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

October 9, 2018

Ms. Tanya M. Hamilton
Site Vice President
Shearon Harris Nuclear Power Plant
5413 Shearon Harris Road
M/C HNP01
New Hill, NC 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0034)

Dear Ms. Hamilton:

By application dated February 1, 2018 (Agencywide Documents Access and Management System Accession No. ML18033B768), Duke Energy Progress, LLC (the licensee) submitted a license amendment request for the Shearon Harris Nuclear Power Plant, Unit 1, requesting to revise the licensing basis to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations*, Part 50, Section 69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors."

The U.S. Nuclear Regulatory Commission staff has determined that additional information is needed in order to complete its review. The enclosed request for additional information (RAI) was e-mailed to the licensee in draft form September 10, 2018, and a clarification call was held on September 19, 2018. An RAI response is due by October 19, 2018. Please note that if a response to this letter is not received by this date, or an acceptable alternate date is not provided in writing, we may deny the application for amendment under the provisions of Title 10 of the *Code of Federal Regulations*, Part 2, Section 108, "Denial of application for failure to supply information."

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

Sincerely,

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

Docket No. 50-400

Enclosure:
Request for Additional Information

cc: Listserv

DUKE ENERGY PROGRESS, LLC

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

REQUEST FOR ADDITIONAL INFORMATION

REGARDING A LICENSE AMENDMENT REQUEST PROPOSING TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS"

DOCKET NO. 50-400

EPID L-2018-LLA-0034

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, internal flooding PRA, 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 dated February 1, 2018, 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, including technical adequacy of the PRA models. The NRC staff requests additional information (RAI) for the following areas in order to complete its review.

2.0 REQUEST FOR ADDITIONAL INFORMATION

PRA RAI 01 – PRA Technical Adequacy Determined By Capability Category (CC) II

Section 3 of the LAR, dated February 1, 2018, requested the NRC staff to utilize the review of the PRA technical adequacy of previous LAR submittals. The information can be found in "Issuance of Amendment Regarding Risk-Informed Justifications for the Relocation of Specific Surveillance Frequency Requirements to a Licensee-Controlled Program," dated November 29, 2016 (ADAMS Accession No. ML16200A285), and "Request for License Amendment to Adopt National Fire Protection Association (NFPA) Standard 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition), dated May 29, 2008 (ADAMS Accession No. ML081560641).

Attachment 3 of the LAR provides a list of open and partially resolved Facts and Observations (F&Os) and their associated supporting requirements (SRs).

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" dated November 29, 2016, 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 met at CC-II will be resolved prior to implementation of the 10 CFR 50.69 categorization process.

PRA RAI 02 – 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 F&Os that are applicable to the parts of the PRA required for the application. This discussion 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, dated February 1, 2018, "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 past LAR to adopt Technical Specification Task Force (TSTF)-425, "Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program, (Relocate Surveillance Frequencies to Licensee Control-Risk Informed Technical Specifications Task Force Initiative 5b)," dated August 18, 2015 (ADAMS Accession No. ML15236A254) and in the May 29, 2008 LAR to adopt NFPA Standard 805 (ADAMS Accession No. ML081560641). For the following F&O

dispositions, provide additional 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, summarize 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 prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model or documentation to resolve the issue.

b. 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 prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model or documentation to resolve the issue.

c. 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 prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model or documentation to resolve the issue.

d. 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 10 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 Sandia National Laboratory (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 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 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 prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model or documentation to resolve the issue.

e. 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 American Society of Mechanical Engineers/American Nuclear Society (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, dated February 1, 2018, 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 that 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.

f. <u>F&O FSS-F3-01</u> regarding exposed structural steel fire scenarios

The disposition provided in the "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," provides a qualitative description of viable fire scenarios related to exposed structural steel near a large fire source in the turbine building. It states that a dedicated water spray (deluge) suppression system is provided near the large oil related fire hazards and provides assurance the fire affects remain localized. It was assessed not to have any measureable impact on core damage frequency/large early release frequency (CDF/LERF). The NRC staff notes that minimal impacts to CDF and LERF could increase certain SSCs above the NEI 00-04 risk.

Attachment 3 of the LAR, dated February 1, 2018, lists this F&O having been assessed at CC-I, since a qualitative assessment was performed and retained in a licensee document. The disposition continues by stating a quantitative assessment has been performed in another licensee calculation and is sufficient to meet CC-II. Neither the TSTF-425 nor the NFPA-805 documents referenced in PRA RAI 01 provide details of the calculation and if the scenarios have been implemented in the FPRA model. The NRC staff seeks clarification on the current status for this F&O. Therefore:

1. If currently assessed as CC-I, provide justification that this capability category has no impact on the 10 CFR 50.69 categorization process. Include in this discussion how the Harris 10 CFR 50.69 process will categorize the SSCs related to these excluded fire scenarios, or

2. If currently assessed as CC-II, confirm that the related fire scenarios have been included in the FPRA model used for the 10 CFR 50.69 categorization process, or

3. Alternatively, propose a mechanism that ensures F&O FSS-F3-01 will be resolved prior to implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model or documentation to resolve the issue.

PRA RAI 03 – Qualitative Function Categorization

The LAR dated February 1, 2018, 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 LSS [low safety significant]." LAR Table 3-1 contains the entry "Allowable" at the intersection of the "IDP change HSS [high safety significant] 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." Explain how the IDP will collectively assess the seven specific questions to identify a function/SSC as LSS as opposed to HSS including a clarification of the "Allowed" entry in LAR Table 3-1.

PRA RAI 04 – Passive Categorization Process

The LAR, dated February 1, 2018, 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 HNP Class 1 SSCs and how sufficient defense-in-depth and safety margins are maintained.

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.

PRA RAI 05 - 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, dated February 1, 2018, 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 Decision Making," 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 13 key assumptions/uncertainties from three PRA models, whereas industry guidance documents such as NUREG-1855, Revision 1, and EPRI TR-1026511 address a larger number of potential assumptions and uncertainties.

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 E of NUREG-1855 (Revision 1) provides guidance on how to identify key sources of uncertainty relevant to the application.

To address the observations above, provide the following:

a. Provide a detailed summary of the process used to determine the 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 uncertainties and assumptions were developed from the initial comprehensive list of PRA model(s) uncertainties and assumptions.

b. 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.

c. Describe specific assumptions and sources of uncertainty that are key to this LAR for the entries in Attachment 6 of the LAR 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 b.

PRA RAI 06 – Very Early Warning Fire Detection Systems (VEWFDS) Utilized in the PRA

Assumption/Uncertainty No. 5 in Attachment 6 of the LAR, dated February 1, 2018, 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 Frequently Asked Question (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.

Therefore, 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.

PRA RAI 07 - 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 are presented in Attachment 6 of the LAR for key assumptions and modeling uncertainties. 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. The staff requests HNP address the following:

a. <u>Cable types and their associated fire damage temperature</u>

Attachment 6 (page 52) of the LAR states that "[a]fter the Harris Fire PRA was completed, NRR [Office of Nuclear Reactor Regulation] 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. <u>Human Reliability Analysis (HRA) Modeling and Dependency</u>

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty also discusses Joint HEP. For performing HRA dependency analysis, NUREG-1921, "EPRI/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 HEP values should not be below 1E-5. Table 4-3 of EPRI TR 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. Assigning joint HEPs that are less than a minimum value should be individually reviewed for timing, cues, etc., to check the dependency between all the operator actions in the cutset.

Attachment 6 of the LAR provides several statements about lower bound values for joint HEPs but includes some discussion about the number of HFEs in the cutsets.

1. Provide an estimate of the number of joint HEP values below 1E-5 in the Fire PRA, discuss the range of values and confirm that justification is documented for each of these joint HEPs.

2. Provide an estimate of the number of these joint HEP values below 1E-6 in the Internal Events PRA, discuss the range of values and confirm that justification is documented for each of these joint HEPs.

PRA RAI 08 – 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 and Fussell-Vesely importance measures that are sufficiently different from those in the original categorization so as to suggest a potential change in the categorization. Therefore, provide the following:

a. Explain how this periodic review will be administered. Provide a discussion to include 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.

PRA RAI 09 – SSCs Categorization Based on Other External Hazards

Section 3.2.4 of the LAR, dated February 1, 2018, states:

> As part of the categorization assessment of other external hazard risk, an evaluation is performed to determine if there are components being categorized that 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.

The last sentence implies that the assessment has been completed and concludes that all other external hazards will never need evaluation during categorization. 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: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2018-LLA-0034) DATED OCTOBER 9, 2018

DISTRIBUTION:
Public
RidsACRS_MailCTR Resource
RidsNrrDorlLpl2-2 Resource
RidsRgn2MailCenter Resource
RidsNrrPMShearonHarris Resource
RidsNrrLABClayton Resource
RidsNrrDraApla Resource
BHartle, NRR

ADAMS Accession No. ML18264A028 *by e-mail

OFFICE	NRR/DORL/LPL2-2/PM	NRR/DORL/LPL2-2/LA	NRR/DRA/APLA*
NAME	MBarillas	BClayton	SRosenberg
DATE	9/24/18	10/4/18	8/15/18
OFFICE	NRR/DORL/LPL2-2/BC	NRR/DORL/LPL2-2/PM	
NAME	UShoop	MBarillas	
DATE	10/8/18	10/9/18	

OFFICIAL RECORD COPY