NRC 2018-0038
10 CFR 50.90

August 10, 2018

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

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301

Subject: Response to Request for Additional Information Regarding License Amendment Request 287, Application to Adopt 10 CFR 50.69, "Risk informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants"

References:

1. NextEra Energy Point Beach, LLC letter NRC 2017-0043, "License Amendment Request 287, Application to adopt 10 CFR 50.69, 'Risk informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants," August 31, 2017 (ML17243A201)
2. NextEra Energy Point Beach, LLC letter NRC 2017-0052, "Supplement to LAR 287, Application to adopt 10 CFR 50.69, 'Risk informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants'," October 26, 2017 (ML17299A012)
3. NRC e-mail "Final- Point Beach - 50.69 Risk Informed LAR 287, Request for Additional Information, CAC Nos. MG0196 and MG0197, EPID L-2017-LLA-0284," June 27, 2018

In Reference 1 and supplemented by Reference 2, NextEra Energy Point Beach, LLC submitted a license amendment request for the Point Beach Nuclear Plant, Units 1 and 2. The proposed amendment would revise the licensing basis by adding a license condition to allow for 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."

In Reference 3, the NRC staff requested additional information to support its review of the request. The enclosure to this letter provides the requested information. Attachment 1 to the enclosure lists the items that must be completed prior to implementation of the 10 CFR 50.69 categorization process.

NextEra Energy Point Beach, LLC

6610 Nuclear Road, Two Rivers, WI 54214

Attachment 2 provides markups of the Point Beach Units 1 and 2 operating licenses containing a proposed license condition.

This response does not alter the conclusions in Reference 1 that the proposed change does not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with the change.

This letter contains no new or revised regulatory commitments.

Should you have any questions regarding this submittal, please contact Mr. Eric Schultz, Licensing Manager, at 920-755-7854.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on Aug 10, 2018

Sincerely,

Robert Craven
Site Director
NextEra Energy Point Beach, LLC

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
Public Service Commission of Wisconsin

Enclosure

Response to Request for Additional Information

RAI 01 – Facts and Observations (F&O) Closure Process

LAR Section 3.3 states that an F&O closure peer review was performed using the process documented in Appendix X to Nuclear Energy Institute (NEI) 05-04, NEI 07-12, and NEI 12-13, "Close-out of Facts and Observations" as accepted by the U.S. Nuclear Regulatory Commission (NRC) in the letter from Joseph Giitter and Mary Jane Ross-Lee, NRC to Greg Krueger, NEI, dated May 3, 2017 (ADAMS Accession Number ML17079A427). NRC staff provided observations of this F&O Closure on July 2017 (ADAMS Accession Number ML17356A055). Provide the following information to confirm that the July 2017 F&O closure review was performed consistent with the NRC accepted process, as discussed in the May 3, 2017 letter.

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

   i. Summary of the scope of the peer review.

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

b. Confirm that the closure review team was provided with a written assessment and justification of whether the resolution of each F&O, within the scope of the independent assessment, constitutes a probabilistic risk assessment (PRA) upgrade or maintenance update, as defined in American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," as qualified by 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 Number ML090410014). If the written assessment and justification for the determination of each F&O was not performed and reviewed by the F&O closure review team, provide all the finding-level F&Os and the dispositions of these F&Os as it pertains to the impact on the 10 CFR 50.69 application. Alternatively, perform an Independent Assessment F&O closure review consistent with Appendix X, as accepted, with conditions, by the NRC letter dated May 3, 2017, and provide any additional open F&Os and associated dispositions as a result of this review.

c. Appendix X (ADAMS Accession Number ML17086A451), Section X.1.3 includes five criteria for selecting members of the closure review team.

   Describe how the selection of members for the July 2017 independent assessment met the five criteria.

d. 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 (CC-II) for SRs from ASME/ANS RA-Sa-2009, as endorsed, with clarifications and qualifications, by RG 1.200, Revision 2.

e. 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 associated disposition for the application.

f. For any SRs that were found to be only met at CC I by previous peer review team(s), summarize the disposition of these SRs and how it was concluded they now meet CC II. Include discussion of whether all associated F&Os described what was needed to achieve CC II and how the F&O reviewed and closed by the F&O closure team.

Response

a. A focused scope peer review was not conducted concurrently with the F&O Closure process.

b. On the initial F&O closure review, the closure team was not provided a written assessment and justification of whether a resolution constituted a PRA upgrade or maintenance update. The closure team reviewed each resolution independently to judge if the change represented a PRA upgrade or update using the industry guidance. The conclusion for each finding as to whether it was an upgrade was identified by a yes/no.

   On a follow-up closure review, the closure team was provided a detailed written assessment and justification for each finding that was previously assessed as closed, whether the resolution constitutes a PRA upgrade or maintenance update as defined in American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," as qualified by Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities. All the resolutions were assessed as maintenance updates and concurred upon by the independent review team.

c. The closure review team members met the five criteria for selecting team members of Appendix X (ADAMS Accession Number ML17086A451), Section X.1.3. These requirements are contained in fleet procedures. They had the appropriate expertise, experience level, independence, and background knowledge (Supporting Requirements (SRs), review process, etc.). Team member experience and independence is documented in the F&O closure report. A pre-job brief and kick-off meeting was held to re-iterate the overall review process, criteria, and documentation.

d. The closure review team decided if the candidate findings had been adequately resolved and could be considered "closed-out" via consensus, using the appropriate SRs of the ASME/ANS PRA Standard for the review criteria. The independent review team assessed each finding as: (1) Finding is Closed - resolution adequately addresses the finding. The assessed SR capability category is Met or Met at CC-II (or higher) or (2) Finding is not closed - resolution does not adequately address the finding. This is per fleet procedure.

e. All finding level F&Os that had not been previously assessed by an independent review or peer review were in the scope of the F&O closure review.

f. See response to RAI 01d.

RAI 02 – Open/Partially Open Findings in the Process of Being Resolved

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. 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. The NRC staff notes that 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" (ADAMS Accession Number ML14190A267). The NRC staff notes that modelling issues that can cause even small impacts to core damage frequency (CDF) and large early release frequency (LERF) (both increases and decreases) can potentially increase the risk importance values for certain system components above the threshold criteria for determining safety significance specified in Section 5 of NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, (ADAMS Accession Number ML052910035). In light of these observations, address the following:

a. F&O IE-A1-01 regarding special initiating events:

The disposition to F&O IE-A1-01 presented in the TSTF-425 LAR indicates that a number of special initiators related to the 4160 volts alternating current (VAC) Vital Switchgear bus were not included in the internal events PRA (IEPRA) model because they were considered not significant, and estimated the CDF for sequences associated with these initiators as high as 1.9E-07/year. It is not clear to the NRC staff that excluding these sequences cannot increase the risk importance values for specific system components above the threshold criteria for determining safety significance as discussed in NEI 00-04, Section 5. Therefore:

i. Provide justification that exclusion of scenarios associated with the cited 4160 VAC Vital Switchgear bus related initiators has no impact on the 10 CFR 50.69 categorization results, or

ii. Propose a mechanism in response to RAI 04 that ensures F&O IE-A1-01 will be resolved in the PRA model prior to the 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 this F&O.

Response

The loss of a 4,160 VAC bus will be added to the PRA model as a special initiator to resolve this finding. As identified in the response to RAI 04, open finding IE-A1-01 will be resolved in the PRA model prior to implementing the 10 CFR 50.69 categorization process.

b. F&O AS-B6-01 and F&O SY-A21-01 regarding excluded electrical alignment

The dispositions to F&Os AS-B6-01 and SY-A21-01 presented in the TSTF-425 LAR stated that although Emergency Diesel Generator (EDG) load management is a potential failure mode for EDGs, it is not modeled in the PRA. The TSTF-425 LAR further provided an estimate of the low likelihood that the EDG load management would be needed. The estimated low likelihood is based on the assumption that the events used in the estimate are independent. The NRC staff notes that loss of coolant accident (LOCA) initiators can induce loss of offsite power (LOOP) events and LOOP initiators can induce LOCAs (i.e., the need for safety injection (SI)) making these events dependent. Additionally, NRC staff notes that modelling exclusions that cause just small increases to CDF/LERF can impact the categorization of specific structures, systems, and components (SSCs). Therefore:

i. Provide justification that the excluded scenarios involving failure of EDG load management has no impact on the 10 CFR 50.69 categorization results. Include consideration of LOCA induced LOOP events and LOOP induced LOCA events, or

ii. Propose a mechanism in response to RAI 04 that ensures F&Os AS-B6-01 and SY-A21-01 will be resolved in the PRA model 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 this F&O.

Response

A new failure mode associated with EDG load management will be added to the PRA model to resolve this finding. As identified in the response to RAI 04, open findings AS-B6-01 and SY-A21-01 will be resolved in the PRA model prior to implementing the 10 CFR 50.69 categorization process.

c. F&O AS-B7-01 regarding inadequate treatment of time-phased modelling

The disposition to F&O AS-B7-01 presented in the TSTF-425 LAR states that recovery of LOOP events is only credited for station blackout (SBO) scenarios and the direct current (DC) batteries are conservatively assumed to fail at time zero. 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:

i. Provide justification that the conservative modelling associated with LOOP recovery and not crediting DC batteries has no impact on the 10 CFR 50.69 categorization results, or

ii. Propose a mechanism in response to RAI 04 that ensures F&O AS-B7-01 will be resolved in the PRA model 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 this F&O.

Response

The treatment of power recovery after LOOP events and battery modeling in the PRA model will be revised to be more realistic to resolve this finding. As identified in the response to RAI 04, open finding AS-B7-01 will be resolved in the PRA model prior to implementing the 10 CFR 50.69 categorization process.

d. F&O HR-D1-01 regarding detailed assessments for significant human failure events (HFEs)

The disposition to F&O HR-D1-01 in LAR Attachment 3 states that no further changes are required. Also, based on the disposition presented in the TSTF-425 LAR, the F&O appeared resolved. Yet, the LAR associated with the adoption of 10 CFR 50.69 states regarding the disposition of this F&O, "[p]rior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding." Therefore:

i. Provide the reason why this F&O could not be closed by the F&O closure review in July 2017.

ii. Justify why this F&O has no impact on the 10 CFR 50.69 categorization results or propose a mechanism in response to RAI 04 that ensures F&O HR-D1-01 will be resolved in the PRA model 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 this F&O.

Response

Since appropriate documentation was not provided to the closure review team in a timely fashion, this finding was not closed. As identified in the response to RAI 04, independent agreement that this finding is resolved in the PRA model will be obtained prior to implementing the 10 CFR 50.69 categorization process.

e. F&O IFQU-A6-01 regarding HFEs for internal flooding scenarios

The description for F&O IFQU-A6-01 states, "HFEs from internal events are 'adjusted' with inadequate basis for those adjustments." The disposition for this F&O presented in the TSTF-425 LAR states that "stress multipliers" from Table 20-16 of NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, Final Report," 1983 (ADAMS Accession Number ML071210299), which are referred to as "stress modifiers" in NUREG/CR-1278, were used to account for the stress associated with an internal flooding event. However, Table 20-16 of the cited NUREG lists modifiers to be applied to human error probabilities (HEPs) for different stress levels and they are not specific to internal flooding. In light of these observations:

i. Justify that the HRA method used is adequate for use in the internal flooding PRA and the values used from NUREG/CR-1278 (that do not seem to relate to internal flooding stress) are appropriate.

ii. Confirm that scenario specific internal flooding HFEs were developed.

iii. Alternatively to items I, and ii above, propose a mechanism in response to RAI 04 that ensures F&O IFQU-A6-01 will be resolved in the PRA model 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 this F&O.

Response

No stress modifiers were applied to the HEP values used in the internal flood PRA model. The internal flood documentation will be updated to justify the HEP values actually used in the model to resolve this finding. As identified in the response to RAI 04, open finding IFQU-A6-01 will be resolved in the PRA model prior to implementing the 10 CFR 50.69 categorization process.

f. <u>F&O PRM-B2-01</u> regarding the impact of internal events findings on the fire PRA

The description for F&O PRM-B2-01 states that resolution of internal events findings could impact fire PRA evaluations. The disposition to this F&O states, "[a]s of the time of this submittal, the only remaining open internal events peer review findings identified…are AS-B6-01 and SY-A21-01." NRC staff notes that AS-B6-01 and SY-A21-01 are the subject of RAI 02.b above. Therefore:

i. Provide justification that not updating the fire PRA to include the resolution of internal events F&Os AS-B6-01 and SY-A21-01 has no impact on the 10 CFR 50.59 categorization results, or

ii. Propose a mechanism in response to RAI 04 that ensures F&O PRM-B2-01 will be resolved in the PRA model 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 this F&O.

Response

The identified internal events findings will be resolved in the manner described in the response to RAI 02.b above. Specifically, a new failure mode associated with EDG load management will be added to the PRA model to resolve this finding. As identified in the response to RAI 04, open findings AS-B6-01 and SY-A21-01 will be resolved in the PRA model prior to implementing the 10 CFR 50.69 categorization process.

g. F&O HRA-B2-01 regarding credit for graphically distinct procedural steps

The description for F&O HRA-B2-01 found that credit for graphically distinct factors is taken for all HRA events, as opposed to taking credit for graphically distinct procedural steps that stand out from the other steps. The disposition for this F&O states, "[o]nly about 10% of the HEPs that credited graphically distinct procedure steps would be increased by more than a factor of 2." The disposition concludes "[b]ased on this review, the impact on the model from this finding is judged minimal." NRC staff notes that modelling issues that can cause even small increases to CDF and LERF can potentially increase the risk importance values for specific system components above the threshold criteria for determining safety significance specified in NEI 00-04, Section 5. Therefore:

i. Provide justification that not performing the cited correction to the HRA has no impact on the 10 CFR 50.69 categorization results, or

ii. Propose a mechanism in response to RAI 04 that ensures F&O HRA-B2-01 will be resolved in the PRA model 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 this F&O.

Response

The HEPs developed for the Fire PRA model will be updated to remove the graphically distinct credit in the cognitive portion of the HEP. The dependency analysis will be updated and the Fire PRA quantified using these updated Human Error Probabilities. As identified in the response to RAI 04, open finding HRA-B2-01 will be resolved in the PRA model prior to implementing the 10 CFR 50.69 categorization process.

h. F&O FQ-A1-01 regarding FRANX and CAFTA discrepancies

The description for F&O FQ-A1-01 states, "some basic events that have been mapped to scenarios, components, or cables are not found in the CAFTA model." The associated disposition states that "information in the mapping table should be reviewed to eliminate the extraneous information and eliminate the discrepancies." The disposition for this F&O presented in the PBNP LAR for transition to the National Fire Protection Association (NFPA) Standard 805 (ADAMS Accession Number ML13182A353) indicates that this review has been performed which identified six failure events that were excluded from the PRA associated with the MCR. The PBNP NFPA 805 LAR for this disposition also states that a sensitivity study was conducted which determined that the exclusion of these basic events have a negligible impact on fire risk. In light of these observations:

i. Identify which fire PRA modeling actions remain to be performed in order to fully resolve F&O FQ-A1-01, and justify that completion of the remaining actions has no impact on the 50.69 categorization results, or

ii. Propose a mechanism in response to RAI 04 that ensures F&O FQ-A1-01 will be resolved in the PRA model 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 this F&O.

Response

The basic event mapping tables will be reviewed and compared to the present basic event mapping associated with each equipment or cable. Those items that are no longer needed will be removed and any incorrect mapping will be updated. The Fire PRA model will be quantified using this updated mapping table. As identified in the response to RAI 04, open finding FQ-A1-01 will be resolved in the PRA model prior to implementing the 10 CFR 50.69 categorization process.

RAI 03 – PRA maintenance versus PRA upgrade

Section 3.2 of the LAR states "[a]ll the PRA models described below have been peer reviewed and there are no PRA upgrades that have not been peer reviewed." Justify that any PRA update performed to resolve any F&Os discussed in RAIs 01 and 02 or PRA modelling uncertainties identified in RAI 08, does not constitute a "PRA upgrade" as defined in ASME/ANS RA-Sa-2009, Section 1-5.4, as qualified by RG 1.200, Revision 2. If an upgrade has been identified, provide the summary and results of the focused-scope peer review performed on the upgrade, including all finding-level F&Os and a disposition for each F&O as it pertains to the impact on the 10 CFR 50.69 application.

Response

As confirmed by the independent review team, there have been no PRA upgrades for any of the closed findings reviewed to date. All open findings discussed in RAIs 01 and 02 or PRA model changes to address uncertainty items in RAI 08 will be independently reviewed to determine if the resolution of those items in the PRA model constitutes a PRA upgrade. If that review identifies a change is a PRA upgrade, a focused-scope peer review will be performed for that change. The closure of these items will be done in accordance with approved NRC methodology.

RAI 04 – Implementation Items to be Completed Prior to Implementing 10 CFR 50.69

LAR Section 3.2.3 states that "[a]n evaluation will be performed of the as-built, as-operated plant against the SMA [Seismic Margin Assessment] SSEL [Safe Shutdown Equipment List]. The evaluation will compare the as-built, as-operated plant to the plant configuration originally assessed by the SMA. Differences will be reviewed to identify any potential impacts to the equipment credited on the SSEL. Appropriate changes to the credited equipment will be identified and documented."

Further, Attachment 3 of the LAR indicates a number of planned updates to the PRA model before implementation of the 10 CFR 50.69 program.

a. Provide a list of each activity and PRA change, including all items from RAIs 01, 02, 03 and 08, that will be completed prior to implementing the 10 CFR 50.69 categorization process (i.e., implementation items). A table of "implementation items" has been used in previous risk-informed licensing actions to formally identify issues requiring resolution before implementation of the amendment.

b. Provide a method to ensure that all implementation items under part a. will be addressed and any associated changes will be made, that focused-scope peer reviews will be performed on any changes that are PRA upgrades as defined in the PRA standard, and any resulting findings will be closed via 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 (for example, a license condition that all applicable implementation items will be completed prior to categorization).

Response

a. Attachment 1 provides a table of implementation items that must be completed prior to implementing the 10 CFR 50.69 categorization process.

b. Attachment 2 provides a proposed license condition that stipulates the activities that must be completed prior to implementation of the 10 C FR 50.69 categorization process.

RAI 05 – Overall Categorization Process

LAR Section 3.1.1, "Overall Categorization Process," has two different sets of bulleted elements and concludes with an additional list of ten elements. The elements discuss: training that will be provided, the different hazard models, and PRA model results. However, it is not clear to the NRC staff what the sequence of evaluations will be in the categorization process, what information will be developed and used, and what guidance on acceptable decisions by the Integrated Decision-Making Panel (IDP) will be followed during the categorization of each system.

a. Summarize, in the order they will be performed, the sequence of elements or steps that will be followed to categorize a respective system. A flow chart, such as that provided in the NEI presentation (ADAMS Accession Number ML17249A072) for the September 6, 2017, public meeting with NEI regarding 10 CFR 50.69 LARs (ADAMS Accession Number ML17265A020) may be provided instead of a description. The steps should include:

   i. The input from all PRA evaluations such as use of the results from the internal events, internal flooding, seismic, and fire PRAs;

   ii. The input from non-PRA approaches (other external events, and shutdown);

   iii. The input from the responses to the seven qualitative questions in NEI 00-04, Section 9.2;

   iv. The input from the defense-in-depth (DID) matrix;

   v. The input from the passive categorization methodology.

b. Clarify the difference between "preliminary high safety significant (HSS)" and "assigned HSS" and identify which inputs can, and which cannot, be changed from preliminary HSS to low safety significant (LSS) by the IDP. Confirm that the approach is consistent with the guidance in NEI 00-04, as endorsed by RG 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," (ADAMS Accession Number ML061090627).

c. Clarify which steps of the process are performed at the function level and which steps are performed at the component level. Describe how the categorization of the component impacts the categorization of the function, and vice-versa. Describe any instances in which the final safety significance of the function would differ from the safety significance of the component(s) that support the function, and confirm that the approach is consistent with the guidance in NEI 00-04 as endorsed by RG 1.201, Revision 1.

d. NEI 00-04, Section 7, states that "if any SSC is safety significant, from either the PRA-based component safety significance assessment (Section 5) or the defense-in-depth assessment (Section 6), then the associated system function is preliminary safety significant." Describe whether your categorization process is consistent with or differs from the guidance in NEI 00-04, Section 7, where functions supported by any HSS component(s) will be assigned as HSS. If your categorization process differs from the guidance in Section 7 of NEI 00-04 cited above where functions supported by any HSS component(s) will be assigned HSS, justify the approach.

e. The industry flow chart presented at the September 6, 2017, public meeting shows that the passive categorization would be undertaken separately from the active categorization.

   i. Explain how the results from the passive categorization will be integrated with the overall categorization results.

   ii. If the results from the passive categorization can be changed by the IDP, explain and justify the proposed approach.

f. NEI 00-04, Section 9.2.2, "Review of Safety Related Low Safety-Significant Functions/SSCs," states "in making their assessment, the IDP should consider the impact of loss of the function/SSC against the remaining capability to perform the basic safety functions." This section also provides seven questions that should be considered for making the final determination of the safety-significance for each system function/SSC. However, it is unclear in the LAR how the IDP will collectively assess these seven specific questions. For example, is a function/SSC considered HSS when the answer to any one question is false (e.g., failure of the function/SSC will directly cause an initiating event or adversely affect the defense-in-depth remaining to perform the function). Explain how the IDP will collectively assess the seven specific questions to identify a function/SSC as LSS as opposed to HSS.

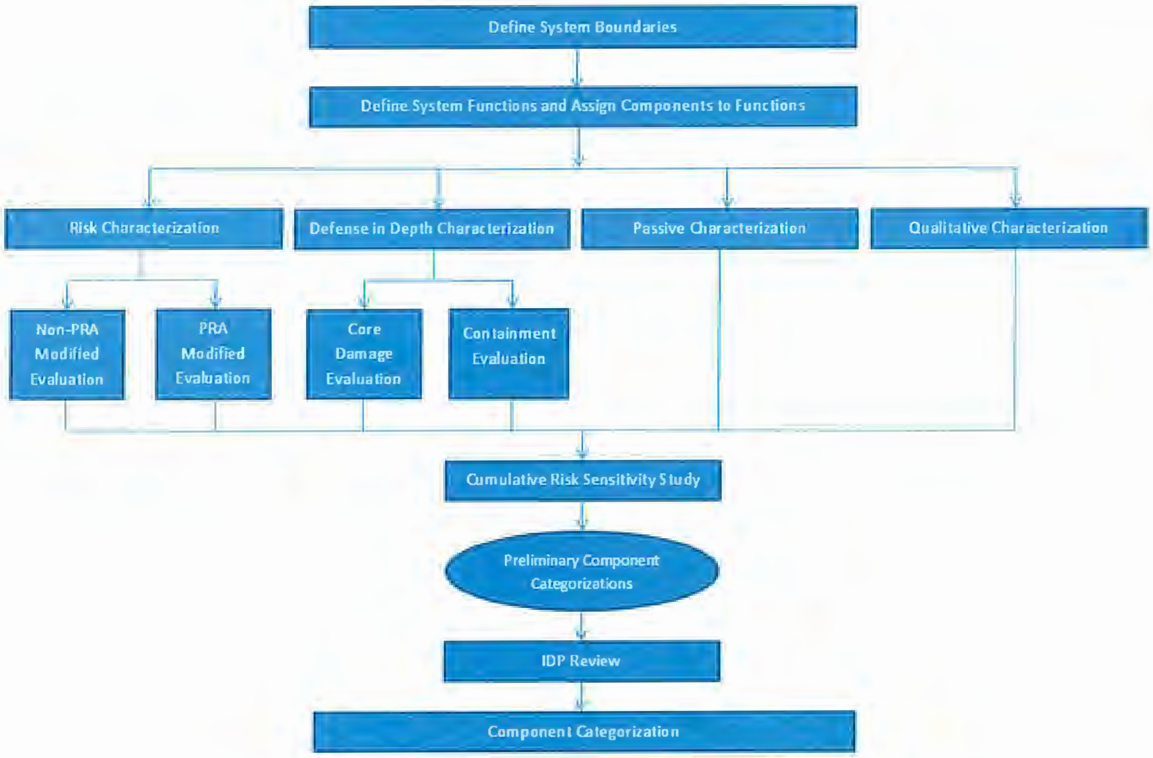
g. NEI 00-04, Section 7.1 states, "[d]ue to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC or part thereof should be assigned the highest risk significance for any function that the SSC or part thereof supports." Section 4 of NEI 00-04 also states that a candidate LSS SSC that supports an interfacing system should remain uncategorized until all interfacing systems are categorized. Confirm that the cited guidance in NEI 00-04 will be followed and that any functions/SSCs that serve as an interface between two or more systems will not be categorized until the categorization for all of the systems that they support is completed and that SSCs that support multiple functions will be assigned the highest risk significance for any of the functions they support, or otherwise justify your proposed approach.

Response

a-c. 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)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. Therefore, the order in which each of the elements of the categorization process (listed below) is completed is flexible and as long as they are all completed; they may even be performed in parallel. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as LSS by all other elements. Similarly, NEI 00-04 only requires Item 4 to be completed for safety related active components/functions categorized as LSS by all other elements.

1. PRA-based evaluations (e.g., the internal events, internal flooding, and fire PRAs)
2. Non-PRA approaches (e.g., seismic safe shutdown equipment list (SSEL), other external events screening, and shutdown assessment)
3. Seven qualitative criteria in Section 9.2 of NEI 00-04
4. Defense-in-depth assessment
5. Passive categorization methodology

On the following page is an example of the major steps of the categorization process described in NEI 00-04.



Categorization of SSCs will be completed per the NEI 00-04 process, as endorsed by RG 1.201, which includes the determination of safety significance through the various elements identified above. The results of these elements are used as inputs to arrive at a preliminary component categorization (i.e., High Safety Significant (HSS) or Low Safety Significant (LSS)) that is presented to the Integrated Decision-Making Panel (IDP). Note: the term "preliminary HSS or LSS" is synonymous with the NEI 00-04 term "candidate HSS or LSS." A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 1 below. The safety significance determination of each element, identified above, is independent of each other, and therefore the sequence of the elements does not impact the resulting preliminary categorization of each component or function. Consistent with NEI 00-04, the categorization of a component or function will only be "preliminary" until it has been confirmed by the IDP. Once the IDP confirms that the categorization process was followed appropriately, the final Risk Informed Safety Class (RISC) category can be assigned.

The IDP may direct and approve detailed categorization of components in accordance with NEI 00-04, Section 10.2. The IDP may always elect to change a preliminary LSS component or function to HSS; however, the ability to change component categorization from preliminary HSS to LSS is limited. This ability is only available to the IDP for select process steps as described in NEI 00-04 and endorsed by RG 1.201. Table 1 summarizes these IDP limitations in NEI 00-04. The steps of the process are performed at either the function level, component level, or both. This is also summarized in Table 1 below. A component is assigned its final RISC category upon approval by the IDP.

Table 1

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	Drives Associated Functions	IDP Change HSS to LSS
Risk (PRA Modeled)	Internal Events Base Case - Section 5.1	Component	Yes	Not Allowed
	Fire, Seismic, and Other External Events Base Case		No	Allowable
	PRA Sensitivity Studies		No	Allowable
	Integral PRA Assessment - Section 5.6		Yes	Not Allowed
Risk (Non-Modeled)	Fire, Seismic, and Other External Hazards	Component	No	Not Allowed
	Shutdown - Section 5.5	Function/Component	No	Not Allowed
Defense-in-Depth	Core Damage - Section 6.1	Function/Component	Yes	Not Allowed
	Core Damage - Section 6.2	Component	Yes	Not Allowed
Qualitative Criteria	Considerations - Section 9.2	Function	N/A	Allowable[1]
Passive	Passive - Section 4	Segment/Component	No	Not Allowed

Notes:

[1] The assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2. In some cases, a 50.69 categorization team may provide preliminary assessments of the seven considerations for the IDP's consideration; however, the final assessments of the seven considerations are the direct responsibility of the IDP.

The seven considerations are addressed preliminary by the 50.69 categorization team for at least the system functions that are not found to be HSS due to any other categorization step. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that consideration. If the 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS. Conversely, if all the seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 50.69 team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.

The mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components. Specifically, functions with mapped components that are determined to be HSS by the PRA-based assessment (i.e., Internal events PRA or Integrated PRA assessment) or defense-in-depth evaluation will be initially treated as HSS. However, NEI 00-04, Section 10.2 allows detailed categorization which can result in some components mapped to HSS functions being treated as LSS; and Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with a HSS function but which do not support the critical attributes of that HSS function. Note that certain steps of the categorization process are performed at a component level (e.g. Passive, Non-PRA-modeled hazards – see Table 1). These components from the component level assessments will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped. Therefore, if a HSS component is mapped to a LSS function, that component will remain HSS. If an LSS component is mapped to an HSS function, that component may be driven HSS based on Table 1 above, or may remain LSS.

d. Section 5 defines categorization process considerations for both PRA-based and non-PRA-based (i.e., deterministic) assessment methods. Section 5.3, for example, describes the process for categorization from seismic risk considerations using either a seismic PRA (i.e., PRA-based) or using a seismic margin assessment (SMA, i.e., deterministic and not PRA based). Section 7 requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5, but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. The interpretation of this requirement is further clarified in the Vogtle SER (ML14237A034), which states "…if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS."

   The reason for this is that the application of non-PRA-based assessments results in the default safety significance categorization of any SSCs associated with the safe shutdown success paths defined in those deterministic assessments to be HSS regardless of its risk significance. Therefore, there is no risk basis for assigning the SSC-associated functions to be HSS, since the deterministic analyses from which the associated safe shutdown equipment lists are derived do not define functions equivalent to those used in the categorization process. This is the reason that the guidance in Section 7 of NEI 00-04 clearly notes "PRA-based" in reference to Section 5 of NEI 00-04. The categorization process is consistent with the guidance in NEI 00-04 as endorsed by RG 1.201.

e. Consistent with the responses to RAI 5a, b, and c, if the results of the passive categorization are HSS, then the SSC is categorized as preliminary HSS regardless of the other categorization elements. A HSS determination by the passive categorization process cannot be changed by the IDP.

f. The seven considerations are addressed preliminary by the 50.69 categorization team for at least the system functions that are not found to be HSS due to any other categorization step. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that consideration. If the 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS. Conversely, if all the seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 50.69 team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.

g. NEI 00-04, Section 7.1 and Section 4 will be followed and SSC(s) that supports an interfacing system will remain uncategorized until all interfacing systems are categorized as captured in industry 50.69 categorization procedures.

RAI 06 – SSCs Categorization based on Other External Hazards

NEI 00-04 provides guidance on including external events in the categorization of each SSC to be categorized. The process begins with the SSC selected for categorization, as illustrated in NEI 00-04, Section 5.4, Figure 5-6 and proceeds through the flow logic for each external hazard. According to Figure 5-6, if a component participates in a screened scenario, then in order for that component to be considered candidate LSS, it has to be further shown that if the component was removed, the screened scenario would not become unscreened. The LAR does not address this aspect of the guidance, but appears to indicate "other external hazards" (i.e., besides seismic events) are screened from consideration in the categorization process.

a. Identify the external hazards that will be evaluated according to the flow chart in NEI 00-04, Section 5.4, Figure 5-6.

b. Identify the external hazards for which all credited SSCs will be considered HSS.

c. Describe and justify any additional method(s) different from (a) or (b) above that will be used to evaluate individual SSCs against external hazards and identify the hazards that will be evaluated with these methods.

d. Confirm that all external hazards not included in the categorization process (a), (b), or (c) above, will be considered insignificant for every SSC and, therefore, will not be considered during the categorization process.

e. <u>Extreme winds and tornado hazards</u>

   Attachment 4 of the LAR, as supplemented, indicates that the extreme wind or tornado hazard is screened on the basis that the high winds CDF is estimated to be less than 1E-6/year. This implies that there are certain mitigating SSCs that, if removed, could increase the CDF above 1E-6/year, and so, these SSCs would become HSS, per the guidance in Figure 5-6 of NEI 00-

04. Explain and justify how the guidance in Figure 5-6 of NEI 00-04 will apply to the high winds hazard and whether this hazard will or will not be considered during the categorization process.

f. External Flooding Hazard

Attachment 4 of the LAR indicates that external flooding hazards are screened from consideration in the 10 CFR 50.69 process. The LAR states that the external flooding hazard was screened because events associated with this hazard are bounded by the current licensing basis or in the case of a local intense precipitation (LIP) event there is "an acceptable method of assuring safe shutdown." The LAR states that for LIP "implementing the FLEX strategy provides an acceptable method of assuring safe shutdown." Section 5.4 of NEI 00-04 states that "after identifying the design basis and severe accident functions of the component, the external hazard analysis is reviewed to determine if the component is credited as part of the safe shutdown paths evaluated. If the component is credited, it is considered safety-significant." Further, Figure 5-6 of NEI 00-04 shows that if a component participates in a screened scenario, then in order for that component to be considered candidate LSS, the licensee has to show that if the component was removed, the screened scenario would not become unscreened.

i. Identify all SSCs that are credited in the screening of the LIP that should be designated safety significant per the guidance in Section 5.4 of NEI 00-04, including passive and/or active components. It should be noted, according to the LAR, it appears that FLEX strategy is relied upon for LIP mitigation. Additionally NRC notes that in a letter from NextEra to the NRC dated June 22, 2017 regarding a focused evaluation for LIP events (ADAMS Accession Number ML17173A082) that PBNP has committed to providing flood protection for the "B" train emergency diesel generator exhaust stacks.

ii. Identify any SSCs that are credited in the screening of all other external flooding mechanisms (other than LIP), including passive and/or active components.

iii. Explain and justify how the guidance in Figure 5-6 of NEI 00-04 will be applied to external flooding. Specifically, Figure 5-6 shows that if a component participates in a screened scenario, then in order for that component to be considered candidate LSS, the licensee has to show that if the component was removed, the screened scenario would not become unscreened.

Response

a. The external hazards that will be evaluated according to the flow chart in NEI 00-04, Section 5.4, Figure 5-6 are any hazards listed in Attachment 4 of the LAR, "External Hazards Screening," that have not been screened in accordance with ASME/ANS PRA Standard RA-Sa-2009.

For Point Beach, all "other" external hazards (i.e., other than internal events, internal flood, internal fire, and seismic) have been screened as noted in the LAR. As part of the external hazard screening, an evaluation was performed to determine if there are components that participate in screened scenarios and whose failure would result in an unscreened scenario. Consistent with the flow chart in Figure 5-6 of NEI 00-04, these components would be considered HSS.

b. The statement "All SSCs credited in other IPEEE external hazards are considered HSS" was intended to be consistent with the flow chart in Figure 5-6 of NEI 00-04. There are no Other External Hazards that will be evaluated using a method other than depicted in the flow chart.

c. There are no additional methods different from items a or b that will be used to evaluate individual SSCs against external hazards.

d. All external hazards not included in the categorization process items a, b, or c are considered insignificant for every SSC and therefore will not be considered during the categorization process.

e. The basis for screening extreme wind or tornado hazards in Attachment 4 of the LAR (Reference 1) is discussed below. The screening process followed the guidance in Figure 5-6 of NEI 00-04. The screening process includes an evaluation of whether SSCs participate in screened scenarios; and also considers whether, if credit for SSCs were removed relative to the hazard being evaluated, the hazard would then become unscreened. More specifically, for each external hazard in Attachment 4 of the LAR, an assessment was performed to determine if equipment (i.e., SSCs) is relied upon to mitigate a hazard based on the design basis and severe accident functions of the component. Such SSCs would be considered HSS.

   Extreme wind and tornado hazards are screened based on the results documented in PBNP IPEEE section 5.1 table 5.1.7-2 on the following page. The "Prior to Fix" column is the core melt probability contribution with recovery incorporated due to both straight winds and tornado winds and missiles. The "After Fix" column is the contribution to core melt probability after the G01 and G02 diesel generator exhaust stacks were fixed (subsequent to TAP A-45, the G01 and G02 diesel generator stacks were modified to accommodate higher winds).

   Two of the six SSCs vulnerable to high winds and tornados have a CDF contribution greater than 1E-08, Diesel Generator Exhaust Stacks (G01 and G02), and Openings in Control Room Walls. Other vulnerable SSCs have a CDF contribution less than 1E-08. These results do not include credit for the G03 and G04 diesel generators.

TABLE 5.1.7-2

CORE MELT PROBABILITY CONTRIBUTION OF THE VULNERABLE EQUIPMENT/STRUCTURE FAILURES BEFORE AND AFTER PROTECTING DIESEL GENERATOR EXHAUST STACKS FROM FAILING DUE TO WINDS

Equipment/Structure	Prior to Fix	After Fix
1. Service Water Pumphouse Roof	∈*	∈
2. Two Doors in Side of Service Water Pumphouse (both doors)	∈	∈
3. Opening in Control Room Walls	8.45E-8	8.45E-8
4. Diesel Generator Exhaust Stacks	6.3-5	2.6E-7
5. Condensate Storage Tanks	∈	∈
6. Refueling Water Storage Tanks	∈	∈

* ∈ = epsilon $< 10^{-8}$

After the IPEEE was submitted, modification MR 91-116 added two new B-train emergency diesel generators (G03 and G04), their necessary support systems, and a new building to house these systems. G03 and G04 were not included in the IPEEE high winds evaluation. This modification also added new 4160 VAC B-train buses to distribute power from the new emergency diesel generators and re-assigned the original B-train emergency diesel generator (G02) and its 4160 VAC buses to A-train. This modification also improved the design and operation of the emergency diesel generator fuel oil system.

Either G03 or G04 can provide the power required to address design basis accident mitigation. However, unlike G01 and G02, G03 and G04 are not vulnerable to high wind and tornado hazards. Given the added redundancy provided by G03 and G04, removal of one of the four emergency diesel generators will not change the original IPEEE conclusions; i.e. total CDF from high winds will remain well below 1E-06. In summary there are no SSCs credited for screening of high winds, per the guidance in Figure 5-6 of NEI 00-04.

f. i. There are no SSCs that are credited in the screening of the LIP. Although the LAR states that FLEX strategies are relied upon for LIP mitigation, per Evaluation 2015-0016, Revision 3 "LIP Flooding Coping Strategies (Flood Levels)" all required key safety functions (KSFs) will remain available during a LIP event after the EDG vent stack issue was corrected. FLEX strategies can be used as a defense in depth method to the plant features that are credited for the CLB floods and are sufficient to assure all KSFs remain available in the LIP flooding event. [Reference ML17173A082]

ii. As noted in NextEra Energy Point Beach, LLC, Response to NRC 10 CFR 50.54(f) Request for Information Regarding Near-Term Task Force Recommendation 2.1, Flooding-Submittal of Flooding Hazards Revaluation Report [ML15071A413] there are no SSCs that are credited in the screening of all other external flooding mechanisms.

iii If a SSC participates in a screened scenario, failure to credit the SSC could result in an unscreened external flooding scenario and, as such, the SSC would be HSS if the system were categorized. As noted in 6.f.i and 6.f.ii, there are no SSCs that fall in this category for external flooding.

RAI 07 – Shutdown Risk

LAR Section 3.2.5, "Low Power & Shutdown," states the categorization process will use the shutdown safety management plan described in NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991, (ADAMS Accession Number ML14365A203) for categorization of safety significance related to low power and shutdown conditions. However, the LAR does not cite the other criteria specified in NEI 00-04, Section 5.5, "Shutdown Safety Assessment," pertaining to low power shutdown events (i.e., DID attributes and failures that would initiate a shutdown event). Clarify and provide the basis for how the categorization of SSCs will be performed for low power and shutdown events, and how it is consistent with the guidance in NEI 00-04 as endorsed by RG 1.201, Revision 1.

Response

For plants without a shutdown PRA, such as Point Beach, NEI 00-04, as endorsed by RG 1.201, allows the use of a process based on the NUMARC 91-06 program. Point Beach's categorization process will follow the guidance and criteria in Section 5.5 in NEI 00-04 to address shutdown risk. Below is a summary of the NEI 00-04 process and requirements.

The overall process for addressing shutdown risk is illustrated in Figure 5-7 of NEI 00-04. NUMARC 91-06 specifies that a defense-in-depth approach should be used with respect to each defined shutdown key safety function. The shutdown key safety functions defined in NUMARC 91-06 are evaluated for categorization of SSCs. NEI 00-04 provides two criteria for SSCs to be considered preliminary HSS.

1. If a system/train being categorized supports a shutdown key safety function as the primary or first alternate means, then it is considered to be a "primary shutdown safety system" and is categorized as preliminary HSS. The station's Shutdown Safety Review and Safety Assessment, which is consistent with NUMARC 91-06, is used as a guide to identify primary and first alternative means. NEI 00-04 defines a "primary shutdown safety system" as also having the following attributes:

   - It has a technical bases for its ability to perform the function.
   - It has margin to fulfill the safety function.
   - It does not require extensive manual manipulation to fulfill its safety function.

2. If a failure of the SSC being categorized would initiate an event during shutdown plant conditions (e.g., loss of shutdown cooling, drain down), then that SSC is categorized as preliminary HSS.

As stated in NEI 00-04, "If the component does not participate in either of these manners, then it is considered a candidate as low safety significance with respect to shutdown safety."

RAI 08 – Key Assumptions and Uncertainties that could Impact the Application

LAR Section 3.2.7, "PRA Uncertainty Evaluations," explains that PRA model assumptions and sources of uncertainty have been identified for this application using guidance from NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," (ADAMS Accession Number ML090970525). LAR Section 3.2.7 indicates that no additional sensitivity analyses are required to address PBNP PRA model specific assumptions or sources of uncertainty beyond what is already required by Sections 5 and 8 of NEI 00-04.

The dispositions presented in Attachment 6 of the LAR for key assumptions and modeling uncertainties appear to fall into one of three categories: (1) the assumption is conservative, (2) the impact is small (negligible), or (3) the assumption realistically models the plant design. NRC staff notes that modelling issues that represent small impacts to CDF and LERF (both increases and decreases) could potentially increase the risk importance values for certain system components above the threshold criteria for determining safety significance specified in NEI 00-04, Section 5. In light of these observations, address the following:

a. Operator action to control Auxiliary Feedwater (AFW) flow late in the accident sequence

Attachment 6 (page 51) of the LAR explains that a sensitivity analysis performed evaluating the impact of not controlling AFW flow for the full PRA mission time shows that exclusion of operator action to control AFW flow late in the accident sequence has a "small" impact. It is not clear to the NRC staff how the sensitivity analysis demonstrates that the exclusion of this operator action has no impact on the categorization of SSCs under 10 CFR 50.69; therefore:

i. Justify that the exclusion of this operator failure to control the AFW flow late in the accident sequence has no impact on the 10 CFR 50.69 categorization results, or

ii. Propose a mechanism that ensures that the operator failure to control the AFW flow late in the accident sequence will be incorporated into the PRA prior to implementation of the 10 CFR 50.69 categorization process.

b. Expansion joint failures not in the PRA model

Attachment 6 (page 51) of the LAR explains that failures of expansion joints are not modelled in the fire protection system for the fire PRA. The LAR explains that the results of a sensitivity analysis on this failure mode demonstrates that the impact of crediting this action has a "negligible impact on the results." It is not clear to NRC staff what "negligible impact on the results" means quantitatively and notes that just small increases to CDF/LERF can impact the categorization of specific SSCs. Therefore:

i. Justify that the exclusion of the cited failure mode has no impact on the 10 CFR 50.69 categorization results, or

ii. Propose a mechanism that ensures that the expansion joint failures will be incorporated into the fire PRA prior to implementation of the 10 CFR 50.69 categorization process.

Response

a. The Internal Events Quantification Notebook discusses the assumption/uncertainty. The conservative results obtained by increasing an existing AFW HEP value in the model by 1E-4 to account for this issue show ~2E-7 increase in CDF. The documentation notes that the impact would be reduced if the HRA dependency analysis were to be redone. Since the exclusion of this operator failure results in such a small change, it is judged to not impact the 10 CFR 50.69 categorization results.

b. Note that the assumption/uncertainty regarding not modeling failures of the expansion joints in the fire protection system is for the internal events PRA, not the fire PRA. The internal events PRA model is the basis of the fire PRA model. The Internal Events Quantification Notebook discusses the assumption/uncertainty. An estimated failure rate for fire protection expansion joints of 1E-7 /hr (or 2.4E-6 /day) is negligible compared to other failure modes associated with the system. For example, the diesel fire pump fail to run probability is ~5E-2 and the motor-driven fire pump fail to run probability is ~9E-3. Since the expansion joint failure mode probability is overshadowed by other failure modes in the system, its exclusion is judged to not impact the 10 CFR 50.69 categorization results.

RAI 09 – 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, dated April 22, 2009 (ADAMS Accession Number 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: 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; 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; justification for crediting operator actions for success and failure of pressure boundary; guidelines and justification for selecting the appropriate break size (e.g. double ended guillotine break or smaller break); and include supporting examples of types of Class 1 SSCs that would be assigned low safety significance.

As mentioned in the meeting summary from the February 20, 2018 Risk-Informed Steering Committee (RISC) meeting (ADAMS Accession Number ML18072A301), 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 license amendment (ADAMS Accession Number 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 SSCs.

Response

The passive categorization process is intended to apply the same risk-informed process accepted by the NRC in the ANO2-R&R-004 for the passive categorization of Class 2, 3, and non-Code class components. This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662 as published in Regulatory Guide 1.147, Revision 15. Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be designated as high-safety-significant, HSS, for passive categorization which will result in HSS for its risk-informed safety classification, and cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at Point Beach for 10 CFR 50.69 SSC categorization.

RAI 10 – Modeling of the Reactor Coolant Pump (RCP) Shutdown Seals

In letter dated December 16, 2015, "NextEra Energy Point Beach, LLC's Notification of Full Compliance with Order EA-12-049 Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design Basis External Events and Submittal of Final Integrated Plan" (ADAMS Accession No. ML15350A085), it is stated that Point Beach Units 1 and 2 have installed the Westinghouse SHIELD Generation III low leakage/shutdown RCP seals.

The PRA model for the Generation III seals was approved by the NRC in the August 23, 2017, Topical Report [TR] PWROG-14001-P, Revision 1, 'PRA Model for the Generation Ill Westinghouse Shutdown Seal'" and the associated NRC Safety Evaluation (ADAMS Package Accession No. ML17200A116).

Consistent with the RG 1.174 guidance that the PRA scope, level of detail and technical acceptability be based on the as-built and as-operated and maintained plant, and reflect operating experience at the plant, please address the following:

a) Clarify whether the current internal events and fire PRA models include credit for the Westinghouse Generation Ill RCP seals.

b) If the PRA models include credit for the Westinghouse Generation Ill RCP seals, address the following:

i. Confirm that the TR PWROG-14001-P was followed and that the limitations and conditions in the NRC safety evaluation for PWROG-14001-P, Revision 1, are met.

ii. If exceptions to the limitations and conditions exist, identify all the exceptions and justify impact on the application.

iii. Describe how the Generation Ill Westinghouse RCP seal model has been peer-reviewed as part of the internal events PRA and fire PRA peer-reviews and whether any changes were required after the peer review.

iv. Justify why the addition of this model to the internal events and fire PRAs is not considered a PRA upgrade requiring a focused-scope peer review. For example, if asymmetric cooling was not included in the peer reviewed PRA, explain how including asymmetric cooling is not an upgrade.

v. If the addition of RCP seal model qualifies as a PRA upgrade, provide the results from the focused-scope peer review including the associated F&Os and their resolutions.

vi. If the PWROG-14001-P was not followed, describe and justify the approach used.

c) Alternatively to item b) above or if the PRA models do not include credit for the Westinghouse Generation III RCP seals, propose a mechanism to implement the RCP seals model in the PRA models and ensure adherence to the TR PWROG-14001-P, Revision 1, the associated NRC safety evaluation, and associated limitations and conditions, prior to implementation of the 10 CFR 50.69 categorization process.

Response

a) Although the current internal events PRA model does not include credit for the Westinghouse Generation III RCP seals, the internal events PRA model that will be used for categorization does include the credit. The current fire PRA model does include credit for the Westinghouse Generation III RCP seals.

b) The model outlined in PWROG-14001-P, Revision 1 as well as the limitations and conditions addressed in the Safety Evaluation will be incorporated into the Seal LOCA model used in the Point Beach PRA models.

i. Limitations and conditions from the SER that may impact the current PRA model are items 2, 4, and 5.

ii. For item 2, where identified conditions might occur, the current PRA model accounts for it by treating such condition as a failure of the shutdown seals. For item 4, the additional failure contribution of the SDS Bypass failure mode will be added to the model that will be used for the 10 CFR 50.69 risk categorization process. For item 5, plant-specific human error probabilities for both of those requirements exist in the current model.

iii. There will be no exceptions to the limitations and conditions outlined in the Safety Evaluation for PWROG-14001-P

iv. The Generation III Westinghouse RCP seal model was peer reviewed as part of the Fire PRA Peer Review as well as the subsequent focused scope peer reviews.

v. The peer-reviewed model did include an RCP seal leakage model (WOG 2000) to assess the plant response to events that result from a total loss of cooling to the RCP seals. Implementation of the Generation III Westinghouse RCP seal model was performed consistent with the existing peer-reviewed PRA methods already in the model for the WOG 2000 model.

The change in the seal leakage model is not a new methodology because the new seal leakage model is simply an expansion of the current peer-reviewed model with additional failure probabilities and associated human actions. There is no change in the model scope because the equipment, dependencies, and types of accident sequences remain the same. Finally, there is no change in PRA modeling capability. The peer reviewed PRA model can still evaluate the risk associated with station blackout and total loss of cooling events related to RCP seal failures. Therefore, implementation of the new seal leakage model is a change implemented within the framework of the existing peer-reviewed PRA model structure.

The addition of the Generation III Westinghouse RCP seal model is only a change in the expected seal leakages associated with the new seals. The framework of the model remains essentially the same, and the High Level and Supporting Requirements (HLRs) in the PRA Standard for the Technical Elements associated with RCP seal models (e.g., those within the Accident Sequence Analysis, Data Analysis, Human Reliability Analysis, and Quantification technical elements) will continue to be Met or Not Met regardless of addition of the shutdown seal model. Although the lower seal failure rates affect the ordering of the associated accident sequences and reduce CDF and LERF overall, the associated sequences were not significantly changed and the new sequences that were not already modeled in the PRA and peer-reviewed were not generated.

v. Implementation of the new RCP seal model is not an upgrade, per the discussion in item iv above.

vi. Not applicable, the approach in PWROG-14001-P will be followed

Attachment 1

Table of 10 CFR 50.69 Implementation Items

The table below identifies the items that are required to be completed prior to implementation of the 10 CFR 50.69 categorization process at Point Beach Generating Station, Units 1 and 2. All issues identified below 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.

NextEra Energy Point Beach, LLC 50.69 PRA Implementation Item	
Description	**Resolution**
NextEra will perform an evaluation of the as-built, as-operated plant against the SMA SSEL.	The evaluation will compare the as-built, as-operated plant to the plant configuration originally assessed by the SMA. Differences will be reviewed to identify any potential impacts to the equipment credited on the SSEL.
The disposition to F&O IE-A1-01 presented in the TSTF-425 LAR indicates that a number of special initiators related to the 4160 volts alternating current (VAC) Vital Switchgear bus were not included in the internal events PRA (IEPRA) model because they were considered not significant, and estimated the CDF for sequences associated with these initiators as high as 1.9E-07/year. It is not clear to the NRC staff that excluding these sequences cannot increase the risk importance values for specific system components above the threshold criteria for determining safety significance as discussed in NEI 00-04, Section 5	The loss of a 4,160 VAC bus will be added to the PRA model as a special initiator to resolve this finding.
The dispositions to F&Os AS-B6-01 and SY-A21-01 presented in the TSTF-425 LAR stated that although Emergency Diesel Generator (EDG) load management is a potential failure mode for EDGs, it is not modeled in the PRA. The TSTF-425 LAR further provided an estimate of the low likelihood that the EDG load management would be needed. The estimated low likelihood is based on the assumption that the events used in the estimate are independent. The NRC staff notes that loss of coolant accident (LOCA) initiators can induce loss of offsite power (LOOP) events and LOOP initiators can induce LOCAs (i.e., the need for safety injection (SI)) making these events dependent. Additionally, NRC staff notes that modelling exclusions that cause just small increases to CDF/LERF can impact the categorization of	A new failure mode associated with EDG load management will be added to the PRA model to resolve this finding.

NextEra Energy Point Beach, LLC 50.69 PRA Implementation Item	
Description	**Resolution**
specific structures, systems, and components (SSCs).	
The disposition to F&O AS-B7-01 presented in the TSTF-425 LAR states that recovery of LOOP events is only credited for station blackout (SBO) scenarios and the direct current (DC) batteries are conservatively assumed to fail at time zero. 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.	The treatment of power recovery after LOOP events and battery modeling in the PRA model will be revised to be more realistic to resolve this finding.
The disposition to F&O HR-D1-01 in LAR Attachment 3 states that no further changes are required. Also, based on the disposition presented in the TSTF-425 LAR, the F&O appeared resolved. Yet, the LAR associated with the adoption of 10 CFR 50.69 states regarding the disposition of this F&O, "[p]rior to implementation, either this finding will be closed or a sensitivity study case will be performed to determine the impact on the CDF and LERF results for those categorizations that could be adversely affected by this finding."	Since appropriate documentation was not provided to the closure review team in a timely fashion, this finding was not closed. Independent agreement that this finding is resolved in the PRA model will be obtained.
The description for F&O IFQU-A6-01 states, "HFEs from internal events are 'adjusted' with inadequate basis for those adjustments." The disposition for this F&O presented in the TSTF-425 LAR states that "stress multipliers" from Table 20-16 of NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, Final Report," 1983 (ADAMS Accession Number ML071210299), which are referred to as "stress modifiers" in NUREG/CR-1278, were used to account for the stress associated with an internal flooding event. However, Table 20-16 of the cited NUREG lists modifiers to be applied to human error probabilities (HEPs) for different stress levels and they are not specific to internal flooding.	No stress modifiers were applied to the HEP values used in the internal flood PRA model. The internal flood documentation will be updated to justify the HEP values actually used in the model to resolve this finding.
The description for F&O PRM-B2-01 states that resolution of internal events findings could impact fire PRA evaluations. The disposition to this F&O states, "[a]s of the time of this submittal, the only remaining open internal events peer review findings	The identified internal events findings will be resolved in the manner described in the response to RAI 02.b above. Specifically, a new failure mode associated with EDG load management will be added to the PRA model to resolve this finding.

NextEra Energy Point Beach, LLC 50.69 PRA Implementation Item	
Description	**Resolution**
identified…are AS-B6-01 and SY-A21-01." NRC staff notes that AS-B6-01 and SY-A21-01 are the subject of RAI 02.b above.	
The description for F&O HRA-B2-01 found that credit for graphically distinct factors is taken for all HRA events, as opposed to taking credit for graphically distinct procedural steps that stand out from the other steps. The disposition for this F&O states, "[o]nly about 10% of the HEPs that credited graphically distinct procedure steps would be increased by more than a factor of 2." The disposition concludes "[b]ased on this review, the impact on the model from this finding is judged minimal." NRC staff notes that modelling issues that can cause even small increases to CDF and LERF can potentially increase the risk importance values for specific system components above the threshold criteria for determining safety significance specified in NEI 00-04, Section 5	The HEPs developed for the Fire PRA model will be updated to remove the graphically distinct credit in the cognitive portion of the HEP. The dependency analysis will be updated and the Fire PRA quantified using these updated Human Error Probabilities.
The description for F&O FQ-A1-01 states, "some basic events that have been mapped to scenarios, components, or cables are not found in the CAFTA model." The associated disposition states that "information in the mapping table should be reviewed to eliminate the extraneous information and eliminate the discrepancies." The disposition for this F&O presented in the PBNP LAR for transition to the National Fire Protection Association (NFPA) Standard 805 (ADAMS Accession Number ML13182A353) indicates that this review has been performed which identified six failure events that were excluded from the PRA associated with the MCR. The PBNP NFPA 805 LAR for this disposition also states that a sensitivity study was conducted which determined that the exclusion of these basic events have a negligible impact on fire risk.	The basic event mapping tables will be reviewed and compared to the present basic event mapping associated with each equipment or cable. Those items that are no longer needed will be removed and any incorrect mapping will be updated. The Fire PRA model will be quantified using this updated mapping table.

Attachment 2

Markup of the Point Beach Units 1 and 2 Operating Licenses

INSERT CONDITION E

E. Adoption of 10 CFR 50.69, "Risk-Informed categorization and treatment of structures, systems, and components for nuclear power plants"

1. NextEra Energy Point Beach is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) specified in the license amendment dated [DATE].

2. Prior to implementation of the provisions of 10 CFR 50.69, NextEra Energy Point Beach shall complete the items below:

   a. Item A in Attachment 1, List of Categorization Prerequisites, to NextEra Energy Point Beach letter NRC 2017-0043, "License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components (SSCs) for Nuclear Power Plants'," and

   b. Attachment 1, Point Beach 10 CFR 50.69 PRA Implementation Items, in NextEra Energy Point Beach letter NRC-2018-0038, "Response to Request for Additional Information Regarding License Amendment Request 287, Application to Adopt 10 CFR 50.69, 'Risk informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants'," August 10, 2018

3. 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).

D. Physical Protection

NextEra Energy Point Beach shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Point Beach Nuclear Plant Physical Security Plan, (Revision 4)," submitted by letter dated May 10, 2006. NextEra Energy Point Beach, LLC shall fully implement and maintain in effect all provisions of the Commission-approved Point Beach Nuclear Plant Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NextEra Energy Point Beach CSP was approved by License Amendment No. 243 as supplemented by a change approved by License Amendment No. 247 and License Amendment No. 252.

E. ~~Deleted~~ INSERT CONDITION E

F. NextEra Energy Point Beach Unit 1 shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment request dated June 26, 2013, and supplements dated September 16, 2013, July 29, 2014, August 28, 2014, September 25, 2014, November 14, 2014, December 19, 2014, January 16, 2015, May 12, 2015, August 26, 2015, February 22, 2016, April 07, 2016, and May 3, 2016, and as approved in the safety evaluation report dated September 8, 2016. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or license condition, and the criteria listed below are satisfied.

1. Risk-Informed Changes that May Be Made Without Prior NRC Approval

   A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

D. Physical Protection

NextEra Energy Point Beach shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Point Beach Nuclear Plant Physical Security Plan, (Revision 4)," submitted by letter dated May 10, 2006. NextEra Energy Point Beach, LLC shall fully implement and maintain in effect all provisions of the Commission-approved Point Beach Nuclear Plant Cyber Security Plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The NextEra Energy Point Beach CSP was approved by License Amendment No. 247 as supplemented by a change approved by License Amendment No. 251 and License Amendment No. 256.

E. ~~Deleted~~ ← INSERT CONDITION E

F. NextEra Energy Point Beach Unit 2 shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment request dated June 26, 2013, and supplements dated September 16, 2013, July 29, 2014, August 28, 2014, September 25, 2014, November 14, 2014, December 19, 2014, January 16, 2015, May 12, 2015, August 26, 2015, February 22, 2016, April 07, 2016, and May 3, 2016 and as approved in the safety evaluation report dated September 8, 2016. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or license condition, and the criteria listed below are satisfied.

1. Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact