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10 CFR 50.90

April 23, 2019

Serial: RA-19-0153

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400 / Renewed License No. NPF-63

Subject: Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors"

References:

1. Duke Energy letter, *Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors"*, dated February 1, 2018 (ADAMS Accession No. ML18033B768).

2. Duke Energy letter, *Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors"*, dated October 18, 2018 (ADAMS Accession No. ML18291A606).

3. NRC letter, *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"*, dated March 18, 2019 (ADAMS Accession No. ML19060A091).

Ladies and Gentlemen:

By letter dated February 1, 2018 (Reference 1), as supplemented by letter dated October 18, 2018 (Reference 2), Duke Energy Progress, LLC (Duke Energy) submitted a license amendment request (LAR) for Shearon Harris Nuclear Power Plant (HNP), Unit No. 1. The proposed amendment would modify the licensing basis, by the addition of a License Condition, 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 reactors."

By letter dated March 18, 2019 (Reference 3), the Nuclear Regulatory Commission (NRC) staff requested additional information from Duke Energy that is needed to complete the LAR review.

The enclosure to this letter provides Duke Energy's response to the Reference 3 RAI related to this amendment request. Attachment 1 contains PRA implementation items which must be completed prior to implementation of 10 CFR 50.69 at HNP. Attachment 2 contains proposed markups of the HNP Renewed Facility Operating License. The markups supersede those provided in Reference 2.

The conclusions of the original No Significant Hazards Consideration and Environmental Consideration in the original LAR are unaffected by this RAI response.

There are no regulatory commitments contained in this letter.

In accordance with 10 CFR 50.91, Duke Energy is notifying the State of North Carolina of this LAR by transmitting a copy of this letter and enclosure to the designated State Official.

Should you have any questions concerning this letter and its enclosure, or require additional information, please contact Art Zaremba, Manager – Fleet Licensing, at (980) 373-2062.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 23, 2019.

Sincerely,

Steve Snider
Vice President - Nuclear Engineering

JLV

Enclosure: Response to NRC Request for Additional Information

Attachments:

1. HNP 50.69 PRA Implementation Items
2. Markup of Proposed Renewed Facility Operating License

cc: Ms. C. Haney, NRC Regional Administrator, Region II
Ms. M. Barillas, NRC Project Manager, HNP (Electronic Copy Only)
Mr. J. Zeiler, NRC Sr. Resident Inspector, HNP
Mr. W. L. Cox, III, Section Chief, N.C. DHSR (Electronic Copy Only)

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Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400 / Renewed License No. NPF-63

Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors”

Enclosure

Response to NRC Request for Additional Information

NRC Request for Additional Information

By letter dated February 1, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18033B768), as supplemented by letter dated October 18, 2018 (ADAMS Accession No. ML18291A606), Duke Energy Progress, LLC (Duke Energy, the licensee), submitted a license amendment request (LAR) for Shearon Harris Nuclear Power Plant, Unit 1. 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 reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on a method of categorizing SSCs according to their safety significance. The Nuclear Regulatory Commission (NRC) staff has determined the following request for additional information (RAI) is needed to complete its review.

Regulatory Basis

Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (ADAMS Accession No. ML052910035), describes a process for determining the safety-significance of SSCs and categorizing them into the four Risk Informed Safety Class categories defined in 10 CFR 50.69. This categorization process is an integrated decisionmaking process that incorporates risk and traditional engineering insights.

NUREG-1855, Revision 1, "Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking (ADAMS Accession No. ML17062A466), provides guidance on how to treat uncertainties associated with probabilistic risk assessment (PRA) in risk-informed decisionmaking.

Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML090410014) describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors. It endorses, with clarifications, the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sa-2009 ("ASME/ANS 2009 Standard" or "PRA Standard") (ADAMS Accession No. ML092870592).

RAI 5.01:

The February 1, 2018, LAR states:

> *The process to categorize each system will be consistent with the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201. RG 1.201 states that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by 50.69(c)(1)(iv)."*

NEI 00-04 references RG 1.200 as the primary basis for evaluating the technical adequacy of the PRA. RG 1.200 references the ASME/ANS RA-Sa-2009 Standard which requires the identification and documentation of assumptions and sources of uncertainty during a peer review. RG 1.200 also references NUREG-1855 as one acceptable means to identify key assumptions and key sources of uncertainty. RG 1.200, Revision 2 defines a key uncertainty as "one that is related to an issue in which there is no consensus approach or model and where the choice of the approach or model is known to have an impact on the risk profile such that it influences a decision being made using the PRA." RG 1.200, Revision 2 defines a key assumption as "one that is made in response to a key source of modeling uncertainty in the knowledge that a different reasonable alternative assumption would produce different results." The term "reasonable alternative" is also defined in RG 1.200, Revision 2.

RAI 5 requested the licensee to clarify how key assumptions and (key) uncertainties that could impact the results are identified and included in the evaluation. In a letter dated October 18, 2018, in the licensee's response to RAI 5, the licensee refers to the integrated risk sensitivity as described in Section 8 of NEI 00-04. For this integrated risk sensitivity study, the unreliability of all low safety significant (LSS) SSCs is increased by a factor of 3 (consistent with NEI 00-04) and the subsequent total risk increase is compared to the RG 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ADAMS Accession No. ML17317A256) acceptable risk increase guidelines. The licensee stated that this integrated risk sensitivity study, and the subsequent performance monitoring of LSS SSCs, could be used directly to address most of the "in excess of 1000" assumptions and sources of uncertainty instead of identifying and evaluating key assumptions and key uncertainties as described in NUREG-1855, Revision 1. The response also included a table titled "Uncertainties and assumptions not addressed by 10 CFR 50.69 factor of 3 sensitivity/performance monitoring" with 28 entries. The licensee recognized that assumptions and uncertainties that cause SSCs to be excluded from the PRA cannot be addressed by the integrated risk sensitivity. The entries in the Table are apparently identified and included because they cause SSCs to be excluded. The dispositions in the Table include dispositions consistent with the NUREG 1855, Revision 1 options of (1) refining the PRA if needed, (2) redefine the application (e.g., add a sensitivity study), or (3) add compensatory measure and monitoring specific to that assumption of uncertainty. However, the title of the table implies that all the unreported assumptions and uncertainty are evaluated and dispositioned as not being key solely using the factor of 3. Furthermore, most dispositions included in the Table also include the phrase "[a]ny impact of the exclusion of these scenarios on acceptance criteria for categorizations of other components is addressed by the factor of 3 sensitivity and performance monitoring."

The NRC staff finds that the licensee's proposed method is a deviation from the guidance of NEI 00-04 and NUREG-1855, Revision 1, for the following reasons. Figure 1-2 in Section 1.5, Categorization Process Summary, of NEI 00-04 illustrates the available paths through the accepted categorization process. The categorization provides the appropriate LSS/high safety significant (HSS) category. The integrated risk sensitivity study is only performed after all steps in the categorization have been completed and it is not intended to be a change in the risk estimate. The study simply verifies that the combined impact of any postulated simultaneous degradation in reliability of all LSS SSCs would not result in significant increases in core damage frequency and large early release frequency. Therefore, the aggregate risk sensitivity study is intended to capture the uncertainty from relaxation of "special treatment" for candidate LSS SSCs. Other assumptions and uncertainties are related to models and methods used in the PRA and the impact of these assumptions and uncertainties is not considered or included in the integrated risk sensitivity study.

NUREG-1855 identifies that one key source of uncertainty is the unknown increase in unreliability associated with the reduced special treatment requirements on LSS SSCs allowed by 10 CFR 50.69. The NUREG states that one acceptable technique to address this specific key source of uncertainty is to increase the unreliability of LSS SSCs by a multiplicative factor in an integrated risk sensitivity study. NEI 00-04 discusses using a factor of 3 to 5 as an acceptable multiplicative factor to address this uncertainty and the licensee selected to use the factor of 3. In contrast, addressing key assumptions and key sources of uncertainty in the PRA might require that SSCs be added to the PRA, might require changes to the model logic, or might require changes in the unreliability (e.g., unreliability increases for unusual uses of SSCs and for consequential failures) greater than the factor of 3 used in the integrated risk sensitivity study. Even for components that are modeled, the integrated risk sensitivity study only addresses the impact of SSCs as they are included in the PRA logic models without addressing any changes to the logic model itself that might be needed to address the key assumption (i.e., because of limitations in scope or level of detail). In addition, the use of the integrated risk sensitivity will result in the licensee identifying potential categorization of a LSS SSC as HSS only if the RG 1.174 risk acceptance guidelines are exceeded. However, addressing key assumptions and sources of uncertainty, can result in a change in categorization even if the RG 1.174 guidelines are not exceeded. NEI 00-04 guidance in Tables 5-2 through 5-5 recognizes such occurrences and Figure 7-2 in NEI 00-04, "Example Risk-Informed SSC Assessment Worksheet," captures such a change in categorization due to the sensitivity studies recommended in Tables 5-2 through 5-5.

The licensee's response simply states and does not justify that the use of the factors in the integrated risk sensitivity study are sufficient to capture the impact of all assumptions and uncertainties on the categorization of SSCs modeled in the current PRA. The approach proposed by the licensee represents a substantial deviation from the endorsed guidance for categorization in NEI 00-04 and the RAI response does not provide sufficient justification for the appropriateness of the deviation. It is unclear to the NRC staff whether the evaluation of assumptions and uncertainties proposed by the licensee can determine the effect of the key assumptions and uncertainties on the categorization of an indeterminate number of components. Therefore, the staff is unable to conclude that the components place in LSS accurately reflect the approved risk-informed process. Based on the above, provide the following information:

RAI 5.01.a:

a. Clarify which process is used and is meant by the RAI 5 Table title "Uncertainties and Assumptions Not Addressed by 10 CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring" (i.e., which types of uncertainties and assumptions have been addressed by the factor of 3).

Duke Energy Response to RAI 5.01.a:

The following RAI responses in parts b through f supersede the response to RAI 5 (ADAMS Accession No. ML18291A606). Accordingly, the table titled "Uncertainties and Assumptions Not Addressed by 10 CFR 50.69 Factor of 3 Sensitivity/Performance Monitoring" that was provided in response to RAI 5 is also being superseded by the following response. Additionally, this response supersedes Attachment 6 of the original LAR.

RAI 5.01.b:

b. Describe the approach used to identify the assumptions and uncertainties that are used in the base PRA models.

Duke Energy Response to RAI 5.01.b:

To identify the assumptions and uncertainties used in the Internal Events and Internal Flood base PRA models supporting the categorization, the generic issues identified in Table A.1 of EPRI 1016737 were reviewed, as well as the PRA documentation for plant-specific assumptions and uncertainties. This identification process is consistent with NUREG-1855 Revision 1 Stage E.

To identify the assumptions and uncertainties used in the Fire base PRA model supporting the categorization, the generic issues identified in EPRI 1026511 were reviewed, as well as the PRA documentation for plant-specific assumptions and uncertainties. This identification process is consistent with NUREG-1855 Revision 1 Stage E.

RAI 5.01.c:

c. Describe the approach(es) used to evaluate each assumption and uncertainty to determine whether each assumption and uncertainty is key or not for this application.

Duke Energy Response to RAI 5.01.c:

To determine whether each assumption or uncertainty is key or not for this application, the assumption or uncertainty was individually assessed based on the definitions in RG 1.200 Revision 2, NUREG-1855 Revision 1, and related references (i.e. EPRI 1016737, EPRI 1013491, and EPRI 1026511). These documents provide definitions and guidance to identify if a specific assumption or uncertainty is key for an application and requires further consideration of the impact to the application.

This assessment was applied to all uncertainties and assumptions identified via the methods in part b for the internal hazards (including fire).

RAI 5.01.d:

d. Provide a summary of the different types of dispositions used for those assumptions and uncertainties determined not to be key for this application.

Duke Energy Response to RAI 5.01.d:

Assumptions or uncertainties determined not to be key are those that do not meet the definitions of key uncertainty or key assumption in RG 1.200 Revision 2, NUREG-1855 Revision 1, or related references. Specifically, the following considerations were used to determine those assumptions and uncertainties that do not require further consideration as key to the application:

- The uncertainty or assumption is implementing a “consensus model” as defined in NUREG 1855 Rev 1.
- The uncertainty or assumption will have no impact on the PRA results and therefore no impact on the decision of HSS or LSS for any SSCs.
- There is no different reasonable alternative to the assumption which would produce different results and/or there is no reasonable alternative that is at least as sound as the assumption being challenged. (RG 1.200 Rev 2)
- The uncertainty or assumption implements a conservative bias in the PRA model, and that conservatism does not influence the results. These conservatisms are expected to be slight and only applied to minor contributors to the overall model. EPRI 1013491 uses the term “realistic conservatisms.” Thus, uncertainties/assumptions that implement realistic [slight] conservativisms can be screened from further consideration.
- EPRI 1013491 elaborates on the definition of a consensus model to include those areas of the PRA where extensive historical precedence is available to establish a model that has been accepted and yields PRA results that are considered reasonable and realistic. Thus, uncertainties/assumptions where there is extensive historical precedence that produces reasonable and realistic results can be screened from further consideration.

If the assumption or uncertainty does not meet one of the considerations above, then it is retained as “key” for the application and is presented in part e.

This assessment was applied to all uncertainties and assumptions identified via the methods in part b for the internal hazards (including fire).

RAI 5.01.e:

Provide a summary list of the key assumptions and uncertainties that have been identified for the application, and discuss how each identified key assumption and uncertainty will be dispositioned in the categorization process. The discussion should clarify whether the licensee is following NEI 00-04 guidance by performing sensitivity analysis or other accepted guidance such as NUREG-1855 Stages A and F.

Duke Energy Response to RAI 5.01.e:

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):

Index	Assumption/ Uncertainty	Discussion	Disposition
1.	**Assumptions within HEP calculations** **Model : Internal Events/Flood/Fire**	In the HNP internal events model, there are several assumptions made when developing the calculation of human error probabilities (HEP) that have the potential to have a more than negligible impact on the HEP values. These include the following 5 items: 1. Based on the diversity of the instrumentation, the unavailability of the condensate storage tank (CST) level indication is not modeled in the fault tree. These level transmitters are only required for a single human reliability analysis (HRA) event to align emergency service water (ESW) to the auxiliary feedwater (AFW) pumps when the CST drains. 2. It assumed that the operators have 10 minutes to respond to spurious opening of	These uncertainties associated with HRA development will be addressed by the NEI 00-04 Table 5-2 and Table 5-3 sensitivity to evaluate human error basic events at their 5th and 95th percentile for all system categorizations under 50.69 and presented to the IDP. There is no additional sensitivity required to evaluate these uncertainties. The sensitivity shows the impact on SSC importance in light of unknowns regarding human error probabilities. As such, SSC importance with respect to the 50.69 application is assessed in light of this uncertainty. Implementation of this sensitivity study is consistent with NEI 00-04 guidance.

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
		a component cooling water (CCW) relief valve and would receive indications such as low CCW surge tank level and low pump suction pressure. An operator action is included for the potential to isolate the non-essential header flow prior to pump failure. 3. Potential common restoration error events involving errors within a single procedure were judged to have moderate dependence when calculating the pre-initiator HEP, as the activities typically are performed by a single operator (or pair of operators) within a single day. 4. An 8-inch loss of coolant accident (LOCA) was selected to establish the time available for operators to manually start the residual heat removal (RHR) pumps. The range established for medium LOCAs is 5 inches to 13 inches, such that the time could be longer or shorter. 5. A 3-inch LOCA was selected to establish the time available for operators to establish an alternate high head safety injection (HHSI) path. The range established for the	

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
		small (S2) LOCA is 3 inches to 5 inches, such that the time could be shorter.	
2.	**Common Cause modeling** **Model : Internal Events/Flood/Fire**	In the HNP internal events model, there are several assumptions made when developing the calculation of common cause events. These three have the potential to have a more than negligible impact on the results. 1. In the calculations, multiple greek letter (MGL) factors for group size four were available, so the fifth valve was conservatively assumed to fail in common cause. 2. Common cause failures (CCFs) were considered only for those combinations of components which would disable both trains of the emergency safety features actuation system (ESFAS), since the probability of a lesser CCF disabling one train in conjunction with another random failure is considered probabilistically insignificant. 3. The method used to determine the common cause factor for the CCW pumps,	These uncertainties associated with CCF development will be addressed by the NEI 00-04 Table 5-2 and Table 5-3 sensitivity to evaluate CCF basic events at their 5th and 95th percentile for all system categorizations under 50.69 and presented to the IDP. There is no additional sensitivity required to evaluate these uncertainties. The sensitivity shows the impact on SSC importance in light of unknowns regarding CCF probabilities. As such, SSC importance with respect to the 50.69 application is assessed in light of this uncertainty. Implementation of this sensitivity study is consistent with NEI 00-04 guidance.

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
		while necessary, has a number of assumptions.	
3.	**Requirement to isolate accumulators after injection** **Model : Internal Events/Flood/Fire**	After the accumulators have emptied, the operator is required by emergency operating procedures to close the three accumulator discharge valves and lock the breakers open in order to prevent injecting nitrogen into the RCS. This action is not assumed to be required in the PSA. This is not an issue for large and medium LOCAs where any N2 is likely to be swept out of the break. However, for small LOCAs or transients in which the RCS must be depressurized to get to shutdown conditions, the insertion of N2 into the RCS could be an issue.	The action to isolate the accumulators is part of the action to cooldown and depressurize the RCS for transients and SGTRs, which is modeled in the PRA via HEP events OPER-9 “Failure to initiate RCS cooldown to use LPSI/RHR” and OPER-41 “Failure to initiate RCS cooldown to use LPSI/RHR (SGTR)”. However, the specific execution steps to isolate the accumulators are not included in the development of the HEP. The execution steps to isolate the accumulators will be added to these HEP event calculations prior to implementation of 50.69. Additionally, any uncertainty from these operator actions will also be addressed by the NEI 00-04 Table 5-2 and Table 5-3 sensitivity to evaluate human error basic events to their 5th and 95th percentile for all system categorizations under 50.69, and the results are presented to the IDP. Implementation of this model change and sensitivity study is consistent with

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
			NUREG-1855 Rev. 1 Stage F (i.e., update the PRA model) and NEI 00-04 guidance (i.e., HRA sensitivity).
4.	**CFC system is not impacted by LOCAs** **Model : Internal Events/Flood/Fire**	The Containment Fan Cooler (CFC) system is assumed to be protected from damage due to the LOCA initiator. Although failure of the CFCs due to LOCA effects for small LOCAs can be discounted, no specific spatial analysis has been performed for larger LOCAs.	A spatial analysis was performed which shows that two of the CFCs (AH-3 and AH-4) are on the 286 ft level of containment, while the other two (AH-1 and AH-2) are on the 236 ft level, such that a LOCA event would not impact CFCs on both floors. On the 286 ft level, the two CFCs are on the containment wall approximately 60 degrees apart such that a single large LOCA would not impact both CFCs. Similarly, on the 236 ft level the two CFCs are on the containment wall approximately 60 degrees apart such that again, a single large LOCA would not impact both CFCs. Based on this, a sensitivity was performed in which all large and medium LOCA events fail a single CFC, while the other 3 are unaffected. LERF was then calculated (CFCs do not impact CDF since they are only credited to prevent containment overpressure), and importance measures were generated. No basic events increased from LSS in

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
			the base case to HSS in the sensitivity case. As such, this sensitivity study shows 10 CFR 50.69 categorization is not sensitive to this uncertainty. Implementation of this sensitivity is consistent with the guidance in NUREG-1855 Stage E to quantify the impact of an uncertainty with respect to the application acceptance criteria.
5.	**Modeling of S/G SRVs** **Model : Internal Events/Flood/Fire** **Table # 128**	Although there are five safety relief valves (SRVs) on each of three steam generators (S/G) for a total of 15 valves, any one of which can perform the steam relief function to remove reactor decay heat, the model conservatively assumes that if any relief valve on a steam generator fails, then all relief valves on that steam generator also fail. A common cause failure of all 15 SRVs is also included. This is a very conservative assumption that overstates the likelihood of losing steam relief/decay heat removal. Taking credit for more of the SRVs, and consideration of more appropriate CCF groupings (such as all relief valves that are intended to open at the same	The HNP models will be updated to credit all SRVs, and appropriate common cause groups will be added to the model to include all relief valves that are intended to open at the same pressure, prior to implementation of 50.69. Any uncertainty from the new CCF events will also be addressed by the NEI 00-04 Table 5-2 and Table 5-3 sensitivity to evaluate CCF basic events to their 5th and 95th percentile for all system categorizations under 50.69, and the results are presented to the IDP.

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
		pressure - on each of the steam lines) would provide a more realistic result.	Implementation of this model change and sensitivity study is consistent with NUREG-1855 Rev. 1 Stage F (i.e., update the PRA model) and NEI 00-04 guidance (i.e., CCF sensitivity).
6.	**Modeling of pressurizer sprays** **Model : Internal Events/Flood/Fire**	During steam generator tube rupture (SGTR) scenarios requiring reactor coolant system (RCS) cooldown/depressurization, the pressurizer power operated relief valves (PORVs) are required to reduce RCS pressure. The PRA model assumes that the spray valves and/or the reactor coolant pumps are unavailable, and the RCS PORVs are always required to function.	A sensitivity has been performed to address the impact of including of pressurizer sprays to mitigate SGTR events. The sensitivity showed inclusion of sprays would decrease the CDF by approximately 0.3%. This extremely small change in CDF will have a negligible impact on component importance measures, and 10 CFR 50.69 categorization is not sensitive to this uncertainty. This approach is consistent with NUREG-1855 Rev. 1, Stage E (sensitivity study).
7.	**Failure of ESW due to backflow through NSW** **Model : Internal Events/Flood/Fire**	A failure of Emergency Service Water (ESW) due to backflow through the Normal Service Water (NSW) system if NSW fails to isolate is not postulated since a motor-operated valve (MOV) and a check valve would both need to fail to close if the NSW pump is unavailable or fails to run.	Further evaluation of the interconnection between the ESW and NSW systems shows that additional failures, beyond the MOV and check valve failures noted in the assumption, would be required to get backflow through the NSW system. When an NSW pump trips or is stopped, its discharge MOV automatically closes. Additionally, when an ESW pump starts,

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
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			the ESW cross-tie MOV (1SW-39 or 1SW-40) between the NSW supply and the ESW supply automatically closes. Thus, to get backflow through the NSW system on ESW start would require the running NSW pump to fail to run, failure of its discharge MOV to close, failure of the common NSW supply check valve (1SW-59) to close, and failure of an ESW cross-tie valve (1SW-39 or 1SW-40) to close. In the HNP PRA model the failure rate for an MOV to close on demand is 3.5E-03. Since there is no common cause between the NSW pump discharge MOV and the ESW cross-tie MOV, the probability of failure of both is 1.2E-05. The probability of the running NSW pump failing over the 24 hour mission time is 1.4E-04. Therefore, even ignoring the check valve, the likelihood of this event is approximately 1.7E-09. Therefore, this assumption has a negligible impact on component importance measures, and 10 CFR 50.69 categorization is not sensitive to this uncertainty. This sensitivity study is consistent with NUREG-1855 Rev. 1, Stage E (sensitivity study), to quantify the impact of an

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
			uncertainty with respect to the application acceptance criteria.
8.	**Use of Generic data for PAL** **Model : Internal Events/Flood/Fire**	The model uses generic screening values for the personnel air lock (PAL) failures. Plant-specific information may significantly vary from these values.	A sensitivity was performed to evaluate the impact of this uncertainty by increasing the PAL mechanical failure rate and leakage failure rate by a factor of 2. LERF was then calculated (the PAL has no impact on CDF) and importance measures were generated. Only one component basic event out of all component basic events in the model increased from LSS in the base case to HSS in the sensitivity case (the basic event for the PAL was already HSS in the base case). A similar sensitivity was performed by setting the mechanical failure rate and leakage failure rate for the PAL to 0, calculating LERF, and generating importance measures. In this case, no component basic events changed from LSS in the base case to HSS in the sensitivity case. As such, this sensitivity study shows 10 CFR 50.69 categorization is not sensitive to this uncertainty. This sensitivity is consistent with the guidance in NUREG-1855 Rev. 1 Stage

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
			E, to quantify the impact of an uncertainty with respect to the application acceptance criteria.
9.	**Exclusion of common cause failure of breakers** **Model: Internal Events/Flood/Fire**	There are breakers connecting the non safety-related 6.9 kV bus supply to the emergency buses which open to allow the emergency diesel generators (EDGs) to close onto the bus. Since there are two in-series breakers, both of which would have to fail to trip, failure of these breakers was not included in the model due to low probability. The impact of a common cause event on the model was not determined.	Exclusion of a common cause failure of the breakers to open (124-SB and 125-SB or 104-SA and 105-SA) as a failure of the EDG supply to the bus may impact the acceptance criteria for 10 CFR 50.69 categorization for these breakers. The independent failures of each breaker, and appropriate CCF events will be added to the model prior to implementation of 50.69. Any uncertainty from the new CCF events will also be addressed by the NEI 00-04 Table 5-2 and Table 5-3 sensitivity to evaluate CCF basic events to their 5th and 95th percentile for all system categorizations under 50.69, and the results are presented to the IDP. Implementation of this model change and sensitivity study is consistent with NUREG-1855 Rev. 1 Stage F (i.e., update the PRA model) and NEI 00-04 guidance (i.e., CCF sensitivity).

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
10.	**Number of SI Accumulators required for large LOCAs** **Model: Internal Events/Flood/Fire**	One high volume, low-pressure injection pump and two passive safety injection accumulators are required for successful RCS makeup after a large break LOCA during injection to meet the licensing design basis. This assumes a double-ended pipe break at the upper range of the possible large LOCA events. Lesser events would not be as extreme and would not necessarily require accumulator injection.	A sensitivity was performed by modifying the fault tree to require only 1 Accumulator to inject during a large LOCA, and re-running the CDF and LERF results. No basic events increased from LSS in the base case to HSS in the sensitivity case for either CDF or LERF. As such, this sensitivity study shows 10 CFR 50.69 categorization is not sensitive to this uncertainty. Implementation of this sensitivity is consistent with the guidance in NUREG-1855 Rev. 1 Stage E to quantify the impact of an uncertainty with respect to the application acceptance criteria.
11.	**Floor value for HRA combinations** **Model: Internal Events/Flood**	For those cutsets with three or more HEPs a lower bound of 1E-06 was used. This lower bound was decreased one decade [from the NUREG-1972 recommendation] to account for the fact that many of the third and fourth HEPs are actions that occur many hours after the initiating event and thus new individuals are there to not only respond but they also perform an independent review of the actions and diagnosis of the event.	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. All HEP

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
			combinations which were determined to be below 1.0E-05 were evaluated to determine the level of dependency between the actions and it was confirmed that the dependencies were low enough to support the lower value. Additionally, HEP combination events were never assigned a value of less than 1.0E-06. Based on the above, the method used to develop dependent HEP values is a consensus method and eliminates the need to explore an alternative hypothesis. This approach is consistent with the guidance in NUREG-1855 Rev. 1 Stage E, section 7.2.4. Additionally, this will be addressed by the NEI 00-04 Table 5-2 sensitivity to evaluate human error basic events at their 5th and 95th percentile for all system categorizations under 50.69 and presented to the IDP.
12.	**Manipulation time for flood Isolation HRAs is 30 minutes**	The Tm (manipulation time) is based on FSAR analysis that states a 30 minute time for rupture location determination, isolation, and securing of the applicable pump. The training observed on internal flooding referenced this document as the	This uncertainty associated with HRA development will be addressed by the NEI 00-04 Table 5-2 sensitivity to evaluate human error basic events at their 5th and 95th percentile for all system

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
	Model: Internal Flood	source of the 30 minutes criteria. This value could be more or less depending on the particular flood scenario.	categorizations under 50.69 and presented to the IDP. There is no additional sensitivity required to evaluate these uncertainties. The sensitivity shows the impact on SSC importance in light of unknowns regarding human error probabilities. As such, SSC importance with respect to the 50.69 application is assessed in light of this uncertainty.
13.	**HRAs not considered blocked in spray areas** **Model: Internal Flood**	Blocked HRAs were only considered for flood and HELB events. Spray events were assumed not to result in conditions that would prevent operator actions from being performed, since it is very likely an operator would be able to complete actions in a spray area. However, no actions involving electrical equipment were credited in spray areas.	This uncertainty associated with HRA development will be addressed by the NEI 00-04 Table 5-2 sensitivity to evaluate human error basic events at their 95th percentile failure rate for all system categorizations under 50.69 and presented to the IDP. There is no additional sensitivity required to evaluate these uncertainties. The sensitivity shows the impact on SSC importance in light of unknowns regarding human actions in spray areas. As such, SSC importance with respect to the 50.69 application is assessed in light of this uncertainty.
14.	**Credit for Incipient Detection**	Incipient detection in low voltage cabinets provides additional 60 minutes for manual suppression.	See response to RAI 06 (ADAMS Accession No. ML 18291A606)

Table 1 – Key Assumptions and Uncertainties from the Internal Hazard Models (Internal Events, Internal Flood, and Fire):			
Index	**Assumption/ Uncertainty**	**Discussion**	**Disposition**
	Model: Fire		
15.	**Assumption related to manual detection** **Model: Fire**	In the HNP fire model, it is assumed that if no detection system is installed in an area, manual detection will occur in 15 minutes. Although this assumption is probably realistic, some fire compartments may have a relatively low potential of fire detection, especially if they are closed and have low occupancy levels.	This uncertainty associated with manual fire suppression will be addressed by the NEI 00-04 Table 5-3 sensitivity to take no credit for manual suppression for all system categorizations under 50.69 and presented to the IDP. There is no additional sensitivity required to evaluate these uncertainties. The sensitivity shows the impact on SSC importance in light of unknowns regarding manual suppression. As such, SSC importance with respect to the 50.69 application is assessed in light of this uncertainty.

RAI 5.01.f:

If NEI 00-04 or NUREG-1855 guidance is not used (e.g. all of the Stages A through F in NUREG 1855, Revision 1) provide justification that the licensee's approach is adequate to identify, capture the impact, and disposition key assumptions and uncertainties to support the categorization process.

Duke Energy Response to RAI 5.01.f:

The response provided in subparts b through e of this RAI are consistent with the guidance in NUREG-1855 Rev 1 and NEI 00-04.

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Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400 / Renewed License No. NPF-63

Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors"

Attachment 1

HNP 50.69 PRA Implementation Items

The table below identifies the items that are required to be completed prior to implementation of 10 CFR 50.69 at Shearon Harris Nuclear Power Plant (HNP), Unit No. 1. Issues identified below will be addressed and any associated changes made, focused scope peer reviews performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and findings resolved and reflected in the PRA of record prior to implementation of 10 CFR 50.69.

Harris 50.69 PRA Implementation Items	
Description	Resolution
i. In the Fire PRA model, detailed analysis is needed for four significant HFE's identified in open finding HRC-C1-3. This condition is described in response to RAI 02.e in Duke letter dated October 18, 2018.	Duke Energy will perform detailed analysis in accordance with current methods for the four significant HFE's identified and incorporate the analysis into the Harris Fire PRA model as indicated in the Duke letter dated October 18, 2018.
ii. Update the HNP Fire PRA model to incorporate NUREG-2180 or other NRC acceptable methodology for incipient detection credit. If this update is determined to be a PRA model upgrade per the 2009 ASME/ANS PRA standard, then conduct a focused scope peer review. Any findings from the focused scope peer review will be resolved and closed per an NRC approved process, or the findings will be dispositioned for the application and submitted for NRC review and approval prior to implementing 10 CFR 50.69. This condition is described in response to RAI 06 in Duke letter dated October 18, 2018.	The Fire PRA model will be updated to credit incipient detection per NUREG-2180 or other NRC acceptable methodology, as described in Duke letter dated October 18, 2018.

iii. Update the HNP Fire PRA model to address finding FSS-F3-01 to meet Capability Category II of the ASME/ANS RA-Sa-2009 as endorsed by RG 1.200, Revision 2. If this update is determined to be a PRA model upgrade per the 2009 ASME/ANS PRA standard, then conduct a focused scope peer review. Any findings from the focused scope peer review will be resolved and closed per an NRC approved process, or the findings will be dispositioned for the application and submitted for NRC review and approval prior to implementing 10 CFR 50.69. This condition is described in response to RAI 02.f in Duke letter dated October 18, 2018.	The fire PRA model will be updated to account for scenarios to address fire induced failure of structural steel in the Turbine Building, as indicated in response to RAI 02.f contained in Duke letter dated October 18, 2018.
iv. Update the HNP Internal Events, Internal Flood and Fire PRA models to resolve uncertainties. a. The execution steps to isolate RCS accumulators as detailed in the EOPs will be added to the appropriate HEP event calculations. b. The HNP models will be updated to credit all safety relief valves (SRVs) and appropriate common cause groups will be added to the model to include all relief valves that are intended to open at the same pressure. c. The independent failures of breakers 124-SB and 125-SB or 104-SA and 105-SA to open as a failure of the EDG supply to the emergency buses along with their common cause failure events will be added to the model. These conditions are described in response to RAI 5.01 in Duke Energy letter dated April 23, 2019.	The HNP PRA models will be updated to account for isolation of the RCS accumulators and steam generator SRVs, as indicated in response to RAI 5.01 of Duke Energy letter dated April 23, 2019.

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Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400 / Renewed License No. NPF-63

Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Reactors”

Attachment 2

Markup of Proposed Renewed Facility Operating License

L. This license is effective as of the date of issuance and shall expire at midnight on October 24, 2046.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Attachment 1 – TDI Diesel Engine Requirements
2. Appendix A – Technical Specifications
3. Appendix B – Environmental Protection Plan
4. Appendix C – Antitrust Conditions
5. Appendix D – Additional Conditions

Date of Issuance: December 17, 2008

APPENDIX D

ADDITIONAL CONDITIONS

RENEWED LICENSE NO. NPF-63

Duke Energy Progress, LLC shall comply with the following conditions on the schedule noted below:

Amendment Number	Additional Conditions	Implementation Date
[NUMBER]	Duke Energy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit 1 License Amendment No. [XXX] dated [DATE]. Duke Energy will complete the implementation items list in Attachment 1 of Duke Energy letter to the NRC dated April 23, 2019 prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process. Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).	Prior to implementation of 10 CFR 50.69.