paragraph III.B of Option B of 10 CFR Part 50, Appendix J, exempting leakage from the valve packing and the body-to-bonnet seal of valve 1E51-F374 associated with containment penetration 1MC-44 from inclusion in the combined leakage rate for penetrations and valves subject to Type B and C tests (SER supporting Amendment 62 to Facility Operating License No. NPF-62). The special circumstances regarding each exemption, except for item (a) above, are identified in the referenced section of the safety evaluation report and the supplements thereto.

An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC Material License No. SNM-1886, issued November 27, 1985, and relieved the licensee from the requirement of having a criticality alarm system. Exelon Generation Company is hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The exemptions in items (b) and (c) above are granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: “Clinton Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 2,” submitted by letter dated May 17, 2006.

   Exelon Generation Company 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 Exelon Generation Company CSP was approved by License Amendment No. 194 and modified by License Amendment No. 206.

F. Exelon Generation Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report as amended, for the Clinton Power Station, Unit No. 1, and as approved in the Safety Evaluation Report (NUREG-0853) dated February 1982 and Supplement Nos. 1 thru 8 thereto subject to the following provision:

   > Exelon Generation Company may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

G. Deleted.

H. Exelon Generation Company 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.

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

I. This license is effective as of the date of issuance and shall expire at midnight on April 17, 2027.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Attachments 1 (Deleted) and 2
2. Appendix A - Technical Specifications (NUREG-1235)
3. Appendix B - Environmental Protection Plan
4. Appendix C - Deleted

Date of Issuance: April 17, 1987

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 237 TO

FACILITY OPERATING LICENSE NO. NPF-62

EXELON GENERATION COMPANY, LLC

CLINTON POWER STATION, UNIT NO. 1

DOCKET NO. 50-461

1.0 INTRODUCTION

By application dated April 30, 2020 (Reference 1), as supplemented by letter dated November 24, 2020 (Reference 2), Exelon Generation Company, LLC (Exelon or licensee) submitted a license amendment request (LAR) for Clinton Power Station, Unit 1 (CPS).

The licensee proposed to add a new license condition to Facility Operating License No. NPF-62 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 a method of categorizing SSCs according to their safety significance.

To support its review, the U.S. Nuclear Regulatory Commission (NRC or Commission) staff conducted an audit as described in the audit plan dated August 27, 2020 (Reference 3). Based on its review of the LAR and information reviewed during the audit, the NRC staff transmitted requests for additional information (RAIs) to the licensee dated October 27, 2020 (Reference 4). Since the October 27, 2020, letter contained the RAIs resulting from the audit, no separate audit summary was issued. By letter dated November 24, 2020, the licensee responded to the RAIs.

The supplemental letter dated November 24, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 16, 2020 (85 FR 36435).

2.0 REGULATORY EVALUATION

2.1 Risk-Informed Categorization and Treatment of SSCs

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 Risk-Informed Safety Class (RISC) categories.

The SSC categorization requirements do 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 Licensee's Proposed Changes

The licensee proposed the addition of the following conditions to the facility operating license for CPS to allow the implementation of 10 CFR 50.69.

> Exelon Generation Company, LLC (EGC) is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE [individual plant examination of external events] Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS [American Society of Mechanical Engineers/American Nuclear Society] PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in EGC's submittal letter dated April 30, 2020, and all its subsequent associated supplements as specified in License Amendment No. 237 dated May 19, 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).

2.3 Regulatory Guides and NRC Staff Review Plans

As discussed above, the provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on an integrated and systematic risk-informed process that includes several approaches and methods for categorizing SSCs according to their safety significance. The NRC staff considered the following regulatory guidance during its review of the proposed changes:

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

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

- RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," January 2018 (Reference 7)

- NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision-making," March 2017 (Reference 8)

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition" (SRP), Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," September 2012 (Reference 9)

NRC-Endorsed Guidance

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (Reference 10), as endorsed by RG 1.201, Revision 1, with clarifications, limitations, and conditions, which describes a process acceptable to the NRC for determining the safety significance of SSCs and categorizing them into the 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 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).

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

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

These key principles are:

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

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

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

- Principle 4: When the proposed licensing basis 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 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 is 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 Basis Change Meets the Current Regulations

Paragraph 50.69(c) of 10 CFR requires licensees to use an integrated decision-making process to categorize safety-related and nonsafety-related SSCs according to the safety significance of the functions they perform 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, 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, 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 Decision-making 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 of the LAR, the licensee stated that it will implement the risk-informed categorization process in accordance with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. In Sections 2.3 and 3.1.1 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 1.

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, ensures 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 NRC staff reviewed the licensee's SSC categorization program and finds that it includes the appropriate steps/elements prescribed in NEI 00-04, Revision 0, to assure that the SSCs specified are appropriately categorized consistent with 10 CFR 50.69. 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.

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 proposed LB 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 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, the licensee clarified that it will 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 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, 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 meets 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) requires 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. 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 LB (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 SSCs design basis function as described in the plants' LB, including the updated UFSAR and technical specifications (TS) bases do not change and continue to be met. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant LB. 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 and, therefore, meets 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 decision-making, pertaining to the assessment for change in risk and monitoring the impact of the LB 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, the licensee described that the CPS 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) Topical Report (TR) 3002017583 dated February 29, 2020 (Reference 11), and qualitative insights about seismic risk at CPS
- External Hazards: Screening analysis performed for IPEEE in Generic Letter 88-20, Supplement 4, dated June 28, 1991 (Reference 12), updated using criteria from Part 6 of the ASME/ANS RA-Sa-2009, "Addendum A to RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," (Reference 13) (the PRA Standard), as endorsed by the NRC
- Other Hazards: Screening analysis performed for the IPEEE updated using criteria from Part 6 of the PRA Standard, 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," December 1991 (Reference 14)
- Passive Components: ANO-2 passive categorization methodology dated April 22, 2009 (Reference 15)

The approaches and methods proposed by the licensee to address internal events, 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. The non-PRA method for the categorization for passive components is consistent with the ANO-2 methodology for passive components approved for risk-informed safety classification and treatment for repair/replacement activities in Class 2 and 3 moderate- and high energy systems. 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 CPS PRA is comprised of a full-power, Level 1, internal events (IEPRA) and fire PRA (FPRA) which evaluate the CDF and LERF risk metrics. The licensee discussed in Section 3.3 of the LAR, that the IEPRA (includes internal floods) model has been assessed against RG 1.200, Revision 2. Furthermore, LAR Section 3.3, states that a finding closure review was conducted on the identified PRA model in December 2018 and November 2019 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 16).

The NRC staff finds that the LAR provides sufficient information to support the staff review of the IEPRA (includes internal flooding) and FPRA for technical acceptability, and therefore, meets the requirements set forth in paragraph 50.69(b)(2)(iii) of 10 CFR.

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 FLEX in the PRA, and (4) assessment of assumptions and approximations. In e-mail correspondence to the licensee on October 27, 2020, the NRC staff issued RAIs to further assess the acceptability of CPS IEPRA (includes internal floods) and FPRA for consistency with RG 1.200, Revision 2, and NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. The 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, the licensee stated that the IEPRA (includes internal floods) model was subjected to a full-scope peer review in October 2009 against ASME/ANS 2009 Standard supporting requirements (SRs) at capability category (CC II). Subsequently, in December 2018 and November 2018, Exelon conducted Independent Assessments for closure of the finding-level Facts and Observations (F&Os) and closed all but two F&Os. An NRC staff review of this Independent Assessment is included below in this section of the SE.

In the LAR Attachment 3, the licensee submitted the remaining two open F&Os from the IEPRA (includes internal floods) peer reviews. For each F&O, the licensee provided a disposition for this application. In its e-mail dated October 27, 2020, the NRC staff requested the licensee provide additional information to further assess the dispositions for some of the findings as described in the following paragraphs.

The LAR Attachment 3, presented the dispositions for two F&Os (i.e., F&Os 1-32 and 1-34) that remained open after the internal events Independent Assessment for closure of F&Os was performed in November 2019. These F&Os both addressed a similar concern, stating for F&O 1-32, in the CPS Technical Specification Task Force (TSTF)-505 application dated April 30, 2020 (Reference 17), that "potentially risk significant combinations of HFEs [Human Factor Events] are not captured throughout the current approach, due to the chosen truncation level for the dependency identification (5E-9 and 5E-10 for CDF and LERF, respectively) in conjunction with the elevated human error probability (HEP) level chosen (0.1)." Accordingly, in APLA/DRA/ RAI 01.a, the NRC staff requested justification that an adjustment of additional HEP combinations revealed using a lower quantification truncation level would not adversely impact the SSC categorization. In its letter dated November 24, 2020, the licensee explained that its HEP dependency analysis approach involved initially setting all HEPs to 0.1, to ensure that HEP combinations that could fall below the truncation level using their nominal values would be captured. The licensee also performed a sensitivity study in which unadjusted HEP combinations (because of the significant time between actions) were set at High Dependence (HD) (i.e., complete dependence) in the sensitivity case using the baseline case minimum joint HEP values (see discussion below for DRA/APLA RAI 01.b on application of minimum joint HEP values). In further response to the RAI, the licensee provided the results of the sensitivity study that demonstrated SSC categorization does not change for the set of impacted SSCs, because these SSCs were already identified as risk significant based on their support of risk significant functions. Therefore, the sensitivity study reasonably concludes that the truncation level used for identification of dependencies has no impact on SSC categorization. Based on the information provided above, the NRC staff finds the resolution to F&Os 1-32 and 1-34 acceptable for use in the 10 CFR 50.69 program.

In APLADRA/RAI 01b, the NRC staff requested the licensee provide information involving the treatment of minimum joint HEP values used in the IEPRA, and justification for those instances where values less than 1E-06 were used. In its letter dated November 24, 2020, the licensee explained that for the IEPRA dependency analysis a minimum joint HEP of 1E-06 was used unless the timeframe for completing one or more actions in the combination was longer than 15 hours. In these cases, a lower minimum joint HEP of 5E-07 was used. The NRC notes, per the guidance in Figure 6-1 of NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis

Guidelines Final Report," July 2012 (Reference 18), which is applicable to IEPRA and FPRA, that when an operator action is performed by a different crew this leads to Low Dependency for even high stress scenarios. Therefore, according to Table 6-1 of NUREG-1921, the credit taken by the licensee for applying a lower joint HEP for combinations that include a long- term action is consistent with the guidance.

The licensee also explained that two sensitivity studies were performed in which a minimum joint HEP of 1E-05 was applied. In one sensitivity case (refer to (Reference 2), Table APLA-01-B.2), three basic events, that previously had low importance in the baseline case, were identified as important in the sensitivity case and their importance values were compared. The results of this sensitivity study demonstrated that SSC categorization does not change for the set of impacted SSCs because those SSCs are already risk significant based on their support of risk significant functions. In the other sensitivity case related to the licensee's response to DRA/APLA RAI 01a, the unadjusted HEP combinations (because of the significant time between actions) were set at HD (i.e., complete dependence) in the sensitivity case using a minimum joint HEP value of 1E-05. Fifteen basic events, that had low importance in the baseline case, were identified as important in the sensitivity case and their importance values were compared. The results of this sensitivity study also demonstrated that SSC categorization does not change for the set of impacted SSCs, because those SSCs are already risk significant based on their support of risk significant functions.

Based on the above, the NRC staff finds the licensee's application of minimum joint HEP values is consistent with NUREG-1921 because the licensee established an appropriate minimum joint HEP value consistent with the level of dependency of HEPs in the combination and provided results of the sensitivity studies performed to demonstrate that the minimum joint HEP values used has no impact on risk categorization.

In Section 3.2 of the LAR for the PRAs, Exelon states, in part, "there are no PRA upgrades that have not been peer reviewed." In DRA/APLA RAI 02, the NRC staff noted the length of time between the last full-scope peer review in October 2009 and the F&O closure review performed in November 2019. Accordingly, the NRC staff requested a summary of significant IEPRA (includes internal floods) model changes that have been made since October 2009 and justification for whether the changes meet the definition of a PRA upgrade as defined in the ASME/ANS R-Sa-2009 PRA Standard. In its letter dated November 24, 2020, the licensee explained that there have been four IEPRA model updates since the October 2009 peer review of its 2006 PRA model (i.e., version CL06C0). The licensee provided a listing of the important model changes made since the 2009 peer review, dispositions of each change as either a "maintenance" update or a PRA "upgrade," and a basis for each disposition. In all cases, the licensee identified the change as a "maintenance" update and furnished example(s) for each model change. The NRC staff reviewed the identified changes, along with the licensee's dispositions and concludes that the changes to the PRA model were consistent with the criteria for PRA maintenance as described in the ASME/ANS RA-Sa 2009 PRA standard; therefore, no additional peer reviews were required.

Based on the above, the NRC staff concludes that the CPS IEPRA (including internal floods) was appropriately peer reviewed, consistent with RG 1.200, Revision 2, and the F&O's have been adequately dispositioned to assess the impact on the risk-informed application.

Internal Fire PRA Peer Review History

In Section 3.2.2 of the LAR the licensee stated that the CPS categorization process for fire hazards will use a peer reviewed plant-specific FPRA model. Furthermore, the licensee confirmed that the CPS FPRA was subject to a full-scope industry peer review in April 2018 consistent with RG 1.200, Revision 2 and ASME/ANS RA-Sa-2009 PRA Standard SRs at CC II.

The finding-level F&Os from the April 2018 full-scope peer review were considered fully resolved by Independent Assessment review teams in December 2018 and November 2019. Therefore, in accordance with RG 1.200, Revision 2, no F&Os associated with the FPRA were provided in the LAR. Based on NRC staff review as discussed above for the IEPRA in this SE, the NRC concluded that no new methods or upgrades were inadvertently incorporated into the FPRA without a peer review in accordance with the ASME/ANS RA-Sa-2009 PRA Standard as endorsed by the NRC.

The NRC staff has reviewed the FPRA peer review results and the licensee's resolution of the results provided in Attachment 3 of the LAR and concludes that the CPS FPRA was appropriately peer reviewed, consistent with RG 1.200, Revision 2, and the F&O's have been addressed for closure using Appendix X to NEI 05-04, 07-12 and 12-13, as accepted, with conditions by the NRC staff (Reference 19).

Appendix X, Independent Assessment Process for F&O Closure

Section X.1.3 of Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, provides guidance to perform an Independent Assessment for the closure of F&O identified from a full-scope or focused-scope peer review.

Based on the NRC staff review of the LAR, which included the peer review history and the response to DRA/APLA RAI 01 and DRA/APLA RAI 02 above, the NRC staff concludes that all F&Os were appropriately assessed by the Independent Assessment team to assure that no new methods or upgrades were inadvertently incorporated into the IEPRA without a peer review in accordance with the ASME/ANS RA-Sa-2009 PRA Standard as endorsed by the NRC.

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 20), provides the NRC staff’s assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision-making in accordance with the guidance of RG 1.200, Revision 2.

In the LAR Attachment 6, concerning the IEPRA modeling uncertainty, the licensee indicated that FLEX equipment and actions have been credited in the PRA models for Station Blackout (SBO) and extended loss of alternating current (AC) power. However, the LAR did not identify the specific FLEX equipment and actions that were credited in the IEPRA and FPRA models and did not describe how those systems and actions were modeled. In DRA/APLA RAI 05, the NRC staff requested: (1) clarification of which PRA models credit FLEX strategies, (2) discussion of the extent to which equipment and actions were credited, (3) discussion of the data and failure probabilities used to support the FLEX modeling, (4) discussion of the methodology used to assess FLEX operator actions, and (5) justification that incorporation of FLEX into the PRA models does not meet the definition of a PRA upgrade.

In its letter dated November 24, 2020, the licensee explained that the deployment and alignment of two portable FLEX 480 volt (V) AC diesel generators and two portable diesel-driven low-pressure high-capacity self-prime pumps are credited in the IEPRA (including internal flooding) and FPRA models. The two diesel-driven pumps have the capacity to provide 2000 gallons per minute for residual heat removal and 1000 gallons per minute to the reactor pressure vessel. The licensee also explained that the credited FLEX strategies for CPS are inclusive of reactor core injection cooling and suppression pool cooling. The licensee applied a factor of two to the FLEX equipment "failure probabilities" (i.e., failure rates) to the CPS-specific equipment failure rates for similar equipment to account for data uncertainty. The licensee also confirmed that the same methodology used to assess FLEX actions was also used to assess non-FLEX actions previously incorporated in the PRA models. The licensee further discussed that for actions described in Sections 7.5.4 and 7.5.5 of NEI 16-06 that engineering judgment and conservative assumptions were used (e.g., transportation of portable equipment, installation of equipment at staging location, routing of cables and hoses and complex actions). The licensee also explained that the models include an event in which "FLEX fails after alignment" that is assigned a "bounding" failure probability of 0.1 to account for the uncertainty associated with implementing FLEX strategies. The NRC staff finds the licensee's modeling of FLEX in the PRAs for this application acceptable because the licensee used conservative decisions and limited the FLEX credit including a failure event that is assigned a failure probability of 0.1 to account for the uncertainty associated with implementing FLEX strategies.

Identification of Key Assumptions and Sources of Uncertainty

In Section 3.2.7 of the LAR, the licensee stated that NUREG-1855, Revision 1, 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 the guidance 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. The licensee confirmed that for the CPS uncertainty analysis, some uncertainties and assumptions were screened based on the use of a consensus method. The licensee presented identified key assumption and sources of uncertainty in the LAR Attachment 6. Based on the above, the NRC staff finds that the assessment performed to identify the key assumptions/sources of uncertainty is consistent with the guidance provided in NUREG-1855, Revision 1.

Section 6.4 of NUREG-1855, Revision 1, states, "[f]or a CC II risk evaluation, the standard requires providing the mean values of the risk metrics and the guidance further discusses the need of these risk metrics to be compared against the risk acceptance guidelines. The risk values provided in Attachment 2 of the LAR are point values, not mean values. The total CDF of 8.8E05 per year provided in the LAR is a relatively small margin with respect to the RG 1.174 threshold for total CDF (i.e., 1.0E-06) and did not consider the risk increase due to state-of-knowledge correlation (SOKC). Accordingly, in DRA/APLA RAI 04,e NRC staff requested a summary of how the licensee considered SOKC in the propagation of parametric uncertainty in the PRA models that support the 10 CFR 50.69 application consistent with guidance in NUREG-1855, Revision 1.

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

In its letter dated November 24, 2020, the licensee described its refined propagation of parametric uncertainty and consideration of the SOKC, and provided the results of the analysis to support the IEPRA and FPRA models. The results of the analysis for the IEPRA and FPRA demonstrated that the mean CDF and mean LERF values were slightly greater than the point-estimate values. The licensee further confirmed that for the combined impact on the IEPRA and FPRA models, adequate margin of 1.5 percent and 0.6 percent for CDF and LERF, respectively, exists between the total mean CDF and LERF values (i.e., 8.86E-05 and 7.11E-06, respectively) and the RG 1.174 thresholds. The NRC staff notes that the impact of the SOKC is spread over a large number of parametric contributors and is unlikely to have more than a minimal impact on the importance values for specific SSCs. Therefore, the NRC staff concludes that consideration of the SOKC using point-estimate values is consistent with NUREG-1855, Revision 1, and has a minimal impact on the 10 CFR 50.69 application, and is, therefore, acceptable.

In the LAR Section 3.2.7, the licensee confirmed that sensitivity studies will be performed consistent with Section 5 of the NEI 00-04 guidance. In accordance with Section 9 of NEI 00-04, as endorsed by RG 1.201, Revision 1, the licensee’s integrated decision-making panel (IDP) will use information and risk insights compiled in the initial categorization process, including awareness of the limitations and assumptions of the PRA, and combines 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 sensitivity studies consistent with Section 5 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 the context of the decision-making under consideration for the categorization of the SSC at the time of the risk analysis being performed.

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) stipulates the process for feedback and adjustment to assure configuration control is maintained for these routine changes and updates to the IEPRA and FPRA.

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 and FPRA 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 provides guidance for determining the acceptability of the PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer review process.

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

Based on the above, the NRC staff finds the licensee provided the required information, and the IEPRA (includes internal floods) and FPRA are acceptable and, therefore, meet the requirements set forth in paragraphs (c)(1)(i) and (ii) of 10 CFR 50.69.

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

Alternate 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 seismic approach for considering seismic risk in the categorization process and described how the proposed alternative seismic approach would be used in the categorization process in Section 3.2.3 of the Enclosure to its letter dated April 30, 2020, and its supplement dated November 24, 2020. In part, the licensee based its plant-specific evaluation on the case studies performed in EPRI TR 3002017583, and stated that the case studies are applicable to CPS and are used in the alternative seismic approach; how the licensee's proposed alternative seismic approach would be used in the categorization process; and 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 TR 3002017583 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from seismic PRAs (SPRAs). In its supplement, the licensee indicated that aside from updates included in an RAI submittal for the Calvert Cliffs 50.69 LAR into the previous version of this report, EPRI 3002012988, the technical criteria in EPRI Report 3002017583 is unchanged from its predecessor report EPRI TR 3002012988. The NRC staff's review confirmed that the case studies in EPRI TR 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 CPS as compared to the amendment approved by the NRC for Calvert Cliffs on February 28, 2020 (Reference 21). The information presented in the LAR 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 the EPRI TR 3002017583 Case Studies

In its supplement dated November 24, 2020, the licensee responded to the NRC staff's RAIs concerning the approach used in the Calvert Cliff's amendment including the case studies, mapping approach, and conclusions on the determination of unique HSS SSCs from the case studies which were used by the licensee 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 CPS. The NRC staff reviewed and evaluated the technical acceptability of the PRAs used in the case studies for Plants A, C, and D, in EPRI

TR 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.

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 TR 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 TR 3002017583; and applicable to CPS 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 CPS is at or below approximately 0.2g, or where the GMRS is below or approximately equal to the safe shutdown earthquake (SSE) between 1.0 Hz and 10 Hz.

The licensee further states that the GMRS to SSE comparison demonstrates that CPS qualifies as a Tier 1 plant under the criteria in EPRI TR 3002017583 and that this comparison confirms the expected seismic risk at CPS 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 22). The NRC staff reviewed the licensee's submittal and supplements and plant-specific evaluation and concludes that the proposed criteria in EPRI TR 3002017583 to determine the applicability and use of the proposed seismic Tier 1 approach is acceptable.

Evaluation of Applicability of Criteria for 10 CFR 50.69

In Section 3.2.3 of the Enclosure to its April 30, 2020, letter, the licensee compared the CPS 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 Figure A4-1 of Attachment 4, of the Enclosure to the April 30, 2020, letter, to demonstrate that the site meets the criteria for application of the proposed alternative seismic approach. In Section 3.2.3 of the Enclosure to its April 30, 2020, letter, 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 CPS 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 Enclosure to its letter dated April 30, 2020, the licensee stated that the small percentage contribution of seismic 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. The NRC staff's evaluation of seismic risk to total risk was based on information in the CPS TSTF-505 submittal (Section 3 of Enclosure 4 to the letter dated April 30, 2020), and the supplement dated November 24, 2020.

The NRC staff verified the licensee's estimate of seismic CDF by mathematically convolving the median seismic capacity with composite uncertainty provided in the CPS TSTF-505 submittal and the site-specific reevaluated seismic hazard for the mean peak ground acceleration. The staff reviewed the licensee's description of its seismic LERF (SLERF) estimate in the CPS TSTF-505 submittal, which indicated a large contribution from seismic-induced failures that led directly to core damage and large early release (e.g., containment structure failure and reactor pressure vessel supports). The purpose of the SLERF determination in the CPS TSTF-505 submittal is to provide a conservative estimate for use in calculating risk-informed completion times for technical specifications. The NRC staff determined that the licensee used conservatively biased median fragility values in calculating the SLERF estimate. This includes the values used for seismic-induced failures that led directly to core damage and large early release. Further, as noted in Section 3.6.5 of EPRI TR 3002017583, containment defense-in-depth assessment addresses containment failures and containment bypass situations. Section 3.6.6 of EPRI TR 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 for the licensee 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 TR 3002017583 case studies performed for GMRS to SSE ratios significantly higher than CPS, 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 TR 3002017583 were appropriately included and implemented.

In Section 3.2.3 of the Enclosure to its letter dated April 30, 2020, 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 CPS.

The licensee explained that the qualitative characterization of seismic risk performed for the independent 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 Figure 3-1 in the LAR, is to identify plant-specific seismic insights derived from the components in the system being categorized.

In its supplement dated November 24, 2020, the licensee stated that its plant-specific evaluation considered differences in the proposed alternative seismic approach between CPS and the approach in the Calvert Cliffs amendment previously reviewed and approved by the NRC staff on February 28, 2020. The NRC staff's review of the licensee's proposed alternative seismic approach determined that the approach used in the Calvert Cliff's amendment is applicable to this licensee's proposed alternative seismic approach and that the plant-specific evaluation on the 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 IDP as part of the integrated, systematic process for categorization.

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

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

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

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

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

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 April 30, 2020, the licensee stated that "U.S. nuclear power plants that utilize the 10 CFR 50.69 Seismic Alternative ([TR] 3002017583) will continue to compare GMRS to SSE." Since the alternative seismic approach explicitly cites and is based on EPRI TR 3002017583, the continued comparison of GMRS to SSE applies to the CPS. 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 finds that the consideration of changes to the seismic hazard in the licensee's plant-specific proposed alternative seismic approach is the same as that approved in the Calvert Cliffs amendment. Consequently, the NRC staff finds that the consideration of changes to the seismic hazard at CPS that exceed the criteria for use of the proposed alternative seismic approach is acceptable for the proposed approach because: (1) the criteria for use of the proposed alternative seismic approach is clear and traceable, (2) the proposed alternative seismic approach includes periodic reconsideration of the seismic hazard as new information becomes available, (3) the proposed alternative seismic approach satisfies the requirements in 10 CFR 50.69 discussed above, and (4) the licensee has included a proposed license condition in the LAR to require NRC approval for a change to the specified seismic categorization approach.

Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach

In Section 3.5 of the Enclosure to its letter dated April 30, 2020, the licensee stated that its configuration control process ensures that changes to the plant, including a physical change and changes to documents, are evaluated.

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, and nearby facility accidents, and other hazards. The licensee discussed its consideration of other external hazards in Section 3.2.4 of the Enclosure to its letter dated April 30, 2020. The licensee stated that as part of the categorization assessment of other external hazard risk, all other external hazards were screened for applicability to CPS per a plant-specific evaluation in accordance with Generic Letter 88-20 and updated to use the criteria in ASME/ANS PRA Standard RA-Sa-2009.

In Attachment 4 of its letter dated April 30, 2020, the licensee provided the results of the plant-specific evaluation that assessed the IPEEE results using endorsed criteria in the ASME/ANS RA-Sa-2009 PRA Standard and current plant hazard information. The NRC notes, this plant-specific evaluation or its results were not peer reviewed against Part 6 of the ASME/ANS Ra-SA-2009 PRA Standard as endorsed in Regulatory Guide (RG) 1.200, Revision 2.

In Attachment 4 of the LAR, the licensee addressed the external flooding hazard and determined that criterion “C1,” defined in Attachment 5, as event damage potential is less than events for which the plant is designed, applies to this hazard. The licensee further noted that all external flooding mechanisms were screened and that there are no SSCs credited for screening this hazard. The licensee also determined that criterion “C1” applies to the extreme wind or tornado hazard, and noted that the hazard screening evaluation determined that a demonstrably conservative estimate of CDF is much less than 1E-6/yr. The licensee stated that the risk categorization process will be implemented in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201, with the exception of the evaluation of impact of the seismic hazard. Therefore, Section 5 of NEI 00-04 will be followed for the external flooding hazard and the extreme wind or tornado hazard as part of the categorization of SSCs. The NRC staff’s review finds that the licensee’s SSC categorization process will evaluate the safety significance of SSCs for other external hazards consistent with the guidance provided in NEI 00-04, as endorsed by the NRC, and is therefore acceptable.

In the supplement dated November 24, 2020, the licensee discussed that the screening of these hazards considers the as-built, as-operated, plant and will remain valid during implementation of 10 CFR 50.69. The licensee explained that for the snow hazard the CPS updated safety analysis report (USAR) addresses design snow and ice loads using “the latest hazard information” and that snow or ice loading is a slow developing event and criterion “C5” defined in Attachment 5 to the LAR, applies to this hazard. The licensee explained that for the sand or dust storm hazard CPS is not located in an area that is impacted by sand or dust storms and that criterion “C3” also applies to this hazard. The licensee explained for the ice cover hazard the CPS USAR stipulates that the submerged location of the ultimate heat sink pond suction prevents ice formation or ice jams from affecting the performance of the ultimate heat sink. The licensee also explained that as discussed in NEI 00-04 scheduled periodic reviews will be conducted to evaluate new insights from available information. If it is determined that there are changes that can impact categorization, then the risk information and the categorization process will be updated.

The NRC staff finds that the bases for screening the other external hazards are acceptable because they consider the as-built, as-operated, plant use current hazard information, and will be updated if changes that can impact categorization are identified from periodic reviews. In summary, the use of the CPS IPEEE results described by the licensee in its letter dated April 30, 2020, its supplement dated November 24, 2020, 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. Based on the above, 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

In Section 3.1.2 of the LAR, the licensee proposed using a categorization method for passive components not cited in NEI 00-04, Revision 0, or RG 1.201, Revision 1, for passive component categorization, but was approved by the NRC for ANO-2. The ANO-2 methodology is a RISC 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," July 2002 (Reference 23). 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.

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 AN0 2-R&R-004 for the passive categorization of Class 2, 3, and non-Class components." The licensee also stated that consistent with AN0-2-R&R-004 "[a]ll ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned HSS, for passive categorization which will result in HSS for its RISC and cannot be changed by the IDP." Based on its review, the NRC staff finds the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process for CPS.

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 the all aspects of the 10 CFR 50.69 program (i.e., includes traditional engineering analyses) and PRA models used to perform the risk assessment continue to reflect the as-built-as-operated plant and that plant modifications and updates to the PRA 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 CPS as-built, as-operated plant.

The NRC staff finds the risk management process described by the licensee in the LAR is consistent with Section 12 of NEI 00-04, Revision 0, 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 CHANGES TO THE OPERATING LICENSE

The licensee proposed the addition of a license condition to Facility Operating License No. NPF-62 for CPS to allow the implementation of 10 CFR 50.69 as is identified in Section 2.2 of this SE.

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 Section 3.3.1.2 of this SE, finds that the non-PRA methods for assessing risk for seismic, shutdown, and passive components which are deviations from NEI 00-04 are acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment on March 30, 2021. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility's components located within the restricted area as defined in 10 CFR Part 20. 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 that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (85 FR 36435, June 16, 2020). 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

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

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Date of Issuance: May 19, 2021

SUBJECT: CLINTON POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 237 REGARDING ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS OF NUCLEAR POWER REACTORS" (EPID L-2019-LLA-0098) DATED MAY 19, 2021

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