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PSEG Nuclear LLC

10 CFR 50.90

LR-N20-0043
LAR H19-07

June 25, 2020

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

Hope Creek Generating Station
Renewed Facility Operating License No. NPF-57
NRC Docket No. 50-354

Subject: Response to Request for Additional Information, Re: Adopt 10 CFR 50.69 LAR

References:
1. PSEG letter to NRC, "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors'" dated November 25, 2019 (ADAMS Accession No. ML19330C961)
2. NRC email to PSEG, "Hope Creek - Final RAI RE: Adopt 10 CFR 50.69 LAR (L-2019-LLA-0265)" dated June 15, 2020 (ADAMS Accession No. ML20167A330)

In the Reference 1 letter, PSEG Nuclear LLC (PSEG) submitted a license amendment request for Hope Creek Generating Station (Hope Creek). The proposed amendment would modify the Hope Creek 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), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." In Reference 2, the Nuclear Regulatory Commission (NRC) requested PSEG to provide additional information in order to evaluate the proposed License Amendment Request.

Attachment 1 to this letter provides a restatement of the RAI questions followed by our responses. PSEG has determined that the information provided in this submittal does not alter the conclusions reached in the 10 CFR 50.92 no significant hazards determination previously submitted. In addition, the information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no regulatory commitments contained in this letter.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), PSEG is providing a copy of this response, with attachments, to the designated State of New Jersey Official.

Should you have any questions regarding this submittal, please contact Mr. Lee Marabella at 856-339-1208.

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

Executed on June 25, 2020
(Date)

Sincerely,

Paul R Duke

Paul R. Duke, Jr.
Manager - Licensing

Attachment:
1. Response to Request for Additional Information - Application to Adopt Title 10 of the Code of Federal Regulations 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

cc: Administrator, Region I, NRC
Mr. J. Kim, Project Manager, NRC
NRC Senior Resident Inspector, Hope Creek
Mr. P. Mulligan, Chief, NJBNE
Site Compliance Commitment Tracking Coordinator
Corporate Commitment Tracking Coordinator

Attachment 1

Response to Request for Additional Information - Application to Adopt Title 10 of the Code of Federal Regulations 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

By letter dated November 25, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19330C961), PSEG Nuclear, LLC (PSEG, the licensee) submitted a license amendment request (LAR) for the Hope Creek Generating Station (HCGS). The proposed license amendment would modify the HCGS 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* (CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The U.S. Nuclear Regulatory Commission (NRC) staff from Probabilistic Risk Assessment Licensing Branch A and Branch C (APLA and APLC respectively) have reviewed the LAR and request additional information (RAI) in order to complete the review.

APLA RAI 01 – Appendix X, Close-out of Facts and Observations

Section 2 of 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), states for the applicable technical requirements, "the staff anticipates that current good practice, i.e., Capability Category II of the American Society of Mechanical Engineers (ASME)/ANS [American Society of Mechanical Engineers/American Nuclear Society] standard [i.e., ASME/ANS RA-Sa-2009], is the level of detail that is adequate for the majority of the applications," and that a peer review is needed to determine whether the intent of the requirements in the standard is met.

LAR Section 3.3 states that an independent assessment team conducted a facts and observations (F&Os) closure review on the internal events (including internal flooding) probabilistic risk assessment (PRA) model in August 2017. F&Os were reviewed and closed using the process documented in the Nuclear Energy Institute (NEI) letter to the NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-1[3], Close-Out of Facts and Observations (F&O)" (ADAMS Package Accession No. ML17086A431), as accepted by the NRC staff on May 3, 2017 (ADAMS Accession No. ML17079A427). Regarding PRA method, Section 1.1 of the August 2017 F&O closure review report states the following definition for a PRA upgrade: if the method has not been reviewed by the NRC staff, either explicitly or implicitly. However, Section 1.3 of the report states that an upgrade is the incorporation into a PRA model of a new methodology that impact the significant accident sequences. It is unclear to the NRC staff what definition of PRA upgrade was used in the August 2017 F&O closure review. In light of these observations:

a. Confirm that none of the resolutions used to close-out F&Os in August 2017 represented a new method to the HCGS internal events PRA model.

b. If any new methods were introduced into the HCGS internal events PRA model for closure, then describe approval of the new methods, perform a focused-scope peer review for the identified PRA upgrades, and provide the F&Os and associated dispositions of each for their impact on the categorization results.

RESPONSE TO APLA RAI 01:

None of the resolutions used to close-out F&Os in August 2017 represented a new method to the HCGS internal events PRA model.

An evaluation of resolutions to close-out F&Os in August 2017 was performed and documented in the associated F&O Closure Report. In section 1.3 of the F&O Closure Report, the definition of a PRA Upgrade is explicitly presented and is consistent with the ASME/ANS PRA Standard. Section 1.3 of the F&O Closure Report also states that the scope of the review of each resolution includes concurrence with the determination of PRA upgrade or maintenance. Table A-1 of the F&O Closure Report presents the closure review results and shows that there are no resolutions that were identified as an upgrade. Inclusion of the PRA upgrade check in the results table is consistent with Appendix X to NEI 05-04/07-12/12-1, Close-Out of Facts and Observations (F&O) documentation requirements. Therefore, the F&O Closure Report for the close-out performed in August 2017 shows that the definition of upgrade used in that close-out was consistent with the ASME/ANS PRA Standard and none of the resolutions used to close-out F&Os in August 2017 represented a new method to the HCGS internal events PRA model.

Sixteen Findings were assessed for closure as part of the August 2017 F&O Closure. Of those sixteen, eight were identified as “documentation only” and therefore clearly did not utilize a new method. Of the eight remaining Findings, none used a new method. A summary of the new method evaluation for these eight Findings is shown in the table below. All closed Findings also meet the definition described in section 1.1 as no methods were used that were not reviewed by the NRC staff, either explicitly or implicitly.

Therefore, the F&O Closure Report for the close-out performed in August 2017 shows that none of the resolutions used to close-out F&Os in August 2017 represented a new method to the HCGS internal events PRA model.

SR / Assessment	Finding	Description	Basis	Assessment of New Method
DA-D1 (SR MET (CC I))	DA-D1-01	Plant specific data was not collected for the most recent update reliability data. The only plant specific information used was for systems that are monitored by the MSPI program. MSPI systems include the diesel generators, HPCI, RCIC, RHR, SSWS and SACS. No other specific data was used for this update. Individual component random failure data is a vital input to the PSA. Therefore, special attention is paid to ensuring that the best available information is used as input to the PSA.	As outlined in the Component Data Notebook, "individual component random failure data is a vital input to the PSA. Therefore, special attention is paid to ensuring that the best available information is used as input to the PSA." Inadequate data collection and update could have an actual impact on the accuracy of the PRA.	The use of plant specific data and Bayesian update methods are common practice in PRA. These methods were already used in the HCGS PRA prior to the F&O Closure. Therefore, no new method was used to address this finding.
HR-C2 (SR MET (CC I))	HR-C2-01	Tables 4.3-4 and 4.3-5 of the HR Notebook (HC PSA-004, Rev. 0) present the defined restoration and miscalibration Type A HFEs. Additional information provided by PSEG regarding additional errors (e.g., restoration of power supply) referred only to the ACP events in Tables 4.3-4 and 4.3-5. However, these pertain only to bus voltage sensors for undervoltage transfers and restoration of the gas turbine.	This issue must be addressed to include all of the modes of unavailability specified for Category II in this SR.	The basic events identified in response to this finding were screened using SR HR-B1, therefore HR-C2 was not applicable. The use of screening and the criteria applied were already used in the HCGS PRA prior to the F&O Closure. Therefore, no new method was used to address this finding.

SR / Assessment	Finding	Description	Basis	Assessment of New Method
IF-D5 (SR MET)	IF-D5-01	The Service Water failure frequencies should match the EPRI failure guideline but they don't. The frequency used is less conservative than that provided in the EPRI guidance. Table G-1 needs to be updated to reflect the correct Service Water (river) rupture frequencies. Note an incorrect rupture frequency was used for the Service Water (river) calculations. This will require the calculations for those sections to be re-performed using the correct EPRI failure frequency.	This is designated a finding as the wrong frequencies were used for SW failures, which will require correction and update of the calculations.	This resolution of this finding consisted of applying a different source for the underlying failure data. The methods or calculations for pipe failure frequencies were already in the HCGS PRA and were not changed. Therefore, no new method was used to address this finding.
QU-D5a (SR MET (CC I))	QU-D5a-01	Section 6.0 of the HCGS Quantification notebook (HC PSA-014, Rev. 1) and Section 3 of the PRA Summary notebook (HC PSA-013, Rev. 0) present some of the significant contributors, including initiating events (Tables 6.2-4 and 3.2-4) and accident sequence subclass (Tables 6.2-5 and 3.2-5). Appendix F of the Quantification notebook also provides overall event importance measures. Although they are not categorized by initiating event, equipment failures, common cause failures or operator errors, they do appear to include all significant events. (Per the ASME standard, significant events are those that have a F-V importance greater than 0.005 or RAW importance greater than 2.) Similar information is provided for LERF. Section 4.2 of the Summary notebook provides risk rankings for system trains based on RAW, and Section 4.3 provides the risk important operator actions based on F-V . However, the identification of significant contributors does not include SSCs and operator actions that contribute to initiating event frequencies although those that contribute to event mitigation have been. Also ensure Summary notebook discussion matches results from QU notebook (e.g., risk important operator actions).	The information provided is incomplete such that the Cat II SR is not met.	This was determined to be a minor documentation issue. The F&O Closure team concluded that with minor documentation changes, the finding was closed. Therefore, no new method was used to address this finding.

SR / Assessment	Finding	Description	Basis	Assessment of New Method
QU-E4 (SR MET (CC II))	QU-E4-01	Section 3.4 and Appendix B and C of the PRA Summary notebook (HC PSA-013) provide an evaluation of the important model uncertainties and Section 4.5 and Appendix E provide a set of structured sensitivity evaluations based on these uncertainties. Sensitivity calculations were run, with seven cases being identified as important to model uncertainty. Table 4.5-1 of the PSA-013 contains a summary of sensitivity cases to identify risk metric changes associated with candidate modelling uncertainties. The uncertainties are identified based on generic sources of uncertainty provided in EPRI TR10009652. However, no additional plant-specific sources of uncertainty are addressed. Initial clarification on sources of uncertainty was provided in a July 27, 2007 NRC memorandum, which specified that at a minimum for a base PRA the analyst must "identify the assumptions related to PRA scope and level of detail, and characterize the sources of model uncertainty and related assumptions, i.e., identify what in the PRA model could be impacted and how". In addition, "While an evaluation of any source of model uncertainty or related assumption is not needed for the base PRA, the various sources of model uncertainty and related assumptions do need to be characterized so that they can be addressed in the context of an application. Therefore, the search for candidates needs to be fairly complete (regardless of capability category), because it is not known, a priori, which of the sources of model uncertainty or related assumptions could affect an application." So excluding plant-specific sources of uncertainty from characterization because they did not "rise to the level that they would be considered candidates for modelling uncertainty" is not appropriate.	The information provided is incomplete; the most recent industry guidance to address modelling uncertainty in order to meet Cat II for these SRs is not met.	The uncertainty evaluation was performed consistent with NUREG-1855 and the complementary EPRI guidance. The HCGS PRA described sources of model uncertainty for HCGS and, among other things, the impact on the model prior to the F&O Closure. Additional sources of uncertainty were identified, assessed, and included with the HCGS PRA documentation as part of resolution of this finding. Since sources of uncertainty and evaluation of those sources were already evaluated in the HCGS PRA prior to the F&O Closure, no new methods were used to address this finding.
QU-F3 (SR MET (CC I))	QU-F3-01	Section 6.0 of the HCGS Quantification notebook (HC PSA-014, Rev. 1) and Section 3 of the PRA Summary notebook (HC PSA-013, Rev. 0) present some of the significant contributors, including initiating events (Tables 6.2-4 and 3.2-4) and accident sequence subclass (Tables 6.2-5 and 3.2-5). Appendix F of the Quantification notebook also provides overall event importance measures, for what appears to include all significant events. Section 6.3 discusses the top 10 accident sequences (68% of the total CDF and at least 2.5% individually) Per the ASME standard, significant accident sequences are those that combine to represent 95% of the CDF or individually represent 1% of the overall CDF. However, there is not a detailed discussion of the significant accident sequences, and the summary table of Accident Classes does not provide a detailed description of significant functional failure groups and does not provide a full, clear picture of the combinations of system or functional failures to which the plant is vulnerable and why they are significant; which is required to distinguish CC II from CC I.	The information provided is incomplete such that the Cat II SR is not met.	The Closure Team agreed that this Finding was closed with minor documentation changes. Therefore, no new method was used to address this finding.

SR / Assessment	Finding	Description	Basis	Assessment of New Method
SC-A6 (SR MET)	SC-A6-01	The basis for the fire pump as a low pressure source of makeup to the vessel after depressurization is based on flow rate inputs, which lack rigor. The flow input to MAAP merely reduces the published pump curve by 20% to account for flow friction losses. The MAAP input also does not correct the pump curve for elevation difference between the fire pump water source and the injection point. Success of the fire pump as a low pressure source of makeup solely depends on its ability to provide adequate makeup. The fire pump flow rate should be based on a flow calculation that considers the piping and fire hose friction and elevation differences.	There is not sufficient rigor and information provided on use of the fire pump as a low pressure makeup source to determine if appropriate analyses were used to develop this success criteria.	This finding resolution corrected an error in the flow rate used in the thermal hydraulic calculations for the fire pumper truck as an external injection source. The resolution included updated MAAP thermal hydraulic calculations and slight modifications to success criteria and success timing regarding use of the fire pumper truck. The use of MAAP for PRA thermal hydraulic calculations was already included in the HCGS PRA prior to the F&O Closure, as was the fire pumper truck as an injection source. Therefore, no new method was used to address this finding.
SY-B14 (SR NOT MET)	SY-B14-01	The standard requires that failure of common piping be modelled if the failure affects more than one system. The common piping failure between HPCI/FW/CS and RCIC/FW have not been modelled.	The modelling provided is incomplete such that the SR is not met, and it cannot be demonstrated that components / failure modes which fail multiple systems have been included.	After review and discussion with the Closure Team, it was determined that common piping system failures were already appropriately modeled, and an extent of condition review showed no additional omissions in the PRA model. Therefore, no new method was used to address this finding.

APLA RAI 02 – Overlap of Functions and Components

Section 7.1 of NEI 00-04, Revision 0, “10 CFR 50.69 SSC Categorization Guideline” (ADAMS Accession No. ML052910035), states, "[d]ue to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC [structure, system, and component], 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 states that a candidate low-safety-significant structure, system, and component (SSC) that supports an interfacing system should remain uncategorized until all interfacing systems are categorized. The LAR does not discuss consideration or implementation of the guidance in Section 7.1 of NEI 00-04.

Explain how the categorization process will be implemented to ensure 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.

RESPONSE TO ALPA RAI 02

NEI 00-04 guidance involving interfacing systems allows for categorization of any functions/SSCs that serve as an interface between two or more systems as long as all impacts to the interfacing system(s) are considered. Functions/SSCs that serve as an interface between two or more systems do not necessarily have to remain uncategorized prior to completing the categorization of all the interfacing systems. This approach to categorization meets NEI 00-04 as well as 10 CFR 50.69(c)(1)(v) for the reasons discussed below.

NEI 00-04 Section 4 discusses system selection and boundary definition which “includes defining system boundaries where the system interfaces with other systems.” One of the initial steps in system categorization is defining the boundaries and developing a list of functions for the system being categorized. If the system being categorized includes interfacing functions/SSCs, then additional functions are created to 1) identify the interfacing system(s) and 2) to describe the support functions to the interfacing system(s).

Component and functional importance assessments would be performed and documented which includes the impact of the failure of interfacing SSCs using all process steps of the NEI 00-04 categorization process (e.g., active and passive categorization, defense-in-depth, etc.). The outcome of each of the individual process steps’ significance determination would be identified and the highest significance from any process step is assigned to the interfacing SSC.

If an SSC supports two functions, one being LSS and the other being HSS, the SSC will receive a categorization of HSS. If an SSC supports only an LSS function but is considered HSS from a component level evaluation, the SSC will be considered HSS. The only exception applies to those components that support an HSS function but do not have a credible means to fail the HSS function. These components may be considered LSS in accordance with the guidance in Section 10.2 of NEI 00-04.

If an interface component is found to be HSS for the system being categorized, then it will be categorized RISC-1 or RISC-2 (and will not receive alternative treatments) even if it interfaces with other systems.

Per NEI 00-04 Section 4, for interfacing systems, all impacts an SSC has on interfacing systems will be considered. In the event an SSC in a system being categorized supports multiple systems, its risk significance will be reflected in the System Categorization Document (SCD) for the system defined by its boundaries and its impact on all interfacing systems. Section 4 includes identification of all system functions (including support functions to interface system SSCs) and includes mapping of system components into the system functions they support.

Per Section 7.1 of NEI 00-04, an SSC being considered for categorization will be assigned the highest risk significance of any portion of the process, whether it be from a function or component level evaluation. Consistent with NEI 00-04 Section 12, periodic reviews of the SSC categorization results will be conducted no longer than every two refueling outages to ensure that the categorization process and results are maintained valid.

PSEG expects to encounter two different options when categorizing an SSC that interfaces with multiple systems:

Option 1: An interface SSC is categorized along with the system it is assigned to.

An SSC's impact on all systems it interfaces with will be considered in the SCD for the system the SSC belongs to. Consistent with NEI 00-04 Section 7.1, component and functional importance assessments would be performed and documented. This includes identification of all functions of the interface SSC in its assigned primary system as well as all functions of the interface SSC to all other systems outside this primary system.

The safety-significant SSCs from the component safety significance assessment (NEI 00-04 Section 5) are mapped to the appropriate function(s) for which they have safety significance. If any SSC is safety significant, from either the PRA-based component safety significance assessment (NEI 00-04 Section 5) or from the defense-in-depth assessment (NEI 00-04 Section 6), then the associated system function is preliminarily safety significant. All other functions/SSCs can be preliminarily assigned low safety significance. All preliminary categorizations assigned as candidate safety significant or low safety significant are then taken to the IDP for final review and approval. The overall process used in integrating the various categorization inputs is depicted in NEI 00-04 Figure 7-1.

Once a system function has been identified as safety-significant, then all components that support this system function are assigned a preliminary safety-significant categorization. All other components are assigned a preliminary LSS categorization. Due to the overlap of functions and components, a significant number of components may support multiple functions. In this case, the SSC is assigned the highest risk significance for any function that the SSC supports. Consistent with NEI 00-04 Section 7, for safety-significant functions/SSCs, the critical attributes that make the function/SSC safety-significant will be identified.

An example of this approach is described below for a portion of the core spray system.

The stop check valve F006A (and associated piping) belongs to the core spray (CS) system's injection flow path to the reactor vessel, but also provides an injection

pathway from the high-pressure coolant injection (HPCI) system. When categorizing CS in this first option, this valve cannot be categorized until the interfacing role with the HPCI system is considered. This would be accomplished by creating a HPCI support function for the valve F006A that fully describes its role, including all SSCs mapped to the HPCI support function and to the CS system function.

All component and functional importance assessments would be performed and documented which includes the impact of failure of the valve on both the CS and HPCI systems using all process steps of the NEI 00-04 categorization process (e.g., active and passive categorization, defense-in-depth, etc.).

The outcome of each of the individual process steps' significance determination would be identified and the highest significance from any process step is assigned to the valve. Critical attributes for any safety-significant function/SSC will be identified. The safety significance of all components interacting with the support function will have been assessed since all functional analysis steps will have been performed. Only the stop check valve F006A in the HPCI system will get assigned a RISC at this time because other HPCI components were not in the defined system boundary. All remaining SSCs and functions of the HPCI system could later be categorized when the HPCI system is categorized, and the shared interface components and piping categorized as part of the CS categorization would retain the same significance level for both systems. Periodic reviews of the SSC categorization results per NEI 00-04 will be conducted no longer than every two refueling outages to ensure that the categorization process and results of the categorization are maintained valid.

Option 2: An interface SSC is left uncategorized until after all interfacing or supported systems are categorized in their entirety.

This option highlights the "example" listed in NEI 00-04 Section 4 concerning cooling water system piping on a ventilation system cooler. Existing treatments will continue to be applied to SSCs that are uncategorized.

For Option 1, an SSC that interfaces with multiple systems will be categorized as HSS if it supports an HSS function. In Option 1, once HPCI is subsequently categorized, the process will confirm the same RISC result for the interface SSC as was achieved when the CS system was earlier categorized. As mentioned above, periodic reviews of the SSC categorization results for all systems categorized will be conducted no longer than every two refueling outages to ensure that the categorization process and results of categorized systems are maintained valid.

PSEG interprets this approach to be consistent with 10 CFR 50.69 Section (c)(1)(v) and the guidance provided in NEI 00-04. The choice of which option to apply is a business decision based on which systems the station believes would benefit from categorization. There may be situations where an interface system would not benefit from categorization and current treatments for the interface system would remain in place. There may be other situations where the interface system may benefit from categorization. The choice of which systems to categorize and when is up to each individual licensee. From Option 1, the initial categorization results of the interfacing SSC will not change during subsequent categorization of other interface systems because all steps of NEI 00-04 will have been performed.

APLA RAI 03 – Masking of Risk Insights due to Conservative Modelling Choices

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

Section 3.2.7 of the LAR states, "[i]f the HCGS PRA model used a non-conservative treatment, or methods that are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine its impact on this application." The NRC staff notes that conservative modelling choices can potentially mask the importance of other SSCs or, in other words, artificially lower the risk importance values of other SSCs below the safety significance threshold criteria. In light of these observations:

a. Discuss how the potential for masking due to conservative modelling choices will be addressed in the categorization process.

b. LAR Attachment 6 identifies a modelling conservatism where SSCs for which cable routing is unknown are assumed to fail in the fire PRA. The LAR cites the results of a sensitivity study that assumes none of these SSCs fail due to a fire and yields a reduction in core damage frequency (CDF) and large early release frequency (LERF) of less than 18 percent. However, this result does not address whether categorization of SSCs is impacted by this modelling assumption.

    i. Justify that the conservatism associated with not modelling SSCs for which cable routing is unknown has no impact on 10 CFR 50.69 categorization results.

    ii. If the modelling conservatism addressed in part (i) above cannot be justified to have minimal impact on the 10 CFR 50.69 categorization results, then propose a mechanism that ensures a sensitivity study is performed during 10 CFR 50.69 categorization that specifically addresses the uncertainty associated with SSCs for which cable routing is unknown.

c. Identify any other major conservatisms in the PRAs and justify they cannot mask or skew the importance of certain SSCs. If a modelling conservatism cannot be justified to have minimal impact on the 10 CFR 50.69 categorization results, then propose a mechanism that ensures a sensitivity study is performed during 10 CFR 50.69 categorization that specifically addresses the uncertainty.

RESPONSE TO APLA RAI 03:

Part a: All sources of PRA modelling uncertainty, including conservatisms, were reviewed in Attachment 6. Individual treatment of these items is discussed further in part c of this response. In addition, NEI 00-04 [1] requires that several sensitivity studies be performed to ensure that assumptions or uncertainties in the probabilistic risk assessment are not masking the importance of an SSC. Sensitivity studies required to be performed include increasing and decreasing human error rates, increasing and decreasing common cause failure rates, increasing and decreasing maintenance unavailability, and increasing the failure rate of low safety significant components. In addition, each risk contributor is initially evaluated separately during the categorization process in order to avoid reliance on a combined result that may mask

the results of individual risk contributors. Sensitivity studies may also be used to address issues raised during the IDP process and may include other quantitative assessments designed to demonstrate that an SSC is not safety significant.

Thus, Attachment 6 addresses the potential for masking due to conservative modelling choices on a case-by-case basis, and additional sensitivity studies will be performed to provide additional assurance that conservative modelling will not mask necessary risk insights.

Part b: The importance lists from the base model were compared with the importance lists that resulted when UNLs were completely removed from the modelling. The comparison of FV importances is analyzed and dispositioned in Table B-1 and the comparison of RAW importances is documented in Table B-2. Events with a FV >= 0.005 or RAW >= 2 were considered important and common cause events with a FV >= 0.005 or RAW >= 20 were considered to be important. Credit was taken for the inclusion of events in the base risk-ranking calculations with importance metrics within 10% of these values. These metrics are consistent with the guidance provided in NEI 00-04 [1] and NEI 16-09 [2], and will be used for the Hope Creek 10 CFR 50.69 application submittal.

No component was found to have its risk-ranking affected by this removal of UNLs. Most of the entries in Tables B-1 and B-2 represent relatively minor adjustments in risk values that were near the defined thresholds and would be captured by the 10% adjustment to risk metrics described above. The drywell shell rupture event, which is a structure rather than a component, became newly risk-significant for this sensitivity. However, this is a seismic Class I structure, and thus will be assumed to have high safety significance without further analysis, so the sensitivity of its Fussell-Vesely value to the UNL assumption does not impact its treatment for the 50.69 application. Failure to perform a turbine trip also became important, but this represents a high safety significance system and human action, not a component, and thus will not affect risk ranking. Note that if the cables were fully routed there would still be cable failures in some fire scenarios and thus the impact on importance results is overstated by the bounding case.

Thus, the conservatism associated with not modelling SSCs for which cable routing is unknown has negligible impact on 10 CFR 50.69 categorization results.

Table B-1: FV Discrepancies Between the Base Importances and the No UNL Importances

BE	Description	Base CDF FV	Base LERF FV	No UNL CDF FV	No UNL LERF FV	Disposition
1--SEQ-LOP-069	SEQUENCE TAG LOP-069	0.00474	0.0251	0.00541	0.02812	The event was previously captured within the FV >= 0.0045 threshold. Therefore, while a minor change to its Fussell-Vesely value raised it above the FV >= 0.005 threshold, it does not represent a new risk-significant component. This is also not a component and thus does not affect the risk ranking of components.

Table B-1: FV Discrepancies Between the Base Importances and the No UNL Importances

BE	Description	Base CDF FV	Base LERF FV	No UNL CDF FV	No UNL LERF FV	Disposition
CIS-MOV-CO-F001-SP-2	MOV F001 SPURIOUS OPENING PROBABILITY DEPENDENCY SET 2	0.00473	0.01401	0.00533	0.01454	The event was previously captured within the FV >= 0.0045 threshold. Therefore, while a minor change to its Fussell-Vesely value raised it above the FV >= 0.005 threshold, it does not represent a new risk-significant component.
DW-SHELL-RUPT	DRYWELL SHELL RUPTURE DISRUPTS INJECTION LINES AND FAILS RB SYS	0.00017	0	0.00702	0	This event does not represent a component, and therefore will not skew the risk ranking of components.
MCA_RB2_AB2	MCA - RB2 to AB2	0.00456	0.02341	0.00503	0.02526	The event was previously captured within the FV >= 0.0045 threshold. Therefore, while a minor change to its Fussell-Vesely value raised it above the FV >= 0.005 threshold, it does not represent a new risk-significant component. This is also not a component and thus does not affect the risk ranking of components.
2--SEQ-3C-39	L2 SEQUENCE TAG 3C-39	0	0.00463	0	0.00509	The event was previously captured within the FV >= 0.0045 threshold. Therefore, while a minor change to its Fussell-Vesely value raised it above the FV >= 0.005 threshold, it does not represent a new risk-significant component. This is also not a component and thus does not affect the risk ranking of components.
ACP-XHP-MC-A0374	MISCALIBRATION OF UV SENSOR FOR UV RELAYING 1A 0374	0.00271	0.00477	0.00304	0.00525	The event was previously captured within the FV >= 0.0045 threshold. Therefore, while a minor change to its Fussell-Vesely value raised it above the FV >= 0.005 threshold, it does not represent a new risk-significant component. This is also not a component and thus does not affect the risk ranking of components.

Table B-1: FV Discrepancies Between the Base Importances and the No UNL Importances						
BE	**Description**	**Base CDF FV**	**Base LERF FV**	**No UNL CDF FV**	**No UNL LERF FV**	**Disposition**
CD62_W2	10-A404 (NON-SEVERE) - Initiator	0.0053	0.00496	0.00499	0.00519	This represents a fire scenario rather than a component, was previously captured within the FV >= 0.0045 threshold, and was above the FV >= 0.005 threshold in the base CDF. Each of these justifications ensure that this event will not impact the risk-ranking of components.
MCA_CD47_CD46	MCA - CD47 to CD46	0.00091	0.00489	0.00101	0.00538	The event was previously captured within the FV >= 0.0045 threshold. Therefore, while a minor change to its Fussell-Vesely value raised it above the FV >= 0.005 threshold, it does not represent a new risk-significant component. This is also not a component and thus does not affect the risk ranking of components.
POR-DDP-FR-1FXE42	FLEX SWIS PUMP C1FLX-1FXE42 FAILS TO RUN	0.00593	0.0019	0.00313	0.00787	While the LERF value became newly risk-significant in the No UNL case, this component was already risk-significant in the base CDF, and thus this does not represent a new risk-significant component.
QUVISL	ALTERNATE MAKEUP SOURCES INADEQUATE (ISLOCA)	0.00089	0.00476	0.001	0.00523	This is a 1.0 tag to represent a non-credited feature in the PRA model, and thus does not represent a new risk-significant component. Also, the event was previously captured within the FV >= 0.0045 threshold.
FWL-XH1-HI-FW8TR-F	PREVENT LEVEL 8 TRIP OF FW DURING TRANSIENT - FIRE PRA VERSION	0	0	0.02879	0.00388	This event does not represent a component, and therefore will not skew the risk ranking of components.
TT-FIRE	FIRE INDUCED TURBINE TRIP	0	0	0.04248	0.03509	This event does not represent a component, and therefore will not skew the risk ranking of components.

Table B-1: FV Discrepancies Between the Base Importances and the No UNL Importances

BE	Description	Base CDF FV	Base LERF FV	No UNL CDF FV	No UNL LERF FV	Disposition
TYPE-TURBTRIP	FIEDT FLAG	0	0	0.04248	0.03509	This event does not represent a component, and therefore will not skew the risk ranking of components.

Table B-2: RAW Discrepancies Between the Base Importances and the No UNL Importances

BE	Description	Base CDF RAW	Base LERF RAW	No UNL CDF RAW	No UNL LERF RAW	Disposition
SWS-MDP-FR-DF03	CCF FAILURE TO RUN ALL SWS MAIN PUMPS	18.26	7.301	20.355	7.925	The event was previously captured within the RAW >= 18 threshold for common cause failures. Therefore, while a minor change to its RAW value raised it above the RAW >= 20 threshold for common cause failures, it does not represent new risk-significant components.
SWS-STR-FR-DF01	CCF FAILURE TO RUN ALL SWS STRNR MOTORS	18.26	7.301	20.355	7.925	The event was previously captured within the RAW >= 18 threshold for common cause failures. Therefore, while a minor change to its RAW value raised it above the RAW >= 20 threshold for common cause failures, it does not represent new risk-significant components.
SWS-TWS-FR-DF01	CCF FAILURE TO RUN ALL 4 TWS	18.26	7.301	20.355	7.925	The event was previously captured within the RAW >= 18 threshold for common cause failures. Therefore, while a minor change to its RAW value raised it above the RAW >= 20 threshold for common cause failures, it does not represent new risk-significant components.
ACP-BAC-ST-1B222	1E DIV B 480 VAC MCC 10B222 FAILURE SHORTS	1.958	1	2.037	1	The event was previously captured within the RAW >= 1.8 threshold. Therefore, while a minor change to its RAW value raised it above the RAW >= 2 threshold, it does not represent a new risk-significant component.

Table B-2: RAW Discrepancies Between the Base Importances and the No UNL Importances						
BE	**Description**	**Base CDF RAW**	**Base LERF RAW**	**No UNL CDF RAW**	**No UNL LERF RAW**	**Disposition**
ACP-BAC-ST-1B420	1E USS BUS 10B420- DIV B FAILURE SHORTS	1.958	1	2.037	1	The event was previously captured within the RAW >= 1.8 threshold. Therefore, while a minor change to its RAW value raised it above the RAW >= 2 threshold, it does not represent a new risk-significant component.
SDP-FLOOR-F	SPURIOUS DURATION PROBABILITY FLOOR VALUE	1.892	2.533	2	2.685	The event was previously captured within the RAW >= 1.8 threshold. Therefore, while a minor change to its RAW value raised it above the RAW >= 2 threshold, it does not represent a new risk-significant component. Additionally, this event was previously above the RAW >= 2 threshold for LERF and does not represent a component.
AB3_D26	TRANSIENT - Initiator	3.028	1.997	2.953	2.096	The event was previously captured within the RAW >= 1.8 threshold. Therefore, while a minor change to its RAW value raised it above the RAW >= 2 threshold, it does not represent a new risk-significant component. Additionally, this component was previously above the RAW >= 2 threshold for CDF, and does not represent a component.
CNT-MDL-FF-WTRCV	FAILURE TO RECOVER A WATER SYSTEM	1	1.998	1	2.071	The event was previously captured within the RAW >= 1.8 threshold. Therefore, while a minor change to its RAW value raised it above the RAW >= 2 threshold, it does not represent a new risk-significant component. Additionally, this event does not represent a component.

Table B-2: RAW Discrepancies Between the Base Importances and the No UNL Importances						
BE	**Description**	**Base CDF RAW**	**Base LERF RAW**	**No UNL CDF RAW**	**No UNL LERF RAW**	**Disposition**
FIREDEPGROUP2-XHD230		1	1.877	1	2.089	The event was previously captured within the RAW >= 1.8 threshold. Therefore, while a minor change to its RAW value raised it above the RAW >= 2 threshold, it does not represent a new risk-significant component. Additionally, this event does not represent a component.

Part c: Attachment 6 was reviewed to identify additional sources of conservatism. Many of these are addressed by other RAIs, but all are included below, along with their disposition:

- The Internal Events entries for FLEX equipment failure and human error probabilities contain conservative estimates. See the response for APLA RAI 04 for more information on this topic.
- The approach used for scoping fire modelling uses the industry consensus approach and is refined as needed for risk significant contributors. Thus, the conservatisms exist in the non-risk-significant fire modelling and have negligible impact on the risk-ranking results.
- While Section 2.2 in Circuit Failure Mode Likelihood Analysis (CFMLA) [3] notebook contains some potentially conservative approaches, risk-significant function states were reviewed in detail and refined. Thus, these conservative approaches are only applied to non-risk-significant function states and have a negligible impact on risk-ranking results.
- The CFMLA techniques used implement the industry consensus approach based on NUREG methodology. Mechanically, the approach taken to implement spurious actions into the model does not subsume non-CFMLA events, so the impact of reducing the conservatism in this modelling would simply be to lower the overall CDF/LERF. This would raise importances marginally across the board, but this would not result in new risk, only increased relative risk. Additionally, there is no alternate method accepted by the NRC for this modelling. Thus, the impact on risk-ranking is expected to be marginal and not feasible.
- The Post-Fire Human Reliability Analysis contains conservative estimates, but as noted in the disposition column, these are addressed by the 5% and 95% sensitivities required by NEI 00-04 [1].
- Conservatism in the LERF truncation level is discussed explicitly in APLA RAI 05c. See the response to that RAI for more information on this topic.

REFERENCES FOR RESPONSE TO ALPA RAI 03:

[1] NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute, July 2005.
[2] NEI 16-09, "Risk-Informed Engineering Programs (10 CFR 50.69) Implementation Guidance," Revision 0, Nuclear Energy Institute, April 2017.

[3] HC-PRA-113, “Hope Creek Generating Station Fire Probabilistic Risk Assessment Circuit Failure Mode Likelihood Analysis,” Revision 2, Public Service Enterprise Group, June 2019.

APLA RAI 04 – Crediting of FLEX in the PRA Model

The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC's staff assessment of identified challenges and strategies for incorporating FLEX equipment into a PRA model in support of risk-informed decisionmaking in accordance with the guidance of RG 1.200.

In the table in LAR Attachment 6 that dispositions the key assumptions/sources of uncertainty for the internal events and internal flooding PRAs, the third and fourth items identify sources of uncertainty regarding the PRA modelling of FLEX equipment and FLEX operator actions. For the NRC staff to determine the acceptability of incorporation of FLEX equipment into the PRA models, provide the following information for the internal events and internal flooding PRAs, as appropriate:

a. A discussion detailing the extent of incorporation, i.e., summarize the supplemental equipment and compensatory actions, including FLEX strategies, that have been quantitatively credited for each of the PRA models used to support this application.

b. Supporting requirement DA-D2 in ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2, states for all capability categories, “[i]f neither plant-specific data nor generic parameter estimates are available for the parameter associated with a specific basic event, use data or estimates for the most similar equipment available, adjusting if necessary to account for differences.” Conclusion 5 of NRC memorandum dated May 30, 2017 states, “[t]he NRC staff does not agree with crediting spare portable equipment not modeled in the PRA in lieu of using appropriate failure rates, because this approach is not consistent with the ASME/ANS PRA Standard [ASME/ANS RA-Sa-2009] and RG 1.200. Furthermore, the potential impact of underestimating failure rates could be larger than the unquantified risk benefits of spare equipment not modeled in PRAs.” Conclusion 6 of the memorandum states, “[t]he failure rates of permanently installed equipment cannot be used for portable equipment even if sensitivity analyses are performed. Licensees should use plant-specific o[r] generic data collected and analyzed using acceptable approaches to estimate the failure rates for portable equipment.” For the portable FLEX equipment credited in the PRA, it is not clear whether the failure probabilities assumed in the PRA are representative of the equipment’s design, procedures, and performance (i.e., as-built, as-operated).

   i. Provide a discussion detailing the methodology used to assess the failure probabilities of any modeled equipment credited in the licensee’s mitigating strategies (i.e., FLEX). The discussion should include a justification of the rational for parameter values, and how the uncertainties associated with the parameter values are considered in the categorization process in accordance with ASME/ANS RA-Sa-2009, as endorsed by RG 1.200.

   ii. Provide a discussion detailing the plant-specific operational experience (e.g., number of failures, number of demands, operational hours) of the Hope Creek portable FLEX

equipment that are credited in the PRA. Discuss any screening or disregarding of plant-specific data (e.g., design modifications, changes in operating practices). Discuss how the failure probabilities assumed in the PRA for this equipment is consistent with the relevant plant-specific evidence/operational experience.

c. A discussion detailing the methodology used to assess operator actions related to FLEX equipment and the licensee personnel that perform these actions. The discussion should include:

   i. A summary of how the licensee evaluated the impact of the plant-specific human error probabilities and associated scenario-specific performance shaping factors listed in (a)-(j) of supporting requirement HR-G3 of ASME/ANS RA-Sa-2009, as endorsed by RG 1.200.

   ii. Whether maintenance or testing procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and whether the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of ASME/ANS RA-Sa-2009, as endorsed by RG 1.200.

   iii. For licensee's procedures governing the initiation or entry into mitigating strategies, identify specific areas which could be ambiguous, vague, or not explicit. Provide a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.

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

   i. Provide an evaluation of the model changes associated with incorporating mitigating strategies, which demonstrates that none of the following criteria is satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences,

   -OR-

   ii. Propose a mechanism to ensure that a focused-scope peer review is performed on the model changes associated with incorporating mitigating strategies, and associated F&Os are resolved to Capability Category II prior to implementation of the 10 CFR 50.69 categorization program.

e. OR, as an alternative to Parts (a), (b), (c), and (d), above:

   Remove credit for FLEX equipment in the PRA used to support this LAR, and provide updated risk results (i.e., LAR Attachment 2) that does not credit FLEX equipment and actions.

<u>RESPONSE TO APLA RAI 04:</u>

<u>Part a</u>: The phased approach to ELAP considers the following three phases:

1. Phase 1 – Use of installed equipment to provide initial plant response and assure time is available for additional mitigating strategies.
2. Phase 2 – Use of onsite portable equipment to supplement or replace the installed equipment to extend the mitigating capability beyond 24 hours.
3. Phase 3 - Use of offsite regional support centers to provide additional equipment that would allow extended coping times well beyond 24 hours.

Hope Creek uses installed plant equipment and procedures for the Phase 1 approach. Phase 2 utilizes portable equipment to supplement the plant equipment used in Phase 1. Phase 3 is not credited in the HCGS PRA. The following equipment and strategies are quantitatively credited in both the FPIE PRA model and the Fire PRA model.

- Deployment and alignment of the FLEX diesel generator and FLEX MCC are credited to provide power to the installed battery chargers, portable FLEX alternate header pumps, and the portable FLEX compressor. Credit is limited to extended loss of AC power (ELAP) scenarios.
- In the event of HPCI or RCIC failure, there are two FLEX alternate header pumps that can provide RPV injection, either through the RHR 'B' flowpath (preferred) or through the RHR 'A' flowpath (alternate). Credit is limited to extended loss of AC power (ELAP) scenarios.
- The portable Service Water Intake Structure (SWIS) FLEX pump is credited for RPV injection either through the RHR 'B' flowpath (preferred) or through the RHR 'A' flowpath (alternate). Upon failure of HPCI or RCIC due to high suppression pool temperatures, the portable diesel-driven SWIS FLEX pump injects river water directly into the RPV. The SRVs are used to reduce RPV pressure to allow for injection. The SWIS FLEX pump is an alternate injection source identified in the Emergency Operating Procedures (EOPs) and is credited as such for non-ELAP scenarios.
- The FLEX compressor is credited to supply instrument gas for the SRVs for RPV pressure control. The portable compressor is aligned to the Primary Containment Instrument Gas System (PCIGS) via a FLEX connection. Credit is limited to extended loss of AC power (ELAP) scenarios.

<u>Part b.</u>

i. The base failure probabilities associated with the credited portable FLEX equipment use generic failure rate data already used in the PRA models. The source of the generic data is NUREG/CR-6928 and is based on the component data most representative of the FLEX components.

   For example, the failure rate for the SWIS FLEX pump failure to start uses the failure rate data for diesel driven pump fail to start. However, given the uncertainty associated with the reliability of the portable FLEX equipment, an assumption is made that two-times the generic reliability values for similar equipment provides a reasonable approximation of the reliability of the FLEX equipment. The factor of two is applied to the following PRA-credited FLEX equipment:

   - SWIS FLEX pump

- FLEX alternate header pumps
- FLEX diesel generator
- FLEX compressor

The uncertainties associated with the data values are based on the uncertainty parameters from the generic data and are in accordance with the ASME/ANS PRA Standard.

ii. The most recent PRA update (HC117A) was scheduled to correspond with the initial implementation of portable FLEX equipment. Thus, no plant-specific operating experience existed, and the parameters described above were used. For most PRA applications, use of the values described above should provide a reasonable approximation of the reliability of the FLEX equipment while plant specific and industry data for FLEX components is being collected.

Hope Creek's FLEX equipment data are still sparse due to the short time that FLEX has been in service. A statistically meaningful comparison of Hope Creek's operational experience to the failure probabilities assumed in the PRA is not possible at this time. Additionally, comparing this experience to the probability estimates described above would not be meaningful because the current probability estimates are not from a database that includes portable FLEX equipment from nuclear power plants.

Part c:

i. A detailed Human Reliability Analysis (HRA) was performed for each credited operator action related to FLEX equipment using the methodologies used for human error probabilities in the HCGS PRA models. This HRA methodology has been developed and commonly used throughout industry in nuclear power plant PRAs for FLEX portable equipment HRA. As such, the HEPs include an evaluation of the scenario-specific performance shaping factors (a)-(j) of supporting requirement HR-G3 of the ASME/ANS RA-Sa-2009 PRA Standard.

ii. Maintenance procedures for the portable equipment were not reviewed for possible pre-initiator human failures. Basic events for maintenance unavailability of the alternate header pumps are included in the model with screening values of 1E-02. Currently, the Hope Creek PRA model does not include maintenance unavailability basic events for the FLEX generator or the SWIS FLEX pump. These maintenance terms, or an equivalent maintenance term for the FLEX system as a whole, will be included in the PRA model prior to its use for categorization and will effectively represent failure of the entire FLEX system. This proposed model change is tracked within the Updating Requirements Evaluation (URE) process, which is the process by which changes to the PRA model are evaluated and tracked for inclusion in a future model update. With the screening values in the current PRA model, the importance measures for the existing maintenance terms do not exceed the risk-significance threshold per the PRA standard. This low risk significance of the maintenance events, combined with the fact that the FLEX equipment failure probability values and combined required post-initiator HEPs used in the model far exceed this maintenance unavailability screening value, allows pre-initiator human failures for the FLEX components to be screened from inclusion in the model.

iii. All the procedures for use of FLEX equipment or mitigating strategies at Hope Creek are clear and explicit. The probability of failure to initiate mitigating strategies is modeled in the PRA via the human error probability for operator failure to declare ELAP and perform load shed. Given a loss of offsite power and failure of the emergency diesel generators to start, the plant is in a station blackout. Procedure HC.OP-AB.ZZ-0135 evaluates whether or not an ELAP should be declared, and this is performed within 1 hour of event initiation. The declaration of ELAP is based on an evaluation if any EDGs can be started and loaded or if offsite power can be restored to the safety related buses within 4 hours. After declaring an ELAP, operators commence load shed. The operator action and associated human error probability in the PRA for declaring ELAP was developed using the same detailed Human Reliability Analysis (HRA) common in industry as described in part (c)(i) above.

The initial cues for operators are obvious (i.e., loss of offsite power and no onsite AC power for 1 hour). Given the evaluation that power cannot be restored within 4 hours, this requires declaration of ELAP and entry into FLEX procedures. This leaves little ambiguity in the procedure governing entry into mitigating strategies.

Part d:

i. As described in the other responses to this RAI, no new methodologies were implemented for the FLEX equipment and mitigating strategies that were added to the PRA model. The impacts of the modelling changes on the results were as expected (i.e., reductions CDF/LERF in the SBO and total loss of AC power accident sequences when FLEX generators are credited). These changes do not represent a change in scope or capability that impacts the significant accident sequences or progression sequences, but merely represent model updates to ensure that the models reflect the as-built, as-operated plant.

ii. Future versions of the PRA models may require a focused-scope peer review of the model changes associated with incorporating mitigating strategies, and associated F&Os will be resolved to Capability Category II prior to implementation of the 10 CFR 50.69 categorization program. The PSEG PRA model update guidance requires an evaluation to determine the need for focused scope peer review as part of the periodic model update process.

Part e:

Because of concerns about the credit for FLEX equipment, PSEG will not use the HC117A PRA model or the HC119F0 fire PRA model with FLEX credit for 10CFR50.69 categorization. The risk results that do not credit ELAP declaration or portable FLEX equipment and will be used for initial categorization activities are tabulated as follows:

Full Power Internal Events / Internal Flooding PRA Model				
Unit	Model	Baseline CDF	Baseline LERF	Comments
1	HC117A January 2018 Peer Reviewed against RG 1.200 R1 in March 2009	6.5E-06 per year	1.9E-07 per year	Gap Assessment to RG 1.200 R2 Documented (Attachment 7) The CDF/LERF results shown remove credit for the FLEX strategy. This includes failure of ELAP declaration and failure of the portable FLEX equipment.
Fire PRA Model				
Unit	**Model**	**Baseline CDF**	**Baseline LERF**	**Comments**
1	HC119F0 June 2019 Peer Reviewed against RG 1.200 R2 in October 2010	3.7E-05 per year	6.5E-06 per year	The CDF/LERF results shown remove credit for the FLEX strategy. This includes failure of ELAP declaration and failure of the portable FLEX equipment.

During each periodic PRA update, PSEG takes steps to improve the PRA models. This routinely includes incorporating updated published data sources, updated plant specific data, and new or improved modelling techniques.

- Currently, there is no published nuclear industry data report on portable FLEX equipment. When such a report is completed, PSEG intends to use this report(s) as a source of generic data.

- Currently, there is no statistically significant plant-specific data on portable FLEX equipment. As this data accrues, PSEG will consider this data in accordance the Data Analysis (DA) supporting requirements of the latest endorsed version of the PRA Standard.
- Currently, there is no industry state-of-the-practice HRA methodology specifically validated for portable FLEX equipment. PSEG will monitor industry publications and other industry PRAs to ensure that the best available HRA approach is used.
- Currently, the PSEG PRA model does not require the inclusion of maintenance terms for the portable FLEX equipment. Once best estimate HRA and data parameters are available, this approach will be evaluated against the guidance in the PRA standard and maintenance unavailability parameters will be added, if necessary.

When the above activities are completed, and peer reviewed in accordance with the latest endorsed PRA standard, as amplified by NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), the PRA model would be used for 50.69 categorization.
PSEG proposes to add the following item to the List of Categorization Prerequisites provided in Attachment 1 to the Enclosure to Reference 1:

> PSEG will not credit portable FLEX equipment in the PRA models used for 10CFR50.69 categorization until focused-scope peer reviews are performed on the model changes associated with incorporating mitigating strategies, and associated F&Os are resolved to Capability Category II.

<u>REFERENCE FOR RESPONSE TO APLA RAI 04:</u>

1. Hope Creek Generating Station, Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors" dated November 25, 2019 (ADAMS Accession No. ML19330C961).

<u>APLA RAI 05 - Dispositions of Key Sources of Uncertainty</u>

Paragraphs (c)(1)(i) and (ii) of 10 CFR 50.69 require that a licensee's PRA be of sufficient quality and level of detail to support the SSC categorization process, and that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience. The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model to address uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask importance of components. NEI 00-04 guidance states that additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered.

The dispositions provided in LAR Attachment 6 for some of the key assumptions or sources of uncertainty appear to potentially impact the SSC categorization process. In light of these observations, provide the following information:

a. In the table in LAR Attachment 6 that dispositions the key assumptions/sources of uncertainty for the internal events and internal flooding PRAs, the impact assessment for

the first item regarding the digital feedwater control failure probabilities implies that the failure probabilities are from a vendor. The NRC staff notes that the general requirement for PRA data is to obtain data from recognized sources (e.g., NUREG/CR-6928), adequate plant-specific data, or expert judgement.

i. Describe the data source(s) used to formulate the digital feedwater control probabilities, including those related to common cause failure (CCF), and justify that this source(s) is in accordance with ASME/ANS RA-Sa-2009, as endorsed by RG 1.200.

-OR-

Provide justification, such as a sensitivity study, that the impact of using an unrecognized data source does not impact the results of the 10 CFR 50.69 categorization process. The justification should also address the CCF contribution, accordingly.

ii. Alternatively to part (i), propose a mechanism to ensure the appropriate failure probabilities for digital feedwater control (including CCF, as applicable) that meet the supporting requirements in ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, are incorporated into the PRA prior to implementation of the 10 CFR 50.69 risk categorization process. Also, explain how these failure probabilities will be developed consistent with the supporting requirements in ASME/ANS RA-Sa-2009, as endorsed by RG 1.200.

b. In the table in LAR Attachment 6 that dispositions the key assumptions/sources of uncertainty for the fire PRA, the disposition to the post-fire human reliability analysis (HRA) uncertainty states that a floor value of 1E-06 or 5E-07 was applied for identified dependent combinations.

NUREG-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report,” dated July 2012 (ADAMS Accession No. ML12216A104), discusses the need to consider a minimum value for the joint probability of multiple human failure events (HFEs) in HRAs. NUREG 1921 refers to Table 2-1 of NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA),” dated April 2005 (ADAMS Accession No. ML051160213), which recommends that joint human error probability (JHEP) values should not be below 1E-5. Table 4-4 of Electric Power Research Institute (EPRI) 1021081, “Establishing Minimum Acceptable Values for Probabilities of Human Failure Events,” provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. Therefore, the available guidance provides for assigning JHEPs that are less than 1E-5, but only through assigning proper levels of dependency. Cutsets with JHEP values less than this value should be individually reviewed (e.g., for timing, cues) to check the dependency between all operator actions in the cutset. In light of these observations:

i. Provide the basis for both floor values (i.e., 1E-06 or 5E-07) and explain why/how the two different values are utilized.

ii. Describe the process used in determining JHEP values in the fire PRA. Include in this discussion an estimate of the number of JHEP values below 1E-05, discuss the range of values, and confirm that each JHEP value below 1E-05 has its own justification that demonstrates the inapplicability of the NUREG-1792

lower value guideline (i.e., that the criteria for independent HFEs are met). Provide at least two different examples where this justification is applied.

iii. Provide justification, such as a sensitivity study, that the JHEP floor values used in the fire PRA do not impact the results of the 10 CFR 50.69 categorization process.

- OR -

Alternatively, propose a mechanism to ensure that the guidance of NUREG-1792 for JHEP floor values is incorporated into the fire PRA prior to implementation of the 10 CFR 50.69 risk categorization process.

c. In the table in LAR Attachment 6 that dispositions the key assumptions/sources of uncertainty for the fire PRA, the disposition to the fire risk quantification states that the LERF convergence is slightly above 5 percent at a truncation value of 1E-11. It continues by stating that quantification at 1E-12 would not significantly alter the risk results and insights. NRC staff notes that the convergence requirement is to ensure no significant accident sequences are inadvertently eliminated. With regard to small changes in risk, the staff notes these changes could potentially increase the risk importance values for certain system components above the threshold criteria for determining high-safety-significant (HSS). In light of these observations:

i. Provide the actual LERF convergence value achieved.

ii. Provide justification, such as a sensitivity study, that using a truncation value that does not achieve convergence does not impact the results of the 10 CFR 50.69 categorization process.

- OR -

Alternatively, propose a mechanism to ensure the use of a truncation value that achieves convergence is incorporated into the fire PRA prior to implementation of the 10 CFR 50.69 risk categorization process.

<u>RESPONSE TO APLA RAI 05:</u>

a.

i. Feedwater system failures are incorporated into the Hope Creek PRA initiating event categorization. Partial feedwater failures are categorized into the Turbine Trip initiator category and complete loss of feedwater is categorized into the Loss of Feedwater initiator category. The reliability values from similar vendor studies demonstrates that the system performance would result in less than 0.1 transients per year and would have very little impact on the final calculated initiator probabilities. Generic initiator data from NUREG/CR-6928 was used for Hope Creek and updated based on plant-specific operational data. The generic initiator data source of NUREG/CR-6928 includes appropriate data to capture such failures of digital feedwater control. Therefore, the initiator failure probabilities are developed consistent with the ASME/ANS PRA Standard.

Basic events representing the reliability values for auto level controller, field buses, false signal from the redundant reactivity control system, and false signal

from the Level 8 trip system are included in the system logic model. These basic events use NUREG/CR-6928 as a data source. Therefore, the failure probabilities are developed consistent with the ASME/ANS PRA Standard.

ii. Item (i) has been responded to. Therefore, this subitem is not applicable.

b.

i. An artificial floor has been placed on the combination of human errors that are explicitly quantified. Based on past PRA experience, it is judged that combinations of operator actions that lead to JHEPs below 1E-7 are extremely difficult to justify. Experience has shown that regardless of the floor value chosen, those JHEPs are not risk significant. This was confirmed for the Hope Creek PRA. Common industry best practices state that using a higher JHEP floor value could obscure risk insights and lead to unmanageable PRAs (e.g., quantification time, computation memory limitations). Therefore, an artificial minimum (i.e., floor) of 1E-6 is used to establish that combinations of errors do not result in a failure probability less than 1E-6 even if the actions are judged to be “independent”. If the time window for the constituent actions within the JHEP is greater than 12 hours (shift length at HCGS), a floor of 5.0E-07 may be used because of the multiple recovery opportunities inherent to long-term actions, including a change of shift.

ii. The HRA dependency analysis methodology used for the Hope Creek Fire PRA has been evaluated as part of a peer review and meets the ASME/ANS PRA Standard. The methodology includes use of the HRA Calculator tool. Quantifications of the model using seed values to artificially inflate the HEPs for the dependency analysis were performed and the results were analyzed using the HRA Calculator.

There are 168 JHEPs associated with the Level 1 (CDF) combinations, and 45 of those are less than 1E-5, with the smallest nominal value being 4.46E-09. There are 71 JHEPs associated with the Level 2 (LERF) combinations, and 13 of those are less than 1E-5, with the smallest nominal value being 3.28E-08.

All JHEP results, particularly those that are near or below the JHEP floor, are manually reviewed to evaluate levels of dependency, order of actions, timing, and other criteria, and then any adjustments necessary are made. This includes review of a sample of cutsets containing the JHEP of interest. After review and any additional adjustments to the JHEP combination, any JHEP that was nominally calculated to be below the floor value was reset to the floor in the recovery file for final quantification. NUREG-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report,” dated July 2012 (ADAMS Accession No. ML12216A104) in Section 6.2 acknowledges that the floor value stated in NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA),” dated April 2005 (ADAMS Accession No. ML051160213) of ~1E-05 is a suggestion. NUREG-1921 further acknowledges that the use of a 1E-05 floor value can introduce skewing of risk metrics and importances as seen in the Significance Determination Process (SDP). Furthermore, the JHEP lower bound application is the same as that used for the internal events PRA, which is recommended in NUREG-1921. Therefore, each JHEP has a justification that demonstrates an applicable floor.

Two examples of the application of the JHEP floor value are combination 390 and combination 457.

JHEP combination 390 has a nominal value of 5.85E-08. This combination consists of four independent HFEs:

- RHR-XH1-FO-RHRDH-F (failure to initiate RHR for decay heat removal within 5 hours),
- CRD-XH1-FO-CRDEN-F (failure to maximize CRD flow to the reactor vessel),
- ADS-XH1-VF-600-F (failure to control RPV pressure at 600 psig for secondary condensate pump injection), and
- ADS-XH1-VF-HCTL-F (failure to depressurize with SRVs after HCTL).

JHEP combination 390 was changed from the nominal value to the floor value of 1.0E-06 since the overall window duration of actions in this JHEP are less than 12 hours (i.e., no shift change).

JHEP combination 457 has a nominal value of 6.90E-08. This combination consists of five independent HFEs:

- CAC-XH1-VF-VNTCT-F (failure to control containment vent),
- RHR-XH1-FO-RHRDH-F (failure to initiate RHR for decay heat removal within 5 hours),
- RHR-XH1-FO-RHRSP-F (failure to initiate RHR for decay heat removal late),
- UV1-XH1-FO-ALDHR-F (failure to align alternate injection flow paths for late injection), and
- POR-XH1-FO-1FXE42-F (failure to align SWIS FLEX pump for RPV injection).

JHEP combination 457 was changed from the nominal value to the floor value of 5.0E-07 since the overall window duration of the actions in this JHEP are greater than 12 hours (i.e., includes a shift change).

iii. A sensitivity was performed where all JHEPs nominally less than 1E-5 were set to 1E-5. The results show that when a JHEP floor value of 1E-5 is applied, the risk metrics (i.e., RAW and F-V) for basic events important to the 50.69 categorization process do not change category (i.e., LSS to HSS) when compared to the results using JHEP floor values of 1E-6/5E-7. Therefore, the JHEP floor values used in the fire PRA do not impact the results of the 10 CFR 50.69 categorization process.

c.

i. The convergence for fire PRA LERF for Hope Creek is 6.8%. This is a comparison between 1E-12 truncation limit and 1E-11 truncation limit.

ii. To ensure convergence, the 50.69 Fire LERF truncation will use a value of 1E-12 for the current model. Future models will be evaluated consistent with the PRA Standard for appropriate truncation. Unfortunately, it is not possible with existing software to directly identify the convergence at this level, as the PRAQUANT software used for the quantification cannot produce an output at 1E-13

truncation. However, this model can be run in FRANX, which produces an almost identical number. There are some differences in the way that PRAQUANT and FRANX perform quantifications, which leads to minor differences between the two. One of the biggest of these is that in the Model of Record used for the PRAQUANT calculation, fire scenarios with very low contribution are screened. This and rounding errors lead to the differences shown in Table 5c-1.

Table 5c-1 shows the PRAQUANT and FRANX Fire LERF calculations over four decades, to demonstrate that the model reaches less than 5% convergence between the 1E-12 and 1E-13 decades. The 1E-13 truncation quantification results for PRAQUANT are unknown due to software limitations, so convergence numbers are shown for the 1E-10 to 1E-11, 5E-11 to 5E-12, 1E-11 to 1E-12 and 5E-12 to 5E-13 decades for both methods, and the 1E-12 to 1E-13 decade for FRANX. Figure 5c-1 graphically maps the truncation decade to the convergence achieved for both PRAQUANT and FRANX for additional clarity. Note that the lines present on Figure 5c-1 are not formulaic, but instead simply connect the points of the graph for improved visualization.

The resulting quantification results between software have <2% difference between them and the FRANX truncation study shows a convergence of well beneath 5% between the 1E-12 and 1E-13 decades. Given that the PRAQUANT results are closer to the 5% convergence for the 1E-11 to 1E-12 decade than the FRANX results, it is reasonable to expect that the PRAQUANT solution at 1E-12 will reach significantly below 5% as well. The full truncation study documented in the model of record also indicates a continuous decrease in convergence levels when the higher decades are taken into account. Thus, these results demonstrate that utilizing a truncation value of 1E-12 for the fire PRA LERF model in the 10 CFR 50.69 risk categorization process will ensure that convergence is achieved. Engineering judgement shows that based on these results and the associated trends, convergence below 5% at 1E-12 is achieved with PRAQUANT.

Table 5c-1: PRAQUANT vs. FRANX Truncation Analysis

Truncation Decade	PRAQUANT Result (/yr)	PRAQUANT Convergence %	FRANX Result (/yr)	FRANX Convergence %
1E-10	5.58E-06	13.6%	5.59E-06	14.7%
5E-11	5.84E-06	11.5%	5.86E-06	12.5%
1E-11	6.34E-06	6.8%	6.41E-06	7.3%
5E-12	6.51E-06	5.2%	6.59E-06	5.8%
1E-12	6.77E-06	(1)	6.88E-06	3.5%
5E-13	6.85E-06	N/A	6.97E-06	N/A
1E-13	(1)	N/A	7.12E-06	N/A

(1) The 1E-13 truncation quantification results for PRAQUANT are unknown due to software limitations.

APLA RAI 05

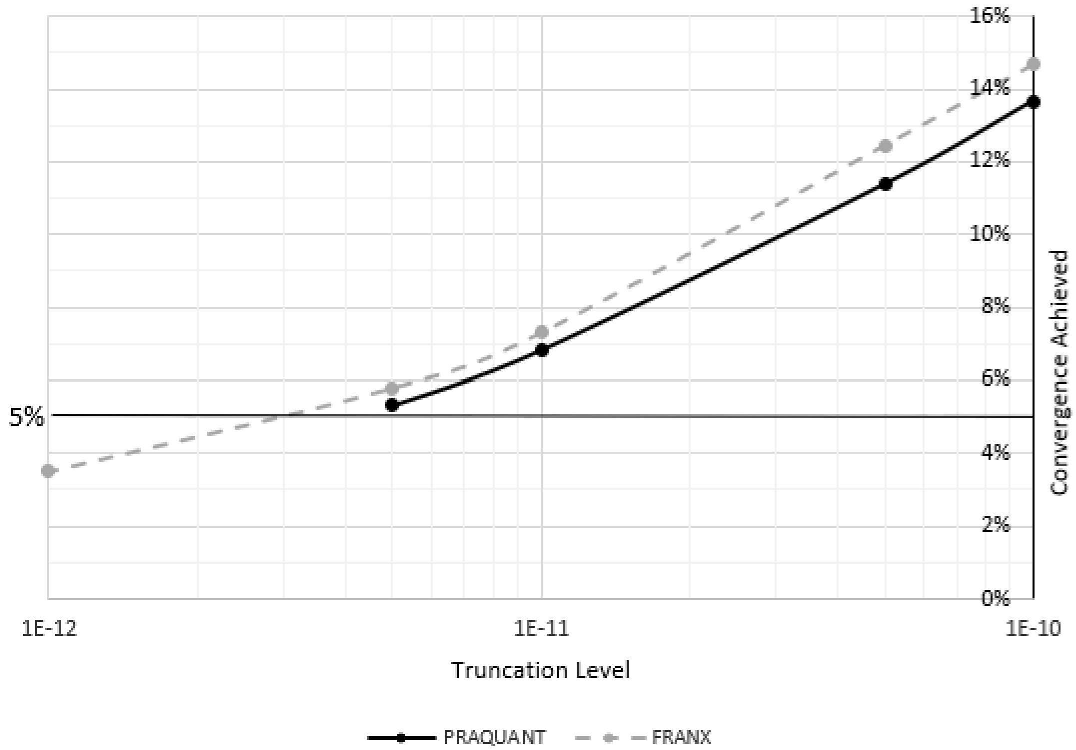


Figure 5c-1: PRAQUANT and FRANX Convergence by Decade

APLA RAI 06 – Implementation Items

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires that a licensee's application contain a description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown are adequate for the categorization of SSCs. If the responses to RAIs 01 through 05 above require any follow-up actions prior to implementation of the 10 CFR 50.69 categorization process, provide a list of those actions and any PRA modelling changes including any items that will not be completed prior to issuing the amendment but must be completed prior to implementing the 10 CFR 50.69 categorization process.

Propose a mechanism that ensures these activities and changes will be completed and appropriately reviewed and any issues resolved prior to implementing the 10 CFR 50.69 categorization process (for example, a license condition that includes all applicable implementation items and a statement that they will be completed prior to implementation of the 10 CFR 50.69 categorization process).

RESPONSE TO APLA RAI 06:

PSEG proposes the addition to the license condition provided in the original LAR application (see paragraph in **bold**) that includes all applicable implementation items with a statement that they will be completed prior to implementation of the 10 CFR 50.69 categorization process as follows:

PSEG 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; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in PSEG submittal letter dated November 25, 2019, and all its subsequent associated supplements, as specified in License Amendment No. [XXX] dated [DATE].

PSEG will complete the implementation items listed in Attachment X of PSEG letter to NRC dated Date 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).

APLC RAI 01 – Risk contribution of a seismic event

In Title 10 of the Code of Federal Regulation (CFR) 50.69(b)(2)(ii) requires a description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs. 10 CFR 50.69(b)(2)(iv) requires a description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv). The SoC on section 50.69(b)(2)(iv) of the rule states that the licensee is required to include information about the evaluations they intend to conduct to provide reasonable confidence that the potential increase in risk would be small. The SoC further clarifies that a licensee must provide sufficient information to the NRC, describing the risk sensitivity study and other evaluations and the basis for their acceptability as appropriately representing the potential increase in risk from implementation of the requirements in the rule.

Section 3.2.3, “Seismic Hazards,” of the enclosure to the LAR states that “small percentage contribution of seismic to total plant risk makes it unlikely that an integral importance assessment for a component, as defined in NEI 00-04, would result in an overall HSS determination…” Section 2.2.2 of the EPRI report identifies the contribution of seismic to total plant risk as a basis for the use of the proposed alternate seismic approach. Further, the insights in the EPRI report are derived from the full spectrum of the seismic hazard (i.e., the entire hazard curve). The LAR does not provide information to support the claim that the plant-specific seismic risk is a small percentage contribution to total plant risk and thereby, the applicability of the proposed alternate seismic approach to the licensee.

Justify that the plant-specific seismic risk is low relative to the overall plant risk such that the categorization results will not be significantly impacted to support the applicability of the proposed alternate seismic approach.

RESPONSE TO APLC RAI 01:

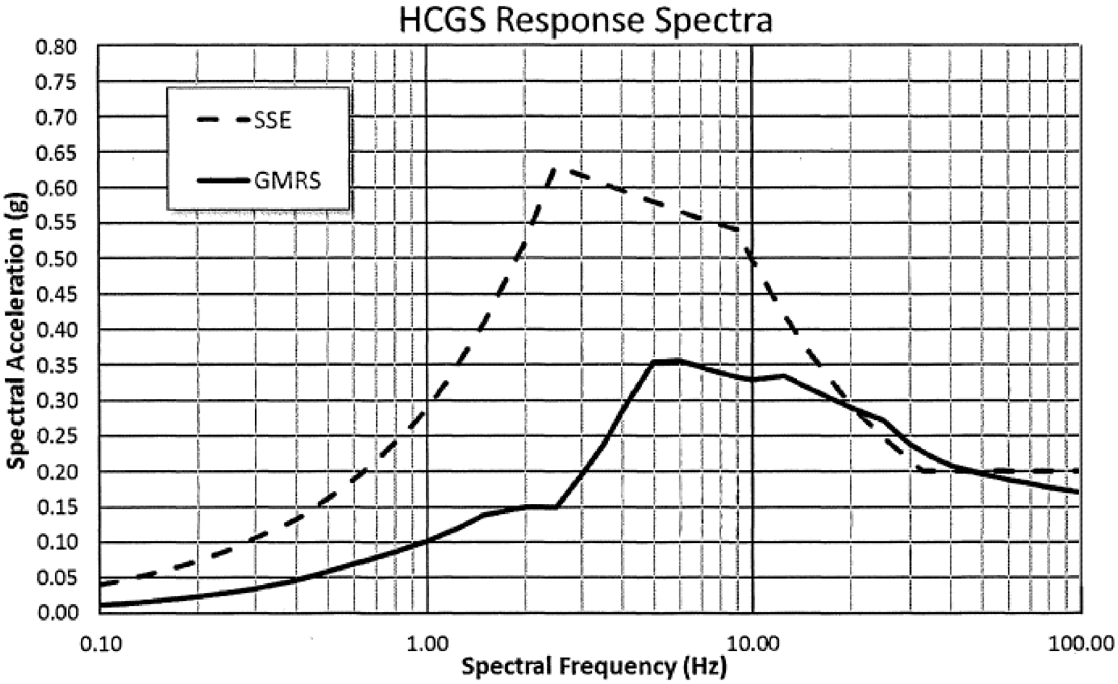
The most direct method to justify that the plant-specific seismic risk is low relative to the overall plant risk would be the use of a modern seismic PRA (SPRA) to calculate the seismic CDF (SCDF). However, Tier 1 plants such as HCGS do not have a current maintained SPRA with which to calculate that SCDF value.

An estimate of seismic core damage frequency (SCDF) for the Hope Creek Generating Station (HCGS) is calculated for use in this LAR. The calculation of the HCGS SCDF estimate is based on a simple convolution of the HCGS 2014 seismic hazard curve, in units of peak ground acceleration (PGA), and the HCGS IPEEE-based plant level seismic fragility, as reported by the NRC in Table C-2 of Reference [1]. In addition, a qualitative assessment to support the applicability of the proposed alternate 50.69 seismic approach followed by an estimate of seismic CDF is provided.

Qualitative Discussion of HCGS Seismic Risk

For the HCGS site, the GMRS-to-SSE comparison taken from the LAR and shown below in Figure 1 demonstrates that the GMRS is well below the SSE between 1 Hz and 10 Hz which qualifies it as a Tier 1 plant under the criteria in the EPRI report. These criteria are similar to the grading process used in EPRI 1025287 (Reference [2]) for the Fukushima 50.54(f) letter responses and in NEI 12-06 (Reference [3]) for the Mitigation Strategy Assessment. This comparison also confirms that the expected seismic risk at HCGS would be very low. Structures, systems, and components (SSCs) designed to the SSE level would have significant seismic capacity margins beyond the GMRS seismic level, which would result in high median capacities, and therefore would produce low seismic risk values.

Figure 1: GMRS and SSE Response Spectra for HCGS
From Reference [4]



Additionally, recent activities have been taken to ensure that seismic risk at HCGS remains low. During the post-Fukushima seismic walkdowns, HCGS verified the plant configuration with the current seismic licensing basis; addressed degraded, nonconforming, or unanalyzed seismic conditions; and verified the adequacy of monitoring and maintenance programs for protective features (Reference [5]). The justification that the plant-specific seismic risk for HCGS is low such that the 10CFR50.69 categorization results will not be significantly impacted includes the following three key conclusions taken from the EPRI report:

1. Most seismic SSCs which would be categorized as HSS for seismic risk would already be categorized as HSS for other reasons under 50.69 (e.g., internal events PRA insights, other external hazard risk insights, pressure boundary insights, Defense-in-Depth considerations, qualitative risk considerations)

2. Comparisons of the HCGS SSE to GMRS show considerable seismic margin in the plant

3. SPRA test cases conducted by EPRI demonstrate that even for plants at sites with high seismic ground motions compared to their design basis, very few if any SSCs would be designated HSS only due to seismic risks.

The text summarized in the question above from EPRI 3002017583 (formerly 3002012988) Section 2.2.2, while accurate, is simply reflecting a supplemental justification referring to the 50.69 Integral Assessment portion of the categorization process. The primary technical basis for the Tier 1 approach is described in Section 2.2.2 as the following.

> The test cases described in Section 3 showed that even for plants with high seismic ground motions compared to their design basis, there would be very few if any SSCs designated HSS for seismic unique reasons. At the low seismic hazard sites in Tier 1,

the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low.

This conclusion holds true even for cases where the seismic risks are not considerably lower than internal events risks. For example, even if the internal events risks are low, the relative risk importance measures applied in 50.69 would still identify the appropriate HSS SSCs. This HSS vs LSS determination, based on internal events risk insights, cannot be changed by the IDP or the integral importance assessment process.

Consistent with Regulatory Guide 1.174 (Reference [6]) and Regulatory Guide 1.201 (Reference [7]), while for some risk-informed applications a quantitative assessment of all plant operating modes and hazard groups may be appropriate, for other risk-informed applications such as HCGS 50.69 implementation, it is not necessary to have a PRA of such scope. A qualitative treatment of the missing modes and hazard groups is sufficient when those modes or hazards would not affect the decision. As such, a key premise of the EPRI approach is that most seismic-related SSCs that would be categorized as HSS in accordance with NEI 00-04 would also be categorized as HSS for other reasons (e.g., internal events PRA insights, other external hazard risk insights, pressure boundary insights, Defense-in-Depth considerations, qualitative risk considerations). Therefore, the EPRI report shows that this qualitative treatment of seismic risk is sufficient to demonstrate that the hazard does not affect the 50.69 HSS vs LSS categorization decision.

SCDF Estimation

An estimate of seismic core damage frequency for the Hope Creek Generating Station is calculated for use in this RAI response. The calculation of the HCGS SCDF estimate is based on a simple convolution of the HCGS 2014 seismic hazard curve from Reference [8] , in units of PGA, and the HCGS IPEEE-based plant level seismic fragility, as reported by the NRC in Table C-2 of Reference [1]. The HCGS IPEEE-based plant level seismic fragility back-calculated by the NRC and reported in Reference [1] is defined by a median seismic capacity (Am) of 1.66g PGA and an associated composite uncertainty ßc of 0.7. These plant fragility values are used in the base case convolution calculation and resulted in an SCDF estimate of 8.5E-7/yr., which is lower than the internal events and fire CDF reported in Attachment 2 of the LAR and in response to APLA RAI 04 above.

In summary:

- The HCGS plant-specific seismic risk is low relative to the overall plant risk, and

- The HCGS 50.69 categorization results will not be significantly or adversely impacted by proposed alternate seismic approach.

REFERENCES FOR RESPONSE TO APLC RAI 01

[1] Generic Issue (GI) 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," U.S. NRC Information Notice (IN) 2010-18, September 2, 2010

[2] EPRI 1025287, "Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," Electric

Power Research Institute (EPRI), February 2013.

[3] NEI 12-06, Rev 4, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," December 2016.

[4] Hope Creek Generating Station Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors," November 25, 2019 (ML19330C961).

[5] Hope Creek Generating Station - Staff Assessment of the Seismic Walkdown Report Supporting Implementation of Near-Term Task Force Recommendation 2.3 Related to the Fukushima Dai-Ichi Nuclear Power Plant Accident, (TAC NO. MF0132), May 14, 2014 (ML14127A006).

[6] NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018 (ML17317A256).

[7] NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006.

[8] PSEG Nuclear LLC LR-N14-0035, "Seismic Hazard and Screening Report (CEUS Sites) Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident - HCGS," March 28, 2014 (ML14087A436).

APLC RAI 02 – Configuration control process

Revision 3 of Regulatory Guide (RG) 1.174, , "*An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17317A256) establish the need for an implementation and monitoring program to ensure that changes proposed do not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms. An implementation and monitoring program is intended to ensure that the impact of the proposed change continues to reflect the reliability and availability of SSCs impacted by the change.

In Section 3.5 of the enclosure to the LAR, the licensee stated that its configuration control process ensured that changes to the plant, including a physical change and changes to documents, are evaluated to determine the impact on design bases, licensing documents, programs, procedures, and training. However, the licensee did not provide how the configuration control program is implemented.

Please provide a list for implementation of the configuration control program that includes

a. A review of the impact on the System Categorization Document for configuration changes that may impact a categorized system under 10 CFR 50.69.

b. Steps to be performed if redundancy, diversity, or separation requirements are identified or affected. These steps include identifying any potential seismic interaction between added or modified components and new or existing safety related or safe shutdown components or structures.

c. Review of impact to seismic loading, safe shutdown earthquake (SSE) seismic requirements, as well as the method of combining seismic components.

d. Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters Seismic Category I mechanical or electrical components.

RESPONSE TO APLC RAI 02:

a. PSEG has not yet developed the management and control systems for the 10 CFR 50.69 program. When developed, the 10 CFR 50.69 governance document will interface with PSEG's design control, risk management and other documents that make up the configuration control process.

An integrated effort will be necessary to ensure that configuration changes are evaluated for impacts on categorized systems. A system of periodic reviews will be implemented that ensures any impacts are understood and brought to the attention of the appropriate systems engineers and eventually Independent Decision-making Panel (IDP). The reviews will include:

- Identification of changes/updates to the Licensing basis that could impact the categorization process, including results affecting functions/components for categorized systems, since the previous Periodic Review.
- Identification of changes to the Emergency Operating Procedures (EOPs) or Abnormal Operating Procedures (AOPs) that could impact the responses previously provided to the system function qualitative essential questions since the previous Periodic Review performance.
- During the periodic PRA update process,
    - o Evaluation as to whether any previously categorized low safety significant (LSS) PRA modeled components should be re-categorized as HSS (HSS) due to PRA Model of Record updates. The current process for PRA Model of Record updates is a mature process for identifying any design or procedural changes that could affect the PRA model.
    - o Perform applicable tasks for Immediate and Periodic Reviews, which includes incorporation of PRA model updates into the categorizations, including updated sensitivity studies results, as applicable.

b. Steps to be performed if redundancy, diversity, or separation requirements are identified or affected. These steps include identifying any potential seismic interaction between added or modified components and new or existing safety related or safe shutdown components or structures.

c. Review of impact to seismic loading, safe shutdown earthquake (SSE) seismic requirements, as well as the method of combining seismic components.

d. Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters Seismic Category I mechanical or electrical components.

Response to b, c and d:

The PSEG configuration control process ensures that changes made to the plant and the plant operation are accurately recorded in documents such as drawings and procedures. PSEG uses the industry Standard Design Process (SDP) for design control and configuration management of plant systems, structures, and components.

Prior to implementation of 10 CFR 50.69, the configuration control program will be updated, as required, to include a list of configuration activities to recognize those systems that have been categorized in accordance with 10 CFR 50.69. Changes to the program will ensure that any physical change to the plant or changes to plant documents are evaluated prior to implementing those changes.

The list will include:

- Steps to be performed if redundancy, diversity, or separation requirements are identified or affected. These steps include identifying any potential seismic interaction between added or modified components and new or existing safety related or safe shutdown components or structures.

- Review of impact to seismic loading, safe shutdown earthquake (SSE) seismic requirements, as well as seismic interactions.

- Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters Seismic Category I mechanical or electrical components.

APLC RAI 03 – Proposed License Condition

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires, for license amendment, a description of measures taken to assure the level of detail of the systematic processes that evaluate the plant. The guidance in NEI 00-04, Revision 0, “10 CFR 50.69 SSC Categorization Guideline” (ADAMS Accession No. ML052910035), allows licensees to implement different approaches, depending on the scope of their PRA (e.g., the approach if a seismic margins analyses is relied upon is different and more limiting than the approach if a seismic PRA is used). RG 1.201, Revision 1, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance” (ADAMS Accession No. ML061090627), states, “[a]s part of the U.S. Nuclear Regulatory Commission (NRC's) review and approval of a licensee's or applicant's application requesting to implement §50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods used in the licensee's categorization approach.”

Section 2.3 of the LAR proposed the following license condition:

> PSEG 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 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the EPRI alternative approach described in EPRI 3002012988 for seismic risk for Tier 1 plants; as specified in License Amendment No. [XXX] dated [DATE].

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

The NRC staff's review reveals that the proposed alternate seismic approach provided in the LAR includes categorization considerations and elements in addition to those in the EPRI 3002012988 report for the so-called Tier 1 approach. Therefore, the proposed license condition does not appear to reflect the licensee's proposed alternate seismic approach as discussed in the LAR.

Provide a revised license condition that refers to the alternate seismic approach provided in the application and any subsequent supplements.

RESPONSE TO APLC RAI 03:

Below is the revised license condition paragraph that refers to the alternate seismic approach provided in the application and any subsequent supplements:

PSEG 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; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in PSEG submittal letter dated November 25, 2019, and all its subsequent associated supplements, as specified in License Amendment No. [XXX] dated [DATE].

APLC RAI 04 – 10 CFR 50.69 Prerequisite Requirement

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires a licensee's application contain a description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown are adequate for the categorization of SSCs.

Section 3.2.4 of the LAR states that a categorization prerequisite for screening of extreme winds or tornado hazard is provided in LAR Attachment 1. LAR Attachment 1 specifies a categorization prerequisite to complete necessary actions to screen tornado missile hazards

prior to the adoption of 10 CFR 50.69. The NRC staff notes this categorization prerequisite may introduce new SSCs or modified SSCs that will enable this hazard to be screened.

Confirm that the categorization prerequisite in LAR Attachment 1 to complete necessary actions to screen tornado missile hazards have been completed. Otherwise, propose a license condition that ensures the analysis and required plant modifications credited for screening of extreme winds or tornados is completed prior to implementation of the 10 CFR 50.69 categorization process and will include any new or modified SSCs used to justify screening to be designated as high-safety-significant.

RESPONSE TO APLC RAI 04:

The categorization prerequisite in LAR Attachment 1 to complete necessary actions to screen tornado missile hazards has been completed.

APLC RAI 05 – Alternate Seismic Approach

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires, for license amendment, a description of measures taken to assure the level of detail of the systematic processes that evaluate the plant. This includes the internal events at power PRA required by §50.69(c)(1)(i), as well as the risk analyses used to address external events.

The proposed alternate seismic approach is based on insights of the EPRI report derives risk insights from four case studies from EPRI alternative approach described in EPRI 3002012988 for seismic risk for Tier 1 plants. Those case studies compare the High Safety Significance (HSS) SSCs determined based on a seismic PRA (SPRA) against HSS SSCs determined from other PRAs used for categorization. Each of the cases studies included a full power internal events (FPIE) PRA but only two of the four case studies used information from a Fire PRA. Sections 3.3 through 3.5 of the EPRI report provide general information about the peer reviews conducted for the PRAs used for in each of the four case studies. However, the level of information is insufficient to determine whether the PRAs used in the case studies supporting this application have been performed in a technically acceptable manner.

The staff has previously requested and reviewed information to support its decision on the technical acceptability of the PRAs used in the case studies as well as details of the conduct of the case studies. This information is included in the supplements to the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, LAR for adoption of 10 CFR 50.69. The supplement to the 10 CFR 50.69 by Calvert Cliffs Nuclear Power Plant LAR dated May 10, 2019 (ADAMS Accession No. ML19130A180), contained additional information related to the alternate seismic approach including incorporation by reference docketed information related to case study Plants A, C, and D; the supplement dated July 1, 2019 (ADAMS Accession No. ML19183A012), further clarified the information related to the alternate seismic approach (see response to RAI 4); the supplement dated July 19, 2019 (ADAMS Accession No. ML19200A216), provided responses to support the technical acceptability of the PRAs used for the Plant A, C, and D case studies as well as technical adequacy of certain details of the conduct of the case studies; the supplement dated August 15, 2019 (ADAMS Accession No. ML19217A143) clarified a response in the July 19, 2019 supplement. The supplement dated July 19, 2019 included modifications to the content of the EPRI report.

Since the above-mentioned information was requested and reviewed by the staff for Calvert Cliffs Nuclear Power Plant's LAR for adoption of 10 CFR 50.69, the staff is unable to use it for

the licensee’s docket unless it is incorporated in the licensee’s LAR. The above-mentioned information is necessary for the staff to make its regulatory finding on the licensee’s proposed alternate seismic approach and has not been provided by the licensee.

1. Provide the above-mentioned information to support the staff’s regulatory finding on the alternate seismic approach by either incorporating the information by reference the identified supplements or responding to the RAIs in the identified supplements.

b. If differences exist between the licensee’s proposed alternate seismic approach and the information in the supplements stated above, identify such differences and either incorporate them in the licensee’s proposed approach or justify their exclusion.

RESPONSE TO APLC RAI 05:

1. As stated in Section 3.2.3 of the LAR, PSEG incorporated by reference plant specific test case information for case studies A, C, and D. PSEG incorporates the information described above and provided in the revised submittal and supplements (References 1 – 4) to the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, LAR for adoption of 10 CFR 50.69.

2. There are no differences between PSEG’s proposed alternate seismic approach and the information in the supplements stated above.

REFERENCES FOR RESPONSE TO APLC RAI 05:

1. Calvert Cliffs Nuclear Power Plant Units 1 and 2, Revised submittal to Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors" dated May 10, 2019 (ADAMS Accession No. ML19130A180).

2. Calvert Cliffs Nuclear Power Plant, Units 1 and 2, "Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors,'" dated July 1, 2019 (ML19183A012).

3. Calvert Cliffs Nuclear Power Plant, Units 1 and 2, "Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors,' " dated July 19, 2019 (ML19200A216).

4. Calvert Cliffs Nuclear Power Plant, Units 1 and 2, "Revised Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,' letter dated July 19, 2019," dated August 5, 2019 (ML19217A143).

5. Calvert Cliffs Nuclear Power Plant, Units 1 and 2- Issuance of Amendment Nos. 332 and 310 RE: Risk-Informed Categorization and Treatment of Structures, Systems, And Components for Nuclear Power Reactors (EPID L-2018-LLA-0482), dated February 28, 2020 (ML19330D909).