SOUTHERN NUCLEAR OPERATING COMPANY,

EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DRAFT REQUEST FOR ADDITIONAL INFORMATION FOR THE

LICENSE AMENDMENT REQUESTS TO ADOPT THE RISK-INFORMED PROVISIONS OF

TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS,* PARAGRAPHS 50.48(c) AND 50.69

(EPID NOS. L-2018-LLA-0107 AND L-2018-LLA-0175)

By separate applications dated April 4, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18096A936), and June 7, 2018 (ADAMS Accession No. ML18158A583), Southern Nuclear Operating Company (SNC) submitted two license amendment requests (LARs) for the Edwin I. Hatch Nuclear Plant (HNP), Unit Nos. 1 and 2. In the first LAR, dated April 4, 2018, SNC proposed to adopt National Fire Protection Association Standard 805 (NFPA 805), "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition (ADAMS Accession No. ML010800360), as incorporated into Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.48(c). In the second LAR, dated June 7, 2018, SNC proposed to adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors."

To support the review of the proposed license amendments, an audit team consisting of U.S. Nuclear Regulatory Commission (NRC) staff from the Office of Nuclear Reactor Regulation will be conducting a regulatory audit at the HNP site the week of March 18, 2019. The audit plan will be sent via separate correspondence.

The NRC staff has developed this draft request for additional information (RAI) in support of its review of the subject LARs, and in preparation for the onsite audit. The NRC staff will use this draft RAI to focus discussions with SNC staff during the audit, and to identify further information to be docketed by the licensee that will be needed by the NRC staff to make final regulatory decisions on the subject LARs. The NRC staff will issue a formal RAI following the audit.

NFPA 805 LAR Audit Questions

NFPA 805 Fire Modeling (FM) Question 01

Section 2.4.3.3, "Fire Risk Evaluations," of NFPA-805, states: "[t]he PSA [probabilistic safety assessment] approach, methods, and data shall be acceptable to the AHJ [authority having jurisdiction] ..."

Based on information provided by the licensee, the NRC staff determined that fire modeling (FM) is comprised of the following:

- A plant-specific Fire Modeling Workbook (FMWB) that was developed in lieu of using NUREG-1805, “Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program” (ADAMS Accession No. ML043290075), or Electric Power Research Institute (EPRI) Fire Induced Vulnerability Evaluation Methodology, Revision 1 (FIVE Rev. 1), to determine the Zone of Influence (ZOI) for ignition sources and the time to Hot Gas Layer (HGL) conditions in all fire areas throughout the plant.

- Heskestad's plume temperature correlation, which was used to determine the vertical separation distance based on temperature to a target in order to determine the vertical extent of the ZOI.

- The Consolidated Fire Growth and Smoke Transport (CFAST) model which was used to assess main control room (MCR) abandonment time and multi-compartment analysis (MCA).

- FLASH-CAT which was used for calculating fire propagation in stacks of horizontal cable trays.

- Heat soak method which was used for evaluating time to cable damage.

NFPA 805 LAR Section 4.5.1.2, “Fire PRA,” states that FM was performed as part of the Fire PRA (FPRA) development (NFPA 805-Section 4.2.4.2) and reference is made to LAR Attachment J, “Fire Modeling Verification and Validation,” for a discussion of the acceptability of fire models that were used to develop the FPRA. Based on the information in the LAR, the NRC staff was unable to fully evaluate the FM performed as part of the FPRA and requests that the licensee:

(a) Identify any applications of FM tools or methods used in the development of the LAR that are not discussed in LAR Attachment J.

(b) Regarding fires in the proximity of a corner or walls, explain how the FM approach was applied. Explain how wall and corner effects for the ZOI and HGL timing calculations were accounted for, or provide technical justification if these effects were not considered.

(c) The NRC staff finds that typically, during maintenance or measurement activities in the plant, electrical cabinet doors remain open for a certain period of time. Describe whether there are any administrative controls in place to minimize the likelihood of fires involving such a cabinet, and describe how cabinets with temporarily open doors are treated.

(d) Describe and provide technical justification for the approach that was used in the FLASH-CAT model to determine the time to ignition, the heat release rate per unit area (HRRPUA), and the flame spread rate for cable trays that contain a mixture of thermoplastic and thermoset cables.

(e) Describe the “Heat Soak Method” that was used to convert the damage times in Appendix H of NUREG/CR-6850 to a percent of damage function for targets exposed to a time-varying heat flux.

(f) Describe how non-cable secondary combustibles were identified and accounted for.

(g) Explain how the model assumptions in terms of location and heat release rate of transient combustibles in a fire area or zone will not be violated during and post-transition.

(h) Describe how high energy arcing fault (HEAF) initiated fires are treated in the HGL development timing. Regarding HEAF generated fires, describe the criteria used to decide whether a cable tray in the vicinity of an electrical cabinet will ignite following a HEAF event in the cabinet. Explain how the ignited area was determined and subsequent fire propagation calculated. Describe the effect of cable tray covers and fire-resistant wraps on HEAF induced cable tray ignition and subsequent fire propagation.

(i) F&O 20-19, "Structural Steel Scenario Selection," was generated and dispositioned with the following:

The F&O identifies a statement in report H-RIEFIREPRA- U00-008D, "Specific ignition sources proximate to or with direct impingement on exposed structural steel were not evaluated." The purpose of this statement is to distinguish that detailed analytical heat transfer and structural analysis of steel members was not performed.

As stated earlier in the report, the following criteria are used to develop structural steel scenarios, which makes the statement identified in the F&O unnecessary:

a) Exposed structural steel is present and,
b) A high-hazard fire source is present.

Following these criteria, the scenarios developed are inherently more conservative than an analysis that relies on detailed fire modeling and analytical heat transfer modeling of individual structural steel members.

The F&O has been resolved by deleting the statement in question.

Describe and provide technical justification for the approach that makes structural steel scenarios more conservative.

(j) Regarding the MCA, describe how the size of the opening between the exposing and exposed compartments assumed in the CFAST HGL calculations was determined, and explain to what extent these vent sizes are representative of conditions in the plant.

(k) Regarding the acceptability of CFAST for the control room abandonment time study, describe whether the volumes of the main control boards (MCBs), electrical panels, raised platforms, ductwork in the interstitial space above the egg-crate ceiling, and other obstructions are excluded from the effective control room volume used in the CFAST calculations.

(l) Explain how transient fires against a wall or in a corner were considered in the MCR abandonment calculations.

(m) Because the MCR abandonment calculations are based on the assumption that all doors would normally remain closed, describe if any natural leakage vents were assumed in the analysis.

NFPA 805 FM Question 02

Section 2.5, “Evaluating the Damage Threshold,” of NFPA 805, requires that damage thresholds be established to support the performance-based approach. Thermal impact(s) must be considered in determining the potential for thermal damage of structures, systems, and components (SSCs). Appropriate temperature and critical heat flux criteria must be used in the analysis.

American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) Standard · RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications," Part 4, indicates that damage thresholds be established to support the FPRA. The standard further indicates that thermal impact(s) must be considered in determining the potential for thermal damage of SSCs and appropriate temperature and critical heat flux criteria must be used in the analysis.

HNP Unit 1 uses thermoplastic and thermoset cables, but due to insufficient cable material data, the FPRA assumes all cables are thermoplastic material, unless a definitive determination could be made that a cable’s insulation is thermoset, or equivalent to thermoset, based on cable material codes contained in the Plant Data Management System (PDMS) or in vendor supplied data. Unit 2 uses all thermoset cables.

Provide the following information:

(a) For Unit 1, explain how raceways with a mixture of thermoplastic and thermoset cables were treated in terms of damage thresholds.

(b) For Unit 2, assumed to have thermoset damage criteria, confirm that the cables are actually thermoset and not just IEEE-383 qualified.

(c) Explain how the damage thresholds for non-cable components (i.e., pumps, valves, electrical cabinets, etc.) were determined. Identify any non-cable components that were assigned damage thresholds different from those for thermoset and thermoplastic cables.

NFPA 805 FM Question 03

Section 2.7.3.2, “Verification and Validation,” of NFPA 805, states that each calculational model or numerical method used shall be verified and validated through comparison to test results or comparison to other acceptable models.

LAR Section 4.5.1.2, “Fire PRA,” states that FM was performed as part of the FPRA development (NFPA 805 Section 4.2.4.2). The LAR further states that the acceptability of the use of these fire models is included in Attachment J.

LAR Section 4.7.3 “Compliance with Quality Requirements in Section 2.7.3 of NFPA 805,” states that calculational models and numerical methods used in support of compliance with 10 CFR 50.48(c) were verified and validated as required by Section 2.7.3.2 of NFPA 805.

Based on the information provided in the LAR, the NRC staff was unable to confirm whether each calculational model or numerical method was properly verified and validated, therefore, the NRC staff requests that the licensee:

(a) Describe how the FMWB was verified, and describe how it was ensured that the empirical equations/correlations were coded correctly and that the solutions are identical to those that would be obtained with the corresponding chapters in NUREG-1805 (FDTs) or FIVE Rev.1.

(b) For any FM tool or method that was used in the development of the LAR, provide the V&V basis if it is not already explicitly provided in the LAR Attachment J.

(c) LAR Attachment J states that the smoke detection actuation correlation (Method of Heskestad and Delichatsios) has been applied within the validated range reported in NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications" (ADAMS Accession No. ML071650546). However, the latter reports a validation range only for Alpert's ceiling jet temperatures correlation. Provide technical details to demonstrate that the temperature to smoke density correlation has been applied within the validated range, or justify the application of the correlation outside the validated range reported in the V&V basis documents.

NFPA 805 FM Question 04

Section 2.7.3.3, “Limitations of Use,” of NFPA 805, states “acceptable engineering methods and numerical models shall only be used for applications to the extent these methods have been subject to verification and validation. These engineering methods shall only be applied within the scope, limitations, and assumptions prescribed for that method”.

LAR Section 4.7.3, “Compliance with Quality Requirements in Section 2.7.3 of NFPA 805,” states that “engineering methods and numerical models used in support of compliance with 10 CFR 50.48(c) were applied appropriately as required by Section 2.7.3.3 of NFPA 805.”

Based on the information provided in the LAR, the NRC staff was unable to completely determine whether the engineering methods used were applied within the scope, limitations, and assumptions prescribed for those methods; therefore, the NRC staff requests that the licensee identify uses, if any, of the FMWBs outside the limits of applicability of the method and for those cases outside of the limits, explain the analysis that was used or why the use of the FMWBs was justified.

NFPA 805 FM Question 05

Section 2.7.3.4, “Qualification of Users,” of NFPA 805, states “cognizant personnel who use and apply engineering analysis and numerical models (e.g., FM techniques) shall be competent in that field and experienced in the application of these methods as they relate to nuclear power plants, nuclear power plant fire protection, and power plant operations.”

LAR Section 4.5.1.2, “Fire PRA,” states that FM was performed as part of the FPRA development (Section 4.2.4.2 of NFPA 805). This requires that qualified FM and PRA personnel work together. Furthermore, LAR Section 4.7.3, “Compliance with Quality Requirements in Section 2.7.3 of NFPA 805,” states:

Cognizant personnel who use and apply engineering analysis and numerical methods in support of compliance with 10 CFR 50.48(c) are competent and experienced as required by Section 2.7.3.4 of NFPA 805.

During the transition to 10 CFR 50.48(c), work was performed in accordance with the quality requirements of Section 2.7.3 of NFPA 805. Personnel who used and applied engineering analysis and numerical methods (e.g. FM) in support of compliance with 10 CFR 50.48(c) are competent and experienced as required by NFPA 805 Section 2.7.3.4.

Post-transition, for personnel performing FM or FPRA development and evaluation, SNC will develop and maintain qualification requirements for individuals assigned various tasks. Position Specific Guides will be developed to identify and document required training and mentoring to ensure individuals are appropriately qualified per the requirements of NFPA 805 Section 2.7.3.4 to perform assigned work. See Attachment S, Table S-3, Implementation Item IMP-8.

Based on the information provided in the LAR, the NRC staff was unable to completely determine how the licensee complies with NFPA 805, Section 2.7.3.4; therefore, the NRC staff requests that the licensee:

a. Describe what constitutes the appropriate qualifications for the staff and consulting engineers to use and apply the methods and FM tools included in the engineering analyses and numerical models.

b. Describe the process/procedures for ensuring the adequacy of the appropriate qualifications of the engineers/personnel performing the fire analyses and modeling activities.

c. Provide the position and qualifications of the personnel who performed the walkdowns for the MCR (abandonment based on damage and inhabitability) and the remaining fire areas in the plant. Address whether the same people who performed walkdowns conduct the FM analysis.

d. Explain the communication process between the FM analysts and PRA personnel to exchange the necessary information and any measures taken to assure the FM was performed adequately and will continue to be performed adequately during post-transition.

e. Explain the communication process between the consulting engineers and plant and corporate personnel to exchange the necessary information. Describe measures taken to assure the FM was performed adequately and will continue to be performed adequately during post-transition.

NFPA 805 FM Question 06

NFPA 805, Section 2.7.3.5, “Uncertainty Analysis,” states: “An uncertainty analysis shall be performed to provide reasonable assurance that the performance criteria have been met.”

LAR Section 4.7.3, “Compliance with Quality Requirements in Section 2.7.3 of NFPA 805,” states that “Uncertainty analyses were performed as required by 2.7.3.5 of NFPA 805 and the results were considered in the context of the application. This is of particular interest in FM and FPRA development."

Based on the information provided in the LAR, the NRC staff was unable to determine how the licensee complies with NFPA 805, Section 2.7.3.5, therefore, the NRC staff requests, for the uncertainty analysis for FM, that the licensee:

a. Describe how the uncertainty associated with the fire model input parameters was accounted for in the FM analyses.

b. Describe how the “model” uncertainty was accounted for in the FM analyses.

c. Describe how the “completeness” uncertainty was accounted for in the FM analyses.

NFPA 805 Safe Shutdown (SSD) Question 01

NFPA 805, Section 2.4.2, "Nuclear Safety Capability Assessment," requires licensees to perform a nuclear safety capability assessment (NSCA). RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, (ADAMS Accession No. ML092730314) endorsed the guidance in NEI 00-01, Chapter 3, as one acceptable approach to perform an NSCA. NEI 00-01, Section 3.5.2, indicates that with respect to the electrical distribution system, the issue of breaker coordination must also be addressed.

The licensee’s cable selection and circuit failure analysis indicates that some devices may not be coordinated or coordination may be undetermined but will be addressed through procedures. Since the probabilistic risk assessment (PRA) treats all credited power supplies as having proper electrical coordination:

a. Discuss whether a comprehensive electrical coordination study for the credited power supplies has been completed and whether all issues have been identified and resolved. If not, provide a proposed path forward to resolve the outstanding issues.

b. Discuss any outstanding issues which should be considered for inclusion in LAR Attachment S, as modifications or implementation items as necessary.

NFPA 805 SSD Question 02

10 CFR 50.48(c)(2)(vii) indicates that the fire protection program elements and minimum design requirements of Chapter 3 may be subject to the performance-based methods permitted elsewhere in NFPA 805, and that licensees who wish to use performance-based methods for these fire protection program elements and minimum design requirements shall submit a request in the form of an application for license amendment under 10 CFR 50.90.

In LAR Attachment L, Approval Request 4, the licensee is requesting NRC approval for the use of a performance-based method regarding non-metallic conduit in embedded applications, and polyvinylchloride (PVC) coated flexible metallic conduits in lengths of up to 6 feet. However, in the Safety Margin and Defense-in-Depth section, the NRC staff found that the licensee also mentions that “PVC conduit that is not embedded introduces a negligible amount of combustibles to an area,” which may indicate that PVC conduit is installed.

Clarify whether exposed PVC conduits are installed, and if so, revise Approval Request 4 accordingly to address these conduits, since their use is not allowed per NFPA 805, Section 3.3.5.2.

NFPA 805 SSD Question 03

NFPA 805 Section 3.11.3, “Fire Barrier Penetrations,” indicates that penetrations in fire barriers shall be provided with listed fire-rated door assemblies or listed rated fire dampers having a fire resistance rating consistent with the designated fire resistance rating of the barrier as determined by the performance requirements established by NFPA 805, Chapter 4, and that passive fire protection devices such as doors and dampers shall conform with NFPA 80, “Standard for Fire Doors and Fire Windows.”

In order to meet the requirements of NFPA 80, the licensee has proposed a plant modification to relocate or install fusible links on certain sliding fire doors, as described in LAR Attachment S, Table S-2, Modification Item 1. The licensee indicates that this modification also addresses the potential of water intrusion into Switchgear 1R23S004 located in Fire Area 1017 from fire suppression activities in Fire Zone 0014K. For a postulated fire in Fire Zone 0014, discuss:

a. The actuation time of the fusible link and the suppression system, and whether the fusible link actuated sliding fire door will close in a timely manner as to preclude potential water damage to Switchgear 1R23S004 from fire suppression activities.

b. Since sliding fire doors are not watertight doors, discuss the impact, if any, on Switchgear 1R23S004 due to water migration into Fire Area 1017.

Describe how this modification ensures Switchgear 1R23S004 will be protected.

NFPA 805 SSD Question 04

NFPA 805 Section 4.2.1 requires that one success path necessary to achieve and maintain the nuclear safety performance criteria (NSPC) shall be maintained free of fire damage by a single fire, and that the effects of fire suppression activities on the ability to achieve the NSPC shall be evaluated.

In LAR Attachment C, Table C-1, the discussion of fire suppression effects in many fire areas includes, “water from some deluge or sprinkler systems and from hose streams might temporarily exceed the capacity of the drain system in some areas. However, safety related equipment is elevated above the floor level by pads or pedestals, such that equipment is protected from flooding.”

Based on the information provided in the LAR, the NRC staff was unable to determine whether the effects of fire suppression activities on the ability to achieve the NSPC have been properly evaluated for areas where flooding is a concern and pads and pedestals are credited to protect SSD equipment.

Provide the internal flooding analysis that demonstrates the raised pads and pedestals are adequate in height.

NFPA 805 Probabilistic Risk Assessment (PRA) Question 01 – Fire PRA F&O Closure Review Process

NFPA 805 Section 2.4.3.3 states that the PRA approach, methods, and data shall be acceptable to the NRC. Regulatory Guide (RG) 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 2 (ADAMS Accession No. ML090410014), describes a peer review process utilizing an associated American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard (currently 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," Februrary 2, 2009.) as one acceptable approach for determining the technical adequacy of the PRA once acceptable consensus approaches or models have been established. The primary results of a peer review are the Facts and Observations (F&Os) recorded by the peer review and the subsequent resolution of these F&Os. In a letter dated May 3, 2017 (ADAMS Accession No. ML17079A427) the NRC staff has accepted, with conditions, a final version of Appendix X to Nuclear Energy Institute (NEI) 05-04, 07-12, and 12-13 (ADAMS Accession No. ML17086A431), which defines a review process for closing finding-level F&Os.

LAR Attachment U and LAR Attachment V state that an F&O independent assessment (IA) was performed on the FPRA peer review results to close finding-level F&Os using the process documented in Appendix X to NEI 07-12.

Based on the information provided in the LAR, the NRC staff was unable to determine if the F&O closure reviews were performed fully consistent with the NRC accepted process described above, therefore, the NRC staff requests that the licensee:

a) Confirm, for each FPRA F&O resolved whether the resolution was determined to be a PRA upgrade or maintenance update and that the specific basis for each determination was documented. Include discussion of how the guidance in Appendix 1-A of the ASME/ANS RA-Sa-2009 standard was used in the basis of each determination.

b) If the request in part (a) above cannot be confirmed based on the current F&O closure review documentation, then provide for each finding-level F&O an indication of whether the resolution was determined to be a PRA upgrade or maintenance update along with the specific bases for those determinations as reviewed by the independent assessment (IA) team.

c) For the internal events (IE) and fire F&O closure reviews, explain whether a focused-scope peer review was performed concurrent with the F&O closure review. If a focused-scope peer review was performed, then address the following:

    i. Describe the scope of concurrent focused-scope peer reviews that were performed.

    ii. Confirm that the focused-scope peer reviews meet the requirements of RG 1.200 Revision 2 and that as part of the review, applicable supporting requirements (SRs) were assessed to be met or to meet capability category-II (CC-II).

iii. If any finding-level F&Os that resulted from a concurrent focused-scope peer review were not closed, then provide the dispositions that resolve those F&Os for this application. Also, confirm that any model updates that are needed to resolve those F&Os have been incorporated into the FPRA.

NFPA 805 PRA Question 02 – Incorporation of Internal Events PRA Updates into the FPRA

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. It appears to the NRC staff that a number of internal event (IEPRA) model additions and revisions were made to resolve F&Os that were then closed in the April 2017 IE F&O closure review. The NRC staff notes that the F&O closure review for the IEPRA was followed closely by a FPRA F&O closure review in October/November 2017. It appears to the NRC staff that a number of updates were made to the IEPRA to resolve F&Os that are relevant to the FPRA's underlying plant response model.

Therefore, the NRC staff requests that the licensee:

a) Confirm that applicable IEPRA model updates that were performed to resolve finding-level F&Os ahead of the IE F&O closure review were also performed for the FPRA model used to determine the fire risk estimates for the NFPA 805 LAR.

b) If the IEPRA model updates that were performed to resolve finding-level F&Os were not also performed for the FPRA model used to determine the fire risk estimates for the NFPA 805 LAR, then justify that these model updates have no impact on NFPA 805 LAR application. Alternatively, perform these updates for the integrated analysis provided in response to PRA Question 03.

NFPA 805 PRA Question 03 – Integrated Analysis

Section 2.4.4.1 of NFPA 805 states that the change in public health risk arising from transition from the current FPP to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 (ADAMS Accession No. ML100910006), provides quantitative guidelines on core damage frequency (CDF) and large early release frequency (LERF); identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis; and describes a general framework to determine the acceptability of RI changes.

Based on other NRC staff questions, the PRA methods discussed in the following questions may need to be revised to be acceptable to the NRC:

- PRA Question 02.b regarding update of the FPRA for IE F&O resolutions
- PRA Question 04.b regarding replacement of methods that deviate from NRC guidance
- PRA Question 07.b regarding treatment of sensitive electronics
- PRA Question 08.b regarding well-sealed cabinets less than 440 V
- PRA Question 09.c.ii regarding minimum joint Human Error Probabilities (HEPs)
- PRA Question 10 regarding obstructed plume modelling
- PRA Question 11.d regarding FM of the MCB enclosure

- PRA Question 11.g regarding credit for MCB partitions
- PRA Question 12.f regarding MCR abandonment due to loss of habitability (LOH)
- PRA Question 13.f regarding MCR abandonment due to loss of control (LOC)
- PRA Question 13.i regarding inclusion of the decision-to-abandon the MCR in LOC scenarios
- PRA Question 16.c regarding the impact of untraced cables on change-in-risk
- PRA Question 16.d regarding the impact of other modeling conservatisms on change-in-risk

This list may be revised following the NRC review of the licensee's response to all the questions (not just those listed here).

a) Provide the results of an aggregate analysis that provides the integrated impact on the fire risk (i.e., the total transition CDF and LERF, and the change (Δ) in CDF (ΔCDF) and ΔLERF, of replacing specific methods identified above with alternative methods which are acceptable to the NRC. In this aggregate analysis, for those cases where the individual issues have a synergistic impact on the results, a simultaneous analysis must be performed. For those cases where no synergy exists, a one-at-a-time analysis may be done. For those cases that have a negligible impact, a qualitative evaluation may be done.

b) For each method above, explain how the issue will be addressed in (1) the final aggregate analysis results provided in support of the LAR, and (2) the PRA that will be used at the beginning of the self-approval of post-transition changes. In addition, provide a method to ensure that all changes will be made, that a focused-scope peer review will be performed on changes that are PRA upgrades as defined in the PRA standard, and that any findings will be resolved before self-approval of post-transition changes.

c) Explain how the RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, (ADAMS Accession No. ML092730314) risk acceptance guidelines are satisfied for the aggregate analysis. If applicable, include a description of any new modifications or operator actions being credited to reduce delta risk as well as a discussion of the associated impacts to the fire protection program.

d) If any unapproved methods or deviations will be retained in the PRA that will be used to estimate the change-in-risk of post-transition changes to support self-approval, explain how the quantification results for each future change will account for the use of these unapproved methods or deviations.

e) Identify and summarize the changes to the FPRA model beyond those associated with the questions cited above that may need to be revised and confirm that the changes do not introduce approaches unacceptable to NRC.

NFPA 805 PRA Question 04 - Use of Unreviewed Methods

LAR Attachment V states that the Hatch full-scope-scope FPRA peer review identified "0 unreviewed analysis methods (UAMs)". Though UAMs, as evaluated by the EPRI/NRC panel were not used, this does not preclude the possibility that 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). Based on the information provided in the LAR, the NRC staff could not determine whether any methods that deviate from NUREG/CR-6850 or other acceptable guidance were used, therefore the NRC staff requests that the licensee:

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. Alternatively, replace those methods with a method acceptable to NRC in the integrated analysis performed in response to PRA Question 03. Include a description of the replacement method along with justification that it is consistent with NRC accepted guidance.

NFPA 805 PRA Question 05 – Implementation Item to Update Fire PRA When Modifications are Complete

LAR Attachment S, Table S-3, presents an implementation item (i.e., IMP-19) to update the FPRA after all plant modifications have been implemented to reflect the as-built, as-operated plant.

This implementation item does not indicate HNP's plan in the event that the updated FPRA results do not meet RG 1.174, Revision 2, risk acceptance guidelines. Also, implementation item IMP-19 does not indicate that updates to the FPRA should include adjustments needed to reflect completion of other implementation items such as update of fire response procedures.

Revise implementation item IMP-19 to include an action to update the FPRA following completion of modifications and implementation items and include a plan of action should the updated as-built as-operated FPRA results risk estimates exceed RG 1.174, Revision 2, risk acceptance guidelines (e.g., this plan could include refining the analytic risk estimates or performing additional modifications to the plant).

NFPA 805 PRA Question 06 – Reduced Transient Heat Release Rates (HRRs)

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA. 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, letter from Joseph Giitter, NRC, to Biff Bradley, 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).

The LAR and detailed FM analysis indicate that although a bounding 98% HRR of 317 kW from NUREG/CR-6850 was typically used, a reduced transient fire HRR seems to have been applied as part of detailed FM for certain fire areas (e.g., the Cable Spreading Room, Intake Structure, and East Cableway).

Discuss the key factors used to justify the reduced heat release rate below 317 kW. 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, 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 for the types and quantities of combustible materials needed to perform maintenance. Also, include a 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.

NFPA 805 PRA Question 07 – Sensitive Electronics

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," Revision 2, (ADAMS Accession No. ML081130188), as providing methods acceptable to the NRC staff for adopting a FPP consistent with NFPA 805. In a letter dated July 12, 2006, to NEI (ADAMS Accession No. ML061660105), the NRC established the FAQ process where official agency positions regarding acceptable methods can be documented until they can be included in revisions to RG 1.205 or NEI 04-02. Methods that have not been determined to be acceptable by the NRC staff or acceptable methods that appear to have been applied differently than described require additional justification to allow the NRC staff to complete its review of the proposed method.

Though LAR Attachment H refers to FAQ 13-0004, “Clarifications on Treatment of Sensitive Electronics” (ADAMS Accession No. ML13322A085), the fire scenario development and detailed FM indicates that guidance from FAQ 13-0004 was not used. For example, it appears that for sensitive electronics enclosed in electrical cabinets, inspection and walkdowns of cabinet configurations as recommended by the guidance in FAQ 13-0004 were not performed. However, it appears that a sensitivity study on sensitive electronics may have been performed. Still, the LAR does not describe a sensitivity study performed for sensitive electronics or present the quantitative results of such a study, and the study does not appear to be included as part of the FPRA uncertainty analysis. In light of these observations:

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 louvers 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 NFPA 805 application. Alternatively, replace the current approach with an acceptable approach in the integrated analysis performed in response to PRA Question 03.

NFPA 805 PRA Question 08 – Well-Sealed Cabinets Less Than 440V

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the staff for adopting a FPP consistent with NFPA 805.

It appears that fire scenarios for well-sealed cabinets that house circuits below 440 V were included in the FPRA. Per Section 6.5.6 of NUREG/CR-6850, fires originating from within "well-sealed electrical cabinets that have robustly secured doors (and/or access panels) and that house only circuits below 440 V," do not meet the definition of potentially challenging fires and, therefore, should be excluded from the counting process for Bin 15. By counting these cabinets, the frequencies applied to other Bin 15 cabinets may be inappropriately reduced. Therefore:

a) Clarify how well-sealed cabinets that house circuits below 440 V were treated in the FPRA and explain whether the treatment of these cabinets is consistent with the guidance in NUREG/CR-6850 (i.e., Section 6.5.6 of NUREG/CR-6850 and Chapter 8 of Supplement 1 of NUREG/CR-6850).

b) If the treatment of well-sealed cabinets that house circuits below 440 V deviates from the guidance in NUREG/CR-6850, then justify that the deviation does not impact the NFPA 805 application. Alternatively, remove well-sealed cabinets that house circuits below 440 V from the Bin 15 count in the integrated analysis provided in response to PRA Question 03.

NFPA 805 PRA Question 09 – Minimum Joint Human Error Probability

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 (HFEs) in human reliability analyses (HRAs). NUREG-1921 refers to Table 2-1 of NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA),” (ADAMS Accession No. ML051160213), which recommends that joint human error probability (HEP) values should not be below 1E-5. Table 4-4 of Electrical Power Research Institute (EPRI) report no. 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 FPRA uncertainty analysis appears to include a sensitivity study to evaluate the impact of the minimum joint HEP on the fire risk estimates. The study concludes that the FPRA CDF and LERF are not sensitive to assumptions made about the joint HEP value. However, the results of this sensitivity study and a description about how the study was conducted are not on the docket. The LAR does not provide this information and does not explain what minimum joint HEP value is currently assumed in the FPRA. Also, even if the assumed minimum joint HEP values are shown to have no impact on the current FPRA 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 supporting post-transition changes. In light of these observations:

a) Explain what minimum joint HEP value was assumed in the FPRA.

b) If a minimum joint HEP value less than 1E-05 was used in the FPRA, then provide a description of the sensitivity study that was performed and the quantitative results (i.e.,

CDF, LERF, ΔCDF, and ΔLERF) that justify that the minimum joint HEP value has no impact on the application.

c) If, in response part (b), if it cannot be justified that the minimum joint HEP value has no impact on the application, then provide the following:

   i. Confirm that each joint HEP value below 1E-5 used in the FPRA includes its own 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 1.0E-5, discuss the range of values, and provide at least two different examples where this justification is applied.

   ii. If joint HEP values used in the FPRA below 1E-5 cannot be justified, set these joint HEPs to 1E-5 in the integrated analysis provided in response to PRA Question 03.

d) If a minimum joint HEP value of less than 1E-05 was used but justified because it has no impact on the FPRA results, then add an implementation item that provides an action to confirm that the impact of joint HEP value continues to have a minimal impact on the FPRA estimates in future FPRA models used for post-transition changes.

NFPA 805 PRA Question 10 – Obstructed Plume Model

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the staff for adopting a FPP consistent with NFPA-805. 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. ML16110A140)16), contains refined peak HRRs, compared to those presented in NUREG/CR-6850, and guidance on modeling the effect of plume obstruction.

The FM performed in support of the FPRA appears to use the guidance from NUREG-2178, Volume 1, though it is not clear whether guidance on modelling the effect of an obstructed plume was used. 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.

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.

Justify any modelling in which the base of an obstructed plume is located at less than one half of the cabinet's height, or remove credit for the obstructed plume model in the integrated analysis provided in response to PRA Question 03.

NFPA 805 PRA Question 11 – Treatment of Main Control Room Fires

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology

for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the staff for adopting a FPP consistent with NFPA-805. In a letter dated July 12, 2006, to NEI (ADAMS Accession No. ML061660105), the NRC established the FAQ process where official agency positions regarding acceptable methods can be documented until they can be included in revisions to RG 1.205 or NEI 04-02. Methods that have not been determined to be acceptable by the NRC staff or acceptable methods that appear to have been applied differently than described require additional justification to allow the NRC staff to complete its review of the proposed method. FPRA FAQ 14-0008, "Main Control Board Treatment" (ADAMS Accession No. ML14190B307) provides guidance on modeling fires in the MCR.

The MCBs in the MCR appear to consist of a front and rear side connected by a single enclosure with a continuous ceiling. However, the FM performed for the MCBs appears to treat the cabinets behind the MCB "horseshoe" as separate electrical cabinets instead of treating them as the rear side of the MCB. The guidance in FAQ 14-0008 indicates that if the front and rear sides of such a configuration are connected together in an enclosure where "the presence of a MCB cabinet ceiling would connote a single cabinet," then the rear cabinets should "classified as an integral part of the MCB." For this MCB configuration, the guidance in FAQ 14-0008 provides three options for applying Appendix L of NUREG/CR-6850 to address fire progression associated with the MCB. HNP's treatment of the MCB appears to deviate from NRC accepted guidance.

In a separate MCB modelling concern, NRC staff notes that a damage delay of 15 minutes was credited due to the presence of solid barriers between MCB cabinets. However, it seems that a number of, or all of, the MCR MCBs have open backs (or backs that are open within the large MCB enclosure). NUREG/CR-6850 Section 11.5.2.8 indicates that the approach described in NUREG/CR-6850 Appendix L may be used for "cabinets separated by a single wall with back covers." It is not clear to NRC staff how the presence of solid barriers between MCB cabinet segments can be credited for a 15 minute damage delay for MCB cabinets with open backs.

In light of the observations above, address the following:

a) Describe the MCB configuration for the MCR and compare its configuration with those elements of FAQ 14-0008. Include discussion of the area between the cabinets that comprise the MCB horseshoe and the cabinets on the backside of the MCB horseshoe that appear to NRC staff to be part of single MCB enclosure.

b) Justify that the cabinets behind the MCB horseshoe are not part of single integral MCB enclosure using the definition in FAQ 14-0008.

c) If it cannot be shown in response to part (b) above that the cabinets behind the MCB horseshoe are not part of single integral MCB enclosure using the definition in FAQ 14-0008, then justify treatment of the cabinets on rear side of the MCB as separate electrical cabinets. Include clarification of how the backside cabinets are modelled 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 horseshoe and the cabinets behind the MCB horseshoe cannot be justified using NRC accepted guidance, then update the treatment of the MCB enclosure to be consistent with the guidance in FAQ 14-0008 in the integrated analysis provided in response to PRA Question 03.

e) Clarify whether the MCB, or whether certain individual cabinets of the MCB, have an open back (or backs that are open within the large MCB enclosure).

f) If the MCB, or individual cabinets of the MCB, have an open back, then justify the credit taken in the FPRA for a damage delay of 15 minutes due to the presence of solid barriers between MCB cabinets. Include a description of the FM that supports the damage delay assumption of 15 minutes.

g) If in the response to part (f) above, the credit for a 15 minute delay in damage cannot be justified, then update the fire propagation assumptions for MCB cabinets to be consistent with NRC guidance concerning cabinets with open backs in the integrated analysis provided in response to PRA Question 03.

NFPA 805 PRA Question 12 – MCR Abandonment on Loss of Habitability

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Revision 2, as providing methods acceptable to the staff for adopting a FPP consistent with NFPA-805. Methods that have not been determined to be acceptable by the NRC staff or acceptable methods that appear to have been applied differently than described require additional justification to allow the NRC staff to complete its review of the proposed method.

The LAR does not describe how MCR abandonment scenarios due to loss of habitability (LOH) were modelled. NCR staff notes that LAR Table W-2 includes among the top fire CDF contributors three MCR abandonment scenarios due to LOH with Conditional Core Damage Probabilities (CCDPs) ranging as large as 2.5E-01 to 8.18E-01. Nonetheless, it is still not completely clear to the NRC staff how the treatment of MCR abandonment due to LOH addresses the complexity associated with the full range of fire impacts that can occur from fires in the MCR. The NRC staff notes that this complexity can present a significant modelling challenge.

In light of the observations above, address the following:

a) Explain how the CCDPs and conditional large early release probabilities (CLERPs) were estimated for MCR abandonment scenarios due to LOH. Include identification of the actions required to execute successful alternate shutdown and how they are modeled in the FPRA, including actions that must be performed before leaving the MCR.

b) Explain how command and control is performed given that Unit 2's Remote Shutdown Panel is divided between four panels in four separate fire zone locations. Include discussion of the challenges of maintaining communication between operators who must perform actions at the four different panels and how this is factored into development of the HEPs that are used to estimate the CCDP and CLERP.

c) Explain how various possible fire-induced failures are addressed in the CCDP and CLERP estimates for fires that lead to abandonment due to loss of habitability. Specifically include in this explanation, a discussion of how the following scenarios are addressed:

i. Scenarios where fire fails only a few functions aside from forcing MCR abandonment and successful alternate shutdown is straightforward;

ii. Scenarios where fire could cause some recoverable functional failures or spurious operations that complicate the shutdown, but successful alternate shutdown is likely; and,

iii. Scenarios where the fire-induced failures cause great difficulty for shutdown by failing multiple functions and/or complex spurious operations that make successful shutdown unlikely.

d) Provide the range of CCDP and CLERP values for MCR abandonment scenarios due to loss of habitability for the appropriate fire areas for the post-transition plant model.

e) Provide the range of frequency of MCR abandonment scenarios due to loss of habitability for the post-transition plant cases.

f) If in the response to parts (b) and (c) above, it cannot be justified that the current modelling of MCR abandonment due to LOH addresses the complexity associated with the full range of fire impacts that can occur from fires in the MCR, then replace the current approach with an approach that does address the full range of fire impacts that can occur from fires in the MCR due to LOH in the integrated analysis provided in response to PRA Question 03.

NFPA 805 PRA Question 13 – MCR Abandonment on Loss of Control

The LAR does not describe how MCR abandonment scenarios due to loss of control (LOC) were modelled in the FPRA. LAR Tables W-2 and W-3 do not include among the top contributors to fire CDF MCR abandonment scenarios due to LOC. Based on the information provided, it is not clear to NRC staff whether the treatment of MCR abandonment due to LOC addresses the complexity associated with the full range of fire impacts that can occur from fires in MCR abandonment areas (which appear to be the MCR, Cable Spreading Room, and Computer Room). The NRC staff notes that this complexity can present a significant modelling challenge. The LAR does not describe what cues and procedures would be used by operators in an actual fire scenario to trigger the decision to abandon the MCR due to LOC. Accordingly, it is not clear to the NRC staff that the failure of operators to make the decision to abandon the MCR and perform alternate shutdown is modeled in the FPRA.

In light of the observations above, address the following:

a) Identify those locations in the plant in which fire could lead to LOC for which MCR abandonment and alternate shutdown actions are credited in the FPRA.

b) Explain how various possible fire-induced failures are addressed in the CCDP and CLERP estimates for fires that lead to MCR abandonment due to LOC. Specifically include in this explanation, a discussion of how the following scenarios are addressed. As a part of this response, indicate if the plant response is fully integrated into the PRA.

i. Scenarios where fire fails only a few functions aside from forcing MCR abandonment and successful alternate shutdown is straightforward;

ii. Scenarios where fire could cause some recoverable functional failures or spurious operations that complicate the shutdown, but successful alternate shutdown is likely; and,

iii. Scenarios where the fire-induced failures cause great difficulty for shutdown by failing multiple functions and/or complex spurious operations that make successful shutdown unlikely.

c) Identify the range of CCDP and CLERP values for MCR abandonment scenarios for the appropriate fire areas due to LOC for the post-transition models. Identify those scenarios which have a CCDP of 1, or explain why there are no such scenarios.

d) Provide the range of frequency of MCR abandonment scenarios due to LOC for the appropriate fire areas for the post-transition plant case.

e) Explain how command and control is performed given that Unit 2's Remote Shutdown Panel is divided between four panels in four separate fire zone locations. Include discussion of the challenges of maintaining communication between operators who must perform actions at the four different panels and how this is factored into development of the HEPs that are used to estimate the CCDP and CLERP.

f) If in the response to parts (b) and (c) above, if it cannot be justified that the current modelling of MCR abandonment due to LOC addresses the complexity associated with the full range of fire impacts that can occur from fires in MCR abandonment areas, then replace the current approach with an approach that does address the full range of fire impacts that can occur from fires in MCR abandonment areas in the integrated analysis provided in response to PRA Question 03.

g) Indicate how the decision to abandon the MCR due to LOC is made procedurally by operators. Include discussion of the cues that would trigger the decision to abandon the MCR due to LOC.

h) Explain how the failure of operators to make the decision to abandon the MCR and perform alternate shutdown actions is modeled in the FPRA. Include in the explanation justification that the modeling is consistent with the guidance in NUREG-1921.

i) If failure of operators to make the decision to abandon the MCR and perform alternate shutdown is not modeled in the FPRA, then justify that this exclusion does not impact the application. Alternatively, incorporate failure of operators to make the decision to abandon the MCR and perform alternate shutdown in the integrated analysis provided in response to PRA Question 03 consistent with the guidance in NUREG-1921.

NFPA 805 PRA Question 14 – PRA Treatment of Dependencies between Units 1 and 2

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current FPP to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of RI changes. Section C of RG 1.174 states that

PRAs supporting RI applications should be “based on the as-built, as-operated and maintained plant.”

The LAR indicates that Units 1 and 2 are adjoined and it makes reference to common areas, a cross-tie (e.g., RHRSW to RHR cross-tie) and systems shared between units (e.g., the Diesel Generator 1B). LAR Attachment W shows contribution by fire area for CDF, LERF, ΔCDF, and ΔLERF, but does not explain how the risk contribution from fires originating in one unit is addressed for impacts to the other unit given the physical proximity of the other unit, common areas, and the existence of shared systems. Therefore, address the following:

a) Explain how the risk contribution of fires originating in one unit is addressed for the other unit given impacts due to the physical proximity of equipment and cables in one unit to equipment and cables in the other unit. Include identification of locations where fire in one unit can affect components in the other unit and explain how the risk contributions of such scenarios are allocated in LAR Attachment W, Tables W-4 and W-5.

b) Explain how the contributions of fires in common areas are addressed, including the risk contribution of fires that can impact components in both units.

c) Explain the extent to which systems are shared by both units and whether shared systems are credited in the PRA models for both units. If shared systems are credited in the PRA models for each unit, then explain how the PRAs address the possibility that a shared system is demanded in both units in response to a single IE or fire initiator.

NFPA 805 PRA Question 15 - Calculation of the Change in Risk

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current FPP to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant’s licensing basis and describes a general framework to determine the acceptability of RI changes.

LAR Attachment W, Section W.2.1 provides a general description of how the change-in-risk associated with variances from deterministic requirements (VFDRs) is determined, including discussion about setting fire-induced failures events to “false” in the FPRA model as a way to “mimic” the compliant plant condition. Based on the information provided in the LAR, the NRC staff was unable to fully understand how the change in risk is calculated, therefore, the NRC staff requests that the licensee:

a) Describe the kinds of model adjustments (if there is more than one type) made to remove different types of VFDRs from the compliant plant model, such as adding events or logic, or the use of surrogate events. Clarify whether the approach used is consistent with guidance in FAQ 08-0054, “Demonstrating Compliance with Chapter 4 of NFPA 805” (ADAMS Accession No. ML15016A280 and associated references therein). In addition, identify any major changes made to the GPRA models or data for the purpose of evaluating VFDRs.

b) Because the determination of the change-in-risk for MCR abandonment scenarios can be more complex than for other scenarios in the FPRA:

i. Describe the model adjustments that were made to remove the VFDRs to create the compliant plant model for MCR abandonment scenarios due to both LOH and LOC.

ii. Explain whether VFDRs were identified differently for fire areas in which MCR abandonment (alternate shutdown) may be required compared to fire areas where MCR abandonment would not occur. If VFDRs were identified differently for MCR abandonment scenarios compared to other areas of the plant, then describe that difference.

iii. If assumptions were made, specific to MCR abandonment scenarios, about modeling the compliant plant (e.g., assumptions about how the CCDP values were determined), then describe and justify those assumptions. As part of the justification, provide an indication of the impact that those assumptions make on the NFPA 805 transition change-in-risk.

c) Describe the types of VFDRs identified, and discuss whether and how the VFDRs identified but not modeled in the FPRA impact the risk estimates. Describe the qualitative rationale for excluding VFDRs from the change-in-risk calculations.

d) Explain, for both the compliant and transition plant PRA models, whether plant modifications are credited in the model. Clarify whether plant modifications that do not resolve VFDRs are credited in the transition (variant) plant model, but not in the compliant plant model, as a way to reduce risk (i.e., indicative of a “combined change” as discussed in Section 1.1 of RG 1.174). If modifications are credited in the transition plant model to reduce risk but do not resolve a VFDR, then provide the total risk increase associated with unresolved VFDRs and the total risk decrease associated with non-VFDR modifications.

NFPA 805 PRA Question 16 – Assumed Cable Routing and Other Conservative Modeling

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current fire protection program to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant’s licensing basis and describes a general framework to determine the acceptability of RI changes.

Table 6-3, “Fire PRA Sources of Uncertainty,” of the LAR to adopt 10 CFR 50.69, risk-informed (RI) categorization and treatment of structures, systems, and components for nuclear power reactors (ADAMS Accession No. ML18158A583), states that that a sensitivity study was performed to address the uncertainty associated with un-located/untraced secondary-side cables, given the conservative assumption made in the FPRA that secondary-side systems are failed in all fires. The LAR does not discuss this sensitivity study nor does it provide the quantitative results of the sensitivity study. The assumption that untraced cables are failed in all fire sequences is a conservative approach for modeling untraced cables in the post-transition plant model, but can lead to underestimation of the change-in-risk when used in the compliant plant model

a) Describe the extent of untraced FPRA cables and how they were treated in the FPRA. Include an explanation of how they were modelled in both the compliant and post-transition plant FPRA models.

b) Justify that assumptions made about untraced cables do not contribute to underestimation of the transition change-in-risk. Include a description of the sensitivity study that was performed to address un-located/untraced cables as well as the quantitative results of that sensitivity study.

c) If failing all untraced cables in the FPRA leads to underestimation of the transition change-in-risk, then demonstrate that the application is not impacted by the underestimation of the transition change-in-risk. Alternatively, replace this conservative approach with an acceptable approach that does not underestimate the change-in-risk in the integrated analysis requested in PRA Question 03.

d) If other conservative treatments used in the compliant plant model can be identified as contributing to the underestimation of the total change-in-risk, then identify those conservatisms and demonstrate that the application is not impacted by the corresponding underestimation of the transition change-in-risk. Alternatively, replace such approaches with more realistic approaches in the integrated analysis requested in PRA Question 03 that do not underestimate the change-in-risk.

NFPA 805 PRA Question 17 – Defense-in-Depth (DID) and Safety Margin

NFPA 805, Section 1.2 indicates that defense-in-depth (DID) shall be achieved when an adequate balance of each of the following DID elements is provided: (1) Preventing fires from starting, (2) Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage, and (3) Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

LAR Section 4.5.2.2 provides a high-level description of how impacts on DID and safety margin were reviewed for the transition to NFPA 805, but did not provide sufficient information for the NRC staff to determine whether each DID element was properly addressed. Also, LAR Section 4.5.2.2 states that "[f]ire protection features and systems relied upon to ensure DID were identified as a result of the assessment of DID," but LAR Attachment C, Table C-2 does not identify any fire protection systems or features to be credited for DID. Based on the above identified issues, the NRC staff requests that the licensee:

a) Explain the criteria used to determine when a substantial imbalance between DID echelons exist in the fire risk evaluations (FREs), and identify the types of plant features and administrative controls credited for providing DID for each of the three DID echelons.

b) Clarify what fire protection features and systems were relied upon to ensure DID and explain why none are identified in LAR Attachment C, Table C-2.

c) Discuss the approach for reviewing safety margin using the NEI 04-02, Revision 2, criteria for assessing safety margin in the FREs.

NFPA 805 PRA Question 18 – Impact of a Key Source of Uncertainty on Application

Section 2.4.3.3 of NFPA 805 states that the PRA approach, methods, and data shall be acceptable to the NRC. Section 2.4.4.1 of NFPA 805 further states that the change in public health risk arising from transition from the current FPP to an NFPA 805 based program, and all future plant changes to the program, shall be acceptable to the NRC. RG 1.174 provides quantitative guidelines on CDF, LERF, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis and describes a general framework to determine the acceptability of RI changes. With regard to model uncertainty, Section 2.5.3 of RG 1.174 states that "In many cases, the industry's state of knowledge is incomplete, and there may be different opinions on how the models should be formulated." It also states that understanding the impact of key assumptions may be addressed by "performing the appropriate sensitivity studies."

Attachment 6, "Disposition of Key Assumptions/Sources of Uncertainty," of your application to adopt 10 CFR 50.69, provides dispositions for candidate key assumptions and sources of uncertainty for RI categorization. One uncertainty identified that may impact the NFPA 805 application concerns the assumed conditional probability of 1E-02 used account for the loss of net positive suction head (NPSH) following emergency containment venting which leads to failure of the of low pressure emergency core cooling system pumps. The 50.69 LAR did not explain the basis for the 1E-02 value or indicate how much uncertainty may exist in this assumption. NFPA 805 LAR Attachment C, Table C-1 identifies VFDRs associated with spurious opening of the Safety Relief Valves (SRVs) which suggests that assumptions made regarding loss of NPSH following containment venting might have an impact on the estimated change-in-risk. Accordingly, the NRC staff observes that the sensitivity of the fire change-in-risk results for the NFPA 805 application may be sensitive to the same modeling uncertainty as the 10 CFR 50.69 application. In light of these observations:

a) Describe the basis for the assumed conditional probability of 1E-02 for loss of NPSH given containment venting and indicate the degree of uncertainty that exists.

b) Justify why the assumed probability for loss of NPSH following containment venting has a minimal impact on fire risk estimates (i.e., CDF, LERF, ΔCDF, and ΔLERF).

c) If it cannot be qualitatively justified that the impact from the assumed probability for loss of NPSH following emergency venting has a minimal impact the fire risk estimates, then perform a sensitivity study demonstrating that the uncertainty associated with the assumed conditional probability of 1E-02 does not impact the NFPA 805 application.

NFPA 805 Fire Protection Engineering (FPE) Question 01

The compliance strategy for NFPA 805 Section 3.3.5.1, in LAR Attachment A, Table B-1, is identified as "Complies with Required Action," with the actions being revising plant documentation and submitting for NRC approval. In LAR Attachment L, Approval Request 3, the licensee indicated that fire zones contain wiring above suspended ceilings that is not in compliance with NFPA 805, Section 3.3.5.1.

The licensee indicated that one of the bases for their approval request is that "…there are small quantities of low voltage video, communication, and data cables, which are not susceptible to self-ignition." However, the licensee provided no justification for its statements that these types of cables are not susceptible to self-ignition.

Provide the technical basis for this statement and discuss whether the quantity and material properties of the cables impact the basis for the request.

NFPA 805 FPE Question 02

The compliance strategy for NFPA 805, Section 3.3.5.2, "Electrical Raceway Construction Limits" in LAR Attachment A, Table B-1, is identified as “Complies.” However, the licensee is requesting approval for the use of polyvinylchloride coated flexible conduit in lengths up to 6 feet and embedded non-metallic conduit in LAR Attachment L, Approval Request 4. It is not clear whether there are any non-embedded, non-metallic conduits installed at the plant.

Describe whether there are any non-embedded, non-metallic conduits installed at the plant. In addition, provide an explanation as to whether approval is being requested for future installations, or if future installations will be installed in accordance with the requirements of NFPA 805.

NFPA 805 FPE Question 03

Fire protection systems and features that require NFPA code compliance are reflected in NFPA 805, Chapter 3, "Fundamental Fire Protection Program and Design Elements of NFPA 805." The LAR does not contain a list of codes of record that establishes whether or how the licensee meets Chapter 3 of NFPA 805, therefore, the NRC staff requests that the licensee provide a complete list of the applicable NFPA codes and standards designated as the code of record, including identification of the edition (years), that will be in place post-transition. For codes and standards with numerous editions, identify which editions pertain to which particular plant areas and systems.

NFPA 805 FPE Question 04

In LAR Attachment A, Table B-1, the compliance strategies for NFPA 805 Sections 3.3.7.1, 3.3.8, and 3.4.1(a)(1), for example, are identified as “Complies” with the use of an existing engineering equivalency evaluation (EEEE) but the LAR does not describe whether or how non-compliances were resolved. Therefore, the NRC staff requests that the licensee describe how any non-compliances identified during these evaluations were addressed. If any non-compliances are still outstanding, describe how these will be addressed prior the completion of NFPA 805 implementation.

NFPA 805 Health Physics (HP) Question 01

LAR Attachment E, “Radioactive Release Transition” states that the potential release of contaminated effluents resulting from a fire involving radioactive contents in the Bounded Areas compartment is bounded by Vendor Document S77684. In addition, Attachment E states, “This calculation demonstrates that releases are below 10 CFR 20 limits and satisfies the acceptance criteria of FAQ 09-0056.” For Vendor Document S77684, please provide a summary of the assumptions, methodology, input parameters, resulting doses and conclusions.

NFPA 805 HP Question 02

To meet the radioactive release performance criteria for NFPA 805, licensees must demonstrate that radiation released to any unrestricted area due to the direct effects of fire suppression

activities remains as low as is reasonably achievable (ALARA), not to exceed the limits in 10 CFR Part 20.

SNC has performed bounding analyses to demonstrate that the doses from the airborne and liquid pathways resulting from fire suppression activities will not exceed the limits of 10 CFR Part 20. In Vendor Document S77684, the calculated bounding doses are provided in terms of total effective dose equivalent (TEDE), which is consistent with the limits specified in 10 CFR Part 20. The limits in 10 CFR Part 20 are specified in terms of TEDE because the regulations in 10 CFR Part 20 are based on the International Commission on Radiation Protection's (ICRP) recommendations in ICRP Reports 26 and 30. However, when using RASCAL to perform the bounding analysis, the licensee chose to use ICRP 60/72 inhalation dose coefficients. In addition, when HotSpot Version 3.0.2 was used, the licensee selected the Federal Guidance Report (FGR) No. 13 dose conversion factors (DCFs). As a result, both the RASCAL and HotSpot calculations provided doses in terms of total effective dose (TED). Likewise, the use of the ICRP 60/119 DCFs for the liquid pathway calculations also resulted in doses in terms of TED. Nevertheless, the results provided in the licensee's conclusions were provided in terms of TEDE. While both TEDE and TED calculate dose for external and internal exposure, the underlying dosimetry models used to develop the DCFs are not the same. The DCFs selected for the gaseous and liquid bounding analyses results in the use of dosimetry models and DCFs that differ from those used in ICRP Reports 26 and 30. Dose conversion factors acceptable to the NRC staff are derived from data and methodologies provide in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" and can be found in FGR No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," and FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," for exposure to radionuclides in air, water, and soil. Please provide a summary explaining why the use of the TED DCFs is acceptable, even though the dose limits in 10 CFR Part 20 are specified in terms of TEDE.

<u>10 CFR 50.69 LAR Audit Questions</u>

50.69 Question 01 (APLA/APLB) – Appendix X, Independent Assessment Process

Paragraph 50.69(c)(1)(i) of 10 CFR requires the PRA must be of sufficient quality and be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

Section 3.3 of the 10 CFR 50.69 LAR states that resolutions to the F&Os resulting from the peer reviews were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, in a letter dated February 21, 2017 from Victoria K. Anderson, NEI, to Stacey Rosenberg, NRC (ADAMS Package Accession No. ML17086A431), and as accepted by the NRC in a letter dated May 3, 2017 from Joseph Giitter and Mary Jane Ross-Lee, NRC, to Greg Krueger, NEI (ADAMS Accession No. ML17079A427).

Based on the information provided in the LAR, the NRC staff was unable to determine if the F&O closure reviews were performed fully consistent with the NRC accepted process described above; therefore, the NRC staff requests that the licensee provide the following information to confirm that the independent assessments for closure of F&Os performed for the IEPRA (April 2017), the seismic PRA (SPRA) (June 2017), and the FPRA (October 2017), were performed consistent with the NRC accepted process, as discussed in the NRC letter dated May 3, 2017.

a) Regarding closure of each F&O, confirm that the independent assessment team was provided with a written assessment and justification of whether the F&O resolution constitutes a PRA upgrade or maintenance update, as defined in ASME/ANS RA-Sa-2009 and qualified by RG 1.200, Revision 2.

b) If the request made in part (a) above cannot be confirmed, then perform a subsequent independent assessment for F&O closure and/or addendum to the independent assessment report to address any identified inconsistencies with Appendix X, as accepted, with conditions, by the NRC staff in letter dated May 3, 2017. Provide any F&Os that remain open as a result of this review. For each F&O and/or item that remains open, provide its associated disposition to demonstrate that it has no adverse impact on the 10 CFR 50.69 application.

50.69 Question 02 (APLA) – Internal Flooding Open F&O 4-5, Credit for Manual Flood Isolation

Internal flooding F&O 4-5 related to SR IFSN-A10 and IFQU-A5 identified that no credit is taken for the manual isolation of floods.

The independent assessment team identified that this F&O identifies a major modeling issue with internal flooding, and that it constitutes significant changes from the previously peer reviewed model. The licensee's disposition states that this is a documentation issue with no impact on the application, and:

> "[T]he original flooding evaluation credited manual isolation of flooding using some screening values and some detailed HRA analysis. A subsequent revision removed all credit for isolation but performed a flooding screening analysis. Then a third revision re-applied the previous HRA analysis to the scenarios that passed the screening."

a) Clarify whether the internal flooding PRA (IFPRA) model that will be used for the 10 CFR 50.69 categorization process credits manual flooding isolation. If not modelled in the PRA, provide justification for why this exclusion has no adverse impact on the 10 CFR 50.69 application.

b) If credit for manual flooding isolation is taken in the PRA(s), provide justification for why this change is either a PRA upgrade or maintenance update in accordance with the ASME/ANS RA-Sa-2009 PRA standard.

c) If a PRA upgrade is determined in part (b) above, describe how this model has been peer reviewed; or alternatively, conduct a focused-scope peer review and provide the resulting F&Os along with the disposition for each F&Os' impact on the 10 CFR 50.69 application.

50.69 Question 03 (APLA) – Hatch FPRA under Review for Adoption to NFPA-805

Paragraphs 50.69(c)(1)(i) and (ii) of 10 CFR require a licensee's PRA be of sufficient quality and level of detail to support the SSC categorization process, and that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience.

Section 3 of the 10 CFR 50.69 LAR states that a LAR was submitted requesting transition to the National Fire Protection Association (NFPA) 805 Performance-Based Standard for Fire Protection (ADAMS Accession No. ML18096A955). Attachment 1 of the 10 CFR 50.69 LAR lists several plant modifications that are credited in the FPRA risk estimates for the NFPA 805 LAR to meet the risk acceptance guidance of RG 1.174. Attachment S of the NFPA 805 LAR lists several implementation items (such as updating of the fire response procedures) that are also credited in the FPRA to meet the risk acceptance guidelines discussed in RG 1.174.

Because there is a potential for additional FPRA model changes to resolve questions associated with the NFPA 805 LAR that is currently under NRC staff review, the NRC staff requests that the licensee propose a mechanism to ensure that the FPRA model used for the 10 CFR 50.69 categorization process reflects the as-built, as-operated plant at the time of the implementation of the 10 CFR 50.69 categorization. The proposed mechanism should ensure that the PRA models (i.e., IEPRA, FPRA and SPRA) used for SSC categorization have incorporated all FPRA updates needed to resolve questions from the NFPA 805 review. This mechanism should also provide an explicit description of changes that will be made to the PRA model(s) and/or documentation to resolve this issue. If the changes to the PRA are determined to involve a PRA upgrade, the mechanism should include a focused-scope peer review and require resolution of all generated finding-level F&Os from the peer review prior to implementation of the 10 CFR 50.69 categorization process. An example would be a table of listed implementation items referenced in a license condition.

50.69 Question 04 (APLA/APLB) – Process for Identification of Key Assumptions and Sources of Uncertainties

Paragraphs 50.69(c)(1)(i) and (ii) require a licensee's PRA be of sufficient quality and level of detail to support the SSC categorization process, and that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience. The guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," (ADAMS Accession No. ML052910035), specifies sensitivity studies to be conducted for each PRA model. The sensitivity studies are performed to ensure that assumptions associated with these uncertainty parameters (e.g., human error, common cause failure, and maintenance probabilities) do not mask the SSC(s) importance.

LAR Section 4.1 identifies RG 1.174, Revision 2, as an applicable regulatory requirement/criterion. RG 1.174 has been updated to Revision 3, dated January 2018 (ADAMS Accession No. ML17317A256). Regulatory Guide 1.174, Revision 3, cites NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (ADAMS Accession No. ML17062A466), as related guidance. In Section B of RG 1.174, Revision 3, the guidance acknowledges specific revisions of NUREG-1855 to include changes associated with expanding the discussion of uncertainties. LAR Section 3.2.7 states that the detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855, March 2009, Revision 0 (ADAMS Accession No. ML090970525) and Section 3.1.1 of EPRI Technical Report (TR)-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments". The NRC staff notes that for the IEPRA (includes internal flooding), FPRA, and SPRA models, only three, two, and five sources of uncertainty were identified, respectively.

NUREG-1855 has been updated to Revision 1 as of March 2017 (ADAMS Accession No. ML17062A466). The NRC staff notes that NUREG-1855, Revision 1, provides guidance in

stages A through E for how to treat uncertainties associated with PRA models in RI decision-making. Revision 1 of NUREG-1855 cites EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainties." Based on the information provided in the LAR, the NRC staff was unable to determine whether key assumptions and sources of uncertainty were properly addressed, therefore, the NRC staff requests that the licensee provide the following:

a. A detailed summary of the process used to identify the key assumptions and sources of uncertainty presented in LAR Attachment 6. The discussion should include the following:

   i. How the process is consistent with NUREG-1855, Revision 1, or other NRC-accepted methods (e.g., NUREG-1855, Revision 0). If deviating from the current guidance provided in NUREG-1855, Revision 1, provide a basis to justify the method's use in the 10 CFR 50.69 categorization process (e.g., exclusion/consideration of EPRI TR-1026511).

   ii. A brief description of how the key assumptions and sources of uncertainties provided in LAR Attachment 6 were identified from the initial comprehensive list of PRA model(s) (i.e., base model) uncertainties and assumptions, including those associated with plant specific features, modeling choices, and generic industry concerns. This can include an identification of the sources of plant-specific and applicable generic modeling uncertainties identified in the uncertainty analyses for the base IEPRA (includes internal flood) and the base FPRA and include a disposition for each of the assumptions and/or uncertainties addressing their impact on the 10 CR 50.69 risk categorization. For any source of uncertainty or assumptions judged not to be key to the application, provide discussion for why it is not pertinent to the application and therefore does not need to be addressed (i.e., sensitivity does not need to be performed).

b. If the process used to identify, characterize, and assess the key assumption(s) and sources of uncertainty provided in LAR Attachment 6 cannot be justified for use in the 10 CFR 50.69 categorization process, provide the results of an updated assessment of the key assumptions, sources of uncertainty, and treatment of the sources of uncertainty performed in accordance with NUREG-1855, Revision 1, and NEI 00-04, Revision 0. If sensitivity studies are proposed to be performed for the treatment of a specific key assumption or source of uncertainty, include a detailed description of the proposed sensitivity studies, and justify how it is bounding to address the specific key assumption and/or source of uncertainty.

50.69 Question 05 (APLA/APLB) – Dispositions of Key Assumptions and Sources of Uncertainties

Paragraph 50.69(c)(1)(i) of 10 CFR requires the licensee to consider the results and insights from the PRA during categorization. The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model. 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 importance of components. NEI 00-04 guidance states that additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered.

LAR Section 3.2.7 states that “a few system specific sensitivity analyses may be required to address Hatch model specific assumptions or sources of uncertainty.” Multiple dispositions provided in LAR Attachment 6, Table 6-3 appear to indicate that no key assumptions or sources of uncertainty were identified that will require a sensitivity study for the 10 CFR 50.69 categorization process.

The NRC staff observes that for even small impacts on modelled PRA hazard(s), the CDF and LERF (i.e., increase and/or decrease) could potentially change the risk importance measures for certain SSCs, thus impacting the RG 1.174 threshold values for total CDF and LERF and the criteria (importance measures) used to determine the categorization of SSCs potentially. Modelling conservatisms can mask the importance measures of other SSCs. Considering these observations, address the following:

a) The disposition for the first item of LAR Attachment 6, Table 6-3 states in part that the uncertainty associated with untraced secondary-side cables was addressed for the FPRA using a sensitivity study to assess the assumption that secondary-side systems fail in all fires. This sensitivity study and its results are not discussed in either the NFPA 805 LAR or the 10 CFR 50.69 LAR. Accordingly, it is not clear that this sensitivity study (i.e., the assumption that all secondary-side systems fail in a fire) performed for the FPRA is applicable to 10 CFR 50.69 categorization. Considering these observations:

   i. Provide the quantitative results of the sensitivity study and/or justification to support the conclusion that the uncertainty associated with untraced secondary-side cables has no impact on the 10 CFR 50.69 categorization process. Include in the justification the following: (1) a description of the sensitivity study that was performed for the FPRA, (2) an explanation of how it considered the potential to mask or skew the importance of certain SSCs, and (3) an explanation of how the sensitivity study bounds the source of uncertainty being addressed.

   OR

   ii. If the uncertainty addressed in part (i) above cannot be justified to have no adverse impact on the 10 CFR 50.69 categorization process, then propose a mechanism that ensures that a sensitivity study is performed during the 10 CFR 50.69 categorization process. Include a brief discussion of the sensitivity study proposed and how it addresses the uncertainty associated with untraced secondary-side cables. Include in the justification the following: (1) a description of the sensitivity study that is proposed, (2) an explanation of how it considers the potential to mask or skew the importance of certain SSCs and (3) an explanation of how the sensitivity study bounds the uncertainty associated with untraced secondary-side cables. An example would be a table of listed implementation items referenced in a license condition.

   OR

   iii. Propose a mechanism that eliminates the uncertainty associated with the untraced secondary-side cables. This mechanism should also provide an explicit description of changes that will be made to the PRA model(s) and/or documentation to resolve this issue. If these changes are determined to involve a PRA upgrade, the mechanism should include a focused-scope peer review and require resolution of all generated finding-level F&Os from the peer review prior to implementation of the 10 CFR 50.69

categorization process. An example would be a table of listed implementation items referenced in a license condition.

b) The disposition for the first item in LAR Attachment 6, LAR Table 6-2 states that the uncertainty associated with credit taken for diverse and flexible coping strategies (FLEX) implementation in the SPRA will be addressed by the human reliability analysis (HRA) sensitivity study performed as part of 10 CFR 50.69 categorization. The HRA sensitivity study performed as part of the categorization process increases all Human Error Probabilities (HEPs) to their 95th percentile value. The disposition also states that the results of a sensitivity study, in which FLEX was assumed to completely fail, showed a significant increase in seismic CDF (41%) and LERF (57%) demonstrating that credit for FLEX is important to seismic risk. The LAR does not describe how the FLEX implementation actions and equipment were modelled in the SPRA or discuss assumptions and sources of uncertainty associated with the modeling. The NRC staff notes that the HRA used to calculate HEP values for FLEX actions, which may occur outside the main reactor buildings and may not be part of a normal or emergency operating procedure, should be consistent with technical elements of the NRC endorsed ASME/ANS PRA Standard (e.g., consideration of environmental conditions). Accordingly, it is not clear to the NRC staff that a sensitivity study using the 95th percentile value for the failure probability of FLEX operator actions is sufficient to address the uncertainty associated with FLEX actions and the lack of industry failure rate information for FLEX equipment.

The disposition for this uncertainty issue also states that treatment of this uncertainty has minimal impact on the application because the seismic risk is small compared to the overall risk. However, the NRC staff notes that for 10 CFR 50.69 categorization the uncertainty associated with crediting FLEX equipment and actions impacting the seismic importance of SSCs and could skew the integrated importance of certain SSCs.

In light of these observations:

i. Describe the modeling assumptions and sources of uncertainty associated with crediting FLEX.

ii. Provide justification that the HEP 95th percentile value sensitivity study is sufficient to address the uncertainty associated with crediting FLEX actions and equipment in the SPRA.

iii. If the HEP 95th percentile value sensitivity study cannot be justified to be sufficient to address the uncertainty associated with crediting FLEX actions and equipment in the SPRA in response to part (iii) above, then propose a mechanism to ensure that a separate acceptable sensitivity study is performed as part of 10 CFR 50.69 categorization to address the use of FLEX equipment in the SPRA.

iv. 10 CFR 50.69(d)(1) requires the licensee to ensure that RISC-2 SSCs perform their functions consistent with the categorization process. Describe the approach that will be used to ensure that FLEX equipment (explicitly and implicitly modeled) that is categorized as RISC-2 will continue to perform their mitigation functions. Include discussion of any enhancement to current treatment of such equipment that may result from the categorization process.

50.69 Question 06 (APLA) - Addition of FLEX to the PRA Model

There are several challenges associated with incorporating FLEX strategies into PRA models. The NRC memorandum dated May 30, 2017, "Assessment of the NEI 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 identified challenges and strategies for incorporating FLEX equipment into a PRA model in support of RI decision making in accordance with the guidance of RG 1.200. For the NRC staff to assess the potential incorporation of FLEX equipment into the PRA model(s) provide the following:

a. State whether FLEX equipment and strategies have been credited in the IEPRA, FPRA, and/or the SPRA. If not incorporated, or their inclusion is not expected to impact the PRA results used in the 10 CFR 50.69 categorization process, no additional response is requested.

b. If the equipment or strategies have been credited, and their inclusion is expected to impact the PRA results used in the 10 CFR 50.69 categorization process, provide the following information separately for IEPRA (includes internal flooding), FPRA, SPRA, and external hazards screening as appropriate:

   i. A discussion detailing the extent of incorporation (i.e. 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).

   ii. A discussion detailing the methodology used to assess the failure probabilities of any modeled equipment credited in the mitigating strategies (i.e., FLEX). The discussion should include a justification explaining the rationale for parameter values, and whether the uncertainties associated with the parameter values are considered in accordance with the ASME/ANS PRA Standard as endorsed by RG 1.200 Revision 2.

   iii. A discussion detailing the methodology used to assess operator actions related to FLEX equipment and the personnel that perform these 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 supporting requirement HR-G3 of the ASME/ANS RA-Sa-2009 PRA standard were evaluated.

      2. Whether maintenance procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and if 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. If the procedures governing the initiation or entry into mitigating strategies are ambiguous, vague, or not explicit, 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 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 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;

50.69 Question 07 (APLA) – LAR Text Clarification

LAR Section 3.1.1 states in part, the PRA-based evaluations used are IE, internal flooding, and FPRAs, and non-PRA approaches used are fire safe shutdown equipment list (FSEL) and seismic safe shutdown equipment list (SSEL). In addition, LAR Section 3.2.2 states that a FPRA will be used and LAR Section 3.2.3 states that a SPRA will be used in the categorization process.

Confirm that Section 3.1.1 should reflect the use of the FPRA and SPRA, as opposed to the FSEL and the SSEL.

50.69 Question 08 (APLB) – Seismic PRA Screened SSCs

Section 5.1 of NEI 00-04 provides guidance on the use of importance measures for identifying the "candidate safety significance" of components during the categorization process. Based on the information provided in the LAR, the NRC staff was unable to determine whether the guidance regarding importance measures for identifying candidate safety significance were followed, therefore, the NRC staff requests that the licensee describe whether a screening level is used in the SPRA and how the potential use of the screening level maintains consistency with the importance measure criteria in NEI 00-04 or justify any deviations from the guidance by using the selected screening level. This justification may include demonstration of the impact of the selected screening level in the SPRA on the importance measure criteria and the categorization of SSCs.

50.69 Question 09 (APLB) – Use of Addendum B of the PRA Standard (2013)

Paragraph 50.69(c)(1)(i) of 10 CFR requires the PRA must be of sufficient quality and be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

LAR Section 3.2.3 states that the SPRA model was peer reviewed using the requirements in Addendum B of the PRA Standard (2013), (ASME/ANS RA-Sb–2013, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications”, which has not been endorsed by the NRC. LAR Section 3.2.3 references discussion in the Vogtle 10 CFR 50.69 LAR (ADAMS Accession No. ML17173A875), as supplemented, justifying use of Addendum B based on an assessment of the differences between Addendum A and B SRs. That assessment included evaluation of the Vogtle SPRA to

Addendum A for SRs identified to be different from or not encompassed by the requirements in the Addendum B SRs. The 10 CFR 50.69 LAR indicates that the SPRA peer review based on Addendum B can be justified using the Vogtle experience.

Section 3.3.1.1 of the Vogtle 10 CFR 50.69 Safety Evaluation (SE) (ADAMS Accession No ML18144A647) accepted the use of the 2013 PRA Standard based on (1) the discussion provided in the Vogtle LAR, (2) Vogtle's comparison of Addendum B to Addendum A SRs in a report titled "Response to Supplemental Information Needed for Acceptance of Systematic Risk-Informed Assessment of Debris Technical Report (ADAMS Accession No. ML17192A245); and (3) a response to a request for additional information clarifying the acceptability of a practice used in the Vogtle SPRA associated with SR SFR-C6. The assessment of debris report defined four comparison categories: (1) "Addendum B Equates to Addendum A", (2) Addendum B Envelopes Addendum A," (3) "Vogtle Conforms to Addendum A," and (4) "Vogtle Conforms to Accepted Current Practices." The NRC staff notes that the first two comparison categories concern generic resolutions and, therefore, are expected to apply to the Hatch SPRA, but the remaining two comparison categories (i.e., "Vogtle Conforms to Addendum A, and "Vogtle Conforms to Accepted Current Practices") concern plant specific resolutions. The NRC staff also notes that these two later categories were only applied to a limited set of SRs (i.e. SHA-B3, SHA-C3, SFR-C3, SFR-G3, SPR-B1, and SFR-C6).

In light of the observations above, confirm that the generic resolutions are applicable to the SPRA and provide plant specific justification that the SPRA is in conformance with Addendum A SRs SHA-B3, SHA-C3, SFR-C3, SFR-G3, SPR-B1 and SFR-CR (C-II where it applies); or that the SPRA conforms to an industry practice considered more current than the practice required by Addendum A (e.g., like Vogtle did for SR SFR-C6).

50.69 Question 10 (APLB) - Seismic PRA Peer Review Criteria

Paragraph 50.69(c)(1)(i) of 10 CFR requires the PRA must be of sufficient quality and be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. LAR Section 3.3 states that the PRA models have been assessed against RG 1.200, Revision 1.

Section 2.2 of RG 1.200 provides regulatory guidance regarding peer reviews and the staff regulatory position on NEI 00-02, 05-04, and 07-12. NRC letter, "U.S. Nuclear Regulatory Commission Acceptance of NEI Guidance NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," (August 2012)," dated March 7, 2018 (ADAMS Accession No. ML18025C025), provides the staff comments on this guidance for seismic and external hazard PRA peer reviews.

Based on the information provided in the LAR the NRC staff was unable to determine if the SPRA peer review and focused scope peer review considered the NRC staff's comments in the March 7, 2018 letter, therefore the NRC staff requests that the licensee discuss whether the SPRA peer review and focused-scope peer review considered the NRC staff's comments in the March 7, 2018 NRC acceptance letter. In addition, provide justification for not considering specific comments in the acceptance letter in the context of this application if applicable.

50.69 Question 11 (APLB) – Seismic PRA Integrated Risk Sensitivity Study

Paragraph 50.69(b)(2)(iv) of 10 CFR requires that each application include a description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv). The

evaluations shall include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms from both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions). Paragraph 50.69(c)(1)(iv) of 10 CFR requires that the categorization process include evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and any potential increase in CDF and LERF resulting from changes in treatment are small. Paragraph 50.69(e)(3) of 10 CFR requires the licensee to consider the data collected for RISC-3 SSCs to determine if there are any adverse changes in performance such that the SSC unreliability values approach or exceed the values used in the evaluations conducted to satisfy 10 CFR 50.69(c)(1)(iv).

Section 8, "Risk Sensitivity Study," of NEI 00-04 states, "[t]he overall risk evaluation process addresses both known degradation mechanisms and common cause interactions." The sensitivity study is driven by the need to verify "that changes in treatment should not significantly degrade performance for RISC-3 SSCs and should maintain or improve the performance of RISC-2 SSCs."

The NRC staff notes that the categorization of SSCs using the SPRA is dominated by structural failure modes which are dependent on modeling inputs such as the "dominant failure modes" and "fragility curves." These modeling inputs are derived from a number of sources including the SSC design, testing, and as-built installation, all of which can be impacted by alternative treatments. The NRC staff notes that NEI 00-04 Section 8 specifies performing a sensitivity study by increasing the unreliability of all Low Safety Significant (LSS) SSCs by a factor of 3 to 5 to provide an indication of the potential trend in CDF and LERF, if there were degradation in the performance of all LSS SSCs. Based on the information provided in the LAR, it is not clear how the licensee will perform such a sensitivity study to address non-random failures such as structural failures caused by a seismic event

Paragraph 50.69(d)(2) of 10 CFR requires the licensee to ensure that RISC-3 SSCs remain capable of performing their functions under design basis conditions, including conditions that occur during seismic events. The guidance in NEI 00-04, Section 5.6 states "if a seismic PRA is used, SSCs may have been screened out of the PRA due to inherent seismic robustness. For such screened SSCs, regardless of their categorization outcome, it is important that the inherent seismic robustness that allows them to be screened out of the seismic PRA be retained."

Based on the preceding discussion, it is unclear to the NRC staff how the licensee will perform the required risk sensitivity study for the 10 CFR 50.69 categorization using the SPRA to meet the requirements of 10 CFR 50.69(b)(2)(iv) and 10 CFR 50.69(c)(1)(iv), and how the modeling inputs in the SPRA and those used for the risk sensitivity study continue to remain valid to ensure compliance with the requirements of 10 CFR 50.69(e). Provide the following:

a. A description, with justification, of the sensitivity study that will be performed to demonstrate conformance with 10 CFR 50.69(b)(2)(iv) and 10 CFR 50.69(c)(1)(iv) for those SSCs that may be classified as RISC-3 based, in part, on the SPRA results.

b. Explain how the inherent seismic robustness of those SSCs screened out of the PRA for that reason will continue to be ensured.

c. Provide a description of how it will be determined that the modeling inputs in the SPRA continue to remain valid to ensure compliance with the requirements of 10 CFR 50.69(e). Include in this description a discussion of the procedural guidance (e.g. plant

program) that addresses this guidance. Additionally, include the modeling inputs used for crediting FLEX equipment and actions in the SPRA, as well as the methodology used to determine the degradation of seismic robustness of SSCs given the alternative treatment.

50.69 Question 12 (APLA/APLB) - SSC Categorization Based on Other External Hazards

Paragraph 50.69(b)(2)(ii) of 10 CFR requires that the quality and level of detail of the systematic processes that evaluate the plant for external events during operation are adequate for the categorization of SSCs.

LAR Section 3.2.4 states that "[a]ll other hazards were screened from applicability to Hatch Units 1 and 2 per a plant-specific evaluation in accordance with the criteria in Section 6 of ASME PRA Standard RA-Sb-2013." This statement appears to indicate that Hatch proposes to treat all SSCs as low-safety-significant (LSS) with respect to other external events risk. The LAR provides no further explanation of how the risk for other external hazards will be considered in 10 CFR 50.69 categorization (i.e., components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario). LAR Attachments 4 and 5 provide a summary of the other external hazards screening results, but do not appear to address any considerations related to applying Figure 5-6 of NEI 00-04 guidance to those hazards. Considering these observations, address the following:

a. LAR Section 3.2.4 states that external hazards were screened using the criteria in Section 6 of the 2013 PRA Standard ASME/ANS RA-Sb-2013. LAR Attachment 5 however appears to list the criteria from the 2009 PRA Standard ASME/ANS RA-Sa-2009. Clarify and justify the criteria used for the screening of external hazards.

b. Identify the external hazards that will be evaluated according to the flow chart in NEI 00-04, Section 5.4, Figure 5-6. Provide detailed justification for screening external hazards (i.e., external flood, high winds, and tornados) using the criteria described in part a above. As applicable, the justification should include consideration of uncertainties in the determination of demonstrably conservative mean values, as discussed in Section 6.2-3 of the ASME/ANS RA-Sa-2009 PRA Standard.

c. LAR Attachment 4 states, regarding the extreme wind and tornado hazard, that "[c]alculations show that the initiator probability is 3.3E-06 and the CCDP is 1E-03." Provide detailed justification for concluding that for the high winds and tornados hazard, the screening criterion PS3 applies, i.e., the mean frequency is less than 1×10^{-5} per reactor-year and the mean conditional core damage probability is less than 0.1.

d. Figure 5-6 of NEI 00-04 shows that if an SSC is included in a screened scenario, then in order for that SSC to be considered a candidate LSS, the licensee has to show that if the component was removed, the screened scenario would not become unscreened.

    i. Identify and justify what type of SSCs, if any, are credited in the screening of the external hazard(s), including both passive, active, and temporary features.

    ii. If there are any SSCs credited for screening of the external hazard(s), then explain and justify how the guidance in Figure 5-6 of NEI 00-04 will be applied for each of the external hazard(s).

e. If the external hazards (i.e., high winds and tornados) cannot be screened out in item (a) above, discuss, using quantitative or qualitative assessments, how the risk from those hazards will be considered in the categorization program. The discussion should include consideration of and, as applicable, the basis for the following factors:

- The frequency of the external hazard(s),
- The impact of the external hazard(s) on plant SSCs and plant's operation including the ability to respond to the external hazard initiating event,
- The operating experience associated with reliability of the external hazard(s) protection measures, and
- The reliability of operator actions.

50.69 Question 13 (APLA/APLB) – Integrated One-Top PRA Hazards Model

NEI 00-04, Section 5.6, "Integral Assessment," discusses the need for an integrated computation using the available importance measures. It further states, in part, that the "integrated importance measure essentially weights the importance from each risk contributor (e.g., IE, FPRA, and SPRAs) by the fraction of the total CDF [or LERF] contributed by that contributor." The guidance provides formulas to compute the integrated Fussell-Vesely (FV), and integrated Risk Achievement Worth (RAW).

To confirm that the importance measures generated for use in the 10 CFR 50.69 process are consistent with the NEI guidance and do not inadvertently introduce a deviation from the computations for FV and RAW provided in the NEI 00-04 guidance, as endorsed by RG 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" (ADAMS Accession No. ML061090627):

a. Explain whether the PRA model that will be used in the 10 CFR 50.69 categorization process is an integrated one-top model across multiple PRA hazards and if the integrated one-top model includes accident sequence(s) modeling to support quantification of both CDF and LERF.

b. If using an integrated one-top model across multiple PRA hazards for the 50.69 categorization process, provide the following:

   i. Discuss the process used to validate and confirm the integration of the PRA hazards into a one-top model to ensure that after the PRA model change was performed, SRs QU-F2 and SR FQ-F1 continue to be met (e.g., cutset reviews, identification of non-minimal cutsets, peer review).

   ii. Discuss how the individual importance measures (i.e., FV and RAW) for the PRA one-top all hazards model are derived considering the different hazards, specifically for those hazards that discretized SSC functions into 'bins'. The discussion should include justification of why the importance measures generated do not deviate from the NEI guidance. If the practice or method used to generate the integrated importance measures is determined to deviate from the NEI guidance, justify why the integrated importance measures computed are appropriate for use in the categorization process.

50.69 Question 14 (APLA) – Implementation Items

10 CFR 50.69(b)(2)(ii) 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 evaluates 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 of the 50.69 Questions 01 through 13 above 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. An example would be a table of listed implementation items referenced in a license condition.

As an alternative to providing an implementation item for an F&O, demonstrate that the F&O(s) will have no adverse impact and/or insignificant impact on the 10 CFR 50.69 categorization process.

50.69 Question 15 (APLA) – Proposed License Condition

The guidance in NEI 00-04 allows licensees to implement different approaches, depending on the scope of their PRA (e.g., the approach if a seismic margins analyses is relied upon is different and more limiting than the approach if a SPRA is used). Regulatory Guide 1.201, Revision 1 states that "as part of the NRC's review and approval of a licensee's or applicant's application requesting to implement §50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non- PRA methods used in the licensee's categorization approach."

LAR Section 2.3 proposes the following License Condition:

> Southern Nuclear Operating Company is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) specified in the renewed license amendment dated DATE.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a SPRA approach).
>
> The Southern Nuclear Operating Company shall complete all items listed in Attachment 1, List of Categorization Prerequisites, of Southern Nuclear Operating Company letter ML Number, dated DATE, prior to implementation.

The proposed license condition does not explicitly address the PRA and non-PRA methods that were used.

Provide a license condition that explicitly address the approaches, e.g.:

[LICENSEE] is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire, and seismic hazards; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the [DESCRIBE..] for other external hazards; as specified in Unit 1 and Unit 2 License Amendment No. [XXX] dated [DATE].

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

[LICENSEE] shall complete all items listed in Attachment 1, List of Categorization Prerequisites, of [LICENSEE] letter ML Number, dated DATE, prior to implementation.

Note: The license condition may need to be expanded to address any implementation items identified in response to the questions.