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

January 29, 2021

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

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT 2 - ISSUANCE OF AMENDMENT NO. 183 REGARDING RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS (EPID L-2019-LLA-0290)

Dear Mr. Rhoades:

The U.S. Nuclear Regulatory Commission (the Commission, NRC) has issued the enclosed Amendment No. 183 to Renewed Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit 2 (Nine Mile Point 2). The amendment consists of changes to the renewed facility operating license and licensing basis in response to your application dated December 26, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19360A145), as supplemented by letters dated April 30, 2020, and August 28, 2020 (ADAMS Accession Nos. ML20121A005 and ML20241A047, respectively).

The amendment allows the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, “Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors,” and adds a license condition proposed by Exelon Generation Company, LLC to the renewed facility operating license that identifies action items that need to be completed prior to implementing 10 CFR 50.69 at Nine Mile Point 2 and identifies possible changes to the categorization process that would require prior NRC approval.

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

Sincerely,

/RA/

Michael L. Marshall, Jr., Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures:
1. Amendment No. 183 to NPF-69
2. Safety Evaluation
3. Notice and Environmental Findings

cc: Listserv

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

NINE MILE POINT NUCLEAR STATION, LLC

LONG ISLAND LIGHTING COMPANY

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 183
Renewed License No. NPF-69

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

   A. The application for amendment by Exelon Generation Company, LLC (Exelon, the licensee) dated December 26, 2019, as supplemented by letters dated April 30, 2020, and August 28, 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. 183, Renewed Facility Operating License No. NPF-69 is hereby amended to authorize 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 December 26, 2019, as supplemented by letters dated April 30, 2020, and August 28, 2020, and evaluated in the NRC staff's safety evaluation enclosed with this amendment.

   In addition, the license is amended by changes as indicated in the attachment to this license amendment, and paragraph 2.C.(30) of Renewed Facility Operating License No. NPF-69 is hereby amended to read as follows:

   (30) Exelon 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 December 26, 2019, and all its subsequent associated supplements as specified in License Amendment No. 183 dated January 29, 2021.

   Exelon will complete the items listed in Attachment 7 of Exelon letter to NRC dated December 26, 2019, 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 implementation of the 10 CFR 50.69 categorization process.

   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 the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna
Digitally signed by James G. Danna
Date: 2021.01.29 11:03:27 -05'00'

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

Date of Issuance: January 29, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 183

NINE MILE POINT NUCLEAR STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Replace the following pages of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Page	Insert Page
15	15
16	16

(25) Within 14 days of the license transfers, Exelon Generation shall submit to the NRC the Nuclear Operating Services Agreement reflecting the terms set forth in the application dated August 6, 2013. Section 7.1 of the Nuclear Operating Services Agreement may not be modified in any material respect related to financial arrangements that would adversely impact the ability of the licensee to fund safety-related activities authorized by the license without the prior written consent of the Director of the Office of Nuclear Reactor Regulation.

(26) Within 10 days of the license transfers, Exelon Generation shall submit to the NRC the amended CENG Operating Agreement reflecting the terms set forth in the application dated August 6, 2013. The amended and restated Operating Agreement may not be modified in any material respect concerning decision making authority over safety, security and reliability without the prior written consent of the Director of the Office of Nuclear Reactor Regulation.

(27) At least half the members of the CENG Board of Directors must be U.S. citizens.

(28) The CENG Chief Executive Officer, Chief Nuclear Officer, and Chairman of the CENG Board of Directors must be U.S. citizens. These individuals shall have the responsibility and exclusive authority to ensure and shall ensure that the business and activities of CENG with respect to the facility's license are at all times conducted in a manner consistent with the public health and safety and common defense and security of the United States.

(29) Reserved.

(30) Exelon 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 December 26, 2019, and all its subsequent associated supplements as specified in License Amendment No. 183 dated January 29, 2021.

Exelon will complete the items listed in Attachment 7 of Exelon letter to NRC dated December 26, 2019, 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 implementation of the 10 CFR 50.69 categorization process.

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.

i) An exemption from the critically alarm requirements of 10 CFR Part 70.24 was granted in the Special Nuclear Materials License No. SNM-1895 dated November 27, 1985. This exemption is described in Section 9.1 of Supplement 4 to the SER. This previously granted exemption is continued in this operating license.

ii) Exemptions to certain requirements of Appendix J to 10 CFR Part 50 are described in Supplements 3, 4, and 5 to the SER. These include (a) (this item left intentionally blank); (b) an exemption from the requirement of Option B of Appendix J, exempting main steam isolation valve measured leakage from the combined leakage rate limit of 0.6 La. (Section 6.2.6 of SSER 5)*; (c) an exemption from Option B of Appendix J, exempting the hydraulic control system for the reactor recirculation flow control valves from Type A and Type C leak testing (Section 6.2.6 of SSER 3); (d) an exemption from Option B of Appendix J, exempting Type C testing on traversing incore probe system shear valves. (Section 6.2.6 SSER 4)

iii) An exemption to Appendix A to 10 CFR Part 50 exempting the Control Rod Drive (CRD) hydraulic lines to the reactor recirculation pump seal purge equipment from General Design Criterion (GDC) 55. The CRD hydraulic lines to the reactor recirculation pump seal purge equipment use two simple check valves for the isolation outside containment (one side). (Section 6.2.4, SSER 3)

iv) A schedular exemption to GDC 2, Appendix A to 10 CFR Part 50, until the first refueling outage, to demonstrate the adequacy of the downcomer design under the plant faulted condition. This exemption permits additional analysis and/or modifications, as necessary, to be completed by the end of the first refueling outage. (Section 6.2.1.7.4, SSER 3)

v) A schedular exemption to GDC 50, Appendix A to 10 CFR Part 50 to allow the operating licensee until start-up following the "mini-outage," which is to occur within 12 months of commencing power operation (entering Operational Condition 1), to install redundant fuses in circuits that use transformers for redundant penetration protection in accordance with their letter of August 29, 1986 (NMP2L 0860). (Section 8.4.2, SSER 5)

* The parenthetical notation following the discussion of each exemption denotes the section of the Safety Evaluation Report (SER) and/or its supplements wherein the safety evaluation of the exemption is discussed.

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

January 29, 2021

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO
AMENDMENT NO. 183 TO
RENEWED FACILITY OPERATING LICENSE NO. NPF-69
NINE MILE POINT NUCLEAR STATION, LLC
LONG ISLAND LIGHTING COMPANY
EXELON GENERATION COMPANY, LLC.
NINE MILE POINT NUCLEAR STATION, UNIT 2
DOCKET NO. 50-410

Application (i.e., initial and supplements)	Principal Contributors to Safety Evaluation
• December 26, 2019 (ADAMS Accession No. ML19360A145) • April 30, 2020 (ADAMS Accession No. ML20121A005) • August 28, 2020 (ADAMS Accession No. ML20241A047)	• Mihaela Biro • Zach Coffman • Jigar Patel • Keith Tetter

1.0 PROPOSED CHANGE

The proposed amendment would modify the licensing basis of Nine Mile Point Nuclear Station, Unit 2 (Nine Mile Point 2, the licensee), to allow for the voluntary implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) that are 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), the requirements may not be changed.

Additionally, the proposed amendment would add a license condition proposed by the licensee that identifies action items that need to be completed prior to implementing 10 CFR 50.69 at Nine Mile Point 2 and identifies possible changes to the categorization process that would require prior U.S. Nuclear Regulatory Commission (NRC, the Commission) approval.

In the enclosure to its letter dated December 26, 2019 (Reference 1), the licensee stated that the internal and fire probabilistic risk assessment (FPRA) models credited in the amendment request are the same PRA models credited in the licensee's amendment request to adopt risk-informed completion time. The licensee's amendment request to adopt risk-informed completion time at Nine Mile Point 2 was submitted by letter dated October 31, 2019 (Reference 2). Since the licensee is using the same probabilistic risk assessment (PRA) models to support both amendment requests, the NRC staff considered PRA model information that the licensee provided as part of the licensee's amendment request to adopt risk-informed completion times at Nine Mile Point 2.

2.0 REGULATORY EVALUATION

2.1 Applicable Requirements

Section 50.69 of 10 CFR provides an alternative approach for establishing requirements for treatment of SSCs for nuclear power reactors using an integrated and systematic risk-informed process for categorizing SSCs according to their safety significance. Specifically, for SSC categorized as LSS, alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of HSS, requirements may not be changed.

Paragraph 50.69(b) of 10 CFR, "Applicability and scope of risk-informed treatment of SSCs and submittal/approval process," specifies the information required in an application for a license amendment to voluntarily adopt 10 CFR 50.69. Paragraph 50.69(b) indicates that the NRC will approve a licensee's program for implementing 10 CFR 50.69 if it determines it satisfies the requirements of 10 CFR 50.69(c).

Paragraph 50.69(c) of 10 CFR, "SSC Categorization Process," requires the licensee 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 four risk-informed safety classes (RISC) categories defined in the rule.

2.2 Criteria Specified in the Requirements

Paragraph 50.69(b)(2) of 10 CFR requires that a licensee voluntarily choosing to implement 10 CFR 50.69 submit an application for license amendment under 10 CFR 50.90 that contains a description of the process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs; a description of the measures taken to assure that the quality and level of detail of the systematic process that evaluates the plant for internal and external events is adequate for the application; results of the PRA review process; and a description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv). This description would include plant-specific PRA, PRA review process, margin-type approaches, and other systematic evaluation techniques that ensure the systematic evaluation processes are adequate for SSC categorization. The evaluation must include common cause interaction susceptibility and potential impacts from known degradation mechanisms for both active and passive functions.

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). The criteria specified in 10 CFR 50.69(c) are that the SSCs must be categorized by a categorization process that:

(1) uses a process that determines if an SSC performs one or more safety-significant functions, including identification of those functions (10 CFR 50.69(c)(1)),

(2) considers the results and insights from the plant-specific PRA that:

   a. at a minimum, model severe accident scenarios resulting from internal initiating events occurring at full power operation (10 CFR 50.69(c)(1)(i)),

   b. is of sufficient quality and level of detail to support the categorization process (10 CFR 50.69(c)(1)(i)), and

   c. is subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC (10 CFR 50.69(c)(1)(i)).

(3) determines the SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA; and reasonably reflects the current plant configuration and operating practices and applicable plant and industry operational experience (10 CFR 50.69(c)(1)(ii)),

(4) maintains defense in depth (DID) (10 CFR 50.69(c)(1)(iii)),

(5) includes evaluations that provide reasonable confidence that 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 permitted by implementation of 10 CFR 50.69(b)(1) and (d)(2) are small (10 CFR 50.69(c)(1)(iv)),

(6) performs categorization for entire systems and structures, not for selected components within a system or structure (10 CFR 50.69(c)(1)(v)), and

(7) includes an integrated decision-making panel (IDP) staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering (10 CFR 50.69(c)(2)).

2.3 Guidance Used by NRC Staff

Nuclear Energy Institute (NEI) 00-04, Revision 0, “10 CFR 50.69 SSC Categorization Guideline,” July 2005 (Reference 3), describes a method that the NRC considers acceptable for complying with 10 CFR 50.69 with respect to the categorization of SSCs that are considered in risk-informing special treatment requirements.

Regulatory Guide (RG) 1.201, Revision 1, “Guidelines for Categorizing Structure, Systems, and Components in Nuclear Power Plants According to their Safety Significance,” (Reference 4), endorses the categorization process described in NEI 00-04, Revision 0, with clarifications, limitations, and conditions. RG 1.201, Revision 1, states that the applicant is expected to document, at a minimum, the technical adequacy of the internal initiating events PRA. Licensees may use either PRAs or alternative approaches for hazards other than internal initiating events.

RG 1.201, Revision 1, clarifies that the NRC staff expects that licensees proposing to use non-PRA approaches in their categorization should provide a basis in the submittal for why the approach, and the accompanying method employed by the licensee to assign safety significance to SSCs, is technically acceptable.

RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” March 2009 (Reference 5), describes an acceptable approach for determining whether the acceptability of the base PRA, in total or the parts, that is used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decision-making for light-water reactors. It endorses, with clarifications, the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard, ASME/ANS RA-Sa-2009, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” February 2009.

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 6), provides guidance and acceptance guidelines on the use of PRA findings and risk insights in support of changes to a plant’s licensing basis.

NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision-making,” March 2017 (Reference 7), provides guidance on how to treat uncertainties associated with PRA in risk-informed decision-making. The guidance fosters an understanding of the uncertainties associated with PRA and their impact on the results of the PRA and provides an approach to addressing these uncertainties in the context of the decision-making.

NEI 05-04 (Reference 8), NEI 07-12 (Reference 9), and NEI 12-13 (Reference 10) provide guidance to perform an independent assessment for the closure of findings and observations (F&Os) identified from a full-scope or focused-scope peer review. Appendix X to NEI 05-04/07-12/12-16 (Reference 11) includes guidance for the independent assessment process regarding: (i) the qualifications of the independent assessment team members, (ii) pre-review activities, (iii) on-site review activities, and (iv) post-review activities, thus assuring that closure of the F&Os are met at Capability Category (CC) II for the applicable supporting requirements in the ASME/ANS Ra-SA 2009 PRA standard (Reference 12), as endorsed by RG 1.200, Revision 2. The NRC staff has accepted the guidance in Appendix X in a memorandum dated May 3, 2017 (Reference 13).

Nuclear Management and Resources Council (NUMARC) 91-06, “Guidelines for Industry Actions to Assess Shutdown Management” (Reference 14), provides considerations for maintaining DID for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment primary/secondary. NUMARC 91-06 also specifies that a DID approach should be used with respect to each defined shutdown key safety function.

2.4 Previous NRC Approvals Reviewed by NRC Staff (i.e., Precedents)

Arkansas Nuclear One, Unit 2, Alternative Request

An alternative request for Arkansas Nuclear One, Unit 2 (ANO-2), included a risk-informed repair or replacement activities (RI-RRA) methodology for a risk-informed safety classification and treatment program for repair and replacement activities for Class 2 and Class 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." The NRC approved the alternative request by letter dated April 22, 2009, which included the NRC staff's safety evaluation (SE) (Reference 15). In the SE, the NRC staff explains why the RI-RRA is acceptable for determining safety significance of Class 2 and Class 3 passive components.

Calvert Cliffs Nuclear Power Plant, Units 1 and 2, License Amendment Request

A license amendment request (LAR) for Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs), included an alternative seismic approach for plants designated as Tier 1 for a qualitative consideration of seismic risk and SSC safety significance. The NRC approved the alternative request by letter dated February 28, 2020, which included the NRC staff's SE (Reference 16). In the SE, the NRC staff explains why the alternative seismic approach for plants designated as Tier 1 is acceptable for use in an integrated and systematic risk-informed process for categorizing SSCs.

Evaluation of case studies in Electric Power Research Institute (EPRI) Report 3002012988 for Tier 1 application has been conducted by the NRC in the SE for the Calvert Cliffs 50.69 LAR (Reference 16). The conclusions for these case studies are applicable to this 50.69 LAR.

The Calvert Cliffs 50.69 SE evaluated the peer review process and resolution of peer review findings and key assumptions and sources of uncertainty for Plants A, C, and D. It concluded that the PRA models used for these three plants are technically acceptable. In addition, the NRC staff finds that the mapping of SSCs between the seismic PRA (SPRA), full power internal events PRA and, as applicable, FPRA for these case studies, was performed in a technically justifiable manner. Furthermore, the NRC staff finds that the conclusions on the determination of unique HSS SSCs from SPRAs in EPRI Report 3002012988 from these case studies are valid.

The Calvert Cliffs 50.69 SE (Reference 16) evaluated the following criteria for the applicability and use of the proposed seismic Tier 1 approach:

- Ground motion response spectrum (GMRS) peak acceleration is at or below approximately 0.2 gram (g) or where the GMRS is below or approximately equal to the safe shutdown earthquake (SSE) between 1.0 hertz (Hz) and 10 Hz.

- The NRC staff concluded in the Calvert Cliffs 50.69 SE that the proposed criteria in EPRI Report 3002012988 to determine the applicability and use of the proposed seismic Tier 1 approach is acceptable.

The licensee provided a supplement to Section 3.1.1 of the enclosure to its letter dated December 26, 2019 (Reference 17), and stated that Nine Mile Point 2 follows the same categorization approach for seismic risk as approved for Calvert Cliffs, Units 1 and 2, as identified in the Calvert Cliffs 50.69 SE with no deviations.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the licensee's proposed categorization process against (as identified in Section 2.0 of this SE):

- criteria specified in 10 CFR 50.69
- guidance for developing a categorization process endorsed by the NRC
- guidance for risk-informed licensing basis changes issued by the NRC
- guidance for PRA technical adequacy issued or endorsed by the NRC
- methods found acceptable by the NRC in other plant-specific licensing actions approved by the NRC

3.1 Categorization Process

3.1.1 Overall Categorization Process

The NRC staff reviewed the licensee's proposed categorization process against the categorization process in NEI 00-04, Revision 0, as endorsed in RG 1.201. The licensee stated in Section 3.1 of the enclosure to its letter dated December 26, 2019, that it will implement the risk categorization process in accordance with NEI 00-04, as endorsed by RG 1.201. The LAR, as supplemented by letter dated April 30, 2020 (Reference 17), contains a categorization flow chart and categorization evaluation summary table that explains the categorization process and identifies which steps are performed at the component level and which steps are performed at the function level. The licensee clarifies that the execution sequence of steps and elements of the process does not impact the resulting preliminary categorization because the safety determination of each element of the process is independent of each other.

The NRC staff finds that the criterion specified in 10 CFR 50.69(b)(2)(i) concerning a description of the process for categorization is met because the licensee, in its LAR, as supplemented, appropriately described the proposed categorization process. Consistent with the requirements in 10 CFR 50.69(c), the licensee's categorization process is an integrated decision-making process to categorize safety-related and non-safety-related SSCs and can determine whether an SSC performs one or more safety-significant functions, including identification of those functions (10 CFR 50.69(c)(1)).

3.1.2 Defense in Depth

In Section 3.1.1 of the enclosure to its letter dated December 26, 2019, the licensee describes that the DID assessment will be performed consistent with the guidance in Section 6 of NEI 00-04. The licensee states that the IDP cannot change the categorization of a SSC categorized as HSS based on a DID assessment. The NRC staff finds that the criterion specified in 10 CFR 50.69(c)(1)(iii) concerning DID is met because the licensee will perform the DID assessment in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

3.1.3 Categorization for Entire Systems

The regulation in 10 CFR 50.69(c)(1)(v) requires that the SSC categorization be performed for entire systems and structures, not for selected components within a system or structure. Section 2 of NEI 00-04 provides guidance for meeting 10 CFR 50.59(c)(1)(v).

Section 4 of NEI 00-04 states that “there may be circumstance where the categorization of a candidate LSS SSC within the scope of the system being considered cannot be completed because it also supports an interfacing system. In this case, the SSC will remain uncategorized until the interfacing system is considered.”

In Attachment 1 to its letter dated August 28, 2020 (Reference 18), the licensee explained how the proposed process addresses the guidance in NEI 00-04 regarding interfacing components. Interfacing components provide functional capability for two or more systems. The licensee explained that system boundaries can be defined in different ways such as by system function, using piping and instrumentation diagram boundaries, or by maintenance rule functions. However, once the system boundary is defined, all components within the boundary that support a system safety function or the design and licensing basis of that system are included. The licensee explained that a list of functions is developed for the components in the system being categorized, and if the system includes an interfacing SSC, then support functions are also defined. The support functions are functions provided by the interfacing SSC for systems outside the system being categorized. The licensee explained that two options are available for categorizing the interface SSC. Option 1 is to fully categorize the interfacing SSC without fully categorizing the interfacing systems. The licensee stated that all functions of the interface SSC will be identified, including in its assigned primary system, as well as all functions to all other systems it interfaces with. The licensee explained that the component safety significance and the DID assessment will be performed according to the guidance in NEI 00-04, Sections 5 and 6, and that the safety-significant SSCs are mapped to the appropriate functions. The SSC is assigned the highest risk significance for any function that the SSC supports. Option 2 is to categorize the interfacing SSC after all the interfacing systems are categorized in their entirety.

The NRC staff finds the licensee’s approach to categorizing SSCs that support multiple systems and functions acceptable and consistent with the guidance in NEI 00-04 because when Option 1 is used, the functions for the interfacing SSC are fully considered, including those that could cause the SSC to be categorized as HSS, and when Option 2 is used, the interfacing SSC is left uncategorized until the impact on all the interfacing systems is categorized in its entirety.

3.1.4 Integrated Decision-Making Panel

The licensee described the role of the IDP in categorizing SSCs on pages 6 to 9 of Enclosure 1 of its letter dated December 26, 2019. The licensee states that the role of the IDP in its proposed categorization process is consistent with NEI 00-04, Revision 0. The licensee described the expertise of the IDP on page 9 of the enclosure to its letter dated December 26, 2019. The licensee stated that the IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and PRA, and that, “the IDP will be trained in the specific technical aspects and requirements related to the categorization process.”

The NRC staff finds that the criterion specified in 10 CFR 50.69(c)(2) concerning IDP is met because in the licensee’s proposed categorization process, (1) SSCs will be categorized by an IDP as described in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1, and (2) the IDP will comprise the required expertise prescribed in 10 CFR 50.69(c)(2).

3.2 Risk Evaluation

3.2.1 PRA Evaluation

3.2.1.1 Scope of the PRA

As described in the LAR, the Nine Mile Point 2 PRA is comprised of internal events PRA (including internal flooding) and FPRA. The licensee stated that the PRA models credited in the request are the same PRA models credited in the LAR to adopt risk-informed completion time (TSTF-505 LAR) (Reference 2). Therefore, the NRC staff's review of the internal events and flooding PRAs is based on the results provided in the TSTF-505 LAR (Reference 2) and TSTF-505 response to the request for additional information (RAI) (Reference 19).

In its TSTF-505 response to the RAI, the licensee addresses credit for diverse and flexible coping strategies (FLEX) in the licensee's PRA models. This issue is important to 10 CFR 50.69 categorization because modeling treatment of FLEX can impact the risk importance values calculated by the PRA models for use in risk-informed categorization. The licensee explained that the following FLEX capabilities were credited in the internal events PRA (including internal flooding) and FPRA:

- use of portable FLEX diesel generators to provide direct current power and power for long-term nitrogen makeup to the safety relief valves
- use of portable FLEX diesel-driven pumps for injection of cooling water from the ultimate heat sink into the reactor pressure vessel
- use of the hardened containment vent system

In its TSTF-505 response to the RAI, the licensee explained the failure data used. The licensee stated that the failure rates for the portable FLEX diesel generators and diesel-driven pumps were assumed to a factor of 2 greater than the failure rates listed in NUREG/CR-6928 (Reference 20) to account for a level of uncertainty. The licensee explained it compared these assumed values to failure rates presented in an analysis of limited industry FLEX equipment failure data provided in PWROG-18043-P, Revision 0, and found them to be consistent. The licensee also reviewed its plant-specific FLEX equipment failure data (which includes data since January 2015) and calculated preliminary plant-specific failures rates for the diesel generators and diesel-driven pumps. The NRC staff notes that these preliminary values provided in the response are in some cases higher than the assumed rates used in the PRA models. The licensee explained that these preliminary values are based on limited experience and that the FLEX program is still evolving as improvements have been made to the program. The licensee stated that it does not consider this early data to represent any outlier events.

In its TSTF-505 response to the RAI, the licensee identified three operator actions that contain activities identified in Sections 7.5.4 and 7.5.5 of NEI 16-06 (Reference 21) flood (e.g., installation of portable equipment at staging locations and addressing complex actions), for which existing human reliability analysis (HRA) approaches are not explicitly applicable. The licensee stated the results of a composite sensitivity study on failure rates and human error probabilities (HEPs) for different plant configurations whose risk could be impacted by this treatment. The licensee stated that it used the 95th percentile HEP values for the three identified operator errors. The licensee clarified that for the joint HEP, a minimum value of

1E-06 was used. The sensitivity study also assumed FLEX equipment failure rates at a factor of 5 greater than the generic values in NUREG/CR-6928. The licensee presented the CDF and LERF for four plant configurations and showed minimal increases in CDF and LERF.

In its TSTF-505 response to the RAI, the licensee described its evaluation of the modeling performed to incorporate FLEX into the PRA against the definition of a PRA upgrade provided in the ASME/ANS RA-Sa-2009 PRA standard using guidance from RG 1.200, Revision 2. The NRC staff is unable to unequivocally conclude that the licensee's implementation of FLEX credit does not constitute an upgrade. Specifically, the development and modeling of HEP associated with transportation and assembling of portable equipment may constitute PRA upgrade. However, the impact of errors associated with this specific HEP has minimal impact to the application. Given the results of the sensitivity studies conducted by the licensee that showed a small impact on CDF and LERF, the NRC staff finds that the treatment of FLEX in Nine Mile Point 2 PRA is acceptable for the 10 CFR 50.69 application. The NRC staff notes that the 10 CFR 50.69 categorization process does not exclusively rely on the PRA model, as it includes other non-PRA considerations. Further, consistent with the requirements in 10 CFR 50.69(c)(1)(ii), the licensee is required to maintain the PRA and the categorization process to reasonably reflect the operating practices and applicable plant and industry operational experience. Therefore, any future plant and industry operating experience affecting the FLEX equipment failure rates and HEPs is expected to be captured by the PRA maintenance process.

3.2.1.2 Internal Events PRA

The NRC staff reviewed the peer review history for the Nine Mile Point 2 internal events and internal flooding PRAs as described in Section 3.3 of the enclosure to the licensee's letter dated December 26, 2019 and presented in Attachment 3 of the enclosure. A full-scope peer review of the internal events PRA was performed by the licensee in July 2009 using the NEI 05-04 process (Reference 8) and the guidance in the ASME/ANS RA-Sa-2009 PRA standard and RG 1.200, Revision 2 (Reference 5). An F&O closure review was performed in February 2019 on all the internal events (including internal flooding) findings. It was performed by an independent assessment team, consistent with guidance in Appendix X of NEI 05-04 and clarifications in the NRC's acceptance letter dated May 3, 2017 (Reference 13).

Additionally, concurrent with the 2019 F&O closure review, a focused-scope peer review was performed by the licensee on the incorporation of support system initiating event (SSIE) fault trees. This focused-scope peer review resulted in three F&Os (i.e., F&O 5-1, 8-1, and 8-2) associated with supporting requirements that were found not to be met. The NRC staff reviewed these F&Os along with dispositions for this application. The F&O dispositions refer to the licensee's proposed implementation item in Attachment 6 of the enclosure to its letter dated December 26, 2019 (i.e., one of six implementation items related to PRA modeling updates) to resolve these F&Os prior to implementation of the 10 CFR 50.69 categorization. In response to RAI 6 associated with the TSTF-505 LAR, the license described the modeling updates associated with these F&Os and stated that they are now closed in the latest PRA model "via the 2020 F&O Closure Review."

In its TSTF-505 LAR response to the RAI, the licensee provided a summary of the internal events PRA model changes made since the July 2009 full-scope peer review and justification for why each change did or did not meet the definition of a PRA upgrade using the definition in the ASME/ANS R-Sa-2009 PRA standard. The licensee indicated that one PRA change was determined to be a PRA upgrade related to a change from a cognitive reliability HRA model to an accident sequence evaluation program (ASEP) time reliability HRA model). In its TSTF-505 response to the RAI, the licensee stated that a focused-scope peer review was performed in 2020 on this HRA PRA upgrade. This focused-scope peer review led to three additional F&Os (i.e., F&O 20-1, 20-2, and 20-3) for which the licensee provided dispositions in the RAI response. The NRC staff reviewed the dispositions for these new F&Os and concluded that they do not impact the application, as follows.

For F&O 20-1, the resolution involves documentation of a reasonableness check that has already been performed. For F&O 20-2, the resolution involves correcting HRA documentation for post-initiator actions execution errors in which the ASEP method was erroneously referenced. For F&O 20-3, the resolution involves updating the HRA calculator database to be consistent with the modeling performed in the calculator and update of the HRA documentation to clarify the timelines for ASEP calculations.

Based on the NRC staff's review, the NRC staff finds that the internal events and internal flooding PRA has been adequately peer reviewed against the current version of the PRA standard and RG 1.200, and that the licensee has adequately dispositioned the F&Os to support the technical adequacy of the internal events PRA for the Nine Mile Point Unit 2 50.69 risk-informed categorization process.

3.2.1.3 Fire PRA

The NRC staff's review of the Nine Mile Point 2 FPRA was based on the results of a full-scope peer review of the FPRA and the associated F&Os closure review described in Enclosure 2 of the licensee's TSTF-505 letter dated October 31, 2019 (Reference 2). A full-scope peer review of the FPRA was performed in June 2018 using the NEI 07-12 process and the guidance in the ASME/ANS RA-Sa-2009 PRA standard and RG 1.200, Revision 2. An F&O closure review was performed in February 2019 on all the FPRA findings. It was performed consistent with the guidance in Appendix X of NEI 07-12 and clarifications in the NRC's acceptance letter dated May 3, 2017. The F&O closure review closed out all open finding-level F&Os.

Based on the NRC staff's review, the staff finds that the FPRA has been adequately peer reviewed against the current version of the PRA standard and RG 1.200, and that the licensee has adequately dispositioned the F&Os to support the technical adequacy of the FPRA for the 10 CFR 50.69 risk-informed categorization process.

3.2.1.4 Results and Insights from Plant-Specific PRA

Paragraph 50.69(c)(1)(i) of 10 CFR requires the licensee to consider the results and insights from the PRA during categorization. In Section 3.1.1 of the enclosure to its letter dated December 26, 2019, the licensee notes that it will implement the risk categorization process in accordance with NEI 00-04, Revision 0. Therefore, insights and results obtained from Fussell-Vesely and risk achievement worth importance measures for SSCs, as well as insights obtained from sensitivity studies and uncertainty evaluations, are integrated into the categorization process as described in Sections 5 and 8 of NEI 00-04.

In Section 3.2.7 of the enclosure to its letter dated December 26, 2019, and Attachment 6, the licensee provides uncertainty evaluations and a list of key assumptions and sources of modeling uncertainties that were reviewed for the internal events and FPRAs. The licensee stated that it used the detailed process of identifying, characterizing, and qualitative screening of model uncertainties that is found in NUREG-1855, EPRI Topical Report (TR) 1016737 (Reference 22) and EPRI TR-1026511 (Reference 23). In its letter dated August 28, 2020, the licensee further explained the process for reviewing PRA assumptions and uncertainties. The licensee explained that, consistent with Steps E-1.2 and E-1.2 of NUREG-1855, Revision 1, generic sources of uncertainty were identified for (1) the internal events PRA from EPRI TR-1016737, and (2) the FPRA and Level 2 PRA from EPRI TR-1026511. The licensee also indicated that it reviewed the internal events and FPRA documentation for unique plant-specific sources of uncertainty. The licensee explained that the considerations for screening the initial comprehensive list of uncertainties down to the list presented in the LAR were based on whether:

(1) a consensus model as defined by NUREG-1855, Revision 1 was used,
(2) the uncertainty had an impact on the PRA results,
(3) there was a reasonable alternative to the assumption, or
(4) treatment of the uncertainty had a conservative bias impacting the risk results.

For the key assumptions and sources of uncertainty that were not screened, the licensee shows in Attachment 6 of the enclosure to its letter dated December 26, 2019, that its evaluation did not identify the need for additional sensitivity analyses for this application. Based on the above, the NRC staff finds that the licensee appropriately searched for, identified, and evaluated PRA key assumptions and sources of uncertainties consistent with the guidance in NUREG-1855, Revision 1, and RG 1.200.

The NRC staff finds that the criterion specified in 10 CFR 50.69(c)(1)(i) concerning results and insights from the PRA during categorization is met because the licensee addresses importance measures, sensitivity studies, and uncertainty evaluations for internal events and FPRAs consistent with NEI 00-04, Revision 0, as endorsed by NRC in RG 1.201, Revision 1.

3.2.1.5 Degradation Mechanisms and Common Cause Interaction

Section 50.69(b)(2)(iv) of 10 CFR requires the licensee's evaluation to include the effects of common cause interaction susceptibility and potential impacts from known degradation mechanisms. RG 1.201 notes that Section 12.4 of NEI 00-04 regarding corrective action describes an acceptable approach to address degradation mechanism and common cause failure concerns.

Further, Section 8 of NEI 00-04 provides justification of how degradation mechanisms and common cause failure concerns are addressed in the categorization process, including the inherent considerations for common cause interactions typically included in the PRA models. It also requires PRA risk sensitivity studies to be performed for all the LSS components to assure that if the unreliability of the components was increased, the increase in risk would be small.

The NRC staff finds that 10 CFR 50.69(b)(2)(iv) is met because the licensee's proposed categorization process is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

3.2.1.6 PRA 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 internal events PRA and FPRA to support SSC categorization is endorsed by RG 1.201, Revision 1. Under 10 CFR 50.69(c)(1)(i), the PRAs "must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard [...] that is endorsed by the NRC." Revision 2 of RG 1.200 provides guidance for determining the acceptability of the PRA by comparing the PRA against the relevant requirements of the ASME/ANS 2009 standard using a peer review process. As discussed above, the licensee has subjected the internal events PRA (including, internal flooding) and FPRA to the peer review processes and submitted the results of the peer review. The NRC staff reviewed the peer review history (which included the results and findings), the licensee's resolution of peer review findings, and the identification and disposition of key assumptions and sources of uncertainty. The NRC staff finds that (1) the licensee's internal events PRA (including, internal flooding) and FPRA are acceptable to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201, Revision 1, and (2) the key assumptions and sources of uncertainty 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 internal events PRA (including, internal flooding) and FPRA meet the requirements set forth in 10 CFR 50.69(c)(1)(i) because the licensee's proposed process considers integrated importance measures, sensitivity studies, and uncertainty consistent with NEI 00-04, as endorsed by RG 1.201, Revision 1. The NRC staff similarly finds the licensee has complied with the 10 CFR 50.69(b)(2)(iii) requirement to submit the "results of the PRA process conducted to meet [10 CFR] 50.69(c)(1)(i)."

3.2.2 <u>Evaluation of the Use of Non-PRA Methods in SSC Categorization</u>

3.2.2.1 Alternative Seismic Approach

As part of its proposed integrated decision-making process to categorize SSCs according to safety significance, the licensee has proposed to use a non-PRA method to consider seismic hazards. Sections 50.69(c)(1)(ii) and 50.69(b)(2)(ii) of 10 CFR permit the use of non-PRA methods in a risk-informed categorization process. The licensee provided a description of its proposed alternative seismic approach for considering seismic risk in the categorization process and how the proposed alternative seismic approach would be used in the categorization process in LAR Section 3.2.3 (Reference 1) and its supplement dated April 30, 2020 (Reference 17). The information presented in the LAR and supplements, as well as in EPRI Report 3002012988 (Reference 24), taken together, provides sufficient detail for the use of the proposed alternative seismic approach as part of the proposed categorization process. Specifically, it describes 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.

EPRI Report 3002012988 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from SPRAs. The NRC staff's review confirmed that the case studies in EPRI Report 3002012988 and the licensees' supplement (Reference 17) 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 Nine Mile Point 2 as compared to the precedent (Reference 16). The information presented in the LAR and the supplements 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.

3.2.2.1.1 Evaluation of Applicability of Alternative Seismic Approach Criteria for this Application

The licensee compared the GMRS from the reevaluated seismic hazard developed in response to Near-Term Task Force Recommendation 2.1 against the site's design-basis SSE, as shown in Figure A4-1 of Attachment 4 of the enclosure to its letter dated December 26, 2019, to demonstrate that the site meets the criteria for application of the proposed seismic Tier 1 approach. The NRC staff previously evaluated the licensee's response to the 10 CFR 50.54(f) letter associated with Near-Term Task Force Recommendation 2.1 in which the licensee submitted its reevaluated seismic hazard (Reference 25). The NRC staff's previous assessment of the licensee's reevaluated seismic hazard (Reference 26) states that the licensee's methodology was acceptable and that the GMRS determined using the reevaluated hazard adequately characterized the site. Since the same reevaluated hazard is used for comparison against the criteria for use of the proposed alternative seismic approach, the NRC staff's previous assessment on the reevaluated hazard is applicable to this review. The NRC staff's evaluation of the licensee's reevaluated seismic hazard determined that the GMRS peak acceleration is below the plant SSE between 1 Hz to 10 Hz, although the GMRS is higher than 0.2 g. Based on its review, the NRC staff finds that the licensee's seismic hazard meets the criteria for the proposed seismic Tier 1 approach.

In Section 3.2.3 of the enclosure to its letter dated December 26, 2019, 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. However, the licensee did not provide information to support the claim that the plant-specific seismic risk is a small percentage contribution to total plant risk. As the EPRI report uses the NEI 00-04 categorization process in which an integral assessment weights the importance from each risk contributor, evaluation of seismic risk to total risk was evaluated by the NRC staff based on the following:

- Seismic risk was based on the Nine Mile Point 2 TSTF-505 submittal (Section 3.1.1 of Enclosure 4, Reference 2) with seismic CDF of 6.4E-7/year (yr) and seismic large early release frequency of 3.2E-7/yr (and 6.4E-07/yr when containment is deinerted)

- Internal event risk was based on this application (Attachment 2) with CDF of 1.8E-6/yr and LERF of 2.6E-7/yr

- Internal fire event risk was based on this application (Attachment 2) with CDF of 3.1E-5/yr and LERF of 6.2E-6/yr

The NRC staff verified the licensee's seismic CDF and LERF estimates using the median seismic capacity of 1.076 and composite beta factor equal to 0.46 provided in the LAR and determined that the licensee's seismic risk is conservative. In addition, the high confidence in low probability of failure value selected in this application is reasonable and is based on the licensee's individual plant examination for external events (IPEEE) (Reference 27) for the first 24 hours, which is consistent with seismic events. When comparing seismic risk to total plant risk, the seismic CDF was approximately 2 percent of the total plant CDF, and the seismic large early release frequency was approximately 5 percent of the total plant LERF, which demonstrates the small percentage contribution of seismic risk to total plant risk.

The NRC staff's finds that the licensee's basis for applying the proposed seismic Tier 1 approach to its site is acceptable because: (1) the reevaluated hazard meets the criteria for use of the proposed seismic Tier 1 approach, and (2) the seismic risk contribution would not solely result in an SSC being categorized as HSS.

3.2.2.1.2 Evaluation of the Implementation of Conclusions from the Case Studies for This Application

The categorization conclusions from the case studies in EPRI Report 3002012988 and the licensees' supplement (Reference 17) 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 3002012988 and precedent (Reference 16) were applicable to Nine Mile Point 2.

In Section 3.2.3 of the enclosure to its letter dated December 26, 2019, the licensee states 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 Nine Mile Point 2. The objective of the alternative seismic approach, as described in Figure 3-1 of the LAR, is to identify plant-specific seismic insights derived from the components in the system being categorized.

The licensee stated that the case study Plants A, C, and D in EPRI Report 3002012988 illustrated that it would be unusual for moderate seismic hazard sites to exhibit any unique seismic insights, including correlated failures. According to the EPRI report this illustration is also applicable to low seismic hazard sites. 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 Report 3002012988 and the licensee's assertion of plant-specific applicability to the precedent. The NRC staff also evaluated the peer review process and resolution of peer review findings, and key assumptions and sources of uncertainty for Plants A, C, and D. In EPRI Report 3002012988 low hazard sites are defined as sites were the GMRS is either very low or similar to the SSE. In its LAR, the licensee provided justification that the Nine Mile Point site is a low seismic hazard site. Specifically, the licensee stated that the GMRS is less than the SSE in the 1 to 10 Hz range, which is very low. The NRC staff finds that the case studies in the precedent (Reference 16) are applicable to this licensee's proposed plant-specific alternative seismic approach. Therefore, the NRC staff's review of the proposed alternative seismic approach, in conjunction with the requirements in 10 CFR 50.69 and the corresponding statements of considerations, finds that the proposed alternative seismic approach provides reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(ii), as well as 10 CFR 50.69(c)(1)(iv) because:

- It 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.

- It 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.

- 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 Report 3002012988. 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.

- It 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.

- It 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.

3.2.2.1.3 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 December 26, 2019, the licensee stated that it will use the EPRI 3002012988 approach. The license stated that the threshold between Tier 1 (i.e., low seismic hazard sites) and Tier 2 (i.e., moderate seismic hazard sites) is clearly defined and traceable. As part of its proposed approach, the licensee stated that if new information is obtained that would change the seismic hazard at Nine Mile Point 2, it would "follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate."

The NRC staff's review finds that consideration of changes to seismic hazard in the licensee's plant-specific proposed alternative seismic approach is appropriately considered the precedent (Reference 16). Therefore, the NRC staff's evaluation of the consideration of changes to the seismic hazard against the requirements in 10 CFR 50.69(e)(1), 10 CFR 50.69(e)(3), and 10 CFR 50.69(d)(2)(ii) as well as the resulting conclusion on consideration of changes to the seismic hazard in the precedent is applicable to this licensee's proposed alternative seismic approach. Therefore, based on its review, the NRC staff finds that the proposed alternative seismic approach acceptably includes consideration of changes to the seismic hazard that exceed the criteria for use of the proposed alternative seismic 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.

3.2.2.1.4 Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach

In Section 3.5 of the enclosure to the LAR, the licensee describes the feedback and adjustment process for the proposed categorization process that will be used to identify and assess significant changes to the plant risk profile. The licensee stated that its configuration control process ensures that changes to the plant, including physical changes and changes to documents, are evaluated to determine the impact to drawings, design bases, licensing documents, programs, procedures, and training. The licensee explicitly identifies the inputs for the qualitative determinations for the alternative seismic approach and other seismic considerations as a specific area to be monitored.

The NRC staff's review found that the consideration of the feedback and adjustment process in the licensee's proposed alternative seismic approach is acceptable. The NRC staff's review determined that (1) the licensee's programs provide reasonable assurance that the existing seismic capacity of low safety-significant (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 reasonable confidence exists 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.

3.2.2.2 External Hazards and Other Hazards (Non-Seismic)

The licensee discussed its consideration of non-seismic external hazards in Section 3.2.4 of the enclosure to its letter dated December 26, 2019. Non-seismic external hazards include high winds, external flood hazards, and other hazards listed in Appendix 6-A of the 2009 ASME/ANS PRA standard (RA-Sa-2009). The licensee evaluated all non-seismic external hazards for the 10 CFR 50.69 application using a plant-specific evaluation in accordance with Generic Letter 88-20 and the hazard screening criteria in Part 6 of the 2009 ASME/ANS PRA standard. The NRC staff reviewed the licensee's evaluation, which was provided in Attachments 4 and 5 of the enclosure to the letter dated December 26, 2019.

Some of the hazards that have been further evaluated since the industry's individual plant examination for external events are the external flooding and extreme wind and tornado hazards. The licensee discussed its flooding focused evaluation (Reference 28 and Reference 29) in Attachment 4 of the enclosure to its LAR and concluded that all flood-causing mechanisms, except local intense precipitation, are bounded by the current licensing basis. Moreover, the licensee stated that analysis shows a local intense precipitation event will not cause enough water to enter buildings with safety-related SSCs or accumulate to a depth that affects any of the safety-related SSCs.

In Attachment 4 of the enclosure to its letter dated December 26, 2019, the licensee stated that SSCs identified in the Updated Safety Analysis Report (Reference 30) as "key equipment and structures" are designed to withstand tornadoes with a maximum rotation velocity of 290 miles per hour and a resultant velocity of 360 miles per hour. The NRC staff's review of the design basis and NUREG/CR-4461, "Tornado Climatology of the Contiguous United States" (Table 6-1, Reference 31), finds that the occurrence frequency of the design-basis wind speeds for the site is less than 1×10^{-7} per year. Regarding tornado missile risk, in Attachment 4 of the enclosure to its letter dated December 26, 2019, the licensee stated that a review of more recent tornado missile risk evaluations concluded that the tornado missile CDF is much less than 1E-6 per year. In the LAR, the licensee stated that no active or passive SSCs are credited to screen high winds and tornado missile hazards other than seismic Category 1 structures, which are designated as HSS for 10 CFR 50.69 categorization. The licensee also screened out hurricane based on the plant's location.

The NRC staff's review finds that the licensee's SSC categorization process will evaluate the safety significance of SSCs for non-seismic external hazards in Attachment 4 of the enclosure to the letter dated December 26, 2019, consistent with the guidance provided in Figure 5-6 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1.

3.2.2.3 Component Safety Significance Assessment for Passive Components

Passive components and passive function of active components (e.g., the pressure or liquid retention of the body of a motor-operated valve) are not modeled in PRAs. Therefore, a different assessment method is necessary to assess the safety significance of these components. Passive components include Class 2 and Class 3 pressure-retaining items and their associated supports exclusive of Class CC and MC items. In Section 3.1.2 of the enclosure to its letter dated December 26, 2019, the licensee described how passive components safety significance is determined in its proposed process to categorize safety-related and non-safety-related SSCs.

The licensee states safety significance of passive components will be evaluated using the ANO-2, RI-RRA methodology and consistent with the related NRC staff's SE. The licensee states that all ASME Code Class 1 SSCs with a pressure-retaining function, as well as supports, will be assigned HSS for passive categorization. The licensee further stated that consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

Because (1) all Class 1 SSCs and supports will be considered HSS, (2) only Class 2 and Class 3 SSCs will be categorized using the ANO-2 passive categorization methodology consistent with previous NRC staff approval, and (3) passive components assigned HSS cannot be changed by the IDP, the NRC staff finds the licensee's proposed approach for passive categorization, as part of its proposed categorization process, is acceptable.

3.2.2.4 Low Power and Shutdown

Consideration of low power and shutdown risk is necessary to ensure SSC functional importance for different plant operating modes. On pages 10, 18, and 19 of the enclosure to its letter dated December 26, 2019, the licensee describes how low power and shutdown risk is considered as part of the proposed categorization process. The licensee states that the process for addressing lower power and shutdown risk is consistent with NUMARC 91-06 (Reference 14) and NEI 00-04, Revision 0.

The NRC staff notes that the approach for considering low power and shutdown risk uses an integrated and systematic process that could identify HSS components, consistent with the shutdown evaluation process. The NRC staff finds the licensee's consideration of low power and shutdown risk acceptable, because it is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

3.2.3 Reflecting Current Plant Configuration and Operating Practices

Paragraph 50.69(c)(1)(ii) of 10 CFR requires, in part, that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices and applicable plant and industry operating experience. Paragraph 50.69(e) of 10 CFR requires review of changes to the plant, operational practices, industry operational experience, and corresponding periodic updates to the licensee's PRA and SSC categorization. In Section 3.2.6 of the enclosure to its letter dated December 26, 2019, the licensee describes administrative controls to ensure PRA models used to support categorization reflect the as-built, as-operated plant over time. In Section 3.5 of the enclosure to its letter dated December 26, 2019, the licensee describes the feedback and adjustment process for SSC categorization.

The licensee states that administrative controls are in place to ensure the PRA models used to support the categorization reflect the as-built, as-operated plant over time. The licensee's process includes regularly scheduled and interim PRA model updates. The process includes provisions for monitoring issues affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience), for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. Routine PRA updates are performed every two refueling cycles at a minimum, and SSC categorization will be reevaluated if there is significant impact to the PRA model from updates.

In Section 3.5 of the enclosure to its letter dated December 26, 2019, the licensee states that if significant changes to the plant risk profile are identified or if RISC-3 or RISC-4 SSC can (or actually did) prevent a safety-significant function from being satisfied, an immediate evaluation and review will be performed prior to the normally scheduled periodic review, which is once every two refueling outages. In addition, if it is determined that risk information changes, SSC performance change, design changes, or operational changes can more than minimally affect categorization results, then the risk information and SSC categorization process will be updated.

The NRC staff finds the criterion in 10 CFR 50.69(c)(1)(ii) concerning modeling of current plant configuration is met because the licensee's administrative control process maintains an as-built, as-operated PRA.

3.3 Sufficient Safety Margin and Potential CDF and LERF Increases

Paragraph 50.69(c)(1)(iv) of 10 CFR requires reasonable confidence that sufficient safety margins are maintained for SSCs categorized as RISC-3. The engineering evaluation that will be conducted by the licensee under 10 CFR 50.69, consistent with the risk categorization process described in NEI 00-04, Revision 0, as endorsed by RG 1.201, will assess the design function(s) and risk significance of the SSCs to assure that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring the categorization of the SSC does not adversely affect any assumptions or inputs to the safety analysis, or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist.

Consistent with the guidance provided in NEI 00-04 for review of safety margins, and in accordance with the implementation of the SSC categorization program, the only requirements that are relaxed for LSS SSCs (includes RISC-3) are those related to treatment. The SSCs' design-basis function, as described in the plant's current licensing basis, including the Updated Safety Analysis Report and technical specifications (TSs) bases, do not change and should continue to be met. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis.

Paragraph 50.69(c)(1)(iv) of 10 CFR also requires reasonable confidence that any potential increases in CDF and LERF resulting from changes to treatment are small. Sections 3.1.1 and 3.2.7 of the enclosure to the licensee's letter dated December 26, 2019, address sensitivity studies and cumulative risk increases. The categorization process described in Section 8 of NEI 00-04, as endorsed by RG 1.201, includes an overall risk sensitivity study for all the LSS components to assure that if the unreliability of the components was increased, the increase in risk would be small (i.e., meet the acceptance guidelines of RG 1.174, Revision 3). Sections 3.1.1 and 3.2.7 of the LAR clarify that in the sensitivity study, the unreliability of all LSS SSCs modeled by the licensee in the PRAs will be increased by a factor of 3.

The NRC staff concludes that sufficient safety margins are maintained by the proposed methodology in NEI 00-04, as endorsed by RG 1.201, and provide reasonable confidence that any potential increases in CDF and LERF resulting from changes to treatment are small, and therefore, meet the requirements set forth in 10 CFR 50.69(c)(1)(iv). The NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(iv) are met because the licensee's evaluation includes the effects of common cause interaction susceptibility and the potential impacts from known degradation mechanisms.

3.4 Licensee's Proposed License Condition

In its letters dated December 26, 2019 (Reference 1), and April 30, 2020 (Reference 17), the licensee proposed to amend its renewed facility operating license by adding the following license condition:

> Exelon 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 [DATE], and all its subsequent associated supplements as specified in License Amendment No. [XXX] dated [DATE].
>
> Exelon will complete the items listed in Attachment 7 of Exelon letter to NRC dated [DATE] 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 implementation of the 10 CFR 50.69 categorization process.

> 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).

Attachment 7 of the enclosure to the licensee's letter dated December 26, 2019, identifies the following six implementation items that will be completed prior to the implementation of the 10 CFR 50.69 categorization:

- The internal events PRA models will be updated to incorporate the SSCs associated with TS Limiting Condition for Operation (LCO) 3.3.7.2.A concerning the mechanical vacuum pump isolation instrumentation consistent with the design-basis success criteria.

- The internal events PRA models will be updated to incorporate the failure mode associated with TS LCO 3.6.1.7 concerning suppression chamber-to-dry-well vacuum breakers that are inoperable for opening.

- The internal events PRA models will be updated for TS LCO 3.7.1.C to incorporate the success criteria for the service water pumps consistent with the design-basis success criteria when the ultimate heat sink is > 82 degrees Fahrenheit (°F).

- The internal events PRA models will be updated to incorporate the SSCs associated with TS LCO 3.7.1.D concerning the intake deicer heaters consistent with the design-basis success criteria.

- The internal flooding PRA model will be updated to incorporate the pipe break frequencies using the pipe break-length approach in the most current version of EPRI TR-1013141, Revision 3, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments" (Reference 32).

- The internal flooding PRA model will be updated to incorporate resolutions to F&Os 5-1, 8-1, and 8-2 from the focused-scope peer review of the internal events PRA regarding SSIE fault tree modeling. Concerning F&O 5-1, the correction factor that was used in the SSIE fault trees will be replaced with improved modeling. Concerning F&O 8-1, potential recovery actions that could be credited in the SSIE fault trees will be reviewed, and if found to impact the SSIE frequency, the SSIE fault tree modeling will be updated to include credit for the recovery action. Concerning F&O 8-2, the licensee explained that common cause failure events would be incorporated into the SSIE fault tree models with the appropriate mission time for an initiating event.

The NRC staff finds that the proposed license condition is acceptable, because: (1) it adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed by the NRC, and (2) the evaluation in SE Section 3.2.2 finds the non-PRA methods for assessing risk for internal fires, seismic, and passive components, which are deviations from NEI 00-04, to be acceptable.

4.0 REGULATORY FINDING

The NRC staff finds the licensee's proposed categorization process acceptable for categorizing the safety significance of SSCs as it meets the requirements in 10 CFR 50.69(c) as follows:

(1) Uses an integrated decision-making process that determines if an SSC performs one or more safety-significant functions, including identification of those functions (see Sections 3.1.1, 3.1.3, and 3.2.2.3 of this SE);

(2) Considers results and insights from plant-specific internal events PRA that have been subjected to a peer review process against RG 1.200, Revision 2, and with the completion of the implementation items, will be of sufficient quality and level of detail to support the categorization process, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(i) (see Sections 3.2.1.4 and 3.2.1.6 of this SE);

(3) Determines SSC functional importance using an integrated, systematic process that reasonably reflects the current plant configuration, operating practices, and applicable plant and industry operational experience, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(ii) (see Sections 3.2.2.1.2 and 3.2.3 of this SE);

(4) Maintains DID, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(iii) (see Section 3.1.2 of this SE);

(5) Includes evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(iv) (see Sections 3.2.2.1.4 and 3.3 of this SE);

(6) Is performed for entire systems and structures rather than for selected components within a system or structure, and therefore, the requirements in 10 CFR 50.69(c)(1)(v) will be met upon implementation with the exception of an SSC that participates in the function of an interfacing system and can be fully characterized and categorized without performing categorization on the entire interfacing system (see Section 3.1.3 of this SE); and

(7) Includes categorization by IDP, staffed with expert, plant knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering, and therefore, meets the requirements in 10 CFR 50.69(c)(2) (see Section 3.1.4 of this SE).

5.0 CONCLUSION

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

6.0 REFERENCES

1. Rafferty-Czincila, S., Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, "Nine Mile Point Nuclear Station, Unit 2 - Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors,'" December 26, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19360A145)

2. Gudger, D., Exelon Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Nine Mile Point Nuclear Station, Unit 2 - License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times TSTF-505," Revision 2, October 31, 2019 (ADAMS Accession No. ML19304B653)

3. Nuclear Energy Institute, "10 CFR 50.69 SSC Categorization Guideline," NEI 00-04, July 2005 (ADAMS Accession No. ML052900163)

4. U.S. Nuclear Regulatory Commission, "Guideline for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" Regulatory Guide 1.201 (For Trial Use), Revision 1, May 2006 (ADAMS Accession No. ML061090627)

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

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

7. U.S. Nuclear Regulatory Commission, "Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," NUREG-1855, Volume 1, March 2017 (ADAMS Accession No. ML17062A466)

8. Nuclear Energy Institute, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," NEI 05-04, Revision 3, Washington, DC, November 2009

9. Nuclear Energy Institute, "Fire PRA Peer Review Process Guidelines," NEI 07-12, Revision 1, June 2010 (ADAMS Package Accession No. ML102230049)

10. Nuclear Energy Institute, "External Hazards PRA Peer Review," NEI 12-13, August 2012 (ADAMS Accession No. ML12240A027)

11. Andersen, Victoria, Nuclear Energy Institute, letter to Rosenberg, Stacey, U.S. Nuclear Regulatory Commission, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations," February 21, 2017 (ADAMS Package Accession No. ML17086A431)

12. American Society of Mechanical Engineers/American Nuclear Society, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” ASME/AND RA-Sa-2009, February 2009 (ADAMS Accession No. ML092870592)

13. Rosenberg, Stacey, U.S. Nuclear Regulatory Commission, letter to Greg Krueger, Nuclear Energy Institute, “U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 7-12, and 12-13, Close Out of Facts and Observations (F&Os),” May 3, 2017 (ADAMS Accession No. ML17079A427)

14. Nuclear Management and Resources Council, “Guidelines for Industry Actions to Assess Shutdown Management,” NUMARC 91-06, December 1991 (ADAMS Accession No. ML14365A203)

15. Markley, Michael, U.S. Nuclear Regulatory Commission, letter to Vice President, Operations, Arkansas Nuclear One, Entergy Operations, Inc., “Arkansas Nuclear One, Unit 2 – Approval of Request for Alternative ANO-2 R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 & 3 Moderate and High Energy Systems,” April 22, 2009 (ADAMS Accession No. ML090930246)

16. Marshall, Jr. M. L., U.S. Nuclear Regulatory Commission, letter to Hanson, B. C., Exelon Generation Company, LLC, “Calvert Cliffs Nuclear Power Plant, Units 1 and 2 – Issuance of Amendment Nos. 332 and 210 Re: Risk-informed Categorization and Treatment of Systems, Structures, and Components (EPID L-2018-LLA-0482),” February 28, 2020 (ADAMS Accession No. ML19330D909)

17. Gudger, D. T., Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, “Supplemental information No.1 for Nine Mile Point Nuclear Station, Unit 2 - Adopt 10 CFR 50.69, ‘Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors,” April 30, 2020 (ADAMS Accession No. ML20121A005)

18. Gudger, David T., Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission, “Nine Mile Point Nuclear Station, Unit 2 - Renewed Facility Operating License No. NPF-69, NRC Docket No. 50-410, Response to Request for Additional Information by the Office of Nuclear Reactor Regulation to Support Review of Nine Mile Point Nuclear Station, Unit 2, License Amendment Request Regarding Risk-Informed Categorization and Treatment of Structures, Systems, and Components,” August 28, 2020 (ADAMS Accession No. ML20241A047)

19. Gudger, D., Exelon Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, “Request for Additional Information for Nine Mile Point Nuclear Station, Unit 2 - Adopt TSTF-505, ‘Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,” Revision 2, August 28, 2020 (ADAMS Accession No. ML20241A044)

20. U.S. Nuclear Regulatory Commission, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” NUREG/CR-6928, February 2007 (ADAMS Accession No. ML070650650)

21. Reisi-Fard, M., U.S. Nuclear Regulatory Commission, Memorandum to Giitter, J. G., U.S. Nuclear Regulatory Commission, "Assessment of the Nuclear Energy Institute 16-06 Guidance for Risk-Informed Changes to Plants Licensing Basis," May 30, 2017 (ADAMS Accession No. ML17031A269)

22. Electric Power Research Institute, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments," EPRI TR-1016737, Palo Alto, CA, December 2008

23. Electric Power Research Institute, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," EPRI TR-1026511, Palo Alto, CA, December 2012

24. Electric Power Research Institute, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," EPRI Report 3002012988, Palo Alto, CA, July 2018 (Note: This document is no longer available from EPRI.)

25. Korsnick, M. G. Constellation Energy Nuclear Group, LLC, letter to U.S. Nuclear Regulatory Commission, "Nine Mile Point Nuclear Station, Units 1 and 3, Renewed Facility Operating License Nos. DPR-63 and NPF-69, Docket Nos. 50-220 and 50-410, Seismic Hazard and Screening Report (CEUS Sites) - Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Daichi Accident," March 31, 2014 (ADAMS Accession No. ML14099A196)

26. Vega, F. G., U.S. Nuclear Regulatory Commission, letter to Orphanos, P., Nine Mile Point Nuclear Station, "Nine Mile Point Nuclear Station, Units 1 and 2 – Staff Assessment of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Seismic Hazard Reevaluations for Recommendation 2.1 of The Near-Term Task Force Review of Insights from the Fukushima Daiichi Accident (TAC Nos. MF3973 and MF3974)," June 16, 2015 (ADAMS Accession No. ML15153A660)

27. Nine Mile Point Nuclear Station – Unit 2, Individual Plant Examination for External Events (IPEEE), SAS-TR-95-001, June 1995

28. Spina, J. A., Constellation Energy Nuclear Group, LLC, letter to U.S. Nuclear Regulatory Commission, "Nine Mile Point Nuclear Station, Units 1 & 2 - Flood Hazard Reevaluation Report (FHRR)," March 12, 2013 (ADAMS Package Accession No. ML130740943)

29. Kuntz, R. F., U.S. Nuclear Regulatory Commission, letter to Korsnick, M. G. Constellation Energy Nuclear Group, LLC, "Nine Mile Point Nuclear Station Units 1 and 2 – Staff Assessment of Response to 10 CFR 50.54(f) Information Request – Flood-Causing Mechanism Reevaluation (CAC NOS. MF1104 and MF1105)," July 24, 2014 (ADAMS Accession No. ML14153A410)

30. Nine Mile Point Unit 2 - Updated Safety Analysis Report (USAR) Revision 20, October 2012

31. U.S. Nuclear Regulatory Commission, "Tornado Climatology of the Contiguous United States," NUREG/CR-4461, Revision 2, February 2007 (ADAMS Accession No. ML070810400)

32. Electric Power Research Institute, "T Pipe Ruptures Frequencies for Internal Flooding Probabilistic Risk Assessments," TR-1013141, Revision 3, Palo Alto, CA, December 2008

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

January 29, 2021

NOTICE AND ENVIRONMENTAL FINDINGS
RELATED TO
AMENDMENT NO. 183 TO
RENEWED FACILITY OPERATING LICENSE NO. NPF-69
NINE MILE POINT NUCLEAR STATION, LLC
LONG ISLAND LIGHTING COMPANY
EXELON GENERATION COMPANY, LLC.
NINE MILE POINT NUCLEAR STATION, UNIT 2
DOCKET NO. 50-410

1.0 INTRODUCTION

By application dated December 26, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19360A145), as supplemented by letters dated April 30, 2020, and August 28, 2020 (ADAMS Accession Nos. ML20121A005 and ML20241A047, respectively), Exelon Generation Company, LLC (the licensee) submitted a license amendment request for the use of a risk-informed process for the categorization and treatment of structures, systems, and components at Nile Mile Point Nuclear Station, Unit 2 (Nine Mile Point 2).

The supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 25, 2020 (85 FR 10733).

2.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment on November 2, 2020. The State official had no comments.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types of any effluents, that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* (85 FR 10733; February 25, 2020). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT 2 - ISSUANCE OF AMENDMENT NO. 183 REGARDING RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS (EPID L-2019-LLA-0290) DATED JANUARY 29, 2021

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