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

April 22, 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: NINE MILE POINT NUCLEAR STATION, UNIT 2 – AUDIT PLAN SUPPLEMENT IN SUPPORT OF REVIEW OF LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT RISK-INFORMED COMPLETION TIMES (EPID L-2019-LLA-0234)

Dear Mr. Hanson:

By letter dated October 31, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19304B653), as supplemented by letter dated December 12, 2019 (ADAMS Accession No. ML19346F427), Exelon Generation Company, LLC (Exelon, the licensee) requested that the U.S. Nuclear Regulatory Commission (NRC) amend the Technical Specifications (Appendix A) of Renewed Facility Operating License No. NPF-69 for Nine Mile Point Nuclear Station, Unit 2. Exelon's proposed license amendment request (LAR) would revise technical specification requirements to permit the use of risk-informed completion times for actions to be taken when limiting conditions for operation are not met. The proposed changes are based on Technical Specifications Task Force Traveler 505, Revision 2, "Provide Risk Informed Extended Completion Times – RITSTF Initiative 4b," dated July 2, 2018 (ADAMS Package Accession No. ML18269A041).

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 NRC staff is conducting a regulatory audit to support its review of the LAR in accordance with the initial audit plan that was provided to Exelon by email dated February 6, 2020 (ADAMS Accession No. ML20037B654). The audit plan is being supplemented to include additional documentation and specific questions in the scope of the audit. The staff notes that the scope of its audit information needs related to the technical acceptability of the probabilistic risk assessments used to develop insights to support the licensee's proposed approach, and the mapping of components in different probabilistic risk assessment models, can be affected based on the response to questions related to the consideration of seismic events during categorization in the proposed approach.

A regulatory audit is a planned activity that includes the examination and evaluation of primarily non-docketed information. The audit will be conducted to increase the NRC staff's understanding of the LAR and identify information that will require docketing to support the NRC staff's regulatory finding. The audit will be conducted from May 4, 2000, to May 7, 2020, at Exelon's office located at 200 Exelon Way, Kennett Square, Pennsylvania, between

9:30 a.m. and 4:00 p.m. on Monday, May 4, 2020, and 8:30 a.m. and 4:00 p.m. on each subsequent day. However, depending on the need for continuing social distancing, the audit may be conducted using video conferencing software instead of in person at Kennett Square. It should be noted that the audit for this LAR and regulatory audit for the risk-informed categorization and treatment of structures, systems, and components LAR are being conducted concurrently. The logistics and scope of the audit supplement were discussed with your staff on April 14, 2020. The audit plan is enclosed.

If you have any questions, please contact me by telephone at 301-415-2871 or by e-mail to Michael.Marshall@nrc.gov.

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

Enclosure:
Audit Plan Supplement

cc: Listserv

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

AUDIT PLAN SUPPLEMENT

REGARDING RISK-INFORMED COMPLETION TIMES

EXELON GENERATION COMPANY, LLC

NINE MILE POINT NUCLEAR STATION, UNIT 2

DOCKET NO. 50-410

1.0 BACKGROUND

By letter dated October 31, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19304B653), as supplemented by letter dated December 12, 2019 (ADAMS Accession No. ML19346F427), Exelon Generation Company, LLC (Exelon, the licensee) requested that the U.S. Nuclear Regulatory Commission (NRC) amend the Technical Specifications (TSs) (Appendix A) and licensing basis of Renewed Facility Operating License No. NPF-69 for Nine Mile Point Nuclear Station, Unit 2 (Nine Mile Point 2). Exelon's proposed license amendment request (LAR) would revise TS requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk Informed Extended Completion Times – RITSTF Initiative 4b," dated July 2, 2018 (ADAMS Package Accession No. ML18269A041).

2.0 REGULATORY AUDIT BASES

A regulatory audit is a planned license or regulation-related activity that includes the examination and evaluation of primarily non-docketed information. The audit is conducted with the intent to gain understanding, to verify information, and to identify information that will require docketing to support the basis of a licensing or regulatory decision. Performing a regulatory audit is expected to assist the NRC staff in efficiently conducting its review and gaining insights for licensee's processes and procedures. Information that the NRC staff relies upon to make the safety determination must be submitted on the docket.

The audit will continue to be performed consistent with NRC Office Instruction LIC-111, Revision 1 "Regulatory Audits," dated October 31, 2019 (ADAMS Accession No. ML19226A274). The NRC staff is conducting a regulatory audit to support its review of the LAR in accordance with the initial audit plan that was provided to Exelon by email dated February 6, 2020 (ADAMS Accession No. ML20037B654). The audit plan is being supplemented to include additional documentation and specific questions in the scope of the audit. An audit was determined to be the most efficient approach toward a timely resolution of issues associated with this LAR review, since the staff will have an opportunity to minimize the potential for multiple rounds of requests for additional

Enclosure

information (RAIs) and ensure no unnecessary burden will be imposed by requiring the licensee to address issues that are no longer necessary to make a safety determination.

3.0 PURPOSE AND SCOPE

The purpose of this audit is to:

- Gain a better understanding of the calculations, analyses, and bases underlying the LARs. Confirm the staff's understanding of the LARs.
- Gain a better understanding of the approach for developing and implementing nuclear power station risk-managed TS programs.
- Identify information that the licensee should submit for staff to reach a regulatory decision. Discuss potential RAIs.
- Gain a better understanding of the extent that the licensee's proposed amendment to modify TS requirements for RICTs is consistent with TSTF-505, Revision 2, and Nuclear Energy Institute (NEI) 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document," dated November 6, 2006.
- Gain a better understanding of whether the proposed configurations introduce any adverse effects on the ability or capacity of plant equipment to perform its design-basis function(s) when the plant is operated in the proposed TS allowable configuration.
- Gain a better understanding of the technical acceptability of the probabilistic risk assessment (PRA) for use in the application and how plant design features are modeled in the PRA used to support the LAR.

The areas of focus for the regulatory audit are the information contained in the LAR, the audit information needs listed in the following section of this audit plan supplement, and all associated and relevant supporting documentation (e.g., methodology, process information, calculations, etc.). The relevant supporting documents are identified below.

4.0 INFORMATION AND OTHER MATERIAL NECESSARY FOR THE REGULATORY AUDIT

The following documentation should be available to the audit team:

- RICT program procedures (e.g., risk-management action (RMA) procedure, PRA functionality determination procedure, recording LCO procedure, etc.), as available
- All PRA models (e.g., internal events, internal flooding, fire PRA (FPRA), and PRA documentation, including PRA notebooks
- All PRA peer review reports, self-assessments of the PRA models, and facts and observations (closure reports)
- Documentation of changes to the PRA models with justification of upgrades and updates
- PRA configuration control procedures
- Analyses supporting PRA success criteria, which differ from design-basis criteria
- Documentation of review of PRA model assumptions and sources of uncertainty and identification of key assumptions and sources of uncertainty for the application identified in the LAR

- Documentation supporting the development and benchmarking against the PRA of the PARAGON tool
- System diagrams (including, piping and instrumentation diagrams), as applicable to audit questions
- Single line diagram(s) for the electrical power distribution system
- Load list and associated load rating for each safety-related bus

The licensee should be prepared to provide the following examples and demonstrations:

- Demonstration of PARAGON tool
- Example of RICT calculation
- Example of PRA functional definition, development, and use
- Example of RMA determination
- Modeling of the instrumentation and controls LCOs in the PRA

The licensee should be prepared to discuss:

- LAR and RICT program
- PRA technical acceptability
- PRA model assumptions and sources of uncertainty and the process for identification and disposition of the key assumptions and sources of uncertainty
- Calculation of the RICT estimates presented in the LAR
- External events treatment for the RICT program
- How RMAs are determined and implemented
- Reviews and benchmark testing of the PARAGON tool to ensure results are consistent with the baseline PRA model
- PRA modeling for select LCOs
- Why certain LCOs do not constitute a loss of function, and how all design-basis criteria are met when entering the specified LCO
- How cumulative risk (i.e., core damage frequency and large early release frequency) will be evaluated and tracked
- Definitions for electrical train, channel, division, and subsystem
- Design success criteria for TS 3.8.1.B (in Table E1-1 of Enclosure 1 of the LAR)

The specific questions that the audit team would like to discuss with Exelon are attached to the audit plan (Attachment 2). It should be noted that the audit for this LAR and regulatory audit for the risk-informed categorization and treatment of SSCs LAR are being conducted concurrently. Specific questions concerning both the proposed RICT program and proposed risk-informed categorization process are included in the attachment to this audit plan supplement. The audit team will not remove non-docketed information from the audit site.

5.0 AUDIT TEAM

The members of the audit team are anticipated to be:

- Jigar Patel, Team Leader, PRA, NRC
- Mihaela Biro, Reliability and Risk Engineer, NRC

- Keith Tetter, Reliability and Risk Engineer, NRC
- Zach Coffman, Reliability and Risk Engineer, NRC
- Charles Moulton, Fire Protection Engineer, NRC
- Khoi Nguyen, Electrical Engineer, NRC
- Michael Marshall, Project Manager, NRC
- Garill Coles, Principal Engineer, Pacific Northwest National Laboratory

6.0 LOGISTICS

The audit will be conducted from May 4, 2000, to May 7, 2020, at Exelon's office located at 200 Exelon Way, Kennett Square, Pennsylvania, between 8:00 a.m. and 4:00 p.m. each day. However, depending on the need for continuing social distancing, the audit may be conducted using video conferencing software instead of in person at Kennett Square. A proposed agenda for the audit is attached to this audit plan supplement (Attachment 1). The NRC project manager will coordinate any changes to the audit schedule and location with the licensee.

7.0 SPECIAL REQUESTS

The NRC staff would like access to the following equipment and services:

1. Telephone with a speaker or speaker phone
2. Enclosed conference room (or comparable space) with a table, chairs, and white board (flip board, chalkboard, or equivalent)
3. Breakout room (at least one) for NRC staff discussions
4. Wireless internet access (if available in the work space)

The NRC staff would like access to the documents listed in Section 4.0 above through an online portal (electronic reading room, online reference portal) that allows the NRC staff and contractors to access documents remotely at least 30 days prior to the start of the regulatory audit. NRC staff and contractors' access to the online portal should be terminated 14 days after the end of the regulatory audit.

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.

Proposed Audit Agenda

Day 1

Morning

- Kick-off. Opening comments - NRC and Exelon. Introductions and logistics.
- Real-time risk demonstration by Exelon.
- Discussion on preparing and benchmarking the real-time risk model, including how seasonal variations are accounted (Audit Questions 5 and 10).

Afternoon

- External hazards discussion for both the TSTF-505 and 50.69 license amendment requests (Audit Questions 21 and 22 and Audit Questions C to F).
- Electrical (Audit Questions 23 to 25)
- Summary of the day.
- NRC staff meeting.

Day 2

Morning

- Summary of previous day.
- Modeling of probabilistic risk assessment (PRA) systems (Audit Question 4).
- Calculation of risk-informed completion time estimates.
- Electrical (Audit Questions 23 to 25)

Afternoon

- Modeling of Instrumentation and Controls in the PRA (Audit Question 11).
- FLEX credit (Audit Question 9).
- Fire PRA technical adequacy (Audit Questions 12 to 20).
- Summary of the day.
- NRC staff meeting.

Day 3

Morning

- Summary of previous day.
- Fire PRA technical adequacy (Audit Questions 12 to 20).

Afternoon

- Fire PRA technical adequacy (Audit Questions 12 to 20).
- Internal events technical adequacy (Audit Questions 1 and 2).
- Summary of the day.
- NRC staff meeting.

Day 4

Morning

- PRA key assumptions and sources of uncertainty (Audit Questions 6, 7, 8, and 21).
- PRA update process (Audit Question 3).
- Any remaining 50.69 audit questions (Audit Question A and B).

Afternoon

- Summary of audit.
- Exit meeting.

Audit Questions

QUESTION 1 - Disposition of Open Internal Events Probabilistic Risk Assessment (PRA) Facts and Observations (F&Os)

Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090410014), 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 peer review are the 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 05-04/07-12/12-06, Appendix X: Close Out of Facts and Observations (F&Os)" (ADAMS Package Accession No. ML17086A431), which was accepted by the U.S. Nuclear Regulatory Commission (NRC) in a letter dated May 3, 2017 (ADAMS Accession No. ML17079A427).

The license condition proposed in both the risk-informed categorization (i.e., 50.69) and the risk-informed completion time (i.e.,TSTF-505) license amendment requests (LARs) (Attachment 8 of the 50.69 LAR and Attachment 7 of the TSTF-505 LAR, respectively) includes the commitment to complete a number of implementation items prior to implementation of the risk-informed completion time (RICT) program and 10 CFR 50.69 programs. One of these implementation items is to address the open F&Os from the internal events PRA F&O closure report. This implementation item does not describe what updates will be made to the internal events PRA models to resolve the three remaining F&Os associated with the support system initiating event (SSIE) fault trees or cite resolutions described elsewhere in the LAR, such as in the descriptions in the "Disposition" column for F&Os (presented in Enclosure 2, Table E2-1, of the TSTF-505 LAR, or Attachment 3 to the 50.69 LAR). Therefore, address the following:

a) The disposition for F&O 5-1 states that the cited correction factor will be replaced with improved modeling in the PRA. If available, describe the proposed PRA modeling. Describe the proposed PRA modeling and any subsequent modifications to the corresponding implementation item.

b) The disposition for F&O 8-1 states that a systemic review of the cutsets produced by the SSIE fault trees will be performed to identify feasible recovery actions that could impact the frequency of the associated SSIE. The disposition does not commit to updating the PRA if feasible recovery actions that could impact the frequency of the associated SSIE are identified. Provide a description of the actions that will be performed upon identifying feasible recovery actions and any subsequent modifications to the corresponding implementation item.

c) The disposition for F&O 8-2 appears to indicate that the mission time for common cause factors used in the SSIE fault tress will be adjusted to a year-long mission time. Describe how mission time will be adjusted. If applicable, provide an update to the associated implementation item.

d) 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 the ASME/ANS RA-Sa-2009 PRA standard 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.

e) Provide an evaluation of the proposed model changes associated with open F&Os 5-1, 8-1, and 8-2 related to the SSIE modeling, and demonstrate 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, and (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences.

QUESTION 2 – Peer Review History for the Internal Events, Including Internal Flooding, PRA

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 the ASME/ANS RA-Sa-2009 PRA standard 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. Criteria presented to identify PRA upgrades are (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, and (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences.

LAR Enclosure 2 states that the last full scope peer review for the internal events PRA was conducted in July 2009 and that an F&O closure review to close out F&Os from the 2009 review was conducted in February 2019. The LAR does not indicate what internal events and internal flood PRA model changes were made between July 2009 and February 2019 to improve the model or to incorporate changes to reflect the as-built, as-operated plant. Address the following:

a) Summarize the model changes performed for the internal events, including internal flood PRA since July 2009, and for each change, justify why it does or does not meet the definition of a PRA upgrade as defined in the ASME/ANS RA-Sa-2009 PRA standard.

b) Confirm that focused-scope peer reviews have been conducted for any model change performed for the internal events, including internal flood, PRA model since July 2009 that meets the definition of a PRA upgrade as defined in the ASME/ANS RA-Sa-2009 PRA standard. Describe the peer review and status of the resulting F&Os. Provide any remaining open F&Os, along with dispositions for this application.

QUESTION 3 – TSTF 505 - PRA Model Update Process

Section 2.3.4 of Nuclear Energy Institute (NEI) 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document," specifies that "criteria shall exist in PRA configuration risk management to require PRA model updates concurrent with implementation of facility changes that significantly impact RICT calculations."

LAR Enclosure 7 states that if a plant change or a discovered condition is identified and has significant impact on the RICT calculations, then an unscheduled update of the PRA models will be implemented. More specifically, the LAR states that if the plant changes meet specific criteria defined in the plant PRA update procedures, including criteria associated with consideration of the cumulative risk impact, then the change will be incorporated into applicable PRA models without waiting for the next periodic PRA update. The LAR does not explain under what conditions an unscheduled update of the PRA model will be performed or the criteria defined in the plant procedure that will be used to initiate the update. Therefore, describe the conditions under which an unscheduled PRA update (i.e., less than once every two refueling cycles) would be performed and the criteria that would be used to require a PRA update. In the response, define what is meant by "significant impact to the RICT Program calculations."

QUESTION 4 – System and Surrogate Modeling Used in the PRA Models

The NRC staff's safety evaluation (SE) to NEI 06-09 specifies that the LAR should provide a comparison of the technical specification (TS) functions to the PRA modeled functions and that justification be provided to show that the scope of the PRA model is consistent with the licensing basis assumptions. Table E1-1 in Enclosure 1 of the LAR identifies each TS limiting condition for operation (LCO) proposed to be included in the RICT program and describes how the systems and components covered in the TS LCO are implicitly or explicitly modeled in the PRA. For certain LCOs, the LAR did not provide sufficient description of the PRA modeling that will be used in the RICT calculations. Therefore, address the following:

a) For TS LCO Condition 3.6.1.7 (Suppression Chamber-to-Drywell Vacuum Breakers), Condition A (One line with one or more suppression chamber-to- drywell vacuum breakers inoperable for opening), LAR Table E1-1 states that the "PRA model includes one failure mode: lines fail to close after initially opening. The model will be updated to include this failure mode prior to exercising the RICT program for this TS." The meaning of text is not clear. Based on the text, it appears that the PRA model already includes the failure mode that could be used to calculate the RICT. The implementation item table presented in LAR Attachment 6 has the same wording as used in the comment column of this TS LCO Condition.

   Explain and justify the PRA model changes proposed for the implementation item associated with TS LCO Condition 3.6.1.7.A.

b) For TS LCO Condition 3.7.1 (Service Water System and Ultimate Heat Sink), Condition C (One service water subsystem inoperable for reasons other than Conditions A and B), LAR Table E1-1, states that the:

   > […] success criteria are consistent with the design basis except when UHS temperature is > 82 °F. The model is being updated to include this condition prior to exercising the RICT program for this TS.

The implementation item table presented in LAR Attachment 6 has the same wording as used in the comment column of this TS LCO condition. This seems to imply that the success criteria that will be used in the PRA models to complete the implementation item associated with TS LCO Condition 3.7.1.C will be the same as the design-basis success criterion: "four of six pumps during a loss of coolant accident (LOCA) without a loss of off-site power and ultimate heat sink greater than 82 degrees Fahrenheit (°F) and less than or equal to 84 °F." Describe and justify the PRA model update. If applicable, provide an update to the associated implementation item.

c) For TS LCO Condition 3.3.5.1 (Emergency Core Cooling System (ECCS) Instrumentation), Condition E (ECCS Actuation instrumentation for low pressure core spray (LPCS), low pressure coolant injection (LPCI), high pressure core spray (HPCS)), LAR Table E1-1, states that the failure of the HPCS minimum flow valve will be used as a surrogate for HPCS discharge instrumentation failure.

   Explain how failure of the HPCS minimum flow valve is deemed conservative compared to failure of HPCS discharge instrumentation.

d) LAR Table E1-1 indicates for TS LCO Condition 3.5.1 (Low Pressure ECCS Injection/Spray), Condition A (One low pressure ECCS injection/spray subsystem inoperable), that the PRA success criterion is "One of four subsystems," while the design-basis success criterion is "Two of four subsystems." The explanation for this difference was not provided in LAR Table E1-1 and is not clear to NRC staff. The comment column indicates for this LCO condition that the "success criteria are consistent with the design basis for each train." Therefore, address the following:

   Clarify and justify the PRA success criteria used to model systems associated with TS LCO Condition 3.5.1.A, Low Pressure ECCS Injection/Spray, and provide justification for the less demanding success criteria.

e) For TS 3.3.7.2 (Mechanical Vacuum Pump Isolation Instrumentation), Condition A (one or more channels inoperable implementation items), the first implementation item listed in Attachment 6 of the LAR states that "SSCs [structures, systems, and components] are not modeled. The model will be updated to include these SSCs prior to exercising the RICT program for this TS. The PRA Success Criteria will match the Design Success Criteria."

   i. Describe the proposed PRA modeling associated with TS 3.3.7.2.A.

   ii. Explain how the inoperability of the mechanical vacuum pump isolation instrumentation impacts the core damage frequency (CDF) or large early release frequency (LERF) and how a change in CDF and LERF can be calculated for the RICT estimate.

   iii. If applicable, provide an update to the associated implementation item.

f) For TS 3.7.1.D (One division of intake deicer heaters inoperable), the fifth implementation item listed in Attachment 6 of the LAR states that "the intake deicer heaters are not directly modeled in the PRA. The model will be updated to explicitly include these components prior to its use with RICT."

i. Describe the proposed PRA modeling associated with TS 3.7.1D.

ii. Explain how the inoperability of the deicer heater heaters impacts the CDF or LERF and how a change in CDF and LERF can be calculated for the RICT estimate.

iii. If applicable, provide an update to the associated implementation item.

g) For TS LCO Condition 3.7.5 (Main Turbine Bypass System), Condition A (Main Turbine Bypass System – Requirements of the LCO not met), LAR Table E1-1, indicates that the PRA success criterion is "Three of five bypass valves," while the design-basis success criterion is "Five of five bypass valves." The explanation provided in the comment column of table for this entry states that the "PRA success criteria is based on the minimum valves required to prevent major demands on the suppression pool." The function of the main turbine bypass valves as stated in LAR Table E1-1 is to "control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown." Accordingly, it is not clear how preventing major demands on the suppression pool is equivalent to limiting peak pressure in the main streamlines and reactor to acceptable limits. Therefore, address the following:

   i. Explain the PRA modeling for the main turbine bypass system and its impact on CDF and LERF.

   ii. Justify that successful opening of three of five main turbine bypass valves is sufficient to fulfill the safety function of these valves under TS LCO Condition 3.7.5.A in the accident scenarios modeled in the PRAs.

<u>QUESTION 5 - Impact of Seasonal Variations on Real-Time Risk (RTR) Model</u>

Regulatory Position 2.3.3 of RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," states that the level of detail in the PRA should be sufficient to model the impact of the proposed licensing basis change. The characterization of the problem should include establishing a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated. Full-scale applications of the PRA should reflect this cause-effect relationship in a quantification of the impact of the proposed licensing basis change on the PRA elements.

LAR Enclosure 8, Section 2, states that the "impact of outside temperatures on system requirements like seasonal service water pumps were evaluated and found no dependent flags were needed to be addressed in the CRMP model." LAR Enclosure 9, Table E9-1, indicates that the industry data used in the PRA models includes data for weather-related loss-of-offsite power. The NRC staff notes that seasonal variations in weather conditions include environmental factors besides temperature. Therefore:

a) Explain how any changes in initiator frequency due to seasonal variations is accounted for in the RTR model used in the RICT calculations. If changes in initiator frequency due to seasonal variation are not addressed in the RTR model, then provide justification for this simplification.

b) If changes in initiator frequency due to seasonal variation are not addressed in the RTR model but can impact the RICT and cannot be justified, then propose a mechanism to ensure that changes in initiator frequency due to seasonal variation are accounted for in the RTR model prior to implementation of the RICT program.

c) Explain how any changes in plant response success criteria based on seasonal variations are accounted for in the RTR model used in the RICT calculations. If changes in plant response success criteria due to seasonal variation are not addressed in the RTR model, then provide justification for this simplification.

QUESTION 6 - PRA Model Uncertainty Analysis Process

The NRC staff SE to NEI 06-09, Revision 0, specifies that the LAR should identify key assumptions and sources of uncertainty and assess and disposition each as to their impact on the RMTS application.

Section 5.3 of NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking, Final Report,” dated March 2017 (ADAMS Accession No. ML17062A466), presents guidance on the process of identifying, characterizing, and qualitative screening of model uncertainties.

Discussion of the PRA model uncertainty process is provided in Enclosure 9 to the TSTF-505 LAR and Section 3.2.7 of the CFR 50.69 LAR. Both LARs state that the process for identifying key assumptions and sources of uncertainties for the internal events and FPRAs was performed using the guidance in NUREG-1855, Revision 1. The LARs state that the internal events and FPRA models and notebooks were reviewed for plant-specific key assumptions and sources of uncertainty. Further, the LARs state that generic sources of uncertainty for the internal events PRA were identified from Electric Power Research Institute (EPRI) Technical Report (TR)-1016737, “Treatment of Parameter and Modeling uncertainty for Probabilistic Risk Assessments” and for the FPRA from EPRI TR-1026511, “Practical Guidance of the Use of Probabilistic Risk Assessment in Risk-informed Applications with a Focus on the Treatment of Uncertainty.” The NRC staff notes that the list of internal events PRA key assumptions and sources of uncertainty reported in Enclosure 9 of the TSTF-505 LAR is different from the list presented in Attachment 6 of the LAR submitted to adopt 10 CFR 50.69 risk-informed categorization. Neither LAR describes the process and the criteria used to identify, from the initial comprehensive list of assumptions and sources of uncertainty in the base PRA model(s) (including those associated with plant-specific features, modeling choices, and generic industry concerns), the specific key assumptions and sources of uncertainties presented in the TSTF-505 and 50.69 LARs.

In light of these observations, address the following:

a) Provide a brief description of the process and the criteria used to identify, from the initial comprehensive list of assumptions and sources of uncertainty in the base PRA model(s) (including those associated with plant-specific features, modeling choices, and generic industry concerns), the specific key assumptions and sources of uncertainties for the TSTF-505 application presented in LAR Enclosure 9. Include a description of how the key assumptions and sources of uncertainty are determined, consistent with the definitions in RG 1.200, Revision 2.

b) Similarly, provide a brief description of the process and criteria used to identify, from the initial comprehensive list of assumptions and sources of uncertainty in the base PRA model(s) (including those associated with plant-specific features, modeling choices, and generic industry concerns), the specific key assumptions and sources of uncertainties for the 50.69 application presented in Attachment 6 of the 50.69 LAR. Include a description of how the key assumptions and sources of uncertainty are determined consistent with the definitions in RG 1.200 Revision 2.

QUESTION 7 - PRA Model Uncertainty Analysis Results

The NRC staff SE to NEI 06-09, Revision 0, specifies that the LAR should identify key assumptions and sources of uncertainty and assess and disposition each as to their impact on the RMTS application. LAR Enclosure 9, Table E9-1, identifies the key assumptions and sources of uncertainty for the internal events PRA and provides and disposition for each source of uncertainty for this application. LAR Enclosure 9, Table E9-3 identifies the key assumptions and sources of uncertainty for the FPRA and provides and disposition for each uncertainty for this application. The NRC staff reviewed the dispositions provided in LAR Tables E9-1 and E9-3 and the key assumptions and sources of modeling uncertainty and noted sources of uncertainty that appeared to have the potential to impact RICT calculations. Therefore, address the following:

a) LAR Enclosure 9, Table E9-1, states that treatment of suppression pool strainers performance is a modeling uncertainty. The disposition to this modeling uncertainty states:

> Because suction strainer failures impact all ECCS systems as a common-mode failure, any potential extended unavailability via RICT is not relevant. This item does not represent a key source of uncertainty for the RICT Application.

It is not clear why the assumed individual and common cause failure probabilities for the suppression pool strainers have no impact on the RICT calculations. The NRC staff notes that suppression pool strainer plugging contributes to the failure probability of ECCSs and that LCOs exist in the RICT program for the ECCSs. Accordingly, it appears that if the strainer plugging probability is underestimated, then the RICT for an ECCS can be overestimated. Therefore, justify the conclusion that the uncertainty associated with suppression pool strainer performance cannot have an impact on the RICT calculations.

b) LAR Enclosure 9, Table E9-1, states that:

> Since BWRs are designed to maintain 2/3 core height for a very large break LOCA, injection by one LPCI pump into the shroud area may maintain the covered core sub-cooled. Cooling of the top 1/3 core for a substantial time is questionable, since long-term steam cooling effect may not be ensured. Nine Mile Point 2 assumes that a single LPCI pump is adequate, and there is no real evidence yet that this is not acceptable to prevent core melt.

The LAR also states that a set of sensitivity studies has been performed that shows this uncertainty has a minimal impact on the RICT calculation. However, the LAR does not describe those studies or provide the results.

Describe the sensitivity studies that were performed. Include a description of the assumptions that were made in the sensitivity cases and provide the results of the studies that support the conclusion that this uncertainty only has a minimal impact on the RICT calculations.

c) LAR Enclosure 9, Table E9-1, identifies detailed circuit analysis as a source of FPRA modeling uncertainty because of conservatisms in the approach. The NRC staff notes that because detailed circuit analysis is resource-intensive, it is not typically performed on all circuits. The disposition to this source of uncertainty presented in Table E9-1 states that "uncertainty (conservatism) that may remain in the fire (FPRA) is associated with scenarios that do not contribute significantly to the overall fire risk." It is not clear what the phrase "contribute significantly to the overall fire risk" means quantitatively. The NRC staff notes that uncertainties (e.g., assumed failures or assumed hot shorts) that have some impact on total fire risk could impact the RICT calculations for certain SSCs.

   Justify that the conservativism that exists in circuit analysis will not have an impact on RICT calculations.

QUESTION 8 – Evaluating State-of-Knowledge Correlation Uncertainties Impact on the RICT Program – for all Hazards

As provided by the guidance in NEI 06-09-A, changes to CDF and LERF calculated by a PRA that models the current operating configuration are used to support the RICT program. The guidance in NEI 06-09-A provides several quantitative risk management thresholds values: the calculated RICT, the calculated instantaneous risk, and the cumulative risk increase. When a risk threshold value is exceeded, specific actions are required, as summarized in Table 2.2 of NEI 06-09-A.

RG 1.174 clarifies that, because of the way the acceptance guidelines in RG 1.174 have been developed, the appropriate numerical measures to use when comparing the PRA results with the risk acceptance guidelines are mean values. The risk management thresholds values for the RICT program have been developed based on RG 1.174 and, therefore, the most appropriate measures with which to make a comparison are also mean values. Point estimates are the most commonly calculated and reported PRA results. Point estimates do not account for the state-of-knowledge correlation (SOKC) between nominally independent basic event probabilities, but they can be quickly and simply calculated. Mean values do reflect the SOKC and are always larger than point estimates but require longer and more complex calculations. NUREG-1855, Revision 1, provides guidance on evaluating how the uncertainty arising from the propagation of the uncertainty in parameter values (SOKC) of the PRA inputs impacts the comparison of the PRA results with the guideline values.

Summarize how the SOKC investigation was performed for all the PRA models used to support the RICT application, and how the SOKC will be addressed for the RICT program.

QUESTION 9 - Credit for FLEX Equipment and Actions

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's staff assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision-making in accordance with the guidance of RG 1.200, Revision 2 (ADAMS Accession No. ML090410014).

Regarding equipment failsurure probability in the May 30, 2017, memorandum, the NRC staff concludes (Conclusion 8):

> The uncertainty associated with failure rates of portable equipment should be considered in the PRA models consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200. Risk-informed applications should address whether and how these uncertainties are evaluated.

Regarding human reliability analysis (HRA), NEI 16-06, Section 7.5, recognizes that the current HRA methods do not translate directly to human actions required for implementing mitigating strategies. Sections 7.5.4 and 7.5.5 of NEI 16-06 describe such actions to which the current HRA methods cannot be directly applied, such as debris removal, transportation of portable equipment, installation of equipment at a staging location, routing of cables and hoses, and those complex actions that require many steps over an extended period, multiple personnel and locations, evolving command and control, and extended time delays. In the May 30, 2017, memorandum, the NRC staff concludes (Conclusion 11):

> Until gaps in the human reliability analysis methodologies are addressed by improved industry guidance, HEPs [human error probabilities] associated with actions for which the existing approaches are not explicitly applicable, such as actions described in Sections 7.5.4 and 7.5.5 of NEI 16-06, along with assumptions and assessments, should be submitted to NRC for review.

Regarding uncertainty, Section 2.3.4 of NEI 06-09, Revision 0-A, states that PRA modeling uncertainties shall be considered in application of the PRA-based model results to the RICT program and that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties that could potentially impact the results of an RICT calculation. NEI 06-09, Revision 0-A, also states that the insights from the sensitivity studies should be used to develop appropriate RMAs, including highlighting risk significant operator actions, confirming availability and operability of important standby equipment, and assessing the presence of severe or unusual environmental conditions. Uncertainty exists in PRA modeling of FLEX, related to the equipment failure probabilities for FLEX equipment used in the model, the corresponding operator actions, and pre-initiator failure probabilities. Therefore, FLEX modeling assumptions can be key assumptions and sources of uncertainty for RICTs proposed in this application.

The LAR does not address whether FLEX equipment or actions have been credited in the PRA models. The NRC staff notes that the LAR Enclosure 4, Section 5 credits FLEX features for defense-in-depth for the impact of Local Intense Precipitation. To understand the credit that will be taken for FLEX equipment and actions in the RICT Program, address the following separately for the internal events PRA, internal flooding PRA, and FPRA:

a) Discuss whether Exelon has credited FLEX equipment or mitigating actions into the Nine Mile Point Nuclear Station, Unit 2 (Nine Mile Point 2) internal events, including internal flooding, or FPRA models.

   If not incorporated or their inclusion is not expected to impact the PRA results used in the RICT program, no additional response is requested and remainder of this question is not applicable.

b) Summarize the supplemental equipment and compensatory actions, including FLEX strategies that have been quantitatively credited for each of the PRA models used to support this application. Include discussion of whether the credited FLEX equipment is portable or permanently installed equipment.

c) Regarding the credited equipment:

   i. Discuss whether the credited equipment (regardless of whether it is portable or permanently-installed) are like other plant equipment (i.e. SSCs with sufficient plant-specific or generic industry data).

      If all credited FLEX equipment is similar to other plant equipment credited in the PRA (i.e., SSCs with sufficient plant-specific or generic industry data), responses to items ii and iii below are not necessary.

   ii. Discuss the data and failure probabilities used to support the modeling and provide the rationale for using the chosen data. Discuss whether the uncertainties associated with the parameter values are in accordance with the ASME/ANS PRA standard, as endorsed by RG 1.200, Revision 2.

   iii. Perform, justify, and provide results of LCO specific sensitivity studies that assess impact on RICT due to FLEX equipment data and failure probabilities. As part of the response, include the following:

      1. Justify values selected for the sensitivity studies, including justification of why the chosen values constitute bounding realistic estimates.

      2. Provide numerical results on specific selected RICTs and discussion of the results.

      3. Describe how the results of the sensitivity studies will be used to identify RMAs prior to the implementation of the RICT program, consistent with the guidance in Section 2.3.4 of NEI 06-09, Revision 0-A.

d) Regarding HRA, address the following:

   i. Discuss whether any credited operator actions related to FLEX equipment contain actions described in Sections 7.5.4 and 7.5.5 of NEI 16-06.

   If any credited operator actions related to FLEX equipment contain actions described in Sections 7.5.4 and 7.5.5 of NEI 16-06, answer either item ii or iii below:

ii. Perform, justify, and provide results of LCO specific sensitivity studies that assess impact from the FLEX independent and dependent human error probabilities (HEPs) associated with deploying and staging FLEX portable equipment on the RICTs proposed in this application. As part of the response, include the following:

1. Justify independent and joint HEP values selected for the sensitivity studies, including justification of why the chosen values constitute bounding realistic estimates.

2. Provide numerical results on specific selected RICTs and discussion of the results.

3. Discuss composite sensitivity studies of the RICT results to the operator action HEPs and the equipment reliability uncertainty sensitivity study provided in response to item c.iii above.

4. Describe how the source of uncertainty due to the uncertainty in FLEX operator actions HEPs will be addressed in the RICT program. Describe specific RMAs being proposed and how these RMAs are expected to reduce the risk associated with this source of uncertainty.

iii. Alternatively, for item ii above, provide information associated with the following items listed in supporting requirements HR-G3 and HR-G7 of the ASME/ANS RA-Sa-2009 PRA standard to support detailed NRC review:

1. the level and frequency of training that the operators and non-operators receive for deployment of the FLEX equipment (performance shaping factor (a));

2. performance shaping factor (f) regarding estimates of time available and time required to execute the response;

3. performance shaping factor (g) regarding complexity of detection, diagnosis, and decisionmaking, and executing the required response;

4. performance shaping factor (h) regarding consideration of environmental conditions; and

5. human action dependencies as listed in supporting requirement HR-G7 of the ASME/ANS RA-Sa-2009 PRA standard.

e) 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 impacts the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of ASME/ANS RA-Sa-2009 PRA standard 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.

> Provide an evaluation of the model changes associated with incorporating FLEX mitigating strategies, 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, and (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences.

<u>QUESTION 10 – RICT Entry Conditions</u>

The guidance in NEI 06-09-A states the following regarding high-risk configurations:

> RMTS evaluations shall evaluate the instantaneous CDF, instantaneous LERF. If the SSC inoperability will be due to preplanned work, the configuration shall not be entered if the CDF is evaluated to be greater or equal than 10-3 events/year or the LERF is evaluated to be greater or equal to 10-4 events/year. If the SSC inoperability is due to an emergent event, if these limits are exceeded, the plant shall implement appropriate risk management actions to limit the extent and duration of the high-risk configuration.

The guidance in NEI 06-09-A prohibits voluntary entry into a high-risk configuration, but it allows entry in such configuration if entered due to emergent event and requires implementation of appropriate risk management actions.

Table E1-2 of the LAR, “Example RICT Calculations,” provides risk estimates for conditions proposed in the scope of the RICT program. Note 2 states:

> Per NEI 06-09, for cases where the total CDF and LERF is greater than 1E-03/yr [year] or 1E-04/yr, respectively, the RICT program will not be entered.

This note differs from the guidance in NEI 06-09-A in that it implies that involuntary RICT entry into conditions with high instantaneous CDF or LERF would be also prohibited. Clarify the intent of Note 2 and whether the guidance in NEI 06-09-A will be followed regarding involuntary entries in high-risk configurations.

<u>QUESTION 11 (APLA) – TSTF 505 – Instrumentation and Controls</u>

The LAR proposed TS LCOs include those related to instrumentation and controls (I&C).

PRA technical acceptability attributes are provided in Section 2.3.4 of NEI 06-09, Revision 0-A, and in RG 1.200, Revision 2. The LAR does not address whether the I&C is modeled in sufficient detail to support implementation of TSTF-505, Revision 2. The following additional information is requested:

a) Explain how instrumentation is modeled in the PRA. This should include, but not be limited to, the scope of the I&C equipment (e.g., channel, relays logic) and associated TS functions for which an RICT would be applied, and PRA modeling of I&C and associated functions, including the level of detail and inclusion of plant-specific data, etc.

b) Section 2.3.4 of NEI 06-09, Revision 0-A, states that PRA modeling uncertainties be considered in application of the PRA-based model results to the RICT program. The NRC staff’s SE for NEI 06-09, Revision 0, states that this consideration is consistent with Section 2.3.5 of RG 1.177, Revision 1. NEI 06-09, Revision 0-A, further states that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties that could potentially impact the results of an RICT calculation and that sensitivity studies should be used to develop appropriate compensatory RMAs.

Regarding digital I&C, the NRC staff notes the lack of consensus industry guidance for modeling these systems for plant PRAs to be used in risk-informed applications. In addition, known modeling challenges exist due to the lack of industry data for digital I&C components and the complexities associated with modeling software failures, including common cause software failures. Given these needs and challenges, if the modeling of digital I&C system is included in the RTR model, then address the following:

a) Provide the results of a sensitivity study on the SSCs in the RICT program demonstrating that the uncertainty associated with modeling digital I&C systems has inconsequential impact on the RICT calculations.

b) Alternatively, identify which LCOs are determined to be impacted by the digital I&C system modeling for which RMAs will be applied during an RICT. Explain and justify the criteria used to determine what level of impact to the RICT calculation require additional RMAs.

Regulatory Background for QUESTIONS 12 - 20

RG 1.200 states, "NRC reviewers, [will] focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application." The relatively extensive and detailed reviews of FPRAs undertaken in support of LARs to transition to National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," determined that implementation of some of the complex FPRA methods often used nonconservative and over-simplified assumptions to apply the method to specific plant configurations. Some of these issues were not always identified in F&Os by the peer review teams but are considered potential key assumptions by the NRC staff because using more defensible and less simplified assumptions could substantively affect the fire risk and fire risk profile of the plant.

The NRC staff evaluates the acceptability of the PRA for each new risk-informed application and, as discussed in RG 1.174, recognizes that the acceptable technical adequacy of risk analyses necessary to support regulatory decisionmaking may vary with the relative weight given to the risk assessment element of the decisionmaking process. The NRC staff notes that the calculated results of the PRA are used directly to calculate an RICT that subsequently determines how long SSCs (both individual SSCs and multiple, unrelated SSCs) controlled by TSs can remain inoperable. Therefore, the PRA results are given a very high weight in a TSTF-505 application, and the NRC staff requests additional information on the following FPRA issues that have been previously identified as potentially key FPRA assumptions.

QUESTION 12 - Fire PRA Model – Use of Unacceptable Methods

The LAR provides the history of the FPRA peer review but does not discuss methods used in the FPRA. Methods may have been used in the FPRA that deviate from guidance in NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," (ADAMS Accession Nos. ML052580075, ML052580118, and ML103090242), or other acceptable guidance (e.g., frequently asked questions (FAQs), NUREGs, or interim guidance documents).

a) Identify methods used in the FPRA that deviate from guidance in NUREG/CR-6850 or other acceptable guidance.

b) If such deviations exist, then justify their use in the FPRA, any impact on the RICT, and describe and justify any replacement methods to be used.

QUESTION 13 - Fire PRA Model – Reduced Transient Heat Release Rates

The key factors used to justify using transient fire-reduced heat release rates (HRRs) below those prescribed in NUREG/CR-6850 are discussed in the June 21, 2012, NRC letter to NEI, "Recent Fire PRA Methods Review Panel Decisions and EPRI 1022993, Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires" (ADAMS Package Accession No. ML12172A406).

If any reduced transient HRRs below the bounding 98 percent HRR of 317 kilowatts (kW) from NUREG/CR-6850 were used, discuss the key factors used to justify the reduced HRRs. Include in this discussion:

a) Identification of the fire areas where a reduced transient fire HRR is credited and what reduced HRR value was applied.

b) A description for each location where a reduced HRR is credited and a description of the administrative controls that justify the reduced HRR, including how location-specific attributes and considerations are addressed. Include a discussion of the required controls for ignition sources in these locations and the types and quantities of combustible materials needed to perform maintenance. Also, include discussion of the personnel traffic that would be expected through each location.

c) The results of a review of records related to compliance with the transient combustible and hot work controls.

QUESTION 14 - Fire PRA Model – Treatment of Sensitive Electronics

FAQ 13-0004, "Clarifications on Treatment of Sensitive Electronics" (ADAMS Accession No. ML13322A085), provides supplemental guidance for application of the damage criteria provided in Sections 8.5.1.2 and H.2 of NUREG/CR-6850, Volume 2, for solid-state and sensitive electronics.

a) Describe the treatment of sensitive electronics for the FPRA and explain whether it is consistent with the guidance in FAQ 13-0004, including the caveats about configurations that can invalidate the approach (i.e., sensitive electronics mounted on the surface of cabinets and the presence of louver or vents).

b) If the approach cannot be justified to be consistent with FAQ 13-0004, then justify that the treatment of sensitive electronics has no impact on the RICT calculations.

c) If the approach cannot be justified as consistent with FAQ13-004, and it has an impact on the RICT calculations, describe and justify how this issue will be resolved.

QUESTION 15 - Minimum Joint Human Error Probability (HEP)

NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report," (ADAMS Accession No. ML12216A104), discusses the need to consider a minimum value for the joint probability of multiple human failure events in HRAs.

NUREG-1921 refers to Table 2-1 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA), Final Report" (ADAMS Accession No. ML051160213), which recommends that joint HEP values should not be below 1E-5. Table 4-4 of EPRI TR-1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. Therefore, the guidance in NUREG-1921 allows for assigning joint HEPs that are less than 1E-5, but only through assigning proper levels of dependency. The NRC staff notes that underestimation of minimum joint probabilities could result in non-conservative RICTs of varying degrees for different inoperable SSCs.

The LAR does not provide this information and does not explain what minimum joint HEP value is currently assumed in the internal events or fire PRAs. Also, even if the assumed minimum joint HEP values are shown to have no impact on the current risk estimates, it is not clear to the NRC staff how it will be ensured that the impact remains minimal for future PRA model revisions. Considering these observations:

a) Explain what minimum joint HEP value was assumed in the internal events or in the fire PRAs.

b) If a minimum joint HEP value less than 1E-6 was used in the internal events PRA, or less than 1E-5 was used in the FPRA, then provide a description of the sensitivity study that was performed and the quantitative results that justify that the minimum joint HEP value has no impact on the RICT application.

c) If, in response part (b), it cannot be justified that the minimum joint HEP value has no impact on the application, confirm that each joint HEP value used in the internal events PRA below 1E-6 and each joint HEP used in the FPRA below 1E-5 includes its own separate justification that demonstrates the inapplicability of the NUREG-1792 lower value guideline (i.e., using such criteria as the dependency factors identified in NUREG-1921 to assess level of dependence). Provide an estimate of the number of these joint HEP values below 1E-6 for the internal events PRA and below 1E-5 for FPRA, discuss the range of values, and provide at least two different examples, separately for the internal events and the fire PRAs, where this justification is applied.

QUESTION 16 - Fire PRA Model – Obstructed Plume Model

NUREG-2178, Volume 1, "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE -FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume" (ADAMS Accession No. ML16110A14016), contains refined peak HRRs, compared to those presented in NUREG/CR-6850, and guidance on modeling the effect of plume obstruction. Additionally, NUREG-2178 provides guidance that indicates that the obstructed plume model is not applicable to cabinets in which the fire is assumed to be located at elevations of less than one-half of the cabinet.

a) If obstructed plume modeling was used, then indicate whether the base of the fire was assumed to be located at an elevation of less than one-half of the cabinet.

b) Justify any modeling in which the base of an obstructed plume is located at less than one half of the cabinet's height or describe and justify how this inconsistency with the guidance will be resolved.

QUESTION 17 - Fire PRA Model – Systems Not Credited in the Fire PRA

The NRC staff's SE of NEI 06-09, Revision 0-A, states:

> When key assumptions introduce a source of uncertainty to the risk calculations (identified in accordance with the requirements of the ASME standard), TR NEI 06-09, Revision 0, requires analysis of the assumptions and accounting for their impact to the RMTS calculated RICTs.

The NRC staff notes that components for which cable routing information was not provided or those components not included in the FPRA represent a source of uncertainty in the FPRA. Components for which there is no cable routing information are generally assumed to be failed. It is not known in the FPRA how many or which systems were assumed failed due to the lack of cable routing information or what impact this assumption may have on the RICT calculations. Although the assumption used in the base FPRA model from failing components having no cable routing information is conservative, the NRC staff notes that this conservatism in PRA modeling could have a nonconservative impact on the RICT calculations.

If an SSC is part of a system not credited in the FPRA or it is supported by a system that is assumed to always fail, then the risk increase due to taking that SSC out of service is masked. Therefore, address the following:

a) Identify the systems or components that are assumed to be always failed in the PRA or not included in the PRA. Justify that this assumption has an inconsequential impact on the RICT calculations.

b) Alternatively, describe and justify how the impact on the RICT of the non-conservative PRA assumption of failed SSCs and of SSCs not included in the PRA model will be accounted for. Describe any additional Risk Management Actions that will be taken during the RICT to account for this impact.

QUESTION 18 - Fire PRA Model – Well-Sealed Motor Control Centers (MCC) Cabinets

Guidance in FAG 08-0042 from Supplement 1 of NUREG/CR-6850 applies to electrical cabinets below 440 volts (V). With respect to Bin 15, as discussed in Chapter 6, it clarifies the meaning of "robustly or well-sealed." Thus, for cabinets of less than 440 V, fires from well-sealed cabinets do not propagate outside the cabinet. For cabinets of 440 V and higher, the original guidance in Chapter 6 indicates that Bin 15 panels that house circuit voltages of 440 V or greater are counted because an arcing fault could compromise panel integrity (an arcing fault could burn through the panel sides, but this should not be confused with the high energy arcing fault type fires). FPRA FAQ 14-0009, "Treatment of Well-Sealed MCC Electrical Panels Greater than 440V" (ADAMS Accession No. ML15119A176), provides the technique for evaluating fire damage from motor control center (MCC) cabinets having a voltage greater than 440 V.

Therefore, propagation of fire outside the ignition source panel must be evaluated for all MCC cabinets that house circuits of 440 V or greater.

a) Describe how fire propagation outside of well-sealed MCC cabinets greater than 440 V is evaluated.

b) If well-sealed cabinets less than 440 V are included in the Bin 15 count of ignition sources, provide justification for using this approach as this is contrary to the guidance.

QUESTION 19 - Fire PRA Model – Influence Factors for Transient Fires

NUREG/CR-6850, Section 6, and FAQ 12-0064, “Hot Work/Transient Fire Frequency Influence Factors" (ADAMS Accession No. ML12346A488), describe the process for assigning influence factors for hot work and transient fires. Provide the following regarding application of this guidance:

a) Indicate whether the methodology used to calculate hot work and transient fire frequencies applies influencing factors using NUREG/CR-6850 guidance or FAQ 12-0064 guidance.

b) Indicate whether administrative controls are used to reduce transient fire frequency, and if so, describe and justify these controls.

c) Indicate whether you have any combustible administrative control that were not meet and discuss your treatment of not meeting these administrative controls for the assignment of transient fire frequency influence factors. For those cases where you have violations and have assigned an influence factor of 1 (low) or less, indicate the value of the influence factors you have assigned and provide your justification.

d) If you have assigned an influencing factor of “0” to maintenance, occupancy, storage, or hot work for any fire physical analysis units provide justification.

e) If a weighting factor of “50” was not used in any fire PAU, justify this in light of the guidance in FAQ 12-0084.

QUESTION 20 - Fire PRA Model – Fire Scenario Treatment of the Main Control Board (MCB)

Traditionally, the cabinets on the front face of the main control board (MCB) have been referred to as the MCB for purposes of FPRA. Appendix L of NUREG/CR-6850, “EPRI/NRC Fire PRA Methodology for Nuclear Power Facilities” (ADAMS Accession Nos. ML052580075), provides a refined approach for developing and evaluating those fire scenarios. Fire PRA FAQ 14-0008, “Main Control Board Treatment,” dated July 22, 2014 (ADAMS Accession No. ML14190B307), clarifies the definition of the MCB and effectively provides guidance for when to include the cabinets on the back side of the MCB as part of the MCB for FPRA. It is important to distinguish between MCB and non-MCB cabinets, because misinterpretation of the configuration of these cabinets can lead to incomplete or incorrect fire scenario development. This FAQ also provides several alternatives to NUREG/CR-6850 for using Appendix L to treat partitions in an MCB enclosure. Therefore, address the following:

a) Briefly describe the MCB configuration, and use the guidance in FAQ 14-0008, to determine whether cabinets on the rear side of the MCB are a part of the MCB. Provide your justification using the FAQ guidance.

b) If the cabinets on the rear side of the MCB are part of a single integral MCB enclosure using the definition in FAQ 14-0008, then confirm that the guidance in FAQ 14-0008 was used to:

   I. develop fire scenarios in the MCB and
   II. determine the frequency of those scenarios.

c) If the cabinets on the rear side of the MCB are part of a single integral MCB enclosure and the guidance in FAQ 14-0008 was not used to develop fire scenarios involving the MCB, then provide a description of how the fire scenarios for the backside cabinets are developed and an explanation of how the treatment aligns with NRC-accepted guidance.

d) If in response to part c above, the current treatment of the MCB and those cabinets on the rear side of the MCB cannot be justified using NRC-accepted guidance, then justify that the treatment has no impact on the RICT calculations, or describe and justify how this inconsistency with the guidance in FAQ 14-0008 will be resolved.

QUESTION 21 - Bounding Seismic CDF Analysis

Section 2.3.1, Item 7, of NEI 06-09, Revision 0-A (ADAMS Accession No. ML12286A322), states that the "impact of other external events risks shall be addressed in the Risk Managed Technical Specifications (RMTS) program," and explains that one method to do this is by "performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated Risk Informed Completion Time (RICT)." The NRC staff's SE for NEI 06-09 (ADAMS Accession No. ML071200238) states that "[w]here [probabilistic risk assessment] PRA models are not available, conservative or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT."

LAR Enclosure 4, Section 3, states that to determine a bounding seismic CDF, the peak ground acceleration hazard curve for the 50th percentile high confidence of low probability of failure (HCLPF) of 0.42g was used from its Individual Plant Evaluation of External Events (IPEEE) seismic margins analysis. (Note: The seismic margins analysis HCLPF of 0.5g is the 84th percentile value). The IPEEE HCLPF value of 0.42g was used rather than the more recent HCLPF value of 0.23g in Table C-2 of Results of Safety/Risk Assessment of Generic Issue 199 (GI-199), "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants" (ADAMS Accession No. ML100270582). The LAR states that related sensitivity studies were performed on the impact to risk of changing the ground motion frequencies and seismic hazard intervals. The LAR does not describe those sensitivity studies results or how insights from those studies were used. (The exception to this is that sensitivity study results are presented in LAR Table E4-4 to justify the defined seismic hazard interval for the highest seismic bin, %G8.) Address the following:

a) Explain why the IPEEE HCLPF value of 0.42g was used rather than the HCLPF of 0.23g in GI-199, and why it is acceptable for this application.

c) Describe the cited sensitivity studies and results.

d) Explain how insights from the sensitivity studies were used to select to a bounding ground motion and to define the seismic hazard intervals.

<u>QUESTION 22 - Screening the Extreme Winds Hazard</u>

As clarified in the NRC staff's SE on NEI 06-09, Revision 0-A, other sources of risk (i.e., seismic and other external events) must be quantitatively assessed if they contribute significantly to the incremental risk of any RMTS configuration. However, sources of risk shown to be insignificant contributors to configuration risk may be excluded for the RICT calculations.

Regarding extreme winds, LAR Enclosure 4, Section 4, states that "key equipment and structures" are designed to withstand tornadoes with a maximum rotational velocity of 290 miles per hour (mph) (with a maximum transitional velocity, maximum external pressure drop, and a maximum rate of pressure drop that equate to a maximum "resultant wind speed velocity" of 360 mph). The NRC staff notes that the frequency of tornado wind speeds greater 290 mph at the Nine Mile Point site is less than 1E-07 per year based on NUREG/CR-4461, Revision 2, "Tornado Climatology of the Contiguous United States" (ADAMS Accession No. ML070810400). However, it Is not clear whether the phrase "key equipment and structures" used in the LAR applies to all SSCs that are important to mitigation, including SSCs that may or may not be safety-related. Moreover, it is not clear whether all such SSCs are protected from wind damage (excluding damage from tornado missiles, which is discussed separately below).

Regarding tornado missile risk, the LAR states that the results of its IPEEE tornado missile risk evaluation indicate that the tornado missile CDF is less than 1E-07 per year. The LAR explains, however, that recently a tornado missile protection evaluation was performed for Nine Mile Point Unit 2 in response to Regulatory Issue Summary 2015-16, "Tornado Missile Protection" (ADAMS Accession No. ML15020A419). The LAR states that:

> […] these potentially vulnerable SSCs could contribute to tornado missile risk,
>
> and
>
> […] the risk associated with the identified SSCs remaining unprotected from tornado missiles was evaluated."

The LAR also states that:

> […] [o]nly one of the unprotected SSCs is included in the Nine Mile Point 2 internal events PRA, […]
>
> and
>
> […] it was conservatively estimated that the likelihood of a tornado missile strike on that SSC was much less than 1E-06/yr.

Nonconformance for tornado missile protection often involves components like exhaust stacks, vents, ductwork, pipe risers, and cables that are not explicitly modeled in an internal events PRA but can impact components that are modeled in the PRA such as emergency diesel generators.

Considering the observations above, address the following:

a) Clarify what is meant by the phrase "key equipment and structures" and explain if this phrase applies to all SSCs that are important to mitigation, including SSCs that are not safety-related. Include justification that such SSCs are protected from wind damage (excluding damage from tornado missiles) and how the relevant SSCs will be considered in the proposed RICT program, consistent with endorsed guidance.

b) Clarify if all tornado missile protection nonconformances that could impact CDF and LERF were evaluated in the cited tornado missile protection evaluation. Identify the tornado missile protection nonconformances that were identified but not evaluated in the PRA and justify why the nonconformances do not impact risk. Include an explanation of how the nonconformances will be considered in the proposed RICT program, consistent with endorsed guidance.

QUESTION 23 - TS 3.8.1.B – One Required Diesel Generator Inoperable

UFSAR, Section 8.1.4 states, in part, "The emergency ac [alternating current] power system is divided into three physically separate and electrically independent divisions designated Divisions I, II, and III. Any two out of these three divisions has the capacity and capability to safely shut down the reactor in case of a LOCA or any other DBA." However, the design success criteria (DSC) (Enclosure 1) states "One of two non-HPCS EDGs [emergency diesel generators]." Please explain how "one non-HPCS EDG," which represents one division, could provide the capacity and capability to safely shut down the reactor in case of a LOCA or any other DBA.

QUESTION 24 - TS 3.8.1.C – Two Required Offsite Circuits Inoperable

In Table E1-1 of Enclosure 1 of the LAR, the DSC of TS 3.8.1.C is "See 3.8.1.A," which lists the DSC as "one offsite source." With both offsite circuits inoperable, please explain how "one offsite source" could be an appropriate DSC.

QUESTION 25 – TS 3.8.7.A and TS 3.8.8.B

a) How many uninterruptible power supply inverters are required to be operable for each division (TS 3.8.7.A)?

b) How many uninterruptible power supply inverters are required to be operable for each 120 V alternating current (VAC) division (TS 3.8.8.B)?

QUESTION 26 – TS 3.8.1.C, TS 3.8.7A, TS 3.8.8.A, TS 3.8.8.B, and TS 3.8.8.C

As part of its evaluation, the NRC staff reviews the proposed risk management action (RMA) examples for reasonable assurance that the RMAs are considered to monitor and control risk and to ensure adequate defense-in depth. Enclosure 12 of the LAR describes the RMAs examples for TS 3.8.1.A, TS 3.8.1.B, TS 3.8.1.D, and TS 3.8.4.A. However, the LAR does not include the RMA examples for TS 3.8.1.C, TS 3.8.7.A, TS 3.8.8.A, TS 3.8.8.B, and TS 3.8.8.C. Please provide the RMA examples of TS 3.8.1.C, TS 3.8.7.A, TS 3.8.8.A, TS 3.8.8.B, and TS 3.8.8.C.

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT 2 – AUDIT PLAN SUPPLEMENT IN SUPPORT OF REVIEW OF LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT RISK-INFORMED COMPLETION TIMES (EPID L-2019-LLA-0234) DATED APRIL 22, 2020

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