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

July 11, 2018

Mr. Steven Capps
Senior Vice President
Nuclear Corporate
Duke Energy Corporation
526 South Church Street, EC-07H
Charlotte, NC 28202

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 AND SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – AUDIT PLAN QUESTIONS RE: APPLICATION TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0008 AND L-2018-LLA-0034)

Dear Mr. Capps:

By letters dated January 10 and February 1, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18010A344 and ML18033B768, respectively), Duke Energy Progress, LLC (Duke Energy) submitted license amendment requests for Brunswick Steam Electric Plant, Units 1 and 2, and Shearon Harris Nuclear Power Plant, Unit 1. The proposed amendments would modify the licensing basis to allow for the implementation of the provisions of Section 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants," of Title 10 of the *Code of Federal Regulations* (10 CFR).

The U.S. Nuclear Regulatory Commission (NRC) staff will conduct a regulatory audit to support its review of the proposed license amendments. The audit will be conducted at Duke Energy's corporate office in Charlotte, North Carolina, on July 17-19, 2018. The NRC staff's audit plan was issued on July 2, 2018 (ADAMS Accession No. ML18180A418). The questions enclosed are a supplement to the audit plan.

If you have any questions, please contact me at 301-415-2760 or Martha.Barillas@nrc.gov.

Sincerely,

Dennis J Galvin for

Martha Barillas, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-325, 50-324, and 50-400

Enclosures:
1. Brunswick Regulatory Audit Plan Supplement Questions
2. Shearon Harris Regulatory Audit Plan Supplement Questions

cc: Mr. William R. Gideon
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REGULATORY AUDIT PLAN QUESTIONS BY THE OFFICE OF NUCLEAR REACTOR REGULATION TO SUPPORT THE REVIEW OF LICENSE AMENDMENT REQUEST FOR BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS"

DOCKET NOS. 50-325 AND 50-324

1.0 BACKGROUND

By letter dated January 10, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18010A344), Duke Energy Progress, LLC (Duke Energy, the licensee), submitted a license amendment request (LAR) for Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The proposed amendment would modify the licensing basis to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants," and provide the ability to use probabilistic risk assessment (PRA) models, the internal events PRA (IEPRA), internal flooding PRA (IFPRA), and internal fire PRA (FPRA) for the proposed 10 CFR 50.69 categorization process.

Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006 (ADAMS Accession No. ML061090627), endorses, with regulatory positions and clarifications, the Nuclear Energy Institute (NEI) guidance document NEI 00-04, Revision 0, "10 CFR 50.69 SSC [Structure, System, and Component] Categorization Guideline," July 2005 (ADAMS accession No. ML052910035), as one acceptable method for use in complying with the requirements in 10 CFR 50.69. Both RG 1.201 and NEI 00-04 cite RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," February 2004 (ADAMS Accession No. ML040630078), which endorses industry consensus PRA standards, as the basis against which peer reviews evaluate the technical acceptability of a PRA. Revision 2 of RG 1.200 issued March 2009 is available at ADAMS Accession No. ML090410014.

Section 3.1.1 of the LAR states that Duke Energy will implement the risk categorization process of 10 CFR 50.69 in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201. However, the licensee's LAR does not contain enough information for the U.S. Nuclear Regulatory Commission (NRC) staff to determine if the licensee has implemented the guidance appropriately in NEI 00-04, as endorsed by RG 1.201, as a means to demonstrate compliance with all of the requirements in 10 CFR 50.69. The NRC staff has the following questions to be discussed during the regulatory audit.

2.0 REGULATORY AUDIT PLAN QUESTIONS

Question 1 – Facts and Observations Closure

Section 3.3 of the LAR states that the licensee conducted a facts and observations (F&O) closure review in August 2017 for open F&Os from the IEPRA, IFPRA, fire PRA (FPRA), and high winds PRA (HPRA) peer reviews and self-assessments in accordance with Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13, as accepted by NRC letter dated May 3, 2017 (ADAMS Accession No. ML17079A427). In addition, the licensee states the results of the F&O closure review are available to the NRC staff for audit. Provide the following clarifications with respect to implementation of the Appendix X F&O closure process:

a. Confirm that the licensee provided the closure review team a written assessment and justification of whether the resolution of each F&O, within the scope of the independent assessment, constitutes a PRA upgrade or maintenance update, as defined in American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008," as endorsed by RG 1.200 Revision 2.

b. Section X.1.3 of the Appendix X guidance includes the following five criteria for selecting members of the closure review team:

    1. Every member of the independent assessment team should be independent of the PRA associated with the F&Os being reviewed, per the criteria of "independent" in the ASME/ANS PRA Standard. These members may be contractors, utility personnel, or employees of other utilities, and may include members of peer review teams that previously reviewed the models being assessed.

    2. Every member of the independent assessment group should meet the relevant peer reviewer qualifications as stated in the ASME/ANS PRA Standard for the technical elements associated with the F&Os being reviewed.

    3. The overall review team experience includes two qualified reviewers for each F&O. An exception to this is allowed for the closure of an F&O related to a single supporting requirement (SR), in which case, a single independent reviewer is acceptable, in alignment with the peer review guidance in accordance with the ASME/ANS PRA Standard.

    4. Each member of the independent assessment team should be knowledgeable about the F&O independent assessment process used to assess the adequacy of the F&O resolution.

    5. The total number of reviewers is a function of the scope and number of finding F&Os to be reviewed for closure

    Describe how the selection of members for the August 2017 independent assessment met these criteria.

c. Explain how closure of the F&Os was assessed to ensure that the capabilities of the PRA elements, or portions of the PRA within the elements, associated with the closed F&Os now meet Capability Category (CC)-II for all the applicable SRs of ASME/ANS RA-Sa-2009 as endorsed by RG 1.200, Revision 2.

d. Discuss whether the F&O closure review scope included all finding-level F&Os, including those finding-level F&Os that are associated with "Met" SRs. If not, identify and provide detailed descriptions for any F&Os that were excluded from the F&O closure review scope, and their disposition for the application.

e. For the F&Os and self-assessment items that remain open, provide the complete text of the original peer review F&O, self-assessment finding, and Appendix X F&O closure review assessment comments and recommendations. If necessary, provide revised dispositions for each F&O and self-assessment item that addresses how resolution to each element of the F&O/self-assessment item impacts the SSC categorization process.

Question 2 – Open/Partially Open Findings in the Process of Being Resolved

Section 4.2 of RG 1.200 states that the LAR should include a discussion of the resolution of the peer review F&Os that are applicable to the parts of the PRA required for the application. This decision should take the following forms:

- A discussion of how the PRA model has been changed, and
- A justification in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application decision were not adversely impacted (remained the same) by the particular issue.

Attachment 3 of the LAR, "Disposition and Resolution of Open Peer Review Finding and Self-assessment Open Items," provides F&Os and self-assessment findings that are still open or partially resolved following the August 2017 F&O closure review. The dispositions for several of these F&Os do not contain sufficient information to justify the licensee's conclusion that the open F&O has minimal or no impact on the application. The specific F&Os and requested information are as follows:

a. <u>Internal events F&O 1-19</u> pertaining to component failure data. This F&O description states that the component failure data values documented in BNP-PSA-049 were developed during a previous PRA update and that some values may need to be updated to be consistent with changes in the Level 1 PRA data. The licensee's disposition states that "[o]nly 4 events were found and all of them had either a [Fussell-Vesely (FV)] in the x10-3 range or a [Risk Achievement Worth (RAW)] of 1. Because of the small number of events that could have a need to be updated but were not, the relatively low value of FV for three of the retained events, and the relatively low RAW value on the remaining event, the effect on 50.69 applications is negligible." The response does not clarify how it was determined what events might need to be updated, only that there were four events identified. Furthermore, the SSC will be low safety significant (LSS) if the

RAW<2 AND the FV< 0.005, but the conclusion that updates were not needed seems to be based on the argument the RAW OR the FV is low.

1. Clarify how the check was performed to determine if any data needed to be updated including how the conclusion was reached that "only 4 events were found."

2. Provide the RAW and FV of the four events, and some indication of the change in the failure likelihood expected from a data update. Based on the data provided, indicate the expected changes to the RAW and FV values.

3. Alternatively, propose a mechanism that ensures F&O 1-19 will be resolved in accordance with an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review) prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also specify how the F&O will be resolved in the PRA (e.g., provide an explicit description or a reference to the appropriate section of the LAR).

b. Internal events F&O 3-6 pertaining to human reliability analysis (HRA). This F&O description notes a specific issue related to the HRA calculation for event OPER-DCDG, specifically, no execution failure probabilities were assigned to the tasks of starting and connecting the diesel generator (DG). Additionally, the calculation may not have considered all of the necessary breaker manipulations. The licensee's resolution states that the standard is met and this is an opportunity for enhancement to the documentation and does not affect the core damage frequency (CDF) or the risk metrics. However, the licensee's resolution did not directly address this specific issue, which does not appear to be just a documentation issue.

1. Explain how the finding concern related to OPER-DCDG was resolved and clarify if the model of record (MOR) has been updated to incorporate this resolution.

2. If the human failure events (HFEs) for starting and connecting the DG and for the breaker manipulations have not been updated in the MOR, justify that excluding the updated operator actions does not impact the results of the 10 CFR 50.69 categorization process.

3. Alternatively, propose a mechanism that ensures F&O 3-6 will be resolved in accordance with an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review) prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also specify how the F&O will be resolved in the PRA (e.g., provide an explicit description or a reference to the appropriate section of the LAR).

c. Internal flooding F&O IFSN-A8 pertaining to the effects of expansion joint failures. The F&O description notes that no propagation from gaskets or expansion joints was modeled in the IFPRA. The licensee's disposition states that the circulating expansion joints are not risk significant to the BSEP IFPRA risk as circulating water piping does not contribute a significant amount to CDF/large early release frequency (LERF) and circulating water expansion joint ruptures represent a small portion of the total rupture frequency for IFPRA. Although this modeling exclusion may have a small impact on the total risk, its inclusion could potentially increase the risk importance values for certain

system components above the threshold criteria for determining high safety significance (HSS).

1. Justify that the circulating expansion joint modeling exclusion cited above does not impact the results of the 10 CFR 50.69 categorization process.

2. Alternatively, ensure that the circulating expansion joint modeling is incorporated into the PRA and appropriate documentation is provided to the NRC staff for review.

d. Internal events F&O QU-C2-1 pertaining to human reliability analysis. The F&O description indicates that some joint human failure events (JHFEs) may be assigned a floor value of 1E-6, and suggested that these cutsets be evaluated to determine the appropriateness of this value. In contrast, the reported disposition states "in examining the top 95% cutsets, there were some cutsets with 5 and 6 human error probabilities (HEP) events that were not explicitly analyzed for dependencies." It is not clear whether "not explicitly analyzed" means the floor value was assigned, or there was no justification of a result that was less than a floor value.

For performing HRA dependency analysis, NUREG-1921, "EPRI [Electric Power Research Institute]/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report," July 2012 (ADAMS Accession No. ML12216A104), discusses the need to consider a minimum value for the joint probability of multiple HFEs, and refers to NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," April 2005 (ADAMS Accession No. ML051160213) (Table 2-1), which recommends joint human error probability (JHEP) values should not be below 1E-5. Table 4-3 of EPRI Technical Report 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," October 2010, provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. Therefore, the available guidance provides for assigning joint HEPs that are less than a minimum value but only through assigning proper levels of dependency. Cutsets with JHEP values less than the minimum value should be individually reviewed for timing, cues, etc. to check the dependency between all the operator actions in the cutset. Provide the following:

1. Summarization of the evaluation of JHEPs done on the cutsets. If the assignment of JHEPs is different for different order cutsets, identify all different orders and the rules applied to each order.

2. An estimate of the number of these JHEP values below 1E-5 in the FPRA, discuss the range of values, and provide at least two different examples of justification for less than minimum values.

3. Provide an estimate of the number of these JHEP values below 1E-6 in the IEPRA, discuss the range of values, and provide at least two different examples of justification for less than minimum values

e. Internal flooding F&O IFEV-A5 pertaining to internal flood pipe break frequencies. The F&O description states that a new methodology was applied to use pipe length, and flood and major flood frequency based on diameter and flow rate and that the analysis only applied major flood frequencies to large pipe, omitting flood frequency from large pipe which is the dominant frequency. In addition, the description notes that the break

frequencies used in the calculation are applied incorrectly in the analysis. The licensee's disposition states that since the flooding frequency data in the calculation and the EPRI data have different pipe size breakpoints, the pipe size interval were adjusted to match. The corresponding frequencies were then adjusted by the ratio of new EPRI flood and major flood frequency to existing major flood frequency. The appropriate multiplier was then applied to each scenario based on pipe size and fluid system type. The licensee disposition does not address the use of a new method or the incorrect application of the break frequencies.

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

1. A summary description and basis for determining the internal flood scenario initiating event frequencies. Include in this discussion the reference(s) for the methodology.

2. A discussion on whether the methodology used constitutes a "PRA upgrade" as defined in the PRA Standard (i.e., ASME/ANS RA-Sa-2009), as endorsed by RG 1.200. If the use of the methodology is considered a PRA "upgrade," then complete a focused-scope peer review of the upgrade and provide the NRC the disposition of any resulting F&Os prior to implementing the 10 CFR 50.69 categorization process.

f. Fire F&O 1-34 pertaining to fire barrier failure probabilities. The F&O description notes that a screening value for rated barrier probability of 1E-2 was applied in the PRA and that this value may not be bounding depending on the features of the barrier. The licensee's disposition states that the 0.1 barrier failure probability was inappropriately applied for certain fire compartment combinations where the partitioning element was open and that it is expected to have no more than a minimal impact on the 50.69 application. Although this incorrect modeling may have a small impact on the total risk, it could potentially increase the risk importance values for certain system components above the threshold criteria for determining HSS. Provide the following:

1. A justification that use of the incorrect fire barrier failure probability exclusion cited above does not impact the results of the 10 CFR 50.69 categorization process.

2. Alternatively, ensure that the updated fire barrier failure probability modeling is incorporated into the PRA and appropriate documentation is provided to the NRC staff for review.

g. Internal fire F&O 4-1 pertaining to fire severity factors. The licensee's disposition states that the treatment of motor control centers (MCCs) is not in accordance with FAQ 14-0009 and that in lieu of an accepted generic method, BSEP used the analysis method piloted at HNP, but that the impact on the 50.69 application is expected to be small. However, though this modeling may have a small impact on the total risk, its

inclusion could potentially increase the risk importance values for certain system components above the threshold criteria for determining HSS.

1. Justify that the incorrect modeling of MCCs cited above does not impact the results of the 10 CFR 50.69 categorization process.

2. Alternatively, ensure MCC modeling utilizing a method acceptable to the NRC is incorporated into the PRA and appropriate documentation is provided to the NRC staff for review.

h. Internal fire F&O 6-4 pertaining to fire barrier failure for multi-compartment analysis. The F&O description describes issues with calculating failure probability of passive fire barriers in the multi-compartment analysis. The disposition is incomplete and, as a result, the NRC staff is unable to assess the disposition to this F&O and the licensee's conclusion that it has "no more than a minimal impact."

1. Clarify the disposition of this F&O.

2. If the disposition does not update the FPRA to resolve the F&O and meet CC-II for SR FSS-G4, provide justification that the resolution does not impact the results of the 10 CFR 50.69 categorization process. [The NRC staff notes that a small impact on the total risk could potentially increase the risk importance values for certain system components above the threshold criteria for determining HSS.]

3. Alternatively, ensure that appropriate resolution is incorporated into the PRA and appropriate documentation is provided to the NRC staff for review.

Question 3 – Determining No Impact on Categorizations

Several F&O resolutions determine that the resolution "will not impact CDF/LERF or the component categorization under 10 CFR 50.69" and others state there will be "no impact on component categorization under 10 CFR 50.69." Minor or no impact on CDF and LERF is fairly straightforward to determine but it is not clear that minor or no impact on CDF/LERF will always lead to no impact on categorization. In addition, excessively conservative assumptions such as setting probabilities to 1.0 may also impact categorization. Describe how the conclusion is reached that there is no impact on component categorization.

Question 4 – Qualitative Function Categorization

Table 3-1 of the LAR indicates that the evaluation of the seven qualitative criteria defined in Section 9.2 of NEI 00-04 is performed at the function level and prior to the Integrated Decision-making Panel (IDP). The LAR states that NEI 00-04 only requires the seven qualitative criteria to be completed for components/functions categorized as LSS. Table 3-1 of the LAR contains the entry "Allowable" at the intersection of the "IDP change HSS to LSS" column and "Qualitative Criteria" row, which appears to contradict the premise that the seven criteria are only applied to LSS functions. The guidance in NEI 00-04 states that the IDP should consider the impact of loss of the function/SSC against the remaining capability to perform the basic safety functions. Discuss categorization guidance that will be provided to assess the safety significance of a function when there is an impact on, or even a loss of, the capability

described in each of the seven criteria (e.g., Is a single false response sufficient to assign the function HSS?)

Question 5 – Passive Component Categorization Process

Section 3.1.2 of the LAR states that passive components and the passive function of active components will be evaluated using the method for risk-informed repair/replacement activities consistent with the safety evaluation issued by the Office of Nuclear Reactor Regulation, "Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems, Third and Fourth 10-Year In-service Inspection Intervals," for Arkansas Nuclear One, Unit 2 (ANO-2), dated April 22, 2009 (ADAMS Accession No. ML090930246). The LAR further states that this methodology will be applied to determine the safety significance of Class 1 SSCs.

This methodology has only been approved for Class 2 and Class 3 SSCs. Because Class 1 SSCs constitute principal fission product barriers as part of the reactor coolant system or containment, the consequence of pressure boundary failure for Class 1 SSCs may be different from Class 2 and Class 3 SSCs and, therefore, the criteria in the ANO-2 methodology cannot automatically be generalized to Class 1 SSCs without further justification.

The LAR does not justify how the ANO-2 methodology can be applied to Class 1 SSCs and how sufficient defense-in-depth and safety margins are maintained. A technical justification for Class 1 SSCs should address how the methodology is sufficiently robust to assess the safety significance of Class 1 SSCs, including, but not limited to: (1) justification of the appropriateness of the conditional core damage probability (CCDP) numerical criteria used to assign 'High', 'Medium' and 'Low' safety significance to these loss-of-coolant initiating events; (2) identification and justification of the adequacy of the additional qualitative considerations to assign 'Medium' safety significance (based on the CCDP) to 'High' safety significance; (3) justification for crediting operator actions for success and failure of pressure boundary; (4) guidelines and justification for selecting the appropriate break size (e.g., double-ended guillotine break or smaller break); and (5) include supporting examples of types of Class 1 SSCs that would be assigned low safety significance, etc.

As mentioned in the March 13, 2018, meeting summary for the February 20, 2018, Risk-Informed Steering Committee (RISC) meeting (ADAMS Accession No. ML18072A301), the NRC staff understands that the industry is planning to limit the scope to Class 2 and Class 3 SSCs, consistent with the pilot Vogtle Electric Generating Plant, Units 1 and 2, license amendment dated December 17, 2014 (ADAMS Accession No. ML14237A034).

Please provide the requested technical justification or confirm the intent to apply the ANO-2 passive categorization methodology only to Class 2 and Class 3 equipment.

Question 6 – Identifying Key Assumptions and Uncertainties that could Impact the Application

Section 3.2.7 of the LAR states that the detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," March 2009 (Revision 0) (ADAMS Accession No. ML090970525), and Section 3.1.1 of EPRI Technical Report (TR)-1016737. The NRC staff notes that one of these sources has been

superseded by a revision (Revision 1 of NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," March 2017 (ADAMS Accession No. ML17062A466), which references the updated EPRI guidance TR-1026511, "Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty" (2012)).

Attachment 6 of the LAR only contains nine assumptions/uncertainties from five PRA models, whereas industry guidance documents such as NUREG-1855, Revision 1, and EPRI TR-1026511 address a large number of potential assumptions and uncertainties. For example, only one fire modeling assumption/uncertainty (page 56) in the LAR is provided as a source of uncertainty, compared to the 2012 EPRI document, which identifies 71 potential sources of uncertainty. There appear to be no uncertainties or assumptions associated with LERF and internal flooding, and one source that relates to both high winds and external flooding. Some of the discussions in Attachment 6 do not provide enough detail to determine the exact nature of the uncertainty or assumption in relation to this application. For example, the LAR states for direct current power availability assessment of battery depletion times, the associated accident sequence timing, and the related success criteria were included in the uncertainty assessment. However, there is no discussion on what depletion times, sequence timing, and no success criteria assumptions were specifically assessed in the uncertainty assessment that may potentially need to be addressed in a sensitivity study. Provide the following:

a. A detailed summary of the process used to determine the nine sources of uncertainties and assumptions presented in Attachment 6 of the LAR. Include in this discussion an explanation of how the process is in accordance with NUREG-1855, Revision 1, or other NRC-accepted method. Also, include in the discussion a detailed description of how the final set of nine uncertainties and assumptions were developed from the initial comprehensive list of PRA model(s) uncertainties and assumptions.

b. A justification for why just one or no key sources of uncertainty or assumptions were identified for the internal flooding, LERF, fire, high winds, and external flooding PRA models.

c. If the process of identifying key sources of uncertainty or assumptions for these PRA models cannot be justified, provide the results of an updated assessment of key sources of uncertainty or assumptions.

d. A description of the specific assumptions and sources of uncertainty key to this application for the entries in LAR Attachment 6 in enough detail that its impact on the application can be clearly understood and that a specific sensitivity could be defined to examine the risk significance of the issue. Include in this description for any new sources of uncertainty or assumptions identified in Part c.

Question 7 – Key Assumptions and Uncertainties that could Impact the Application

The licensee's dispositions presented in Attachment 6 of the LAR for key assumptions and modeling uncertainties state in each case that they do not represent a key source of uncertainty and will not be an issue for the 50.69 calculations. However, in a number of instances, the licensee did not provide sufficient information for NRC staff to conclude that the uncertainty will

not impact the 10 CFR 50.69 risk categorization. The specific key assumptions and uncertainties, and requested information are as follows:

a. Heating, Ventilation and Air Conditioning (HVAC) model of switchgear ventilation and supporting GOTHIC analysis

Attachment 6 (page 56) of the LAR states that switchgear HVAC requirements is not a bounding case and shows only one of eight HVAC fans needed for room cooling. However, in the disposition for this assumption/uncertainty, the licensee states that screening of HVAC for switchgear rooms needs to consider the level of detail in the GOTHIC analysis. The NRC staff is unclear if HVAC support has been screened from the model or the success criterion modeled is one of eight fans. Provide the following:

1. A clarification of the actual modeling of HVAC support of the switchgear rooms and specify the assumption/uncertainty related to this modeling choice.

2. Justification that the specific assumption/uncertainty does not impact the results of the 10 CFR 50.69 categorization process.

3. Alternatively, ensure that the assumption/uncertainty resolution is incorporated into the PRA prior to implementation of the 10 CFR 50.69 risk categorization process.

b. Equipment Recovery Credit and Modeling of Control Rod Drive Support Systems

Attachment 6 (pages 56 and 57) of the LAR state, "[t]his assumption could result in some SSCs...being classified as HSS...when in fact they are LSS. Therefore this uncertainty will be addresses as individual systems are categorized in this risk-informed application." The NRC staff is unclear if the intent of the last statement is that the associated SSCs that the PRA determines to be HSS could be re-classified as LSS. Table 3-1 of the LAR states for the Internal Events Base Case that the IDP is not allowed to change risk categorization. Provide the following:

1. A clarification that for assumptions/uncertainties associated with internal events and internal flood that the impacted SSCs categorization would not be changed during the categorization process.

2. If the SSCs categorization can be changed due to these assumptions/ uncertainties provide both regulatory and technical justification.

Question 8 – 10 CFR 50.69(e), Feedback and Adjustment Process

The regulation under 10 CFR 50.69 delineates that a licensee voluntarily choosing to implement this section shall submit an application for license amendment under §50.69 that contains the following information. Part (e)(1) of §50.69, for feedback and process adjustment, states, in part, that for RISC-1, RISC-2, RISC-3 and RISC-4 SSCs, the licensee shall review changes to the plant, operational practices, applicable plant and industry operational experience, and as appropriate, update the PRA and SSC categorization and treatment process.

Section 11.2, "Following Initial Implementation," of NEI 00-04 discusses that "a periodic update of the plant PRA may affect the results of the categorization process. If the results are affected,

the licensee must make adjustments as necessary to either the categorization or treatment processes to maintain the validity of the processes." Specifically, NEI 00-04, Section 12.1 discusses cases for which, in some instances, an updated PRA model could result in new RAW and FV importance measures that are sufficiently different from those in the original categorization so as to suggest a potential change in the categorization. To address the observations above, the NRC staff requests the following additional information:

a. Explain how this periodic review will be administered. At minimum, discuss the following:

   1. Participants involved in the review;

   2. Sources of material identified to be reviewed;

   3. Periodicity for when the review will be performed;

   4. Documentation of the review performed (e.g., corrective action program, engineering evaluation, etc.); and

b. Provide the criteria to be used to determine if the change being reviewed has any impact to a modeled PRA hazard(s) and/or any SSC categorized by the 50.69 process.

Question 9 – SSCs Categorization Based on Other External Hazards

Section 3.2.4 of the LAR states, in part, that:

> As part of the categorization assessment of other external hazard risk, an evaluation is performed to determine if there are components being categorized participate in screened scenarios and whose failure would result in an unscreened scenario. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered HSS. All remaining hazards were screened from applicability and considered insignificant for every SSC and, therefore, will not be considered during the categorization process.

a. The first sentence states that assessment of external hazards "is performed" implying that the assessment will be performed. The last sentence implies that the assessment has been completed and concludes that all other external hazards will never need evaluation during categorization. Please clarify if any additional screening of external hazards will be part of the categorization process.

b. The individual plant examination of external events (IPEEE) screening process did not include the additional step illustrated in Figure 5-6 in Section 5.4 of NEI 00-04. Figure 5-6 and its associated text states that an evaluation is performed to determine if there are components being categorized that participate in screened external event scenarios whose failure would result in an unscreened scenario. Please clarify how the screening criteria in Attachment 5, Progressive Screening Approach for Addressing External Hazards, satisfy the guidelines that HSS will be assigned to SSCs whose failure would cause a screened external event scenario to become unscreened.

REGULATORY AUDIT PLAN QUESTIONS BY THE OFFICE OF NUCLEAR REACTOR REGULATION TO SUPPORT THE REVIEW OF LICENSE AMENDMENT REQUEST FOR SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS"

DOCKET NO. 50-400

1.0 BACKGROUND

By letter dated February 1, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18033B768), Duke Energy Progress, LLC (Duke Energy, the licensee), submitted a license amendment request (LAR) for Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed amendment would modify the licensing basis to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants," and provide the ability to use probabilistic risk assessment (PRA) models, the internal events PRA (IEPRA), internal flooding PRA (IFPRA), and internal fire PRA (FPRA) for the proposed 10 CFR 50.69 categorization process.

The requirements for technical adequacy of PRA models used in risk informed applications is provided in Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (ADAMS Accession No. ML090410014), on PRA technical adequacy. The RG 1.200 describes a peer review process utilizing American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008," as one acceptable approach for determining the technical adequacy of the PRA once acceptable consensus approaches or models have been established for evaluations that could influence the regulatory decision.

To complete its review, the U.S. Nuclear Regulatory Commission (NRC) staff has the following questions to be discussed during the regulatory audit.

Question 1 – Facts and Observations (F&O) Closure Process

Section 2 of RG 1.200 states for the applicable technical requirements, "the staff anticipates that current good practice, i.e. Capability Category II of the ASME/ANS standard, is the level of detail that is adequate for the majority of the applications," and that a peer review is needed to determine if the intent of the requirements in the standard is met.

LAR Section 3.3 states that an F&O closure review for internal events and internal flooding PRA was performed in March 2017 using the process documented in the draft Appendix X to Nuclear Energy Institute (NEI) 05-04, NEI 07-12, and NEI 12-13, "Close-out of Facts and Observations," published at the time of the review. The NRC staff observed the pilot application of the

Appendix X closure review process at HNP held January 31 through February 1, 2017 (ADAMS Accession No. ML17095A252).

The final version of Appendix X (February 21, 2017) was accepted by the NRC in the letter from Joseph Giitter and Mary Jane Ross-Lee to Greg Krueger, NEI, dated May 3, 2017 (ADAMS Accession No. ML17079A427), which included conditions of acceptance. This letter of acceptance was issued after the Appendix X closure review process was conducted at HNP.

LAR Section 3.3 also states that an F&O closure peer review for the fire PRA was performed in October 2017 using the process documented in Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13, "Close-out of Facts and Observations," as accepted by the NRC in a letter from Joseph Giitter and Mary Jane Ross-Lee to Greg Krueger, NEI, dated May 3, 2017.

Provide the following clarifications with respect to implementation of the Appendix X F&O closure process for both the internal events/internal flooding PRA and the fire PRA, or provide the F&O closure report to support the LAR review:

a. Clarify whether a focused-scope peer review was performed concurrently with the F&O closure process. If so, provide the following:

    1. Summary of the scope of the peer review.

    2. Detailed descriptions of any new F&Os generated from the peer review and the associated dispositions for the application.

b. The LAR identifies the two major differences between the Appendix X version used for the January 31, 2017, review and the final versions as (1) utility and review teams documentation of upgrade versus maintenance changes and (2) review teams confirmation that the underlying supporting requirements (SRs) are now met or met a Capability Category (CC)-II for closed F&Os.

    1. The LAR states that "utility portion of the upgrade verses maintenance assessment was completed globally." This is not consistent with the final guidance. Confirm that each F&O change has a documented evaluation of upgrade versus maintenance by both the utility and the review team. Provide six examples of the documented evaluations.

    2. The LAR states that "[t]he assessment team confirmed resolution of the findings allowed re-categorization of capability categories to meet or met at CC-II, as applicable." Provide six examples of the documented evaluations.

c. Appendix X (ADAMS Accession No. ML17086A451), Section X.1.3 includes five criteria for selecting members of the closure review team. Describe how the selection of members for the March 2017 independent assessment met the five criteria.

d. Discuss whether the F&O closure review scope included all finding-level F&Os including those finding-level F&Os that are associated with "Met" SRs.

e. Attachment 3 in the LAR, "provides a summary of the remaining findings and open items." Under the column "Disposition for 50.69," most F&Os have F&O closure team comments, but some have no F&O closure team comments. Confirm that the closure

review team closed all utility resolved findings with no comments, or provide the closure teams comments on the remaining findings.

Question 2 – PRA Technical Adequacy Determined By Capability Category II

Section 3 of the LAR requested the NRC staff to utilize the review of the PRA technical adequacy of previous LAR submittals, Relocation of Specific Surveillance Frequency, dated November 29, 2016 (ADAMS Accession No. ML16200A285), and Adoption of National Fire Protection Association Standard 805, dated May 29, 2008 (ADAMS Accession No. ML081560641). Attachment 3 of the LAR provides a list of F&Os and their associated SRs assessed at either Not Met, CC-I, or CC-II.

In Section 3.1.4.1 of, "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee-controlled Program (CAC No. MF6583)," dated November 29, 2016, the licensee states that SRs were identified to be assessed to meet CC-I with no associated F&Os (e.g., FSS-D7, FSS-D9). For any SRs not addressed by the references above in Attachment 3 of the LAR, provide the following:

a. Justification that each of the SRs that were assessed to be CC-I or Not Met, in accordance with Section 2.1 of RG 1.200, Revision 2, does not impact the application.

b. Alternatively, propose a mechanism that ensures those SRs not meet at CC-II will be resolved in accordance with an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review) prior to implementation of the 10 CFR 50.69 categorization process.

Question 3 – Open/Partially Open Findings in the Process of Being Resolved

Section 4.2 of RG 1.200 states, in part, that the LAR should include:

> A discussion of the resolution of the peer review findings... and observations that are applicable to the parts of the PRA required for the application. This decision should take the following forms:
>
> - A discussion of how the PRA model has been changed
> - A justification in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application decision were not adversely impacted (remained the same) by the particular issue

Attachment 3 of the LAR, "Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items," provides finding-level F&Os that are still open or only partially resolved after the F&O closure review. Also, F&O descriptions and their dispositions were previously provided to the NRC in the LAR to adopt for Technical Specification Task Force (TSTF)-425, "Relocate Surveillance Frequencies to Licensee Control – Risk Informed Technical Specifications Task Force (RITSTF) Initiative 5b," and in the LAR to adopt National Fire

Protection Association Standard 805. For a number of F&O dispositions, there is insufficient information for NRC staff to conclude that the F&O is sufficiently resolved for this application.

a. F&O 1-9 regarding internal flooding drain analysis

The F&O closure team determined that drain analysis for the other buildings besides the reactor auxiliary building (RAB) was not performed and no qualitative evaluation was documented. The closure team provided an example of draining from the turbine building upper elevations to the basement that required assessment. The licensee's disposition stated water will not accumulate from backflow through floor drains in these buildings and the assessment of these buildings, "is not expected to impact the results." The implication of this disposition is that the requested drain analysis for buildings other than the RAB have not been completed.

1. Clarify if the drain analysis recommended by the closure team has been performed. If performed, provide details of the analysis including how it impacts the flood analysis and the 10 CFR 50.69 categorization process.

2. Provide justification, such as a sensitivity study, that the exclusion of the completed drain analysis has no impact on the 10 CFR 50.69 categorization results, or

3. Alternatively, propose a mechanism that ensures F&O 1-9 will be resolved in accordance with an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review) prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also specify how the F&O will be resolved in the PRA (e.g., provide an explicit description or a reference to the appropriate section of the LAR).

b. F&O 1-16 and 2-8 regarding human-induced (maintenance) flooding events

The F&O closure team raised the concern for F&O 1-16 that flood frequencies were not appropriately apportioned across flood compartments and recommended a sensitivity be performed on the potential impact of underestimating maintenance-induced flooding frequencies. For F&O 2-8, the F&O closure team stated that insufficient description was provided related to the selection (screening) of these maintenance-induced flooding scenarios. The closure team stated, "unless it can be demonstrated that no open maintenance (including preventative (PM) and corrective (CM)) can be performed on the subject fluid system during power operation," the frequencies may not be apportioned correctly. The licensee's disposition states that this is a documentation issue. The NRC staff notes that the disposition did not directly address the specific issue of selection (screening) and does not appear to be just a documentation issue. Therefore:

1. Summarize the analysis conducted to identify at-power maintenance on fluid systems and if each identified source (compartments) was included in the frequency calculation.

2. Provide justification, such as a sensitivity study, that the exclusion of the appropriate maintenance-induced flood frequencies and compartments has no impact on the 10 CFR 50.69 categorization results, or

3. Alternatively, propose a mechanism that ensures F&O 1-16 and 2-8 will be resolved in accordance with an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review) prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also specify how the F&O will be resolved in the PRA (e.g., provide an explicit description or a reference to the appropriate section of the LAR).

c. F&O 1-18 regarding assessment of door failure modes

The F&O closure team stated the updated analysis did not include all critical failure modes and that the door failure criteria may not be appropriate for all door types. The disposition states that "[re]examination is not expected to significantly change the timing or impacts of any flooding sequence (because of the very large rooms at HNP), and is not expected to affect categorization." The NRC staff notes that changes in door failures can affect propagation path effects and timing, which can affect flood operator actions and impact scenario risk values. Therefore:

1. Provide justification, such as a sensitivity study, that not performing the cited door failure calculations and subsequent flood scenario analysis has no impact on the 10 CFR 50.69 categorization results, or

2. Alternatively, propose a mechanism that ensures F&O 1-18 will be resolved in accordance with an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review) prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also specify how the F&O will be resolved in the PRA (e.g., provide an explicit description or a reference to the appropriate section of the LAR).

d. F&O 1-19 regarding operator action cues used to isolate internal flooding

The F&O closure team found the 5-minute cue timing related to the operators isolating flooding based on specific alarms that indicate floods in each flood area is not sufficient. The licensee's disposition states that Duke Energy performed a sensitivity where the cue times were increased by a factor of 3, "and there was minimal impact on the flooding results." Therefore:

1. Provide justification that a 15-minute delay is a reasonable upper bound for cues indicating floods, or

2. Alternatively, propose a mechanism that ensures F&Os 1-7 and 1-19 will be resolved in accordance with an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review) prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also specify how the F&Os will be resolved in the PRA (e.g., provide an explicit description or a reference to the appropriate section of the LAR).

e. F&O 2-3 regarding automatic and manual actions to terminate internal floods

The F&O closure team stated that no manual actions are identified for many flood compartments. The various descriptions also state that "[n]o considerations were given to isolation of the ruptured system," "[n]evertheless isolation actions are modelled for

many of the flood scenarios but they are just not listed in the documentation." The licensee concludes that this is a documentation issue only. The NRC staff notes that inclusion of complex (isolation) actions in the PRA scenarios but not in the documentation does not provide confidence that the modelled actions and probabilities are appropriately evaluated. Therefore:

1. Identify all the different types of automatic (e.g., sump pumps) and manual actions (e.g., motor-operated valve (MOV) closures) included in the fault trees.

2. For each identified type of automatic and manual actions, reference (or summarize) the method(s) used to model and quantify the action.

3. Alternatively, propose a mechanism that ensures F&O 2-3 will be resolved in accordance with an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review) prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also specify how the F&O will be resolved in the PRA (e.g., provide an explicit description or a reference to the appropriate section of the LAR).

f. F&O 2-4 regarding induced failure mechanisms from high energy line break (HELB) events

The initial Harris HELB analysis was performed using a zone of influence (ZOI) of 20 feet or ten times the inner pipe diameter, whichever is greater, to determine which SSCs may be impacted. The F&O closure team noted that analysis beyond the currently defined ZOI is needed to properly assess these failure mechanisms. The licensee's disposition states that "[t]he ZOI calculation is based on SNL analyses and has been accepted by the NRC in previous industry submittals. The additional analysis is beyond the requirements of the Standard." Therefore:

[

1. Provide references to the SNL [Sandia National Laboratory] analyses and NRC acceptance of previous NRC submittals. Provide the specific guidance from the SNL document related to determining HELB impacts, the applicability of the ZOI determination for the different potential HELB impacts (e.g., jet impingement, pipe whip, high temperature, high humidity, etc.), and describe how the HNP analysis was conducted in accordance with this reference.

2. If the methodology cannot be shown to have been previously accepted by the NRC, provide detailed justification for the use of this methodology and discuss the impact of its use on the results of the 10 CFR 50.69 categorization process, or

3. Alternatively, propose a mechanism that ensures F&O 2-4 will be resolved in accordance with an NRC-accepted process (e.g., full-scope peer review, focused-scope peer review, or F&O closure review) prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also specify how the F&O will be resolved in the PRA (e.g., provide an explicit description or a reference to the appropriate section of the LAR).

g. F&O HRA-C1-3 regarding detailed analyses for significant human failure events (HFEs)

CC-II for SR HR-G1 (previously HRA-C1) of the ASME/ANS PRA standard requires a detailed analysis to estimate human error probabilities (HEPs) for significant HFEs and conservative estimates for non-significant HFEs. In Attachment 3 of the LAR, the disposition to F&O HRA-C1-3 states that "some significant HFEs were not selected for detailed analysis and were instead conservatively assumed to be failed or left at a screening value. However, the significant operator actions that were selected for detailed analysis are sufficient to provide risk insights for the 50.69 application." Therefore:

1. Provide justification, such as a sensitivity study, that the significant HFEs which are assumed failed or left at screening value have no impact on the 10 CFR 50.69 categorization results compared to completion of a detailed analysis.

2. Alternatively, perform a detailed analysis and provide the results for the HFEs that were conservatively assumed to be failed or left at screening value, and demonstrate these significant HFEs have no impact on the 10 CFR 50.69 categorization results.

Question 4 – Determining No Impact on Categorizations

Several F&O resolutions determine that the resolution "will not impact CDF/LERF [core damage frequency/large early release frequency] or the component categorization under 10 CFR 50.69" and others state there will be "no impact on component categorization under 10 CFR 50.69." Minor or no impact on CDF and LERF is fairly straightforward to determine but it is not clear that minor or no impact on CDF/LERF will always lead to no impact on categorization. Also, an excessively conservative assumption, such as setting probabilities to 1.0, may also impact categorization. Please discuss how the conclusion is reached that there is no impact on component categorization.

Question 5 - Qualitative Function Categorization

LAR Table 3-1 indicates that the evaluation of the seven qualitative criteria defined in Section 9.2 of NEI 00-04 is performed at the function level and prior to the Integrated Decision-making Panel (IDP). The LAR states that "NEI 00-04 only requires the seven qualitative criteria in Section 9.2 of NEI 00-04.... to be completed for components/functions categorized as [low safety significant] LSS." LAR Table 3-1 contains the entry "Allowable" at the intersection of the "IDP change HSS to LSS" column and "Qualitative Criteria" row, which appears to contradict the premise that the seven criteria are only applied to LSS functions. The guidance in NEI 00-04 states that the IDP "should consider the impact of loss of the function/structure, system, and component (SSC) against the remaining capability to perform the basic safety functions." Provide details about the categorization guidance that will be provided to assess the safety significance of a function when there is an impact on, or even a loss of, the capability described in each of the seven criteria (e.g., Is a single false response sufficient to assign the function high safety significance (HSS)?).

Question 6 – Passive Categorization Process

LAR Section 3.1.2, "Passive Categorization Process," states that passive components and the passive function of active components will be evaluated using the method for risk-informed repair/replacement activities consistent with the safety evaluation issued by the Office of Nuclear Reactor Regulation, "Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems, Third and Fourth 10-Year In-service Inspection Intervals," for Arkansas Nuclear One, Unit 2 (ANO-2), dated April 22, 2009 (ADAMS Accession No. ML090930246). The LAR further states that this methodology will be applied to determine the safety significance of Class 1 SSCs.

The NRC staff notes that this methodology has been approved for Class 2 and Class 3 SSCs. Because Class 1 SSCs constitute principal fission product barriers as part of the reactor coolant system or containment, the consequence of pressure boundary failure for Class 1 SSCs may be different than that for Class 2 and Class 3 SSCs. Therefore, the criteria in the ANO-2 methodology cannot automatically be generalized to Class 1 SSCs without further justification.

The LAR does not justify how the ANO-2 methodology can be applied to Class 1 SSCs and how sufficient defense-in-depth and safety margins are maintained. An acceptable technical justification for Class 1 SSCs would have to address how the methodology is sufficiently robust to assess the safety significance of Class 1 SSCs, including, but not limited to: (1) justification of the appropriateness of the numerical criteria for conditional core damage probability (CCDP) and conditional large early release probability (CLERP) used to assign 'High', 'Medium' and 'Low' safety significance to these loss-of-coolant initiating events; (2) identification and justification of the adequacy of the additional qualitative considerations to assign 'Medium' safety significance (based on the CCDP and CLERP) to 'High' safety significance; (3) justification for crediting operator actions for success and failure of pressure boundary; (4) guidelines and justification for selecting the appropriate break size (e.g. double-ended guillotine break or smaller break); and (5) include supporting examples of types of Class 1 SSCs that would be assigned low safety significance.

As mentioned in the March 13, 2018, meeting summary for the February 20, 2018, Risk-Informed Steering Committee (RISC) meeting (ADAMS Accession No. ML18072A301), the NRC staff understands that the industry is planning to limit the scope of passive categorization to Class 2 and Class 3 SSCs, consistent with the pilot Vogtle Electric Generating Plant, Units 1 and 2, license amendment dated December 17, 2014 (ADAMS Accession No. ML14237A034).

Provide the requested technical justification or confirm the intent to apply the ANO-2 passive categorization methodology only to Class 2 and Class 3 SSCs.

Question 7 - Identifying Key Assumptions and Uncertainties that could Impact the Application

Section 4.2 of RG 1.200 states for licensee submittal documentation, "[i]dentification of the key assumptions and approximations relevant to the results used in the decision-making process," is to be provided. Section 1.3 of RG 1.200 describes the level of detail of a PRA required and states that "[i]n general, the level of detail for the base PRA needs to be consistent with current good practice." Current good practices are those practices that are generally accepted throughout the industry and have shown to be technically acceptable in documented analyses or engineering assessments.

Section 3.2.7 of the LAR states that, "[t]he detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 (Revision 0) and Section 3.1.1 of [Electric Power Research Institute (EPRI) Technical Report (TR)]-1016737." The NRC staff notes that one of these sources has been superseded by a revision (Revision 1 of NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," March 2017; ADAMS Accession No. ML17062A466), which references the updated EPRI guidance (TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty" (2012)).

Attachment 6 of the LAR contains just 13 key assumptions/uncertainties from three PRA models, whereas industry guidance documents such as NUREG-1855, Revision 1, and EPRI TR-1026511 address a large number of potential assumptions and uncertainties. For example, just four key sources of fire PRA modeling assumptions/uncertainty are provided in the LAR, compared to the 2012 EPRI document which identifies 71 potential sources of uncertainty. There appear to be no uncertainties or assumptions associated with LERF and internal flooding.

The LAR continues, "[t]he list of assumptions and sources of uncertainty were reviewed to identify those which would be significant for the evaluation of this application. Only those assumptions or sources of uncertainty that could significantly impact the risk calculations were considered key for this application."

The NRC staff notes that Stage F of NUREG-1855 (Revision 1) provides guidance on how to identify key sources of uncertainty relevant to the application.

To address the observations above, the NRC staff requests the following additional information:

a. Provide a detailed summary of the process used to determine the nine sources of uncertainty and assumptions presented in Attachment 6 of the LAR. Include in this discussion an explanation of how the process is in accordance with NUREG-1855, Revision 1, or other NRC-accepted method. Also include in the discussion a detailed description of how the final set of nine uncertainties and assumptions were developed from the initial comprehensive list of PRA model(s) uncertainties and assumptions.

b. Provide justification for why just four or no key sources of uncertainty or assumptions were identified for the internal flooding, LERF, and fire PRA models.

c. If the process of identifying key sources of uncertainty or assumptions for these PRA models cannot be justified, provide the results of an updated assessment of key sources of uncertainty or assumptions.

d. Describe specific assumptions and sources of uncertainty key to this application for the entries in LAR Attachment 6 in enough detail that its impact on the application and that a specific sensitivity could be defined to examine the risk significance of the issue. Include in this description any new sources of uncertainty or assumptions identified in Part c.

Question 8 – Very Early Warning Fire Detection Systems (VEWFDS) Utilized in the PRA

Assumption/Uncertainty No. 5 in Attachment 6 of the LAR states "[t]he HNP Fire PRA assumes Incipient Detection System functions as outlined in NUREG 2180 with some exceptions." The

disposition to this uncertainty states "[t]he current methodology is based on NRC FAQ 08-0046." The disposition further explains, based on a qualitative evaluation, that "the overall impact of using the current FAQ 08-0046 method is that it is not expected that any SSCs would be categorized as LSS that would be categorized HSS using the NUREG 2180 methodology". It is not at all clear to the NRC staff that this is the case, especially for SSCs that are not associated with basic events where VEWFDS is not credited.

LAR Section 3.2.2 states "[t]he internal Fire PRA model was developed consistent with NUREG/CR-6580 and only utilizes methods previously accepted by the NRC." However, in a letter dated July 1, 2016, "Retirement of National Fire Protection Association 805 Frequently Asked Question 08-0046 "Incipient Fire Detection Systems" (ADAMS Accession No. ML16167A444), FAQ 08-0046 was retired. In this letter. it was requested of licensees to evaluate the impact of the new guidance on their PRA in accordance with their licensing basis. Furthermore, in a subsequent internal memorandum dated March 29, 2018, "Interim Guidance Memorandum – Review of Procedures, Training, and Operating Experience Related to Very Early Warning Fire Detection Systems in License Applications" (not publicly available), additional guidance was provided to the NRC staff when reviewing an LAR that includes PRA credit for VEWFDS. Therefore:

a. Provide justification, such as a sensitivity study, that use of the FAQ 08-0046 VEWFDS methodology, which is not endorsed by the NRC, has no impact on the 10 CFR 50.69 categorization results. Include in this justification consideration of SSCs that are included in basic events where the VEWFDS is credited and SSCs that are not included in basic events where the VEWFDS is credited.

b. Alternatively, propose a mechanism that ensures the VEWFDS methodology will be updated to the NUREG-2180, or other NRC acceptable, methodology prior to implementation of the 10 CFR 50.69 categorization process. If this update is determined to be a PRA model upgrade per the ASME/ANS PRA standard, include in this mechanism a process for conducting a focused-scope peer review and ensure any findings are closed by using an approved NRC process.

c. Describe HNP procedures, training, and operating experience related to the HNP's response to VEWFDS alerts/alarms. In the response, address how these procedures, training, and operating experience support the VEWFDS credit in the PRA.

Question 9 - Key Assumptions and Uncertainties that could Impact the Application

Section 1.2.10 of RG 1.200 discusses the technical approach in determining the impact of assumptions and sources of uncertainty on the PRA model.

The licensee's dispositions presented in Attachment 6 of the LAR for key assumptions and modeling uncertainties state in each case that: "this does not represent a key source of uncertainty and will not be an issue for the 50.69 calculations." However, in a number of instances, there is not enough information provided in the dispositions for the NRC staff to

determine whether the uncertainty will not impact 10 CFR 50.69 risk categorization. In light of these observations, address the following:

a. Cable types and their associated fire damage temperature

Attachment 6 (page 52) of the LAR states that "[a]fter the Harris Fire PRA was completed, NRR Research tested Kerite cable damage properties and determined they will fail and then ignite 75°C higher than Thermoplastic cables." The licensee's disposition states that "[f]ire PRA may be updated in the future to reduce the ZOI and time to damage to reflect the actual capabilities of the Kerite cables." The disposition assesses that excluding the updated analysis is a conservatism. The NRC staff notes that conservative modeling in the PRA can skew the plant's risk profile and impact the SSCs risk importance values determined as part of 10 CFR 50.69 categorization. Therefore:

1. Provide justification, such as a sensitivity study, that the conservative modelling choice of not updating the cable fire damage analysis has no impact on the 10 CFR 50.69 categorization results.

2. Alternatively, propose a mechanism that ensures cable fire damage analysis will be updated to incorporate the new test data on the damage temperature of HNP Kerite cables prior to implementation of the 10 CFR 50.69 categorization process. If this update is determined to be a PRA model upgrade per the ASME/ANS PRA standard, include in this mechanism a process for conducting a focused-scope peer review and resolution of any findings.

b. Human Reliability Analysis (HRA) Modeling and Dependency

For performing HRA dependency analysis, NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report," July 2012 (DAMS Accession No. ML12216A104), discusses the need to consider a minimum value for the joint probability of multiple human failure events (HFEs), and refers to NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," April 2005 (ADAMS Accession No. ML051160213) (Table 2-1), which recommends joint human error probability (HEP) values should not be below 1E-5. Table 4-3 of EPRI Technical Report 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," October 2010, provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. Therefore, the available guidance provides for assigning joint HEPs that are less than a minimum value but only through assigning proper levels of dependency. Cutsets with joint human error probability (JHEP) values less than the minimum value should be individually reviewed for timing, cues, etc., to check the dependency between all the operator actions in the cutset.

Attachment 6 (page 56) of the LAR provides two statements concerning lower bound values for JHEPs. The first uncertainty states that "[a]ny cutsets with more than four HFE are not evaluated for more than four HFEs and the additional actions are considered completely dependent." The second uncertainty states that "a lower bound of 1 x 10-5 was enforced as the limiting HEP in any two dependent HEPs in a cutset. For cutsets with three HEPs a lower bound of 1E-06 was used." It concludes that "the selection of the lower bounds is based on guidance provided in NUREG-1792." The NRC staff notes that the two uncertainty statements do not provide a floor JHEP value

for fourth (and higher) order combinations (i.e. four or more HFEs in the same cutset as described in the first statement). Therefore:

1. Clarify the JHEP applied to fourth and higher order combinations. If the assignment of JHEPs is different for different order cutsets, identify all different orders and the rules applied to each order.

2. Provide an estimate of the number of these JHEP values below 1E-5 in the FPRA, discuss the range of values, and provide at least two different examples of justification for less than minimum values.

3. Provide an estimate of the number of these JHEP values below 1E-6 in the IEPRA, discuss the range of values, and provide at least two different examples of justification for less than minimum values

Question 10 - Feedback and Adjustment Process

Section 11.2, "Following Initial Implementation," of NEI 00-04 discusses that "a periodic update of the plant PRA may affect the results of the categorization process. If the results are affected, the licensee must make adjustments as necessary to either the categorization or treatment processes to maintain the validity of the processes." Specifically, NEI 00-04, Section 12.1 discusses cases for which, in some instances, an updated PRA model could result in new risk achievement worth (RAW) and Fussell-Vesely (FV) importance measures that are sufficiently different from those in the original categorization so as to suggest a potential change in the categorization. To address the observations above, the NRC staff requests the following additional information:

a. Explain how this periodic review will be administered. At a minimum, discuss the following:

   1. Participants involved in the review;

   2. Sources of material identified to be reviewed;

   3. Periodicity for when the review will be performed; and

   4. Documentation of the review performed (e.g., corrective action program, engineering evaluation, etc.).

b. Provide the criteria to be used to determine if the change being reviewed has any impact to a modeled PRA hazard(s) and/or any SSC categorized by the 50.69 process.

Question 11 – SSCs Categorization Based on Seismic Hazards

Section 3.2.3 of the LAR states:

> The NEI 00-04 approach using the [safe shutdown equipment list] SSEL would identify credited equipment as HSS regardless of their capacity, frequency of challenge or level of functional diversity.

> An evaluation was performed of the as-built, as-operated plant against the SSEL. The evaluation was a comparison of the as-built, as-operated plant to the plant configuration originally assessed by the SMA [seismic margin analysis]. Differences were reviewed to identify any potential impacts to the equipment credited on the SSEL. Appropriate changes to the credited equipment were identified and documented.

The use of "would" in the first sentence implies that the SSEL was not used as completed in the individual plant examination of external events (IPEEE) because that results in equipment assigned HSS, "regardless of their capacity, frequency of challenge or level of functional diversity." The second paragraph concludes that "[a]ppropriate changes to the credited equipment were identified and documented."

a. Summarize the changes to the SSEL and describe the types of changes to the as-built, as-operated plant that caused the changes to the SSEL.

b. Confirm that changes were not made to the SSEL based on changes in capacity, frequency of challenge, or level of functional diversity unless the change was a physical changes to the facility. If changes other than physical changes were credited, describe and justify those changes.

Question 12 – SSCs Categorization Based on Other External Hazards

Section 3.2.4 of the LAR states:

> As part of the categorization assessment of other external hazard risk, an evaluation is performed to determine if there are components being categorized participate in screened scenarios and whose failure would result in an unscreened scenario. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered HSS.
>
> All remaining hazards were screened from applicability and considered insignificant for every SSC and, therefore, will not be considered during the categorization process.

a. The first paragraph states that assessment of external hazards "is performed" implying that the assessment will be performed. The last sentence implies that the assessment has been completed and concludes that all other external hazards will never need evaluation during categorization. Clarify if any additional screening of external hazards will be part of the categorization process.

b. The IPEEE screening process did not include the additional step illustrated in Figure 5-6 in Section 5.4 of NEI 00-04. Figure 5-6 and its associated text states that an evaluation is performed to determine if there are components being categorized that participate in screened external event scenarios whose failure would result in an unscreened scenario. Clarify how the screening criteria in Attachment 5, Progressive Screening Approach for Addressing External Hazards, satisfy the guidelines that HSS will be assigned to SSCs whose failure would cause a screened external event scenario to become unscreened.

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2, AND SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – AUDIT PLAN QUESTIONS RE APPLICATION TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0008 AND L-2018-LLA-0034) DATED JULY 11, 2018

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