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

June 11, 2020

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LA SALLE COUNTY STATION, UNITS 1 AND 2 – SUPPLEMENT TO THE AUDIT PLAN IN SUPPORT OF STAFF REVIEW OF LAR TO ADOPT 10 CFR 50.69 (EPID L-2020-LLA-0017)

Dear Mr. Hanson:

By letter dated January 31, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20031E699), Exelon Generation Company, LLC (Exelon, the licensee) requested that the U.S. Nuclear Regulatory Commission (NRC) modify the licensing basis of Renewed Facility Operating License Nos. NPF-11 and NPF-18 for La Salle County Station, Units 1 and 2 (La Salle).

Exelon's proposed license amendment request (LAR) would modify the La Salle licensing basis to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 69 (50.69), "Risk-Informed Categorization and Treatment of Structures, Systems and Components [SSCs] for Nuclear Power Reactors." The proposed changes are based on Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," dated July 2005 (ADAMS Accession No. ML052910035).

The NRC staff has reviewed Exelon's LAR and determined that a regulatory audit would assist in the timely completion of the LAR review. The initial audit plan consisting of "In-office Audit" and "Site Audit" was provided to the licensee on March 30, 2020 (ADAMS Accession No. ML20090F616). This letter provides an update to the audit plan to reflect a remote audit and the detailed audit agenda along with questions the NRC staff has prepared to be discussed during the remote audit. The staff will conduct a regulatory audit to support its review of the LAR in accordance with the enclosed audit plan.

The audit will be conducted from June 15, 2020, to June 19, 2020, remotely. The logistical details for the remote audit are provided to the licensee in the letter on TSTF-505 Audit.

It should be noted that the audit for this LAR, and regulatory audit for the risk-informed completion time LAR to adopt TSTF-505, are being conducted concurrently. The logistics and scope of this part of the audit were discussed with your staff on June 10, 2020. The audit plan supplement is enclosed.

If you have any questions, please contact me by telephone at 301-415-3308 or by e-mail to Bhalchandra.Vaidya@nrc.gov.

Sincerely,

/RA/

Bhalchandra K. Vaidya, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-373 and 50-374

Enclosure:
Audit Plan Supplement

cc: Listserv

AUDIT PLAN SUPPLEMENT

REGARDING RISK-INFORMED COMPLETION TIMES AND CATEGORIZATION AND

TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS

EXELON GENERATION COMPANY, LLC

LA SALLE COUNTY STATION UNITS 1 AND 2

DOCKET NO. 50-373 AND 50-374

1.0 BACKGROUND

By letter dated January 31, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20031E699), Exelon Generation Company, LLC (Exelon, the licensee) requested that the U.S. Nuclear Regulatory Commission (NRC) modify the licensing basis of Renewed Facility Operating License Nos. NPF-11 and NPF-18 for La Salle County Station, Units 1 and 2 (La Salle). Exelon's proposed process described in the license amendment request (LAR) would allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components [SSCs] for Nuclear Power Reactors." The proposed changes are based on Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," dated July 2005 (ADAMS Accession No. ML0713604560).

Specifically, the proposed amendment would modify the LaSalle licensing basis to allow for the implementation of the provisions of 10 CFR Section 50.69. The amendment proposes an alternative approach for the consideration of seismic risk in the 10 CFR 50.69 categorization process. The proposed alternative approach is a deviation from the approaches in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline" (ADAMS Accession No. ML052910035), as endorsed by the NRC staff in Regulatory Guide (RG) 1.201, "Guidelines for Categorization of Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 0 (ADAMS Accession No. ML061090627). The NRC staff has reviewed Exelon's submittal and determined that a regulatory audit of LaSalle's alternate seismic approach would assist in the timely completion of the subject LAR review process.

2.0 REGULATORY AUDIT BASES

The basis of this audit is Exelon's LAR for LaSalle and the Standard Review Plan, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (ADAMS Accession No. ML071700658).

The audit will be performed consistent with NRC Office of Nuclear Reactor Regulation Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195). An audit was determined to be the most efficient approach toward a timely resolution of potential issues associated with this LAR review, since the NRC staff will have an opportunity to minimize the potential for multiple rounds of requests for additional information

(RAIs) and ensure no unnecessary burden will be imposed by requiring the licensee to address issues that are not needed to render a staff finding. Upon completion of this audit, the staff will develop any RAIs, as determined are needed to allow the staff to complete the LAR review. The final RAIs will be issued after the audit.

3.0 PURPOSE AND SCOPE

The purpose of the audit is to gain a more detailed understanding of the basis and implementation of the licensee's approach for their proposed categorization process and to gain more information relevant to the review of the subject LAR. Specifically, the NRC staff will examine the licensee's material to support the probabilistic risk assessment (PRA) technical acceptability, the alternate approach for seismic to consider the seismic risk, and discuss the technical and regulatory bases of the licensee's proposed approach for this application, and the unique technical aspects associated with using these approaches in the licensee's categorization process. The NRC staff will review the internal events (includes internal floods) PRA, fire PRA, and conformance to NEI 00-04, as endorsed by RG 1.201 for implementation of 10 CFR 50.69.

The areas of focus for the regulatory audit are the information contained in the licensee's submittal, the enclosed audit information needs, and all associated and relevant supporting documentations including methodology, process information, calculations, etc.

4.0 INFORMATION AND OTHER MATERIAL NECESSARY FOR THE REGULATORY AUDIT

The following documentation should be available to the audit team:

1. Reports of peer reviews (full-scope and focused-scope), self-assessments, and Facts & Observations (F&Os) closure reviews for the internal events, internal flooding, and fire PRAs cited in LaSalle's LAR dated January 31, 2020;

2. Uncertainty notebooks for the LaSalle internal events, internal flooding, and fire PRAs related to PRA model assumptions and sources of uncertainty;

3. Documentation on evaluation of the generic and plant-specific uncertainties with respect to the LaSalle LAR dated January 31, 2020;

4. PRA notebooks for the modeling of FLEX equipment and FLEX human error probabilities, if credited in the PRA;
5. Results of the fire PRA and resolution of F&Os;

6. Available 10 CFR 50.69 SSC categorization program procedures (e.g., categorization review and adjustment process, decision criteria for Independent Decision-Making Panel (IDP)); and

7. Any other supporting documentation that the licensee may determine is responsive to the NRC staff's above information requests;

5.0 AUDIT TEAM

The members of the audit team are anticipated to be:

Adrienne Brown, Reliability and Risk Analyst, PRA, NRC
Todd Hilsmeier, Reliability and Risk Analyst, PRA, NRC
Shilp Vasavada, Reliability and Risk Analyst, PRA, NRC
De Wu, Reliability and Risk Analyst, PRA, NRC
Stacey Rosenberg, Branch Chief, PRA Licensing Branch C, NRC
Robert Pascarelli, Branch Chief, PRA Licensing Branch A, NRC
Robert Vettori, Fire Protection Engineer, NRC
Bhalchandra Vaidya, Project Manager, NRC
Garill Coles, Principal Engineer, Pacific Northwest National Laboratory (NRC Contractor)
Mark Wilk, Pacific Northwest National Laboratory (NRC Contractor)
John Bozga, Region III
John Honcharik NRC
G. Bedi, NRC
Carte Norbert, NRC
Andrea Russell, NRC
Victor Cusumano, Branch Chief, Technical Specifications Branch, NRC
Joseph Ashcraft, NRC
Matharu Gurcharan, NRC
Ed Kleeh, NRC

6.0 LOGISTICS

The audit will be conducted from June 15, 2020, to June 19, 2020, remotely, between 8:30 a.m. and 4:00 p.m. each day. An entrance briefing will be held at the beginning of the first part of the audit, and an exit briefing will be held at the end of the second part of the audit. A detailed agenda for the combined audit is included in the enclosure provided for the Technical Specification Task Force (TSTF)-505 audit (ADAMS Accession ML20160A164). The NRC project manager will coordinate any changes to the audit schedule and logistics with the licensee.

7.0 SPECIAL REQUESTS

The NRC staff would like access to the documents listed above in Section IV through an online portal that allows the NRC staff and contractors to access documents via the internet. The following conditions associated with the online portal must be maintained throughout the duration that the NRC staff and contractors have access to the online portal:

- The online portal will be password-protected, and separate passwords will be assigned to the NRC staff and contractors who are participating in the audit.

- The online portal will be sufficiently secure to prevent the NRC staff and contractors from printing, saving, downloading, or collecting any information on the online portal.

- Conditions of use of the online portal will be displayed on the login screen and will require acknowledgement by each user.

- User name and password information should be provided directly to the NRC staff and contractors. The NRC project manager will provide Exelon the names and contact information of the NRC staff and contractors who will be participating in the audit. All other communications should be coordinated through the NRC project manager.

- Visitor access for the plant walkdown to audit aspects of the proposed alternate seismic approach. Alternately, videos and/or photos, to support a virtual walkdown, as a contingency.

- Access to licensee and licensee's contractor personnel knowledgeable in the proposed alternate seismic approach for categorization, plant design, operation and any supporting PRA(s) used to address the staff's audit questions.

8.0 DELIVERABLES

An audit summary, which may be public, will be prepared within 90 days of the completion of the audit. If the NRC staff identifies information during the audit that is needed to support its regulatory decision, the staff will issue RAIs to the licensee after the audit.

9.0 REFERENCES

1. Exelon, 2020, "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors'," letter to NRC from Dwi Murray, Exelon Generation Company, LLC, January 31, 2020, ADAMS Accession No. ML20031E699

AUDIT QUESTIONS

LICENSE AMENDMENT REQUEST TO ADOPT 10 CFR 50.69, RISK-INFORMED

CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS

FOR NUCLEAR POWER REACTORS,

EXELON GENERATION COMPANY, LLC

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

Background:

By letter dated January 31, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20031E699), Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) for the LaSalle County Station (LaSalle or LSCS) to adopt 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 U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the LAR and determined that the following information is needed in order to complete the review.

Section 50.69(c)(i) of 10 CFR requires that a licensee's probabilistic risk assessment (PRA) 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 or set of acceptance criteria that is endorsed by the NRC. Section 50.69(b)(2)(iii) of 10 CFR requires that the results of the peer review process conducted to meet 10 CFR 50.69 (c)(1)(i) criteria be submitted as part of the application.

Regulatory Guide (RG) 1.200, Revision 2 (ADAMS Accession No. ML090410014), "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," provides guidance for addressing PRA acceptability. RG 1.200, Revision 2, describes a peer review process using the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard ASME/ANS-RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," as one acceptable approach for determining the technical acceptability of the PRA. The primary results of a peer review are the Facts and Observations (F&Os) recorded by the peer review team and the subsequent resolution of these F&Os. A process to close finding-level F&Os is documented in Appendix X to the Nuclear Energy Institute (NEI) guidance documents NEI 05-04, NEI 07-12, and NEI 12- 13, titled "NEI 05-04/07-12/12-[13] Appendix X: Close-out of Facts and Observations (F&Os)" (ADAMS Package Accession No. ML17086A431), which was accepted by the NRC in a letter dated May 3, 2017 (ADAMS Accession No. ML 17079A427).

APLA Question 01 – Internal Events PRA Self-Assessment Findings

Defer to Technical Specification Task Force (TSTF)-505, Question 01.

APLA Question 02 – Open Fire PRA Facts and Observations (F&O)

LAR Attachment 3 presents the dispositions for three F&Os that remain open after the Independent Assessment (IA) performed for closure of F&Os; two that remain open (i.e., 1-19, 4-17) and one partially resolved (i.e., 6-11). For F&O 1-19, the finding has been resolved but the resolution has not yet been reviewed by the IA team. For F&Os 4-17 and 6-11, the licensee states the items will be resolved prior to implementation of the provisions of 10 CFR 50.69 (hereafter, "10 CFR 50.69"). In light of these observations, please provide the following:

a) Regarding F&O 1-19, the LAR disposition acknowledges that the update was not reviewed by the IA team, and does not discuss the results of the systemic review.

   i. Describe the results from the systemic review of the circuit evaluation package notes and assumptions, and explain what fire PRA modelling adjustments have been determined to be needed.

   ii. Describe the modelling updates that have been incorporated into the PRAs demonstrating that the identified modelling concerns are addressed.

   iii. Alternatively, propose a mechanism that ensures the results of the systemic review and the updates to the fire PRA are reviewed by the IA team and the F&O is closed prior to implementation of 10 CFR 50.69 (e.g., include as an implementation item in the LAR associated with the proposed license condition).

b) Regarding F&O 4-17, the LAR disposition states that "[t]his item will be resolved prior to 50.69 implementation." Furthermore, the licensee states that the impact of this issue is "judged to be minimal." However, it is not clear to NRC staff the impact of this issue on the categorization of systems, structures, and components (SSCs). Therefore, address the following:

   i. Provide justification (e.g., description and results of a sensitivity study) that any needed adjustments to the fire PRA identified from review of the plant-specific data on the fire suppression and detection systems does not adversely impact SSC categorization.

   ii. Alternatively, propose a mechanism that ensures the review of plant-specific data for fire suppression and detection systems is performed and any update needed to the fire PRA is completed prior to implementation of 10 CFR 50.69 (e.g., include as an implementation item in the LAR associated with the proposed license condition).

c) Regarding F&O 6-11, the LAR disposition discusses that a review will be performed to verify consistency with NEI 00-01, Revision 3, prior to implementation of 10 CFR 50.69. However, no commitment to complete an implementation item for this F&O is made in the LAR. Also, the NRC staff notes that in the event the review cannot verify the circuit analysis was performed in accordance with the requirements of

NEI-00-01, Revision 3, then adjustments to the fire PRA model may be needed.

i. Provide sufficient justification to support the conclusion that any modelling updates determined to be needed for the fire PRA based on from review of the circuit analysis to the requirements of NEI-00-01, Revision 3, does not adversely impact the SSC categorization results.

ii. Alternatively, propose a mechanism that ensures the review of the circuit analysis to the requirements of NEI-00-01 is performed and any needed update to the fire PRA is completed prior to implementation of 10 CFR 50.69 (e.g., include as an implementation item in the LAR associated with the proposed license condition).

APLA QUESTION 03 – Process: PRA Model Uncertainty Analysis

Defer to TSTF-505, Question 09.

APLA QUESTION 04 – Treatment: PRA Model Uncertainty Analysis

The guidance in NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (ADAMS Accession No. ML052910035), specifies sensitivity studies to be conducted for each PRA model to address uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask the importance of components. NEI 00-04 states that additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered.

The dispositions provided in LAR Attachment 6 for some of the key assumptions and key sources of uncertainty appear to potentially impact the SSC categorization results. In light of these observations, provide the following information:

a) The impact assessment for the item pertaining to the digital feedwater control failure probabilities implies that the failure probabilities are from a vendor. The NRC staff notes that the general requirement for PRA data is to obtain data from recognized sources (e.g., NUREG/CR-6928), adequate plant-specific data, or expert judgement. The disposition states that a sensitivity study was performed to address the modeling uncertainties related to the digital feedwater control system and concluded that there was only a "small impact" to the PRA results. However, the NRC staff notes the impact on the SSC categorizations was not provided. The staff also notes that small changes in overall risk can change an SSC categorization from low-safety-significant (LSS) to high-safety-significant (HSS).

   i. For the sensitivity study performed for the digital feedwater control system, provide a justification that the uncertainty associated with modeling the system does not impact the SSC categorization results.

   ii. If the uncertainties associated with the modeling of the digital feedwater control system do impact the SSC categorization results, propose a mechanism to ensure that this PRA assumption/source of uncertainty will be appropriately addressed during the implementation of 10 CFR 50.69 (e.g., include as an implementation item in the LAR associated with the proposed license condition).

APLA Question 05 - Dispositions of Key Sources of Uncertainty

Defer to TSTF-505, Question 10, for the following:

- Cable Selection
- Vapor Suppression Capability
- Hardened Containment Vent
- Target Set Identification
- Emergency Core Cooling System (ECCS) Containment Venting

APLA Question 06 – Overlap of Functions and Components

Section 7.1 of NEI 00-04 states, "[d]ue to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC, or part thereof, should be assigned the highest risk significance for any function that the SSC or part thereof supports." Section 4 of NEI 00-04 states that a candidate LSS SSC that supports an interfacing system should remain uncategorized until all interfacing systems are categorized. The LAR does not discuss consideration or implementation of the guidance in Section 7.1 of NEI 00-04.

Explain how the categorization process will be implemented to ensure that the cited guidance in NEI 00-04 will be followed and that any functions/SSCs that serve as an interface between two or more systems will not be categorized until the categorization for all of the systems that they support is completed and that SSCs that support multiple functions will be assigned the highest risk significance for any of the functions they support.

APLA Question 07 – Masking of Risk Insights due to Conservative Modeling Choices

Section 3.2.7 of the LAR states, "[i]f the LSCS PRA model used a non-conservative treatment, or methods that are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine its impact on this application." The NRC staff notes that conservative modeling choices can potentially mask the importance of other SSCs (i.e., artificially lower the risk importance values of other SSCs below the safety significance threshold criteria). In light of these observations, provide the following information:

a) Discuss how the potential for masking due to conservative modeling choices will be addressed in the categorization process.

b) LAR Attachment 6 identifies a modeling conservatism where SSCs for which cable routing is unknown are assumed to fail in the fire PRA. The LAR cites a sensitivity study that assumes none of these SSCs fail due to a fire and shows a "moderate impact" on fire core damage frequency (CDF) and large early release frequency (LERF). However, this result does not address whether categorization of SSCs is impacted by this modeling assumption.

   i. Provide justification to confirm that the conservatism associated with not modeling SSCs for which cable routing is unknown has no impact on 10 CFR 50.69 categorization results.

   ii. If the modeling conservatism addressed in part (i) above cannot be justified to have no impact on the 10 CFR 50.69 categorization results, then propose a mechanism that ensures a sensitivity study is performed during 10 CFR 50.69 categorization that specifically addresses the uncertainty associated with SSCs for which cable routing is unknown (e.g., include as an implementation item in the LAR associated with the proposed license condition).

APLA Question 08 – Addition of FLEX to the PRA Model

The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC staff's position concerning incorporating mitigating strategies (FLEX) into a PRA in support of risk-informed decision making in accordance with the guidance in RG 1.200, Revision 2 (ADAMS Accession No. ML090410014).

To complete the NRC staff's review of the FLEX strategies modeled in the PRA, the NRC staff requests the following information for the internal events and internal flooding PRAs, and fire PRA, as appropriate.

a) Clarify whether FLEX equipment and associated operator actions are credited in the PRAs used to support this application, identifying the specific PRA(s) that include such credit. If FLEX is not credited in the PRAs, then no response to parts (b) and (c) of this question is requested. If FLEX is credited in the PRAs and this credit is not expected to impact the PRA results used in the categorization process, then provide sufficient justification to confirm this conclusion, and no response to parts (b) and (c) of this question is requested.

b) If the FLEX equipment or operator actions have been credited, and their inclusion is expected to impact the PRA results used in the categorization process, provide the following information separately for the internal events PRA (includes internal floods) and fire PRA, as appropriate:

   i. A discussion detailing the extent of incorporation, i.e. summarize the supplemental equipment and compensatory actions that have been quantitatively credited for each of the PRA models used to support this application.

   ii. Discuss the data and failure probabilities used to support the FLEX modeling and provide the rationale for using the chosen data. Include discussion on whether the uncertainties associated with the parameter values are in accordance with the applicable supporting requirements (SRs) in the ASME/ANS PRA Standard, as endorsed by RG 1.200, Revision 2.

   iii. Discuss the methodology used to assess human error probabilities for the FLEX operator actions. The discussion should include:

      1. A summary of how the impact of the plant-specific human error probabilities and associated scenario-specific performance shaping factors listed in (a)-(j) of SR HR-G3 of the ASME/ANS RA-Sa-2009 PRA Standard were evaluated.

      2. Whether maintenance and testing procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the

equipment unavailable during an event, and whether the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of the ASME/ANS RA-Sa-2009 PRA Standard.

3. For licensee's procedures governing the initiation or entry into mitigating strategies, identify specific areas which could be ambiguous, vague, or not explicit. Provide a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.

c) The ASME/ANS RA-Sa-2009 PRA Standard defines PRA upgrade as the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of ASME/ANS RA-Sa-2009 states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of this standard.

   i. Provide an evaluation of the model changes associated with incorporating non-safety related SSCs that were included following the FLEX mitigation strategies (permanently installed and/or portable), which demonstrates that none of the following criteria is satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences,

   OR

   ii. Propose a mechanism to ensure that a focused-scope peer review is performed on the model changes associated with incorporating mitigating strategies, and associated F&Os are resolved to Capability Category II prior to implementation of the 10 CFR 50.69 categorization process.

APLA Question 09 – Fire Hazards for LSCS 10 CFR 50.69

Section 3.2.2, "Fire Hazards," of the LAR states in part, "[t[he internal fire PRA model was developed consistent with NUREG/CR-6850 and only utilizes methods previously accepted by the NRC. The licensee's risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for LSCS." Furthermore, in Section 3.3 of the LAR, the licensee specifies that a full-scope fire PRA model peer review was performed in December 2015.

There have been changes to the fire PRA methodology since the last full-scope peer review of the LSCS. The integration of NRC-accepted fire PRA methods and studies described below that are relevant to this submittal could potentially impact the 10 CFR 50.69 risk categorization results and/or risk metrics for total CDF and total LERF in LAR Attachment 2:

- NUREG-2178, Volume 1, "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE)," dated April 2016 (ADAMS Accession No. ML16110A140).

- NUREG-2180, “Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE),” dated December 2016 (ADAMS Accession No. ML16343A058).

Section 2.5.5 of RG 1.174 provides guidance that indicates additional analysis is necessary to ensure that contributions from the above influences would not change the conclusions of the LAR.

For each of the above NRC-accepted fire PRA methods and studies, the NRC staff requests the licensee address one of the following:

a) Discuss how the fire PRA method/study had been incorporated into the LSCS and, as applicable, summarize the changes made to the fire PRA model. Indicate whether this change was PRA maintenance or a PRA upgrade as defined in ASME/ANS RA-Sa-2009, Section 1-5.4, as qualified by RG 1.200, Revision 2, along with a justification for the determination. If this change constitutes a PRA upgrade, discuss the focused-scope (or full-scope) peer review(s) that has been performed to evaluate the change, and provide any open F&Os and associated dispositions from this peer review(s) in accordance with RG 1.200, Revision 2.

OR

c) If the fire PRA method/study has not been incorporated into the LSCS fire PRA, provide a detailed justification for why the integration of the fire PRA method/study would not change the conclusions of the LAR, and subsequently not change the categorization process results. As part of this justification, identify any fire PRA methodologies used in the LSCS fire PRA that are no longer accepted by the NRC staff (e.g., guidance provided in frequently asked question (FAQ) 08-0046, “Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046 Incipient Fire Detection Systems” (ADAMS Accession No. ML093220426), has been retired by letter dated July 1, 2016 (ADAMS Accession No. ML16167A444). Provide technical justification for its use and evaluate the significance of its use on the risk metrics for this application provided in Attachment 2 of the LAR.

OR

c) Propose a mechanism that ensures the fire PRA method/study (or other NRC acceptable method) will be integrated into the LSCS fire PRA prior to implementation of the 10 CFR 50.69 categorization process. If this update is determined to be a PRA model upgrade per the ASME/ANS PRA standard, include in this mechanism a process for conducting a focused-scope peer review, and ensure any findings are closed by using an approved NRC process.

<u>Question 10 – Implementation Items</u>

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires that a licensee’s application contain a description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown are adequate for the categorization of SSCs. If the responses to any requests for additional information (RAIs) require any follow-up actions prior to implementation of the 10 CFR 50.69 categorization process, provide a list of those actions and any PRA modeling changes including any items that will not be completed prior to issuing the amendment but must be completed prior to implementing the 10 CFR 50.69 categorization process.

Propose a mechanism that ensures these activities and changes will be completed and appropriately reviewed and any issues resolved prior to implementing the 10 CFR 50.69 categorization process (for example, a license condition that includes all applicable implementation items and a statement that they will be completed prior to implementation of the 10 CFR 50.69 categorization process).

APLA Question 11 – Key Principle 5: Maintenance Rule and Monitoring

Defer to TSTF-505, Question 12

PROPOSED ALTERNATE SEISMIC APPROACH

Background and Regulatory Bases

By letter dated January 31, 2020 (ADAMS Accession No. ML20031E699), Exelon, the licensee, submitted a license amendment request (LAR) regarding the LaSalle County Station, Units 1 and 2 (LaSalle or LSCS). The proposed amendment would modify the LaSalle licensing basis to allow for the implementation of the provisions of 10 CFR, Section 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors."

Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006 (ADAMS Accession No. ML061090627), endorses, with clarifications and qualifications, the Nuclear Energy Institute (NEI) guidance document NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (ADAMS Accession No. ML052910035), as one acceptable method for use in complying with the requirements in 10 CFR 50.69. The NEI 00-04 guidance describes in detail a process for determining the safety significance of SSCs and for categorizing them into the four risk-informed safety class (RISC) categories defined in 10 CFR 50.69. This categorization process uses an integrated decision-making process, incorporating both risk and traditional engineering insights. The NEI 00-04 guidance allows licensees to implement different approaches, depending on the scope of their probabilistic risk assessment (PRA). The proposed amendment includes an exception to the U.S. Nuclear Regulatory Commission (NRC) endorsed categorization process in NEI 00-04 to apply an alternative seismic approach specified in Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," dated July 2018[1].

The regulation 10 CFR 50.69(b)(2)(ii) provides the requirements to describe the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are

[1] All references to the EPRI report in this document refer to EPRI 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10CFR50.69 Risk-Informed Categorization," July 2018. The same report is cited as Reference 4 in the enclosure to the licensee's November 28, 2018 submittal and is publicly available free of cost online at https://www.epri.com/#/pages/product/000000003002012988/?lang=en-US.

adequate for the categorization of SSCs. The regulation 50.69(c)(ii) requires the categorization process to determine 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. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe accidents. 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 operational experience. Finally, the regulation 10 CFR 50.69(e) requires periodic updates to the licensee's PRA and SSC categorization.

RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” March 2009 (ADAMS Accession No. ML090410014), describes an approach for determining whether the base PRA, in total or the parts that are used to support an application, is acceptable for use in regulatory decision making for light water-reactors. RG 1.200 endorses, with staff clarifications and qualifications, the 2009 version of the American Society of Mechanical Engineers (ASME) /American Nuclear Society (ANS) PRA Standard (ASME/ANS RA-Sa-2009).

A. Proposed Alternate Seismic Approach

Implementation of Seismic Assessment

APLC Question 1: Figure 3-2, “Seismic Correlated Failure Assessment for Tier 2 Plants,” in the enclosure to the LAR (reproduced from Figure 2-3 in the EPRI report) depicts decision-making for the proposed alternate seismic approach to account for the insights from the EPRI case studies based on a site-specific plant walkdown. A demonstration of the inputs and basis for important decisions depicted in Figure 3-2 in the enclosure to the LAR is necessary to understand the implementation of the walkdown assessment for the proposed alternate seismic approach. Demonstrate the implementation of Figure 3-2 in the enclosure to the LAR, especially Steps 3b, 5 (a through c), and 6, from that figure, using a site-specific walkdown and example SSCs. It is recognized that the walkdown demonstration will not be part of a formal categorization of the example SSCs.

APLC Question 2: Demonstrate the sensitivity study in the proposed alternate seismic approach using a full power internal events (FPIE) PRA to compare against the categorization results from a seismic PRA (SPRA) for example SSCs. It is preferred that the demonstration is performed for 3 - 4 SSCs that would be assessed through the sensitivity in the proposed alternate seismic approach. It is recognized that the demonstration of the sensitivity will not be part of a formal categorization of the example SSCs. The demonstration should:

a. Show how the surrogate events will be included in the FPIE PRA to reflect all the impacts of the seismic-specific failure modes of the example SSCs and the modeling approach used for SPRAs where the seismically induced failure is modeled under the so-called ‘top gate’ for the SSC and not under a random failure ‘sub-gate’.

b. Support the responses to information needs identified in item 6e, 6g, 6h, and 7.

Seismic Hazard Change and Performance Monitoring

APLC Question 3: Section 3.2.3 of the enclosure to the LAR cites the "Moderate Seismic Hazard / Moderate Seismic Margin" (so-called Tier 2 in the EPRI report) criterion from the EPRI report as being applicable to the licensee. The licensee's criterion relies on the criteria for Tiers 1 and 3 in the EPRI report. According to the EPRI report, Tier 3 criterion is based on prior action by the NRC as part of the 10 CFR 50.54(f) request for information date March 12, 2012 (ADAMS Accession No. ML12053A340).

> Section 3.2.3 of the enclosure to the LAR further states that "In the unlikely event that the LSCS seismic hazard changes from medium risk (i.e., Tier 2) at some future time, EGC will follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e)." In addition, Section 3.5 of the enclosure to the LAR also states that implementation of the EGC design control and corrective action programs will ensure the inputs for the qualitative determinations for seismic continue to remain valid to maintain compliance with the requirements of 10 CFR 50.69(e) to performance monitoring as it pertains to the proposed alternate seismic approach.
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> NRC's decision-making for the 10 CFR 50.54(f) letter was in a different context (i.e., requesting information under 10 CFR 50.54(f)) and considered information available to the staff at the time in that context. The proposed amendment would change the licensee's operating license for the duration of the license. Therefore, it is unclear to the staff how the Tier 3 criterion will provide a clear and traceable boundary that the licensee can be unambiguously apply if new information (e.g., hazard assessment) is available. Further, based on the information presented in Section 3.2.3 and 3.5 of the enclosure to the LAR with regard to compliance with 10 CFR 50.69(e), it is unclear to the staff whether the GMRS [general mobile radio service] to SSE [safe shutdown earthquake] ratio based on the licensee's re-evaluated hazard is being proposed as the clear and traceable boundary beyond which "EGC will follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e)."
>
> Clarify whether a GMRS to SSE ratio based on the licensee's re-evaluated hazard is being proposed as the clear and traceable boundary for compliance of the proposed alternate seismic approach with 10 CFR 50.69(e). If yes, clarify what GMRS to SSE ratio (e.g., in the 1 – 10 hertz frequency range) is used as the boundary with the corresponding bases. If not, provide a clear and traceable boundary with the corresponding bases to justify ensuring compliance with 10 CFR 50.69(e).

Application of the Proposed Alternate Seismic Approach

APLC Question 4: A review of the results of the case studies for Plants A, C, and D in Tables 3-5, 3-9, and 3-11, respectively demonstrate that the fragility groups identified as an HSS uniquely from SPRAs include relay chatter induced failures and failures of passive components, such as tanks and heat exchangers, in addition to correlated and seismic interactions related failures. The NRC staff notes that a fragility group includes multiple components. Section 3.6, "Summary of Sensitivity Study Insights," of the EPRI report also discusses seismically-induced failures not limited to correlated and interaction failures.

Section 3.2.3 of the enclosure to the LAR as well as Section 4 of the EPRI report states that for the licensee's plant there may be a limited number of unique seismic insights appropriate for consideration in determining HSS SSCs and those SSCs would be typically associated with seismically correlated failures. The proposed approach includes a seismic walkdown of certain SSCs to look for correlation and spatial interaction concerns. Section 3.2.3 of the enclosure to the LAR also lists various seismic-specific failure modes that are included in the licensee's review during categorization. Given the insights from the EPRI report and the expected seismic risk for the licensee, the basis for limiting additional evaluation in the proposed approach to only correlated and interaction issues is unclear.

Justify how unique seismic-specific failures other than correlated and interaction failures (i.e., those identified in the case studies and in Section 3.2.3 of the enclosure to the LAR) have been appropriately evaluated in the proposed approach without walkdowns and sensitivity calculations using the internal events PRA. Alternately, discuss how the proposed approach will evaluate unique seismic-specific failures identified in Section 3.2.3 of the enclosure to the LAR.

APLC Question 5: Section 3.2.3 of the enclosure to the LAR includes a summary and flowchart (Figure 3-2 in the LAR) of the process to be followed for additional evaluations under the proposed approach. Additional information on the process is provided in Section 2.3.1 of the EPRI report. The discussion of the "screening" in item 3 of Section 2.3.1 of the EPRI report states that "[t]hese screening decisions…likely will involve cost/benefit decisions in terms of how best to complete the sensitivity study." Information on what "cost/benefit decisions" would be included, how such considerations would alter the implementation of the proposed approach, and the justification for such decisions as well as the resulting changes is unavailable either in the LAR or the EPRI report.

Item 3a of Section 2.3.1 of the EPRI report mentions inherently rugged components, cites Appendix H of Nuclear Energy Institute (NEI) 12-06, Revision 4 (ADAMS Accession No. ML16354B421), and lists certain SSCs as being inherently rugged. Appendix H of NEI 12-06, Revision 4, refers to other documents. It is unclear whether the list under item 3a of Section 2.3.1 of the EPRI report encompasses the inherently rugged components that are included in other documents referenced in Appendix H of NEI 12-06, Revision 4. The basis of consideration of such components as inherently rugged is also not provided.

Item 3b of Section 2.3.1 of the EPRI report states that if the SSCs under consideration is not used in safety functions that support mitigation of core damage or containment performance, it can be screened. The EPRI report also provides an example of screening based on this criterion. However, item 4 of Section 2.3.1 includes additional evaluation of SSCs where correlation and interaction considerations can impact the function of the SSCs. It is unclear whether the impact of the failure of SSCs that are screened out per item 3b of Section 2.3.1 of the EPRI report on SSCs with functions supporting mitigation is considered. In case of the example provided in the EPRI report, a chiller system that is not part of the safety function of the tank, the consideration of seismic-specific failure modes of the chiller system (e.g., correlated or interaction failure of the heat exchangers and or piping resulting in a drain path for the tank) can change the screening decision. In addition, Section 4 of NEI 00-04 includes discussion of categorization of interfacing systems. It is unclear how the screening criterion in item 3b will interface with the guidance in Section 4 of NEI 00-04 for any interfacing systems.

9.1 Discuss the justification for, intent of, and implementation of the consideration of the "cost/benefit decisions" in the screening process for the proposed alternate seismic approach. Include an explanation of how "cost/benefit decisions" would change the licensee's proposed approach and justify such changes.

9.2 Clarify whether the list under item 3a of Section 2.3.1 of the EPRI report encompasses the inherently rugged SSCs that that are included in documents referenced in Appendix H of NEI 12-06, Revision 4 or identify changes to the list to encompasses such SSCs. Include the basis of consideration of SSCs in the list as inherently rugged.

9.3 Clarify whether screening out items per step 3b of the EPRI report (i.e., SSCs that are not used in safety functions that support mitigation of core damage or containment performance) will preclude consideration of the impact of the failure of those SSCs on SSCs with functions supporting mitigation. If the impact of items that are screened out per step 3b is not considered, provide a basis for exclusion of such consideration in the categorization process. Include an explanation of the interaction of the screening in item 3b of the EPRI report with the guidance in Section 4 of NEI 00-04.

APLC Question 6: Section 2.3.1 of the EPRI report provides the process for additional evaluation in the licensee's proposed alternate seismic approach. Item 8 of the process states that the loss-of-offsite power (LOOP) and small loss-of-coolant accident (SLOCA) initiators in the FPIE PRA will be used, in conjunction with surrogate events, to determine the impact of seismic-specific failures of SSCs following the walkdown. The basis for limiting the evaluation to only the LOOP and SLOCA initiators is not provided. It appears that the "success path" approach for seismic margins analysis (SMA) used in the individual plant examination for external events (IPEEE) is used to identify the initiators. However, the proposed approach is based on insights from seismic PRAs and not SMAs. SPRAs include several initiators in addition to LOOP and SLOCA.

The initiating frequencies for the LOOP and SLOCA initiators in item 8 of Section 2.3.1 of the EPRI report are provided as 1.0 and 1E-2 per year, respectively. The assumptions used for the initiating frequency of SLOCA in the proposed approach are unclear and the basis for its generic applicability is not provided. The insights from the Plant A, C, and D case studies that support the licensee's proposed approach do not appear to inform the selection of the initiators and initiating frequencies. In addition, item 8a of Section 2.3.1 of the EPRI report states "[o]ther appropriately justified values for small LOCA frequency may be used" but does not provide any information about the basis for such values. Item 8c indicates that the sequence to be included in the evaluation is the SLOCA-LOOP sequence (i.e., SLOCA with conditional LOOP) rather than the SLOCA sequence.

The discussion in item 8 of Section 2.3.1 of the EPRI report does not provide any information about including changes to key internal events modeling assumptions due to seismic-specific impacts. Examples of such impacts include, but are not limited to, non-recovery of offsite power and non-recovery of DC [direct current] power. Such impacts contribute to the results from an SPRA and therefore, are expected to be contributing factors to the insights from the Plants A, C, and D case studies in the EPRI report. It is unclear why such impacts are not being considered in the sensitivity study for the licensee's proposed approach.

a. Clarify whether the SLOCA-LOOP (i.e., SLOCA conditional with LOOP) or only the SLOCA sequence is proposed to be included for the quantitative evaluation in the proposed alternate seismic approach.

b. Discuss whether and how other seismically induced failures (e.g., SLOCA), modeling assumptions (e.g., non-recovery of LOOP), and all impacts of the seismically induced failure of the SSC being categorized are included in the FPIE [full power internal events] SLOCA and LOOP event trees. If the sensitivity study excludes such failures and assumptions, justify their exclusion from the proposed approach.

c. Justify the selected initiating frequencies for the SLOCA initiator using the insights from seismic PRAs, including those used for Plant A, C, and D case studies to develop the proposed approach.

d. Explain the basis for use of "[o]ther appropriately justified values for small LOCA frequency initiating frequency" SLOCA or SLOCA-LOOP sequences and how the proposed categorization process will ensure that the use of the "other" value is accepted by the NRC staff for the proposed alternate seismic approach (i.e., how the response to item (c) can be provided for such "other" value(s)).

The proposed approach utilizes walkdowns to identify correlated failure and interaction concerns and then use a sensitivity to determine their impact on the categorization of SSCs. The sensitivity calculation is a key step in the proposed alternate seismic approach and the resulting input to the categorization of SSCs. The proposed approach uses a failure probability for the surrogate events of 1E-4. Section 2.3.1 of the EPRI report states that the value is based a "typical" total seismic hazard exceedance frequency above which SPRAs would typically model loss of offsite power and for which correlated failures may be likely. The licensee's proposed approach also states that the failure probability can be "justified based on the hazard." Further, it appears that the same surrogate event failure probability is applied regardless of the SSC and seismic-specific failure mode that the event is supposed to represent.

e. Demonstrate that the proposed sensitivity study using a single failure probability for the surrogate event, the two initiating events, the corresponding initiating event frequencies, and any changes due to response to items (a) through (c) above, results in categorization input that is equivalent, or conservative compared to corresponding results from a SPRA.

f. Explain how failure probability/ies other than a fixed value, determined based on the response to item (d), will be developed and "justified based on the hazard" for implementation in the proposed approach and how the use of the "other" value is accepted by the NRC staff for the proposed alternate seismic approach (i.e., how the response to item (e) can be provided for such "other" value(s)).

NEI 00-04 guidance includes consideration of SSC importance measures from the Level 1 portion of a PRA (i.e., only CDF) as well as the LERF portion. The insights from the case studies in the EPRI report are also based on both CDF and LERF importance measures and the resulting categorization. The discussion of the proposed alternate seismic approach in Section 3.2.3 of the enclosure to the LAR as well as Section 2.3.1 of the EPRI report is unclear on whether the inclusion of the surrogate event and the resulting categorization input will be based on only the Level 1 portion (i.e., only CDF) of the licensee's full power internal events (FPIE) PRA or will include the LERF portion. For certain SSCs identified in the case studies in the EPRI report, surrogate events appear to be necessary for both the CDF and LERF portions of the FPIE PRA.

g. Explain whether the proposed sensitivity study will include both the CDF and LERF portions of the licensee's FPIE model, as necessary, and if the importance measures from both portions will be used to provide categorization input. If applicable, justify the exclusion of the LERF portion given the insights from the case studies in the EPRI report and the guidance in NEI 00-04 for categorization of SSCs.

The proposed alternate seismic approach includes addition of surrogates to the FPIE PRA under the appropriate areas in the logic model (Step 7 of Section 2.3.1 of the EPRI report). The approach likens the surrogates to a common-cause failure mode. The FPIE PRA usually includes common-cause failure events under a particular random failure mode for an SSC (e.g., common-cause failure of pumps to run). However, the seismic correlated or interaction failures fail the SSC independent of a random failure mode of the SSC (i.e., the seismic failure is under the so-called top gate for the SSC). It is unclear where the surrogates will be included in the FPIE PRA.

The proposed alternate seismic approach compares the results of the sensitivity for each surrogate event to the results to the F-V and RAW HSS criteria for common cause failure in the FPIE PRA from NEI 00-04 (that is, F-V > 0.005 or RAW > 20). The proposed approach and the use of sensitivity for seismic-specific failure modes is based on the insights from the case studies in the EPRI report. It appears to the NRC staff that the objective of the sensitivity is to obtain categorization inputs like those that would be obtained from a seismic PRA for seismic-specific failure modes. However, the categorization using the seismic PRAs in the case studies in the EPRI report are not based on the common cause failure but rather the individual SSC failure. As an example, Section 3.4.2.2 of the EPRI report, which discusses the identification of HSS SSCs from the Plant C seismic PRA, states that "[c]omponents are considered high safety significant (HSS) if the group F-V is greater than 0.005 or if the group RAW is greater than 2.0 for CDF or LERF." In addition, based on the modeling in the seismic PRA, which the proposed approach is attempting to reproduce, the seismic failure event is under the so-called 'top gate' for the SSC and not under a random failure 'sub-gate'. Therefore, it appears that using different thresholds from those used for the case studies, which provide the insights supporting the proposed approach, is (i) not supported by the basis for the proposed approach, and (ii) can result in categorization inputs different those that would be obtained from a seismic PRA for seismic-specific failure modes, which is contrary to the purpose of the proposed approach.

h. Justify the use of the criteria for common cause failure for determination of HSS SSCs from the proposed sensitivity study given that the categorization in the case studies providing the underlying insights for the proposed approach used different F-V and RAW criteria. The justification should include a demonstration of the proposed approach of using common cause failure (CCF) criteria for the surrogate event importance measure provides results equivalent to that from a seismic PRA using F-V and RAW criteria for a single SSC. Alternately, discuss any updated F-V and RAW criteria for use with the proposed sensitivity study and discuss their consistency with the case studies in the EPRI report as well as NEI 00-04.

APLC Question 7: Step 10 in Section 2.3.1 of the EPRI report states that if the importance measure criteria are met by the surrogate basic events, then the corresponding SSC should be considered HSS. However, it is unclear whether the comparison against the importance measure criteria and the consequent categorization as HSS will be performed if the results from either of the sensitivities (i.e., for LOOP and for SLOCA) show the criteria being met or both the sensitivities need to show that the criteria is met or some combination shows the criteria is met. Further, the basis for the use of any approach (individual, combined, etc.) for comparison against the criteria is not provided.

Explain how the results from the sensitivity will be used for comparison against the importance measure criteria in NEI 00-04 (i.e., individually, combined, etc.). The explanation should include a justification of how the proposed comparison approach is consistent with the guidance in NEI 00-04 and reflective of the insights from the case studies in the EPRI report.

APLC Question 8: Figure 2-3 in the EPRI report depicts the steps and decisions made as part of the licensee's proposed alternate seismic approach. Certain steps in the flowchart in Figure 2-3 are unclear in their intent and scope. The accompanying discussion does not include information to address the lack of clarity in Figure 2-3.

a. Step 3b indicates a decision related to the SSC function for mitigation of core damage or containment performance. It is unclear if the mitigation function being questioned here are hazard-specific. Clarify whether the question in Step 3b includes SSCs functions for mitigation of core damage or containment performance for seismically-induced design basis and severe accidents. Include justification if the functions exclude mitigation of seismically-induced design basis and severe accidents.

b. The discussion for Step 3b in Section 2.3.1 of the EPRI report includes an example of a chiller system which would be screened out from consideration in the proposed alternate seismic approach. However, neither Step 3b nor the example in the discussion in Section 2.3.1 of the EPRI report considers the indirect impact of seismically-induced failures of SSCs on mitigation of core damage or containment performance. The case study for Plant C demonstrates such an impact. Table 3-9 of the EPRI report shows failures of two SSCs, one of them a chiller, (S_CB-CHLR-NCSW-FLOOD and S_1FC-ACU-FLD) which results in flooding and consequently loss of mitigation. Step 3b does not appear to include such impacts and would therefore, potentially screen out such SSCs. Clarify whether the question in Step 3b includes the indirect impacts of seismically-induced failures of SSCs on mitigation of core damage or containment performance. Include justification if such impacts are excluded given the insights from the Plant C case study.

c. Step 5a through 5c in Figure 2-3 and the accompanying guidance in Appendices A and B of the EPRI report do not discuss the consideration and treatment of the indirect impacts of seismically-induced failures of SSCs on mitigation of core damage or containment performance discussed in item (b) above. If indirect impacts of seismically-induced failures of SSCs on mitigation of core damage or containment performance are included in Step 3b in response to item (b), explain how such SSCs will be considered in the walkdowns in Step 5a through 5c.

APLC Question 9: The steps in Section 2.3.1 of the EPRI report discuss the process for performing the sensitivity study. The surrogates used for the sensitivity study are intended to reflect the impact of seismic-specific failure modes identified in SPRAs. The steps in Section 2.3.1 of the EPRI report do not provide any guidance on whether the surrogates for SSCs that have already been categorized will be retained in the licensee's FPIE PRA model for subsequent sensitivity studies (for other SSCs).

Clarify whether surrogates that were incorporated in the licensee's FPIE PRA for the characterization of an SSC will be retained and included in the sensitivity studies for subsequent (and distinct) SSC categorizations. If the surrogates are not intended to be retained, justify such an approach given that importance of SSCs from SPRAs, which the approach is attempting to emulate, can depend of the seismic failures of preceding SSCs.

APLC Question 10: Appendix A of the EPRI report provides guidance on "identifying seismic correlated or seismic interaction scenarios" as part of the proposed alternate seismic approach. The guidance in that appendix includes statements indicating that the correlation decision is "a judgment of engineers experienced in both seismic capacity and seismic response fields." In addition, steps for determining correlation provided in Appendix A of the EPRI report include consideration of "similar seismic response", "similar failure modes and fundamental frequencies", "significantly different seismic responses", and "interaction sources not deemed credible based on their experience and training." Such determinations appear to require technical knowledge and experience in areas such as seismic response, design, capacity, walkdown, and plant response. The requisite technical knowledge and experience is specialized and expected to be uncommon, especially with the plant's staff. However, neither the LAR nor the EPRI report describes how the proposed approach ensures the qualifications of the personnel performing the walkdowns and using information related to failure modes, fundamental frequencies, plant response etc.

The walkdowns and result of the walkdowns appears to be an important element of the proposed alternate seismic approach. However, neither the LAR nor the EPRI report includes any information of how the walkdowns will be documentation and what the documentation will include. Such documentation appears to be necessary to support the conclusions of the walkdowns and for future regulatory processes, such as audits and inspections.

a. Provide the qualifications that personnel need to possess to perform the walkdowns and implementing the guidance in Appendix A of the EPRI report as part of the proposed alternate seismic approach. Discuss how the proposed approach ensures qualifications for performing the walkdowns and implementing the guidance. If the proposed approach does not explicitly consider such qualifications, justify not including special qualification considerations given that lack of specialized technical knowledge and experience indicated by the guidance in Appendix A of the EPRI report can decrease confidence in the results of the walkdowns.

b. Discuss how the approach ensures that the results of the walkdowns are communicated for incorporation in the proposed alternate seismic approach.

APLC Question 11: Appendix B of the EPRI report provides guidance on "capacity-based screening for high capacity SSCs." The guidance includes a screening value of the seismic core damage frequency and if a single SSC were to contribute that value or less, it could be screened from evaluation under the proposed alternate seismic approach. The proposed screening value appears to be an absolute value and therefore, not relative to the seismic risk for the licensee and the consideration of the importance measures thresholds used for categorization under 10 CFR 50.69. Depending on the seismic risk for the licensee, the proposed screening value can result in a non-trivial contribution relative to the seismic risk. Consequently, its use can result in screening of SSCs that would otherwise have to follow the other steps in Figure 3-2, "Seismic Correlated Failure Assessment for Tier 2 Plants," in the enclosure to the LAR (reproduced from Figure 2-3 in the EPRI report).

Appendix B of the EPRI report additionally provides guidance on the development of fragility values for SSCs to support the capacity-based screening. In addition to the state-of-practice approaches identified in Appendix B via relevant documents (e.g., representative values and conservative deterministic margins analysis [CDFM]), the guidance also discusses "more simplified and conservative approaches when justified by experienced engineers," "simplified approaches documented in ASCE 7," and "assessments made would have to be necessarily conservative." In addition, Step 11 in Section 2.3.1 of the EPRI report discusses refinement to the fragility based on direction from the Integrated Decision-making Panel (IDP). SPRAs, such as those included in the case studies supporting the proposed alternate seismic approach, use state-of-practice approaches and undergo an NRC endorsed peer-review process for the implementation of such approaches. Further, based on reviews of several SPRAs that use the state-of-practice approaches, the NRC staff also has confidence in those analytical methods and their implementation by personnel with specialized knowledge about their use. The EPRI report and the LAR do not appear to include constraints on the type of approaches that can be used to achieve any "refinement" and therefore, such approaches can include those that have not been appropriately vetted even among the practitioners. It is unclear how the NRC staff can determine the acceptability of approaches that are not state-of-practice (either simplified or refined) and for which, as part of the proposed alternate seismic approach, an implementation peer-review will not be conducted.

It is expected that the fragility calculations require specialized knowledge and experience to implement the approaches as well as any caveats for the approaches in the corresponding documents. The qualifications of the personnel performing the fragility development, even using state-of-practice approaches, has not been specified in either the LAR or the EPRI report. Neither the LAR nor the EPRI report includes any information on the documentation of the fragility analysis. Such documentation appears to be necessary to support the conclusions of the analysis and for future regulatory processes, such as audits and inspections.

a. Justify the independence of the screening value of the seismic core damage frequency proposed in Appendix B of the EPRI report from the licensee's seismic risk and the importance measures thresholds used for categorization under 10 CFR 50.69. Alternately, propose a screening approach relative to the licensee's seismic risk which considers the impact on the importance measures thresholds used for categorization under 10 CFR 50.69.

b. Identify the analytical approaches and methods that are consistent with the state-of-practice and will be used for fragility calculations in the proposed alternate seismic approach.

c. Discuss how the proposed approach will ensure NRC staff's review and acceptance of analytical approaches and methods for fragility calculations other than those identified in item (b) above prior to their use in the licensee's alternate seismic approach (e.g., through a LAR controlled by the license condition).

d. Discuss how the proposed approach will ensure that the qualifications of personnel that perform the fragility calculations, such as those using the methods identified in item (b), above are sufficient to support the development of the fragilities.

e. Discuss how the approach ensures that the fragility analyses performed for the proposed alternate seismic approach using the methods discussed in item (b) are documented to support the conclusions of the analyses and support future regulatory processes, such as audits and inspections.

APLC Question 12: Table 3-1 of the enclosure to the LAR provides a "categorization evaluation summary." According to the table, the IDP can "change HSS to LSS" for the "seismic" categorization step. Step 11 in Section 2.3.1 of the EPRI report that is incorporated by reference by the licensee (page 14 of the enclosure to the LAR includes Section 2.3.1 in the incorporation of the EPRI report in the LAR), states that the proposed approach (using the sensitivity study) is "pseudo-deterministic" and, therefore, the "seismic correlated group HSS designations should be treated similar to HSS designations using the IPEEE SMA [safeguards summary event list] SSEL and, in general, not be subjected to reconsideration by the Integrated Decision-Making Panel (IDP)." It appears that the LAR is inconsistent with the EPRI report which it is referencing and supposedly, following. In addition, the basis for the LAR's deviation from the EPRI report is also unclear.

Step 11 of the EPRI report states that "SSCs which are HSS solely due to surrogate events representing seismic induced interactions (such as block walls impacting equipment) may be downgraded to LSS by the IDP with appropriate justification". It appears that the preceding discussion on consideration of seismic correlated group HSS designations similar to IPEEE SMA SSEL and the subsequent possibility of downgrade to LSS are not only unclear but also contradictory.

a. Clarify whether the summary for seismic categorization in Table 3-1 is intended to be a deviation from the approach discussed in the EPRI report. If yes, provide the justification for the deviation based on the insights and guidance in the EPRI report for the proposed approach which is incorporated by reference in the LAR. If no, provide an updated version of Table 3-1 that clarifies the licensee's intent.

b. Explain the apparent contradiction between two statements in the same step (step 11) in Section 2.3.1 of the EPRI report and clarify what is being proposed in terms of changes to HSS designations arising from the sensitivity study in the proposed approach by the IDP.

c. Explain, with examples, what would be "appropriate justification" by the IDP to downgrade an HSS determination arising from the sensitivity study in the proposed approach noting that a walkdown would have been conducted, the flowchart in Figure 2-3 of the EPRI report would have been followed, and a sensitivity would be performed before the HSS determination is made.

Technical Acceptability of PRAs Used for Case Studies in the EPRI Report

APLC Question 13: The proposed alternate seismic approach is based on the insights from the EPRI report which were derived from case studies. Those case studies compare the HSS SSCs determined based on a SPRA against HSS SSCs determined from other PRAs used for categorization. Each of the cases studies included a FPIE PRA but only two of the four case studies used information from a Fire PRA. Sections 3.3 through 3.5 of the EPRI report provide general information about the peer reviews conducted for the PRAs used for in each of the four case studies. However, the level of information is insufficient to determine whether the PRAs used in the case studies supporting this application have been performed in a technically acceptable manner.

The NRC staff has previously requested and reviewed information to support its decision on the technical acceptability of the PRAs used in the case studies as well as details of the conduct of the case studies. This information is included in the supplements to the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, LAR for adoption of 10 CFR 50.69. The supplement to the 10 CFR 50.69 by Calvert Cliffs Nuclear Power Plant LAR dated July 1, 2019 (ADAMS Accession No. ML19183A012), further clarified the information related to the alternate seismic approach (see response to RAI 4); the supplement dated July 19, 2019 (ADAMS Accession No. ML19200A216), provided responses to support the technical acceptability of the PRAs used for the Plant A, C, and D case studies as well as technical adequacy of certain details of the conduct of the case studies; the supplement dated August 15, 2019 (ADAMS Accession No. ML19217A143) clarified a response in the July 19, 2019 supplement. The supplement dated July 19, 2019 included modifications to the content of the EPRI report.

Since the above-mentioned information was requested and reviewed by the NRC staff for Calvert Cliffs Nuclear Power Plant's LAR for adoption of 10 CFR 50.69, the staff is unable to use it for the licensee's docket unless it is incorporated in the licensee's LAR. The above-mentioned information is necessary for the staff to make its regulatory finding on the licensee's proposed alternate seismic approach and has not been provided by the licensee.

Provide the above-mentioned information to support the staff's regulatory finding on the alternate seismic approach by either incorporating the information by reference or responding to the RAIs in the identified supplements.

NON-SEISMIC EXTERNAL HAZARDS CONSIDERATION IN PROPOSED 10 CFR 50.69 PROGRAM

APLC Question 14: NEI 00-04 provides guidance on including external events in the categorization of each SSC to be categorized. Fire and seismic hazards are discussed in Section 5.2 and 5.3 of NEI 00-04, respectively. All other hazards are discussed in Section 5.4, "Assessment of Other External Hazards." Figure 5-6 in Section 5.4 illustrates the process that begins with the SSC selected for categorization and then proceeds through the flow chart for each external hazard.

Section 3.2.4 of the enclosure to the LAR discusses the consideration of non-seismic external hazards in the proposed categorization approach. Attachment 4 of the enclosure to the LAR provides the basis for the consideration of each identified non-seismic external hazard in the proposed categorization approach.

The discussion for turbine missiles states "the speed capability of these rotors is considerably higher than the maximum attainable speed of these turbine generator units. Consequently, the probability of missiles being generated is statistically insignificant." The turbine missile probability analysis evaluates the failure of turbine stop, control, and bypass valves and determines the inspections and frequency of those inspection so that the failure rate and probability of turbine missile damaging safety related equipment is below the threshold of 10^{-7}. It is unclear whether the turbine missile probability analysis is the basis for the screening or the speed capability of the rotors. It is also unclear whether turbine missile hazard will be subjected to the flowchart in Figure 5-6 to SSCs relevant to that hazard (e.g., turbine stop, control, and bypass valves).

The discussion for "extreme winds" states "a demonstrably conservative estimate of CDF associated with high wind hazard (other than wind generated missiles) is much less than 1E-6/yr" and "[i]n addition, based on the plant design for tornado missiles, considering a limited set of SSCs vulnerable to tornado missiles, a demonstrably conservative estimate of CDF associated with tornado missiles is less than 1E-6/yr." Additional information to support the above cited statements is unavailable in the LAR.

Section 5 of Enclosure 4 of Attachment 2 to the licensee's LAR for adoption of Technical Specification Task Force (TSTF)-505 (ADAMS Accession No. ML20035E577), includes information related to the impact of high winds on LaSalle. It is unclear if any or all the information from the licensee's TSTF-505 LAR is applicable to the 10 CFR 50.69 LAR to support the "demonstrably conservative estimates" identified above. In addition, Section 5 of Enclosure 4 of Attachment 2 to the licensee's LAR includes discussion of operator actions and related equipment which can mitigate certain potential impacts from tornado-generated missiles. However, the licensee's 10 CFR 50.69 LAR does not mention those actions and related equipment as being credited to support the consideration of high winds (aka extreme winds) in the proposed 10 CFR 50.69 categorization approach.

a. Identify the external hazards that will be evaluated according to the flow chart in Figure 5-6 of NEI 00-04.

b. Explain how any SSCs, including equipment required by applicable procedures, that are used to support the basis for screening high winds (aka extreme winds) and tornado-generated missiles in Attachment 4 of the enclosure to the 10 CFR 50.69 LAR will be considered in the proposed categorization approach and its consistency with the guidance in NEI 00-04 (e.g., Figure 5-6 in NEI 00-04 discusses SSCs that participate in a screened scenario).

c. Clarify whether the basis for screening the turbine missile hazard is the probability of turbine missile analysis. If not, provide justification for not using that analysis and selecting an alternate approach.

SUBJECT: LA SALLE COUNTY STATION, UNITS 1 AND 2 – SUPPLEMENT TO THE AUDIT PLAN IN SUPPORT OF STAFF REVIEW OF LAR TO ADOPT 10 CFR 50.69 (EPID L-2020-LLA-0017) DATED JUNE 11, 2020

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