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U. S. Nuclear Regulatory Commission
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Edwin I. Hatch Nuclear Plant – Units 1 and 2
Response to Request for Addition Information Regarding Application to Adopt Title 10 of Code of Federal Regulations (CFR) 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

Ladies and Gentlemen:

By letter dated June 7, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18158A583), Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) for Edwin I. Hatch Nuclear Plant (HNP) Units 1 and 2 to adopt Title 10 of Code of Federal Regulations (CFR) 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors."

By email correspondence dated April 17, 2019, the Nuclear Regulatory Commission (NRC) staff requested additional information to complete its review. The Enclosure provides the SNC response to the NRC request for additional information (RAI). Please note that the SNC response to NRC RAI 03 and RAI 12 will be provided shortly after SNC responds to the remaining open items described in the May 28, 2019 NFPA-805 RAI response (SNC letter NL-19-0536).

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

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15TH day of July 2019.

Respectfully submitted,

C. A. Gayheart
Director, Regulatory Affairs
Southern Nuclear Operating Company

CAG/RMJ

Enclosure: SNC Response to NRC RAI

cc: Regional Administrator, Region II
NRR Project Manager – Hatch
Senior Resident Inspector – Hatch
Director, Environmental Protection Division – State of Georgia
RType: CHA02.004

Edwin I. Hatch Nuclear Plant – Units 1 and 2
Response to Request for Addition Information Regarding Application to Adopt Title 10 of Code of Federal Regulations (CFR) 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"

Enclosure

SNC Response to NRC RAI

RAI 01 (APLA/APLB) – Appendix X, Independent Assessment Process

Paragraph 50.69(c)(1)(i) of Title 10 of the Code of Federal Regulations (CFR) requires the Probabilistic Risk Assessment (PRA) to be of sufficient quality and be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the U.S. Nuclear Regulatory Commission (NRC).

Section 3.3 of the 10 CFR 50.69 License Amendment Request (LAR) states that resolutions to the Facts and Observation (F&Os) were reviewed and closed using the process in Appendix X to Nuclear Energy Institute (NEI) 05-04, NEI 07-12 and NEI 12-13, as documented in letter dated February 21, 2017 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML17086A431), and as accepted by the NRC, with conditions in letter dated May 3, 2017 (ADAMS Accession No. ML17079A427).

Provide the following information to confirm that the Independent Assessments for closure of F&Os performed for the internal events PRA (IEPRA) (April 2017), the seismic PRA (SPRA) (June 2017), and the fire PRA (FPRA) (October 2017), were performed consistent with the process accepted by the NRC, with conditions, in letter dated May 3, 2017:

a) Regarding closure of each F&O, confirm that the Independent Assessment team was provided with a written assessment and justification of whether the F&O resolution constitutes a PRA upgrade or maintenance update, as defined in the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009 PRA Standard and qualified by Regulatory Guide (RG) 1.200, Revision 2 (ADAMS Accession No. ML090410014).

SNC Response:

Internal Events (including Internal Flood) PRA:

A full scope peer review for the Internal Events (including Internal Flood) PRA was performed in November 2009. A total of 25 Finding level F&Os were issued.

An Independent Assessment (IA) Team reviewed disposition of these 25 Finding level F&Os in April 2017. The IA Team was provided with a characterization of each Finding level F&O resolution as a PRA upgrade or maintenance update. The characterization was performed using guidance provided in RIE-001 procedure, which is consistent with ASME/ANS guidance.

The IA Team determined that disposition of 21 Finding level F&Os was satisfactory; therefore, these Findings were closed out. The IA Team agreed that the disposition of these 21 Finding level F&Os constituted PRA maintenance. The remaining four Finding level F&Os are not fully closed out as stated in the LAR. Two of these Finding level F&Os were determined to not impact the 50.69 application as stated in the LAR. A focused scope peer review is scheduled for the other two Finding level F&Os, as discussed later in this Enclosure.

Fire PRA:

A full scope peer review for the Fire PRA was performed in May 2016. A total of 61 Finding level F&Os were issued.

An Independent Assessment (IA) Team reviewed disposition of 61 Finding level F&Os in October 2017. The IA Team was provided with a characterization of each Finding level F&O resolution as a PRA upgrade or maintenance. The characterization was performed using guidance provided in RIE-001 procedure, which is consistent with ASME/ANS guidance.

The IA Team determined that disposition of all 61 Finding level F&Os was satisfactory; therefore, all Findings were closed out. The IA Team also determined that resolution of one Finding level F&O (20-18) constituted PRA Upgrade. As a result, a concurrent focused-scope peer review was performed to review a method that calculated time to cable damage due to exposure of a fire environment. The IA Team determined that the method was technically sound and provided a reasonable and realistic method for estimating time to cable damage due to exposure of a fire environment. No additional F&Os were issued as a result of the focused-scope peer review.

Seismic PRA:

A full scope peer review for the Seismic PRA was performed in October 2016. A total of 23 Finding level F&Os were issued.

An IA Team reviewed disposition of these 23 Finding level F&Os in June 2017. The IA Team was provided with a characterization of each Finding level F&O resolution as a PRA upgrade or maintenance. The characterization was performed using guidance provided in RIE-001 procedure, which is consistent with ASME/ANS guidance.

The IA Team determined that disposition of all 23 Finding level F&Os was satisfactory; therefore, all Findings were closed out. The IA Team also determined that a resolution of two Finding level F&Os (6-2 and 6-10) constituted PRA Upgrade. As a result, a concurrent focused-scope peer review was performed. The disposition of two Finding level F&Os affected one high level requirement (HLR) under the technical element SHA, namely HLR SHA-I. The SHA-I has two Supporting Requirements (SHA-I1 and SHA-I2). The focused-scope peer review team determined that both Supporting Requirements met Capability Category II. No additional F&Os were issued as a result of the focused-scope peer review.

b) *If the request made in part (a) above cannot be confirmed, then perform a subsequent Independent Assessment for F&O closure and/or addendum to the Independent Assessment report to address any identified inconsistencies with Appendix X, as accepted, with conditions, by the NRC staff in letter dated May 3, 2017. Provide any F&Os that remain open as a result of this review. For each F&O and/or item that remains open, provide its associated disposition to demonstrate that it has no adverse impact on the 10 CFR 50.69 application.*

SNC Response:

Four Finding level F&Os remain open for Internal Events (including Internal Flood) PRA Model. The impact of these F&Os on 10 CFR 50.69 application has been provided in the LAR submittal. Two of these Finding level F&Os were determined to not impact the 50.69 application as stated in the LAR. A focused scope peer review is scheduled for the other two Finding level F&Os, as discussed later in this Enclosure.

All Finding level F&Os have been closed out for Fire PRA and Seismic PRA models using the Appendix X process.

c) *List all PRA upgrades, if any, identified during the IEPRA, FPRA and SPRA F&O closure. For each upgrade, confirm that a focused-scope peer review was conducted and provide all F&Os that were generated from the focused-scope peer review, along with their associated dispositions for the application.*

SNC Response:

Internal Events (including Internal Flood) PRA:

An Independent Assessment (IA) Team reviewed disposition of 25 Finding level F&Os in April 2017. The IA Team closed out 21 Finding level F&Os. The IA Team agreed that resolution of these F&Os constituted PRA maintenance. The remaining four Finding level F&Os are not fully closed out. The LAR provides a list of the four open Finding level F&Os. Two findings (1-15 and 4-5) have been characterized as a PRA upgrade. As a result, a focused-scope peer review to address findings 1-15 and 4-5 is scheduled to be performed during July 2019. SNC's expectation is that this focused-scope peer review will resolve F&O 1-15 and 4-5, and that there will be no resulting F&Os. If there are any resulting F&Os from the focused-scope peer review that will not be closed prior to NRC approval of this amendment request, then SNC will propose a license condition stating these F&Os shall be closed using the Appendix X Independent Assessment Process, as accepted, with conditions, by the NRC staff in letter dated May 3, 2017, prior to using the Internal Flooding PRA model for the implementation of the 10 CFR 50.69 categorization process.

Fire PRA:

An Independent Assessment (IA) Team reviewed disposition of 61 Finding level F&Os in October 2017. The IA Team determined that resolution of one Finding level F&O (20-18) constituted PRA Upgrade. As a result, a concurrent focused-scope peer review was performed to review a method that calculated time to cable damage due to exposure of a fire environment. The IA Team determined that the method was technically sound and provided a reasonable and realistic method for estimating time to cable damage due to exposure of a fire environment. No additional F&Os were issued as a result of the focused-scope peer review.

Table 1.1: Fire PRA Upgrades

F&O	F&O Description	Peer Review Assessment	SNC Disposition	IA Team Disposition
20-18	Basis: Based on a review of the fire modeling notebooks multiple fire modeling tools were utilized in the analysis to realistically characterize or bound the fire scenarios. These include: fire modeling workbooks as documented in Attachment 2 of H-RIE- FIRE-U00-008A, the application of Appendix L in H-RIE-FIRE-U00-008B, the use of CFAST to support MCR abandonment calculations in H-RIE-FIRE-U00-008B, the use of FDS to support a subset of the HGL calculations in H-RIE-FIRE-U00-008F, and the MQH equation for the HGL calculations performed in non-detailed fire modeling H-RIE-FIRE-U00-008E. However, the application of fire damage delay times provided in Appendix H of NUREG/CR-6850 using the Daphne Tool may not bound	FSS-D3 CC-II Met	The time to damage for targets as determined by DAPHNE will be removed. The time to damage will be determined using an equivalent exposure, damage integral method. This method equates the integrated heat flux of the actual fire to the steady-state cases and thus includes the target heating prior to the exposure reaching the damaging temperature. Reference: TECH-FRA-02, A Heat Soak Method for Evaluating Time to Cable Damage.	**Status:** Closed **Basis:** TECH-FRA-02 was reviewed to verify that this method is reasonable as compared to the methodology presented in NUREG/CR-6850, Appendix H. The method presented in TECH-FRA-02 considers pre- heating of target cables and the effect of that heating on damage time as the fire environment changes. Unlike Appendix H, this method allows for variation in the exposure environment (temperature or heat flux). The approach seems sound and reasonable for the application; however, this is judged to be an “upgrade” due to the use of a “new method.” According to the guidance of Appendix X to NEI 05-04/07-12/12-13, a focus peer review is required for this new method. As such, a focused peer review of the new evaluation method presented in Technical Procedure TECH-FRA- 02, “A Heat Soak Method for Evaluating Time to Cable Damage” was conducted concurrently by Mr. Robert Ladd and Mr. James Lin to ascertain the acceptability of this, previously unreviewed, method. This review determined that the method is technically sound and provides a reasonable and realistic method for estimating time to cable

F&O	F&O Description	Peer Review Assessment	SNC Disposition	IA Team Disposition
	or realistically characterize the fire damage times. Appendix H of NUREG/CR-6850 provides delay times assuming a constant heat flux/temperature from t=0 and therefore consideration must be taken for the impact of a dynamic heat flux/temperature profile on the delay times. Therefore, a finding has been applied relative to the use of the Daphne Tool.			damage due to exposure of a fire environment. Based on this review this F&O is considered closed.

Seismic PRA:

An IA Team reviewed 23 Finding level F&Os in June 2017. The IA Team determined that a resolution of two Finding level F&Os (6-2 and 6-10) constituted PRA Upgrade. As a result, a concurrent focused-scope peer review was performed. The disposition of two Finding level F&Os affected one high level requirement (HLR) under the technical element SHA, namely HLR SHA-I. The SHA-I has two Supporting Requirements (SHA-I1 and SHA-I2). The focused-scope peer review team determined that both Supporting Requirements met Capability Category II. No additional F&Os were issued as a result of the focused-scope peer review.

Table 1.2: Seismic PRA Upgrades

F&O	F&O Description	Peer Review Assessment	SNC Disposition	IA Team Disposition
6-2	The analyses used to evaluate the IPEEE analysis of soil liquefaction, ground settlement, and slope stability are largely qualitative in nature and leave doubt about the veracity of the conclusions.	Not Met SHA-I1	A soil failure evaluation has been performed using state of practice methods. Liquefaction triggering is screened in as a credible soil failure mode. State-of-practice empirical methodologies have been used to	See focused-scope peer review

F&O	F&O Description	Peer Review Assessment	SNC Disposition	IA Team Disposition
	Basis: A rationale is provided for not using more recently developed methods to evaluate liquefaction potential (e.g., Youd et al., 2001; Cetin et al., 2004; Idriss and Boulanger, 2008) based on the observation that there is a lack of consensus on these methods. However, it is noted that these new methods are based on "very carefully crafted databases" assembled by "acknowledged experts" to develop "mathematically correct relationships." In that context, the lack of consensus may be viewed as epistemic uncertainty that should be included in the analysis of liquefaction triggering rather than reasoned away. That statement that the new methods are biased because they include more sites that have liquefied than not contradicts the earlier assertion that regression analysis or other mathematical techniques are used to obtain "the most mathematically correct relationships." The assertion that new methods may introduce conservatisms that bias the results is puzzling. First, as noted above, they are based on more rigorous mathematical		estimate effects (principally vertical liquefaction-induced settlements) as a function of seismic hazard level. Multiple methodologies have been employed to provide epistemic uncertainty. Estimates of liquefaction-induced vertical ground settlement are performed based on weighted averaging of the methodologies employed. The susceptibility of plant structures and buried structures to damage from vertical settlements is evaluated. Slope stability is screened as a credible soil failure mode. Slope stability failure is postulated for one safety-related slope. State-of-practice methodology is used estimate lateral slope deformations as a function of hazard level. The fragility of plant commodities potentially affected by the lateral slope deformations is provided for use in the plant logic model. Finding is judged to be closed.	

F&O	F&O Description	Peer Review Assessment	SNC Disposition	IA Team Disposition
	analyses than used in earlier methods where bounds between sites that have and have not liquefaction were often drawn subjectively. Second, the deterministic methods employed in the IPEEE analyses have been shown to implicitly correspond to a probability of liquefaction equal to approximately 15%. It would preferable to use one or more new methods that explicitly calculate the probability of liquefaction in an effort to develop a median-centered estimate of the liquefaction hazard. The method employed in the IPEEE to estimate Newmark displacements (i.e., Makdisi and Seed, 1978) is a decoupled method. More recently, studies have demonstrated that the decoupled approach may be either conservative or unconservative depending on the ratio of the fundamental period of the sliding mass to the predominant period of the ground motion and the ratio of the yield coefficient to the maximum horizontal equivalent acceleration. These checks should have been performed before using the results from the IPEEE analyses. In addition, several statements were made			

F&O	F&O Description	Peer Review Assessment	SNC Disposition	IA Team Disposition
	with respect to perceived conservatisms in various assessments (e.g., the behavior of the concrete duct bank or shear strength of soils). Such statements are unsupported and speculative in nature. Resolution: The screening level analyses for soil liquefaction, ground settlement, and seismic slope stability/deformation should be updated to reflect advancements in the past 20 years sincethe IPEEE analyses were performed			
6-10	Liquefaction triggering analyses performed for the FLEX travel paths indicate that liquefaction should not be screened out. Basis: Liquefaction triggering analyses performed for the FLEX travel paths at the for borings T-8 and T-9 indicate that the factors of safety against liquefaction (FS) are 1.0 and 1.3, respectively, for a groundwater table at Elevation 77 ft. Although the FS are 1.0 or above, analyses (e.g., Idriss and Boulanger, 2010) show that the probability of liquefaction associated with FS = 1.0 is approximately 15%, i.e.,	Not Met SHA-I1	Liquefaction triggering for the FLEX Travel Paths locations T-1, T-8 and T-9 screens in for localized liquefaction at depths >55 to 60-ft. below round surface. The effects of liquefaction are summarized by seismic hazard level. A study is performed to show that the loose sands at depth in these locations are not present under or adjacent to the Unit 2 Reactor Building and Unit 2 Radwaste	See focused-scope peer review

F&O	F&O Description	Peer Review Assessment	SNC Disposition	IA Team Disposition
	not negligible. Both soil borings are located with approximately 200 ft. of the Reactor Building. The layer(s) with the lowest FS are at a depth of 60-70 ft. below ground surface (bgs); the foundation of the Reactor Building is at a depth of 54 ft. bgs. On this basis alone, liquefaction should not have been screened out from further consideration. Resolution: Liquefaction-related ground failure should screen in. Additional analyses should be performed to assess the "frequency of hazard occurrence and the magnitude of hazard consequences" per SHA-I2		Building. The Unit 2 Reactor Building is assessed for liquefaction triggering in terms of the soil beneath the structure. Finding is judged to be closed.	

d) *Appendix X guidance states in part, "[t]he relevant PRA documentation should be complete and have been incorporated into the PRA model and supporting documentation prior to closing the finding." For closure of F&O(s) after the on-site review, Appendix X guidance explicitly states, "[t]he host utility may, in the time between the on-site review and the finalization of the independent assessment team report, demonstrate that the issue has been addressed, that a closed finding has been achieved, and that the documentation has been formally incorporated in the PRA Model of Record [MOR]."*

i. *Confirm that all model changes associated with the closure of all F&Os reviewed during the Independent Assessment performed in May 2017 were incorporated into the PRA and/or the supporting documentation at the time of the finalization of the Independent Assessment team report, consistent with Appendix X, as accepted, with conditions, by the NRC staff via letter dated May 3, 2017 (ADAMS Accession No. ML17079A427).*

SNC Response:

The seismic PRA Model of Record (MOR) includes all changes associated with the closure of all Finding level F&Os reviewed during the Independent Assessment performed in May 2017.

OR

ii. *Propose a mechanism that assures all the PRA model logic and all documentation changes reviewed by the Independent Assessment team for the closure of all F&Os in* *the final Independent Assessment report are incorporated into the MOR(s) prior to implementation of the 10 CFR 50.69 risk-informed categorization.*

SNC Response:

This is not applicable due to the response in part 1(d)(i).

RAI 02 (APLA) – Internal Flooding Open F&O 4-5, Credit for Manual Flood Isolation

Internal flooding F&O 4-5 related to supporting requirement IFSN-A10 and IFQU-A5 identified that no credit is taken for the manual isolation of floods.

The independent assessment team identified that this F&O identifies a major modeling issue with internal flooding, and that it constitutes significant changes from the previously peer reviewed model. The licensee's disposition states that this is a documentation issue with no impact on the application, and:

> *"[T]he original flooding evaluation credited manual isolation of flooding using some screening values and some detailed [human reliability analysis] (HRA). A subsequent revision removed all credit for isolation but performed a flooding screening analysis. Then a third revision re-applied the previous HRA analysis to the scenarios that passed the screening."*

a) Clarify whether the internal flooding PRA (IFPRA) model that will be used for the 10 CFR 50.69 categorization process credits manual flooding isolation.

SNC Response:

The internal flooding PRA model that will be used for 10 CFR 50.69 categorization credits manual flooding isolation. As a point of clarification, the current Model of Record already credits manual flooding isolation. The resolution of F&O 4-5 has been determined to be PRA upgrade in accordance with the ASME/ANS RA-Sa-2009 PRA standard. A focused-scope peer review to address findings 1-15 and 4-5 is scheduled to be performed during July 2019. SNC's expectation is that this focused-scope peer review will resolve F&Os 1-15 and 4-5, and that there will be no resulting F&Os. If there are any resulting F&Os from the focused-scope peer review that will not be closed prior to NRC approval of this amendment request, then SNC will propose a license condition stating these F&Os shall be closed using the Appendix X Independent Assessment Process, as accepted, with conditions, by the NRC staff in letter dated May 3, 2017, prior to using the Internal Flooding PRA model for the implementation of the 10 CFR 50.69 categorization process.

b) If the IFPRA does not credit manual flooding isolation, provide justification for why this exclusion has no adverse impact on the 10 CFR 50.69 application.

SNC Response:

This is not applicable due to the response in 2(a).

c) If the IFPRA credits manual flooding isolation, address the following:

i. *Provide date of when the version of the PRA model that credits manual flooding isolation was peer reviewed, and a brief description of the PRA model that was peer reviewed (e.g, detailed HEPs developed, screening values applied, etc.).*

SNC Response:

The internal flooding PRA model and associated comprehensive report were developed by a vendor and delivered to SNC in January 2009. The PRA model developed by the vendor contained credit for manual flooding isolation operator action and derivation of detailed HEP for select operator actions.

After SNC took ownership of the Internal Flooding PRA model, the SNC model owner removed credit for the manual flooding isolation operator action. The SNC model owner created an addendum to the Internal Flooding PRA comprehensive report to document changes made to the logic model. However, the applicable chapters in the original report developed by the vendor were not revised to reflect removal of credit for the manual flooding isolation operator action.

A full scope peer review for Internal Events (including Internal Flooding) PRA was performed in November 2009. The peer review team was presented with a logic model that did not contain credit for manual flooding isolation operator action along with an addendum developed by the SNC model owner and the comprehensive report developed by the vendor. After reviewing the supplied information, the peer review team noted that although the SNC model owner removed credit for the manual flooding isolation operator action, additional flood targets were not failed as non-isolated floods would have failed them. Furthermore, the peer review team suggested that operator action should be added to the logic model to reflect how the plant would be operated in the event of flood.

Therefore, a finding level F&O was issued.

ii. *Describe any further modeling changes that were performed in the PRA after the peer review identified in part (i) above (e.g., additional HEPs developed, HEPs removed, screening values applied, etc.).*

SNC Response:

In order to resolve the peer review finding, the operator actions were added to the logic model. The Human Error Probabilities (HEPs) derived in the original vendor report (delivered to SNC in January 2009) were used. In addition, new HEPs were derived for additional flood scenarios that were not screened out when the internal flooding PRA model was updated. The methodology used to derive HEP for new operator actions was the same as that was used by the vendor and presented to the full scope peer review team.

Also, in 2018 the Internal Flooding model was updated to separate it from the Internal Events model and to update the HRA analysis of the flooding isolation actions. For the 10 CFR 50.69 categorization process, the model of record for Internal Flooding will be used.

iii. *Provide justification for why the change to credit manual flooding isolation in the IFPRA is either a PRA upgrade or maintenance update in accordance with the ASME/ANS RA-Sa-2009 PRA standard.*

SNC Response:

The resolution of F&O 4-5 has been determined to be a PRA upgrade in accordance with the ASME/ANS RA-Sa-2009 PRA standard. A focused-scope peer review to address findings 1-15 and 4-5 is scheduled to be performed during July 2019. SNC's expectation is that this focused-scope peer review will resolve F&O 1-15 and 4-5, and that there will be no resulting F&Os. If there are any resulting F&Os from the focused-scope peer review that will not be closed prior to NRC approval of this amendment request, then SNC will propose a license condition stating these F&Os shall be closed using the Appendix X Independent Assessment Process, as accepted, with conditions, by the NRC staff in letter dated May 3, 2017, prior to using the Internal Flooding PRA model for the implementation of the 10 CFR 50.69 categorization process.

iv. *If a PRA upgrade is determined in part (iii) above, confirm that a focused-scope peer review was conducted and provide the resulting F&Os along with the disposition for each F&Os impact on the 10 CFR 50.69 application. Alternatively, propose a mechanism in response to RAI 12 that ensures that, prior to implementation of the 10 CFR 50.69 categorization process, a focused-scope peer review will be conducted for the credit for manual flood isolation in the IFPRA, and any resulting F&Os will be closed using the Appendix X Independent Assessment Process, as accepted, with conditions, by the NRC staff in letter dated May 3, 2017.*

SNC Response:

A focused-scope peer review to address findings 1-15 and 4-5 is scheduled to be performed during July 2019. SNC's expectation is that this focused-scope peer review will resolve F&Os 1-15 and 4-5, and that there will be no resulting F&Os. If there are any resulting F&Os from the focused-scope peer review that will not be closed prior to NRC approval of this amendment request, then SNC will propose a license condition stating these F&Os shall be closed using the Appendix X Independent Assessment Process, as accepted, with conditions, by the NRC staff in letter dated May 3, 2017, prior to using the Internal Flooding PRA model for the implementation of the 10 CFR 50.69 categorization process.

RAI 03 (APLA) – Hatch FPRA under Review for Adoption to NFPA-805

Paragraphs 50.69(c)(1)(i) and (ii) of 10 CFR require a licensee's PRA be of sufficient quality and level of detail to support the system, structures, and components (SSCs) 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 of the 10 CFR 50.69 LAR states that a LAR was submitted requesting transition to the National Fire Protection Association (NFPA) 805 Performance-Based Standard for Fire Protection (ADAMS Accession No. ML18096A955). Attachment 1 of the 10 CFR 50.69 LAR lists several plant modifications that are credited in the FPRA risk estimates for the NFPA 805 LAR to meet the risk acceptance guidance of RG 1.174, Revision 3 (ADAMS Accession No. ML17317A256). Attachment S of the NFPA 805 LAR lists several implementation items (such as updating of the fire response procedures) that are also credited in the FPRA to meet the risk acceptance guidelines discussed in RG 1.174, Revision 3.

Because there is a potential for additional FPRA model changes to resolve requests for additional information (RAIs) associated with the staff determination of acceptability of the FPRA for approval of the Hatch adoption of NFPA 805 LAR that is currently under NRC staff review, address the following:

a) *Confirm that all the NFPA 805 plant modifications, implementation items, and FPRA model changes necessary to resolve questions associated with the NFPA-805 LAR review are complete, or alternatively, propose a mechanism to ensure that all these items are complete prior to the implementation of the 10 CFR 50.69 categorization process. This mechanism should also provide an explicit description of changes that will be made to the PRA model(s) and/or documentation to resolve the identified issues. An example would be a table of listed implementation items referenced in a license condition.*

SNC Response:

SNC is unable to provide this response until after SNC develops the response to remaining open items described in the May 28, 2019 NFPA-805 RAI response letter. The response to this RAI will be provided after SNC responds to remaining NFPA-805 RAIs.

b) *Alternatively to item (a) above, address the following:*

   i. *Provide detailed justification that the NFPA 805 modifications, implementations items and FPRA model changes necessary to resolve questions associated with the NFPA-805 LAR review have no impact the PRA models (i.e., IEPRA, FPRA and SPRA) used for the 10 CFR 50.69 application.*

   SNC Response:

   SNC is unable to provide this response until after SNC develops the response to remaining open items described in the May 28, 2019 NFPA-805 RAI response letter. The response to this RAI will be provided after SNC responds to remaining NFPA-805 RAIs.

   ii. *If any plant modifications, implementations items or FPRA model changes necessary to resolve questions associated with the NFPA-805 LAR review are determined to impact the PRA models (i.e., IEPRA, FPRA and SPRA) and documentation, address the following:*

1. *Provide explicit description for how they will be addressed in the PRA models that will be used for the 10CFR 50.69 categorization.*

   SNC Response:

   SNC is unable to provide this response until after SNC develops the response to remaining open items described in the May 28, 2019 NFPA-805 RAI response letter. The response to this RAI will be provided after SNC responds to remaining NFPA-805 RAIs.

2. *Confirm the status of the plant modifications and how the PRA models will reflect the as-built and as-operated plant at the time of implementation of the 10 CFR 50.69 process.*

   SNC Response:

   SNC is unable to provide this response until after SNC develops the response to remaining open items described in the May 28, 2019 NFPA-805 RAI response letter. The response to this RAI will be provided after SNC responds to remaining NFPA-805 RAIs.

3. *Provide detailed description and justification for any alternative PRA modeling that is not subject to the NRC staff review for the NFPA-805 application or the 10 CFR 50.69 application (e.g., subsequent removal of credit for modifications).*

   SNC Response:

   SNC is unable to provide this response until after SNC develops the response to remaining open items described in the May 28, 2019 NFPA-805 RAI response letter. The response to this RAI will be provided after SNC responds to remaining NFPA-805 RAIs.

iii. Provide any updated Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) estimates resulting from removal of credit for the NFPA-805 modifications and/or implementation items to confirm that the acceptance criteria for total CDF and LERF values provided in RG 1.174, Revision 3 (ADAMS Accession No. ML17317A256) remain met.

SNC Response:

SNC is unable to provide this response until after SNC develops the response to remaining open items described in the May 28, 2019 NFPA-805 RAI response letter. The response to this RAI will be provided after SNC responds to remaining NFPA-805 RAIs.

RAI 04 (APLA/APLB) – Process for Identification of Key Assumptions and Sources of Uncertainties

Paragraphs 50.69(c)(1)(i) and (ii) require a licensee's PRA to 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, "10 CFR 50.69 SSC Categorization Guideline," (ADAMS Accession No. ML052910035), specifies sensitivity studies to be conducted for each PRA model. The sensitivity studies are performed to ensure that assumptions associated with these uncertainty parameters (e.g., human error, common cause failure, and failure probabilities) do not mask the SSC(s) importance.*

LAR Section 4.1 identifies RG 1.174, Revision 2 (ADAMS Accession No. ML100910006), as an applicable regulatory requirement/criterion. RG 1.174 has been updated to Revision 3, dated January 2018 (ADAMS Accession No. ML17317A256). Regulatory Guide 1.174, Revision 3, cites NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (ADAMS Accession No. ML17062A466), as related guidance. In Section B of RG 1.174, Revision 3, the guidance acknowledges specific revisions of NUREG-1855 to include changes associated with expanding the discussion of uncertainties. LAR Section 3.2.7 states that the detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855, March 2009, Revision 0 (ADAMS Accession No. ML090970525) and Section 3.1.1 of EPRI Technical Report (TR)-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments". The NRC staff notes that for the IEPRA (includes internal flooding), FPRA, and SPRA models, only three, two, and five sources of uncertainty were identified, respectively.

NUREG-1855 has been updated to Revision 1 as of March 2017 (ADAMS Accession No. ML17062A466). The NRC staff notes that NUREG-1855, Revision 1, provides guidance in stages A through F for how to treat uncertainties associated with PRA models in RI decision-making. Revision 1 of NUREG-1855 cites EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainties."

Additionally, Section 3.3.2 of RG 1.200 Revision 2 defines key assumptions and sources of uncertainty as follows:

> *A **key assumption** is one that is made in response to a key source of model uncertainty in the knowledge that a different reasonable alternative assumption would produce different results, or an assumption that results in an approximation made for modeling convenience in the knowledge that a more detailed model would produce different results. For the base PRA, the term "different results" refers to a change in the risk profile (e.g., total CDF and total LERF, the set of initiating events and accident sequences that contribute most to CDF and to LERF) and the associated changes in insights derived from the changes in the risk profile. A "reasonable alternative" assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being challenged.*
>
> *A **key source of uncertainty** is one that is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an impact on the risk profile (e.g., total CDF and total LERF, the set of initiating events and accident sequences that contribute most to CDF and to LERF) such that it influences a decision being made using the PRA. Such an impact might occur, for example, by introducing a new functional accident sequence or a change to the overall CDF or LERF estimates significant enough to affect insights gained from the PRA.*

The NRC staff requests the following information to confirm the key assumptions and sources of uncertainty provided in Attachment 6 of the LAR were properly assessed from the base PRAs that have received peer reviews:

a) *A description of the process and the criteria used to identify, from the initial comprehensive list of uncertainties and assumptions for the base PRA model(s) (including those associated with plant specific features, modeling choices, and generic industry concerns), the application specific key assumptions and sources of uncertainties provided in LAR Attachment 6. Describe how the key assumptions and sources of uncertainty are determined consistent with the definitions in RG 1.200 Revision 2. The descriptions should be provided separately for internal hazard PRAs (including internal fire) and external hazard PRAs supporting this application.*

SNC Response:

The process used to identify base PRA internal events and internal flooding PRA model uncertainties and their impact for the LAR is described in PRA-BC-H-10-010, “Hatch PRA Quantification Calculation for Units 1 [and 2]”. The process followed was based on the guidance detailed in NUREG-1855, Revision 0, including consideration of the generic sources of model uncertainty provided in EPRI TR 1016737.

Internal Events and Internal Flooding PRA Assessment As Submitted. The Hatch internal events and internal flooding PRA assessment of uncertainties and assumptions in PRA-BC-H-10-010 started with a review, by PRA technical element, of assumptions made in the analysis for each technical element, and provided a characterization of whether each assumption is related to a source of uncertainty for the PRA. If a technical element assumption was judged to represent a potential source of model uncertainty (e.g., because it may introduce a conservative or non-conservative bias in results, or because the basis for a modeling assumption could not be directly identified as consistent with an industry consensus approach, or because the phenomena or nature of the event or failure mode being modeled is not completely understood) it was identified for additional consideration of overall PRA impact. The technical element review also considered the generic technical element-specific technical topics identified in EPRI-1016737 Appendix A, correlated with the supporting requirements in the ASME/ANS PRA Standard, to determine if any of these might be significant sources of uncertainty for the PRA requiring further evaluation. As such, this evaluation represents a comprehensive review of potentially important uncertainties and assumptions for the base internal events and internal flooding PRA model that captures those associated with plant specific features, modeling choices, and generic industry concerns. For some of the identified potential sources, sensitivity studies were performed for the base model to determine whether or not the source might be significant. The list of potentially important sources resulting from the base model assessment was then reviewed relative to the 50.69 application to determine potentially key sources for that application.

Internal Fire PRA Assessment As Submitted. The process used to identify base internal fire PRA model uncertainties and their impact for the LAR is described in H-RIE-FIREPRA-U00-015, Version 2.0, “Hatch Fire PRA Task 15, Uncertainty and Sensitivity Analysis”. The approach taken in H-RIE-FIREPRA-U00-015 reflects the guidance in NUREG/CR-6850 and in EPRI-1016737. The evaluation examines potential sources of uncertainty for each of the Fire PRA development tasks defined in NUREG/CR-6850, and also evaluates assumptions made in each of the Fire PRA development tasks to identify associated model uncertainties. Although this evaluation does not specifically refer to EPRI-1026511 and therefore does not specifically address each of the Fire PRA entries in Appendix B of that report, the considerations in H-RIE-FIREPRA-U00-015 are aligned with each of the Fire PRA tasks defined in NUREG/CR-6850 including the guidance in that report for evaluation of uncertainties in fire PRA, and are reasonably consistent with the sources of uncertainty identified in EPRI 1026511 Appendix B. Therefore, this evaluation represents a structured review of uncertainties and assumptions for the base fire PRA model that captures those associated with plant specific features, modeling choices, and generic industry concerns. The evaluation concluded that the treatment of the identified sources was consistent with NUREG/CR-6850 guidance but that for some sources, sensitivity studies should be performed for the base model to determine whether or not the source might be significant. The list of potentially important sources resulting from the base model assessment, including the sensitivity studies, was then reviewed relative to the 50.69 application to determine potentially key sources for that application.

Seismic PRA Assessment As Submitted. The process used to identify base seismic PRA model uncertainties and their impact for the LAR is described in H-RIE-SEIS-U00-001-002, Version 3.0, “Seismic Probabilistic Risk Assessment Quantification Report”. The approach taken in H-RIE-SEIS-U00-001-002 was to identify a set of modeling assumptions based on review of the seismic PRA results and knowledge of the model for which sensitivity studies should be performed. The list of potentially important sources resulting from the base model assessment, including the sensitivity studies, was reviewed relative to the 50.69 application to determine potentially key sources for that application.

The result of the above evaluations for the internal events/internal flooding, internal fire, and seismic PRA models were presented in the tables in Attachment 6 of the Hatch 50.69 LAR. Emphasis was given to those items where the disposition for the base hazard model noted that additional sensitivities might be needed to establish impact on the model. The selection process was based on the information available from the evaluations described above, with a focus on those base mode assumptions and potential sources of uncertainty judged to have the potential to significantly affect the relative risk importance results that are the focus of the 50.69 categorization process.

Subsequent to submittal of the LAR, the evaluations of sources of uncertainty for the Hatch PRA models have been updated, as a normal part of the PRA model maintenance process.

Internal Events and Internal Flooding PRA Assessment Update. The updated review of base internal events and internal flooding PRA model uncertainties and their impact is described in H-RIE-IEIF-U00-011, “Hatch Full Power Internal Events and Internal Flooding Uncertainty Analysis Notebook”, Version 1, February 2019. The process followed is consistent with the guidance detailed in NUREG-1855, Revision 1, including consideration of the implementation process in EPRI-1026511 and consideration of generic sources of model uncertainty provided in EPRI-1016737. The overall process is consistent with the stages defined in NUREG-1855 Revision 1, including determining that there are no significant issues of PRA model completeness relative to the 50.69

application and evaluating the impact of parametric uncertainty to confirm no significant impact on calculated mean values that would challenge meeting the RG 1.174 CDF and LERF acceptance criteria. Sources of model uncertainty and completeness have been evaluated and dispositioned relative to the 50.69 application. Based on the updated review of internal events and internal flood PRA uncertainties, the entries in Table 6-1 of the LAR are updated below to encompass the set of sources for the updated model. This table supersedes the version in the LAR. Note that Table 6-1 in the LAR included an entry discussing the modeling of drywell cooler standby fans as a potential source of uncertainty. After further review, this entry has been clarified to better characterize it relative to this application.

Internal Fire PRA Assessment Update. An updated review of base internal fire PRA model uncertainties and their impact is described in H-RIE-FIREPRA-ABAO-U00-015, “Hatch Fire PRA ABAO [as-built, as-operated] Uncertainty,” December 2018. The approach taken in the updated evaluation, which was performed for the as-built/as-operated Hatch fire PRA, supplements the assessment in H-RIE-FIREPRA-U00-015, Version 2.0 and reflects the guidance in NUREG/CR-6850 and in EPRI-1016737. As was the case for the evaluation performed for the LAR, the updated evaluation examines sources of uncertainty for each of the Fire PRA development tasks defined in NUREG/CR-6850. Although this evaluation does not specifically refer to EPRI-1026511 and therefore does not specifically address each of the Fire PRA entries in Appendix B of that report, the considerations in the updated evaluation are aligned with each of the Fire PRA tasks defined in NUREG/CR-6850 including the guidance in that report for evaluation of uncertainties in fire PRA, and are judged to be consistent with the sources of uncertainty identified in EPRI 1026511 Appendix B. Therefore, this evaluation represents a structured review of uncertainties and assumptions for the base fire PRA model that captures those associated with plant specific features, modeling choices, and generic industry concerns. The evaluation includes determining that there are no significant issues of PRA model completeness relative to the 50.69 application and evaluating the impact of parametric uncertainty to confirm no significant impact on calculated mean values that would challenge meeting the RG 1.174 CDF and LERF acceptance criteria. The sources of model uncertainty have been evaluated and dispositioned relative to the 50.69 application. For most of the NUREG/CR-6850 tasks and associated assumptions and potential sources of uncertainty, the approach taken was judged to be consistent with the guidance in NUREG/CR-6850, and for these assumptions and potential sources of uncertainty no application-specific source key assumption or source of uncertainty is defined, given that a consensus approach has been followed. Based on the updated review of internal fire PRA uncertainties, the entries in Table 6-3 of the LAR are updated below to encompass the set of sources for the updated model. This table supersedes the version in the LAR.

Seismic PRA Assessment Update. The updated review of base seismic PRA model uncertainties and their impact is described in RBA-19-001-H, “Identify and characterize the assumptions and sources of uncertainty in the Hatch SPRA”, June 2019. The process followed is consistent with the guidance detailed in NUREG-1855, Revision 1 including consideration of the implementation process in EPRI-1026511. Although the specific list of generic sources of model uncertainty in Appendix C of EPRI TR 1026511 has not been explicitly considered, the assessment documented in RBA-19-001-H is judged to be sufficiently comprehensive to address the recommended sources. The overall process is consistent with the stages defined in NUREG-1855 Revision 1, including determining that there are no significant issues of PRA model completeness relative to the 50.69 application and evaluating the impact of parametric uncertainty to confirm no significant impact on calculated mean values that would challenge meeting the RG 1.174 CDF and LERF acceptance criteria. Sources of model uncertainty and completeness have been evaluated and dispositioned relative to the 50.69 application. In this process, potentially important assumptions from the internal events PRA, on which the seismic PRA is built, were also considered.

The following candidate list of SPRA model uncertainties and associated assumptions was identified, and sensitivities were performed to determine their significance to the 50.69 application, as noted.

- Issue: Credit for portable FLEX equipment can have an impact on SPRA results. Although the results do not appear to be sensitive to assumptions regarding portable FLEX equipment reliability, the results are sensitive to crediting vs. not crediting portable FLEX.
    - This item is resolved for this application by removing credit for portable FLEX from the SPRA model, consistent with not crediting portable FLEX for this application in the internal events/internal flooding and internal fire PRA models.
- Issue: The use of a capacity-based screening criterion to determine which SSCs are directly modeled in the seismic PRA.
    - The capacity screening is more of a completeness uncertainty consideration than a model uncertainty, as it deals with the inclusion in or exclusion from the model of specific components with capacities above the screening value that are demonstrated to have only small impact on the results.
    - This item is resolved as discussed in the response to NRC RAI 07 regarding the Hatch 50.69 LAR, and a categorization process sensitivity as listed in Table 6-2 (Update) will be incorporated into the Hatch 50.69 process.
- Issue: Shared diesel generator 1B is assumed to be available 50% of the time to each unit. For some scenarios with unit 1 PSW the 1B DG would favor unit 1 and for random failures 1B DG would favor the unit that has the random failure.
    - A sensitivity was performed adjusting the likelihood of 1B DG being available to unit 1 depending on whether a PSW or random failure occurs. Cases were run varying the availability given PSW pump failure, availability given random failure, and accounting for other model impacts.
    - The results of the sensitivities indicated an impact of approximately 14% reduction from base case unit 1 seismic CDF (8% reduction for unit 2) and approximately 12% reduction in unit 1 seismic LERF (2% reduction for unit 2).
- Issue: Seismic failure of the Control Building and Reactor Building are conservatively assumed to result in core damage and LERF. However, the calculated seismic

capacities for these buildings are for failures less severe than collapse. This is a conservative assumption.

- The impact of using more realistic fragility values for each building on CDF and LERF results was examined, based on a review of FV values for RB and CB in the base model.

- The results of the review indicate insignificant (less than 1%) reductions in CDF, and less than 4% reductions in LERF for each building.

- Issue: Impact of slope stability failure due to the seismic event is conservatively modeled and is assumed to fail piping from the intake structure, which in turn fails cooling to the DGs and also fails the FLEX staging area. This is further assumed to result directly in LERF.

  - A sensitivity was performed that removes the global intake structure impact and replaces it with seismic hazard bin-specific impacts on the CC system.

  - The results of the sensitivity indicate no impact on seismic CDF and an approximately 8% reduction in seismic LERF.

- Issue: Block wall failure impacts are modeled since these walls can collapse in a seismic event and fail components located near the walls. For most of the block walls, it is assumed that 50% of the time the wall would collapse to either side. In cases where there is more stiffness, e.g., due to wall mounted items or wall is separating an elevator vestibule and stairwell, the collapse might favor one side.

  - A sensitivity was performed assuming that the affected walls only fail 25% of the time toward the structurally stiffer side.

  - The results of the sensitivity indicate only approximately 1% reductions in unit 1 seismic CDF and LERF and a reduction of approximately 6% unit 2 seismic CDF and less than 3% reduction in seismic LERF.

- Issue: During the walkdowns for IPEEE it was observed that not all the anchor bolts for several 600v switchgear components were visible or able to be confirmed to be tight enough. The switchgear meets the design basis. The SPRA fragility calculations assume that all anchor bolts will function consistent with other Control Building assumptions. This has not yet been verified by SPRA walkdown.

  - A sensitivity was performed in which the fragility for the 600v switch gear was recalculated crediting only bolts that were determined to be tight in the IPEEE.

  - The results of the sensitivity indicate an increase in seismic CDF or LERF of approximately 1% for unit 2, insignificant for unit 1.

- Issue: Modeling of reactor recirculation pump snubber failure. In the SPRA model, failure of the first (most limiting) reactor recirculation pump snubber is assumed to fail the entire recirculation pump support system and fail the entire pump assembly. Without a detailed finite element model (FEM), it is not known how the loads would transfer to remaining snubbers when the first snubber fails. It is possible that the next snubber would fail at a significantly higher fragility than the base value and have an impact on the PRA results.

- A sensitivity was performed in which the fragility of the “next” snubber was calculated assuming that the load redistribution would not significantly change the stress on the second snubber given that there is a large number of snubbers on the loop that the load would be distributed to. This issue only affects unit 1 because the unit 2 “first” and “next” snubber capacities are similar.

- The results of the sensitivity indicate a reduction in seismic CDF of approximately 4% and a reduction in seismic LERF of approximately 2%.

As noted above, for the items other than FLEX and SPRA capacity-based screening, the sensitivity studies indicate that the SPRA model is not overly sensitive to the identified issues. With regard to the 50.69 application, seismic importance is considered numerically as part of the integral assessment specified in the NEI 00-04 guidance. Since the Hatch seismic CDF is a small percentage of total plant CDF, and the Hatch seismic LERF is also a small percentage of total plant LERF, there is unlikely to be a significant impact on 50.69 categorization, and these items are not retained as key to the 50.69 application. That is, for the most significant impact of the sensitivities discussed above (shared DG 1B assumption), while a 14% reduction in seismic CDF could affect some seismic importances, those importances would be weighted as only a small percentage in the integral assessment for CDF. Similarly, for LERF, the 12% reduction would be weighted as only a small percentage in the integral assessment for LERF. Therefore, no additional application-specific consideration of these items is necessary as part of 50.69 categorization.

Based on the above review of seismic PRA uncertainties, the corresponding entries in Table 6-2 of the LAR have been dispositioned and are no longer applicable. A revised Table 6-2 for the updated model is provided below. This table supersedes the version in the LAR.

Table 6-1 (Update) Internal Events/Flooding PRA Model Sources of Uncertainty			
#	Assumption/Uncertainty	Model Sensitivity and Disposition	50.69 Impact
1	Impact of containment venting on core cooling system NPSH. Many BWR core cooling systems utilize the suppression pool as a water source. Venting of containment as a decay heat removal mechanism can substantially reduce NPSH, even lead to flashing of the pool. For the Hatch Unit 1 PRA, MAAP runs show that for medium or large LOCAs other means of containment heat removal the requirements for opening the Hardened Vent blow out plug are met. If all other containment heat removal methods fail the Hardened Vent will protect the drywell from overpressure failure. For the smallest of the MLOCA/largest of SLOCA break range (0.01ft2), conditions to operate the Hardened Vent also exist if no other means of heat removal are used. In these cases NPSH concerns may exist with the low pressure ECCS pumps. This is accounted for in the current model under gate EMERGENCYVENT with event NPSHLOSSPROB. Loss of NPSH following emergency venting is assumed to have a probability of 1E-2 for Unit 1.	With the assumed value for loss of NPSH, the impact of credit for injection systems after containment venting was judged to be a potential source of uncertainty for the Hatch PRA and for applications. The model will be re-structured to properly model the impact of containment pressure on low pressure injection systems.	Because the model is being revised to appropriately model the NPSH impact, it will no be longer a modeling assumption or significant source of uncertainty. The updated model will be used for 50.69 categorization.

Table 6-1 (Update) Internal Events/Flooding PRA Model Sources of Uncertainty			
#	Assumption/Uncertainty	Model Sensitivity and Disposition	50.69 Impact
2	Core cooling success following containment failure or venting through non-hard pipe vent paths; affects long term loss of decay heat removal sequences. The Hatch PRA assumes that if the containment is failed, the injection sources contained in the reactor building would fail due to environmental conditions, steam binding of pumps, or disruption of flow paths due to catastrophic containment failure. Injection is not credited after containment failure in the Level 1 PRA. In the Hatch Level 2 PRA, injection sources external to the reactor building are considered after containment failure. Of these, only condensate/ RHRSW are credited (fire protection and LPCI/CS from CST are not). Condensate/ RHRSW are credited in the Level 2 PRA after containment failure as they would not be impacted by harsh conditions in the reactor building.	No credit for injection sources in the reactor building after containment failure may represent a slight conservative bias.	The assumption may increase the importance of the containment boundary. However, the assumption and modeling are reasonable and not expected to have a significant impact on 50.69 categorization results. Therefore, no further application-specific consideration is warranted.

Table 6-1 (Update) Internal Events/Flooding PRA Model Sources of Uncertainty			
#	Assumption/Uncertainty	Model Sensitivity and Disposition	50.69 Impact
3	The RPS division is modeled in the PRA with exception of the RPS ATTS panels. These RPS ATTS panels are not modeled because failure of these panels would be in the safe condition. Surrogate common cause events are used instead of detailed modeling.	The RPS channel failure probabilities have little impact on the Hatch PRA risk metrics, indicating the addition of events to represent the ATTS panels would have little impact as well.	This is a completeness limitation relative to determination of the importances of the non-modeled ATTS panels, i.e., the PRA model cannot be used to directly determine the importances of these components. However, the 50.69 process allows for categorization of non-modeled components, so there is no application impact.
4	It is assumed that only one fan is available for each Drywell Cooler; standby fans are not modeled	Drywell coolers do not contribute significantly to the PRA results. Therefore, the model is insensitive to this modeling assumption.	This is a completeness limitation relative to determination of importances of the non-modeled drywell coolers standby fans, i.e., the PRA model cannot be used to directly determine the importances of these components. However, the 50.69 process allows for categorization of non-modeled components, so there is no application impact.

Table 6-1 (Update) Internal Events/Flooding PRA Model Sources of Uncertainty			
#	Assumption/Uncertainty	Model Sensitivity and Disposition	50.69 Impact
5	Loss of all intake structure ventilation system fans during the PRA mission time is assumed to cause failure of the PSW and RHRSW pumps unless the operators act to establish alternate cooling within 8 hours. A single basic event is used to account for the probability that ventilation is needed, that the fans fail, and that the operators will fail to establish alternate cooling. The probability for this basic event is assigned a screening value of 0.5. Detailed modeling for this event has not been performed, and the model does not explicitly account for these factors, instead treating this as an operator response with a screening HEP.	A sensitivity has been performed to determine whether this action as modeled is significant to the results. The results of the sensitivity showed that lowering the probability by an order of magnitude did not have a significant impact on basic event importances in the internal events PRA.	This is not a key assumption for the 50.69 application. Based on the sensitivity evaluation performed, the assumption and modeling are not expected to have a significant impact on 50.69 categorization results. Therefore, no further application-specific consideration is warranted.
6	The mission time for SLCS is assumed to be 1 hour. The actual time required to inject the entire contents of the SLCS storage tank is between 30 and 90 minutes, depending upon the amount of solution in the tank.	The individual SLC pump failure to run events have little impact on the Hatch PRA risk metrics.	This assumption has no significant impact on the PRA results and will not significantly affect 50.69 categorization. Therefore, no further application-specific consideration is warranted.

Table 6-1 (Update) Internal Events/Flooding PRA Model Sources of Uncertainty			
#	Assumption/Uncertainty	Model Sensitivity and Disposition	50.69 Impact
7	Operator Action assumptions:		
	a. Time to Align Isolated EHC Cooler: It is assumed that the operators have 2 hours to recover the heat exchanger. This is an engineering judgment that based on the light load on the EHC system due to the turbine generator being off-line.	a. The PRA model is insensitive to this assumption.	a. The PRA results show that the EHC cooler assumption has no significant impact on the PRA results and will not significantly affect 50.69 categorization. Therefore, no further application-specific consideration is warranted.
	b. Alignment of RHRSW for Injection: It is assumed that the operators clearly understand that they cannot recover high head injection, and would act on the initial cue of inability to restore and maintain reactor water level, so they would not delay going to ALC.	b. The HRA uses a time window of 40 minutes. However, if operators believe that they can recover high head injection, they may delay going to ALC until the RWL drops to -155" and then would only have 6 minutes to do the SW/RHR cross-tie, which may not be sufficient.	b. The PRA results, and the sensitivity performed for the RHRSW alignment action, show that this assumption has a potential to impact the PRA results. Therefore, it is retained as a potential key uncertainty for the 50.69 application. However, the NEI 00-04 categorization process includes 5th and 95th percentile HEP sensitivities, which will address this source. No additional sensitivity is needed beyond the existing NEI 00-04 sensitivity.

Table 6-2 (Update) Seismic PRA Model Sources of Model Uncertainty			
#	Assumption/Uncertainty	Model Sensitivity and Disposition	50.69 Impact
1	Seismic Equipment List Screening A capacity based screening criterion is used to determine which SSCs are directly modeled in the seismic PRA.	As discussed in the response to NRC RAI 07 regarding the Hatch 50.69 LAR, the usage of a screening level does not impact relative importances. The justification for the conclusion is: • Although an SSC may have a fragility above the screening value, these fragilities are usually conservative. Once the SSC fragility is evaluated to be above the screening value, no further refinement of the fragility is performed although conservatisms may still be present. • Many of the screened components would not lead directly to core damage or large early release; other failures would also be needed to result in core damage or large early release; the combination of failures needed to result in core damage or large early release would result in a contribution lower than the screening level. • When SSCs were judged to be important to CDF and LERF, they were included in the logic model even though their fragility was greater than the screening level. For instance, structures were included, as were components associated with the diesel generators, switchgear, and the instrumentation and control power.	To address this potential source of uncertainty in the 50.69 process, a review of SSCs with fragilities greater than the screening level will be performed to identify any seismic “singletons.” That is, any SSC (or correlated group of SSCs) that could lead directly to core damage or large early release would be identified, and addressed during the categorization process for the applicable system being categorized, to be consistent with NEI 00-04.

Table 6-2 (Update) Seismic PRA Model Sources of Model Uncertainty			
#	Assumption/Uncertainty	Model Sensitivity and Disposition	50.69 Impact
2	Modeling of Portable FLEX Equipment Unavailabilities and failure probabilities for the portable FLEX equipment were based on those for similar equipment in the IE PRA.	There is uncertainty associated with actual performance history and associated human error probabilities for portable FLEX equipment. A sensitivity study on a previous version of the model demonstrated that credit for portable FLEX has a significant impact on CDF results, but that the assumed unavailability values do not have an impact.	Credit for portable FLEX equipment will not be taken in the SPRA model used for 50.69 categorization, consistent with the approach for the other hazard PRA models. Therefore, there is no impact on the application.

Table 6-3 (Update) Internal Fire PRA Model Sources of Model Uncertainty			
#	Assumption/Uncertainty	Model Sensitivity and Disposition	50.69 Impact
1	Task 3: Fire PRA Cable Selection (CS): For the As-Built As-Operated model, existing cable routing was used for a small number of cables (6 cables) that are expected to be re-routed in the future, but that route is unknown at this time. Therefore, for the current model, these impact of the failure of these cables may change in certain scenarios after the cables are re-routed.	The model currently reflects the as-built and as-operated plant, and includes operator actions to recover from the affected cable failures. Once the cables are re-routed, the model will be updated to reflect the new routing and potentially eliminate the need for the associated recovery actions. Therefore, the modeling is appropriate and does not represent a source of uncertainty beyond the Fire PRA modeling sources already evaluated.	The model reflects an appropriate treatment given the current configuration of the plant. When the plant is modified and the model is updated, any impact on 50.69 component categorization will be identified as part of the 50.69 performance monitoring and feedback process.
2	Task 11b: Main Control Room analysis: The HNP Main Control Boards (MCB) are open back panels. The FPRA used fire modeling consistent with NUREG/CR-6850 App. L (updated to the latest NUREG-2169 fire ignition frequencies and NUREG-2178 heat release rates (HRR)). The panels behind the MCBs were concluded to not fully meet the definition of a MCB per FAQ 14-0008; therefore, App. L was not used for these panels.	Some of these panels are significant contributors and are treated consistent with other Bin 15 cabinets. The MCR abandonment time report was updated to include a fire model that includes a 98% transient HRR and the open back MCB HRR. The model estimated ~13% reduction in MCR abandonment time from the baseline case when postulated damage/ignition of the MCB cables at t = 0. The FPRA incorporates the bounding results in the scenarios. Sensitivity evaluation shows that the overall FPRA is not sensitive to incorporation of the bounding model (e.g., < 1% increase in risk).	The impact of this uncertainty is on time available for operator action (MCR abandonment and subsequent shutdown actions). Given the limited impact from the sensitivity performed, this source of uncertainty is adequately addressed by the existing NEI 00-04 required sensitivity to evaluate human error probabilities at their 5^{th} and 95^{th} percentile values. No further impact evaluation is needed.

Table 6-3 (Update) Internal Fire PRA Model Sources of Model Uncertainty			
#	Assumption/Uncertainty	Model Sensitivity and Disposition	50.69 Impact
3	Task 12: HRA: A minimum floor value is assigned to Joint human error probabilities (JHEP).	Applying a minimum joint HEP may inaccurately skew the results by artificially increasing the risk due to human actions. However, sensitivities performed for the Hatch FPRA show that results are not overly sensitive to the use of a JHEP floor value. The FPRA is being updated to use a JHEP floor value of 1E-5.	This source of uncertainty is addressed by the existing NEI 00-04 required sensitivity to evaluate human error probabilities at their 5th and 95th percentile values. No further impact evaluation is needed.

Table 6-3 (Update) Internal Fire PRA Model Sources of Model Uncertainty			
#	Assumption/Uncertainty	Model Sensitivity and Disposition	50.69 Impact
4	Several systems and components in the FPIE are not credited in the FPRA. These have been classified as “Unknown Location” (UNL) components and have not had circuit analysis performed and cable routing to physical analysis unit (PAU) traced.	As noted in the response to NRC RAI 05b regarding the Hatch 50.69 LAR, the UNL items in the Hatch Fire PRA are primarily the components associated with the main condenser, the condensate and feedwater systems, and the circulating water system. Unavailability of these non-safety systems for pressure relief and low-pressure injection causes the safety systems to be more important. Since power to the UNL components is primarily from the non-safety electrical system, they are lost during LOSP events. Because the main steam isolation valves automatically close on low vessel pressure, the spurious SRV opening would also fail most of the UNL components even if they had detailed circuit analysis and were included in the Fire PRA model. Thus, for the PAUs with the highest CDF and LERF, there is a high likelihood that there is a loss of offsite power or spurious SRV opening, and in turn, the items on the UNL would not be available regardless of having detailed circuit analysis. Therefore, actual impact of the UNL is limited given the limited number of UNL components and their plant function.	Assuming failure of non-safety components increases the importance of safety related components, and would have a slight conservative bias (greater tendency to HSS determination) in the 50.69 categorization application. No further impact evaluation is needed.

b) *Provide a summary list of the key assumptions and sources of uncertainty that have been identified for the application.*

SNC Response:

The set of key assumptions and sources of uncertainty that were originally identified for the Hatch 50.69 License Amendment Request are as listed in Tables 6-1, 6-2, and 6-3 of LAR Attachment 6. These were selected based on review of the assumptions and sources of uncertainty identified in the PRA model notebooks described in the response to Part a, by considering the potential model impacts noted in those notebooks relative to the 50.69 risk importance application.

As noted in the response to Part a, subsequent to submittal of the LAR, the evaluations of sources of uncertainty for the Hatch PRA models have been updated. The updated sets of key assumptions and sources of uncertainty in the models for the 50.69 application are as noted in the updated versions of Tables 6-1, 6-2, and 6-3 in the response to Part a.

c) *Confirm that the process is consistent with NUREG-1855, Revision 1, or other NRC-accepted methods (e.g., NUREG-1855, Revision 0). If deviating from the current guidance provided in NUREG-1855, Revision 1, provide a basis to justify the use of the method(s) in the 10 CFR 50.69 categorization process (e.g., exclusion/consideration of EPRI TR-1026511).*

SNC Response:

The discussion in the response to Part a demonstrates that the process for identification and evaluation of assumptions and sources of uncertainty in the Hatch PRAs for the 50.69 application is consistent with NUREG 1855, Revision 1 for the internal events including internal flooding PRA, and with NUREG-1855, Revision 0 and elements of NUREG-1855 Revision 1 for the seismic PRA. The evaluation of assumptions and sources of uncertainty in the Hatch internal fire PRA is consistent with the NRC-accepted guidance in NUREG/CR-6850, as discussed in Part a.

RAI 05 (APLA) – Dispositions of Key Assumptions and Sources of Uncertainties

Paragraph 50.69(c)(1)(i) of 10 CFR requires the licensee to consider the results and insights from the PRA during categorization. The guidance in Section 5 of NEI 00-04 specifies sensitivity studies to be conducted for each PRA model. 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.

LAR Section 3.2.7 states that "a few system specific sensitivity analyses may be required to address Hatch model specific assumptions or sources of uncertainty." Multiple dispositions provided in LAR Attachment 6, Table 6-3 appear to indicate that no key assumptions or sources of uncertainty were identified that will require a sensitivity study consistent with Section 5 of the NEI 00-04 guidance for the 10 CFR 50.69 categorization process.

The NRC staff observes that modelling conservatisms can mask the importance measures of other SSCs. Considering these observations, address the following:

a) *For any additional key assumptions/sources of uncertainty identified as a result of RAI 04 response, discuss how each identified key assumption and uncertainty will be dispositioned in the categorization process. The discussion should clarify whether the licensee is following the guidance in Section 5 of NEI 00-04 by performing sensitivity analysis or other accepted guidance such as NUREG-1855 Stages A and F. The summaries and descriptions should be provided separately for the identified key assumptions and uncertainties related to internal hazard PRAs (including internal fire) and those related to external hazard PRAs supporting this application.*

SNC Response:

The development of the response to RAI 04 was generally in accordance with the guidance of NUREG-1855 Stages A and F. In addition, one additional 50.69 categorization process step, consistent with the guidance in NEI 00-04 has been identified as a result of this assessment, as described below.

In reviewing the assumptions and sources of uncertainty relative to the 50.69 application for the internal events PRA model, several potential sources were identified as potentially key to this application. For each of these, the assessment and disposition relative to 50.69 application is listed in the last column of Table 6-1 (Update) in the RAI 04 response.

- In the case of item 1 in Table 6-1 (Update), the PRA is being changed to resolve the uncertainty.
- Item 2 is related to item 1. The current modeling assumption was judged to have a slight conservative bias that should not affect 50.69 categorization results. With the resolution of item 1, the impact should be further reduced.
- Items 3 and 4 are assessed as completeness issues that can be accommodated by the 50.69 categorization process.
- For Item 5, a sensitivity was performed that determined that the importance results are not overly sensitive to the probability assigned to the basic event representing the loss of intake structure ventilation scenario.
- Item 6 was assessed as not significant to the application due to its low relative importance.
- Item 7 is addressed through the NEI 00-04 required HEP sensitivity evaluation performed as part of the categorization process.

In reviewing the assumptions and sources of uncertainty relative to the 50.69 application for the seismic PRA model, several potential sources were identified as potentially key to this application. For each of these, the assessment and disposition relative to 50.69 application is listed in the last column of Table 6-2 (Update) in the RAI 04 response.

- Item 1 is addressed by incorporating into the Hatch 50.69 categorization process an additional step to review SSCs with fragilities greater than the SPRA screening level, using an approach already implemented for the Plant Vogtle 50.69 program.
- Item 2 is resolved by removing the impact of portable FLEX equipment from the SPRA model used in the 50.69 categorization process.

In reviewing the assumptions and sources of uncertainty relative to the 50.69 application for the internal fire PRA model, several potential sources were identified as potentially key to this application. For each of these, the assessment and disposition relative to 50.69 application is listed in the last column of Table 6-3 (Update) in the RAI 04 response.

- Item 1 is related to ensuring that the PRA reflects the as-built/as-operated plant, and the fact that certain plant changes are anticipated. This is addressed by following the PRA maintenance process and the 50.69 performance monitoring and feedback process.
- For Item 2 a sensitivity evaluation was performed that indicated the model is not overly sensitive to the approach used, such that the $5^{th}/95^{th}$ percentile HEP sensitivity already specified in the NEI 00-04 categorization process will address any remaining uncertainty.
- Item 3 is addressed through the existing NEI 00-04 $5^{th}/95^{th}$ percentile HEP sensitivity.
- The categorization impact of Item 4 has been addressed in the response to RAI 5b as not requiring further evaluation as part of the 50.69 categorization process.

b) *The disposition for the first item of LAR Attachment 6, Table 6-3 states in part that the uncertainty associated with untraced secondary-side cables was addressed for the FPRA using a sensitivity study to assess the assumption that secondary-side systems fail in all fires. This sensitivity study and its results are not discussed in either the NFPA 805 LAR or the 10 CFR 50.69 LAR. Accordingly, it is not clear that this sensitivity study (i.e., the assumption that all secondary-side systems fail in a fire) performed for the FPRA is applicable to 10 CFR 50.69 categorization. Considering these observations:*

i. *Provide the quantitative results of the sensitivity study and/or justification to support the conclusion that the uncertainty associated with untraced secondary-side cables has no impact on the 10 CFR 50.69 categorization process. Include summary of the systems or components with untraced cables and explain how their functions are assumed to be impacted in the baseline fire PRA and sensitivity study case. Include in the justification the following: (1) a description of the sensitivity study that was performed for the FPRA, (2) an explanation of how it considered the potential to mask or skew the importance of certain SSCs, and (3) an explanation of how the sensitivity study bounds the source of uncertainty being addressed.*

SNC Response:

As documented in H-RIE-FIREPRA-U00-015, one of the quantitative sensitivity runs performed in support of the Hatch Fire PRA development was on the “untraced” cables, referred to in the calculations as ‘unknown location’ or UNL components.

Typically, a Fire PRA model does not include all credited equipment in the IEPRA. In these cases, circuit analysis is not always performed. The sensitivity documented in Section 4.5.1 of H-RIE-FIREPRA-U00-015 shows that the Fire PRA risk would decrease if these components were credited.

However, it should be noted that this sensitivity does not take into consideration the potential for scenario-specific fire-induced impacts of these components. The fire-induced impact may be due to direct fire damage (cables are directly failed in a given fire scenario) or an indirect fire damage (a parent system failure due to fire).

In the case of the Hatch Fire PRA, two of the main risk contributors are loss of offsite power (LOSP) caused by fire-induced loss of the startup transformers or the high voltage switchyard circuit breakers and fire-induced LOCA caused by spurious opening of the SRVs. The details of the Fire PRA risk results are documented in H-RIE-FIREPRA-ABAO-U00-014. A review of this calculation shows that the high-risk fire areas include: Main Control Room, Control Building Corridors, Control Building Working Floor, Switchgear Rooms, and Cable Spreading Rooms.

The UNL components in the Hatch Fire PRA are associated with specific systems. These systems contain a subset of functions that contain UNL components and these are listed in Table 3.1-2 of H-RIE-FIREPRA-U00-002. The functions that do have UNL components for these systems and their impacts to CDF are shown in the table below from H-RIE-FIREPRA-U00-002.

Table 5b.1: System Functions with UNL Components

System	System Description	Function	Comments
B21	Nuclear Boiler System	Automatic initiation of ARI	Minimal impact to CDF.
B21	Nuclear Boiler System	Decay Heat Removal using steam line drains	Minimal impact to CDF.
C11	Control Rod Drive Hydraulic System	CRD Injection	Minimal impact to CDF.
C71	Reactor Protection System	Failure of Drywell and Torus Temp Control (NON ATWS) using 2-inch Emergency Vent path from Suppression Pool	Minimal impact to CDF.
E11	Residual Heat Removal System	Decay Heat Removal using RHR Torus Suction air operated valves (motor operated valves are credited)	Minimal impact to CDF.
G31	Reactor Water Cleanup System	RWCU in Blowdown Mode for Heat Removal Support	Minimal impact to CDF.

System	System Description	Function	Comments
N11	Turbine Valves	Condensate and Feedwater Injection	Minimal impact to CDF. Impacted by loss of IA in many fire zones.
N71	Circulating Water System	Alternate PSW discharge path.	Minimal impact to CDF. Normal PSY path is credited.
N71	Circulating Water System	Main Condenser Heat Removal	Not available for LOSP scenarios.
OSP	Offsite Power	OSP Support to various systems (other sources credited)	Minimal impact to CDF. OSP source redundancy remains.
P42	Reactor Building Closed Cooling Water	CRD Injection and RWCU for Decay Heat Removal.	Minimal impact to CDF.
P70	Drywell Pneumatic	SRV Depressurization	Minimal impact to CDF. Depressurization capability is maintained.
X43	FIRE PUMP HOUSE	Fire Water Injection to the RPV (31EO-EOP-110-1).	Minimal impact to CDF.

The UNL items in the Hatch Fire PRA are primarily the components associated with the main condenser, the condensate and feedwater systems, and the circulating water system. Unavailability of these non-safety systems for pressure relief and low-pressure injection causes the safety systems to be more important. Thus, the systems that would be categorized under the 50.69 process will appear somewhat higher in risk than if the UNL components were credited. Since power to the UNL components is primarily from the non-safety electrical system, they are lost during LOSP events. Because the main steam isolation valves automatically close on low vessel pressure, the spurious SRV opening would also fail most of the UNL components even if they had detailed circuit analysis and were included in the Fire PRA model.

The UNL items account for approximately a 26% increase in CDF for Unit 1 and 24% for Unit 2. For both Unit 1 and 2, the top 10 Physical Analysis Units (PAUs) make up approximately 90% of the CDF and LERF. As stated previously, for the PAUs with the highest CDF and LERF, there is a high likelihood that there is a loss of offsite power or spurious SRV opening, and in turn, the items on the UNL would not be available regardless of having detailed circuit analysis. Therefore, the treatment of the UNL items relative to the importance of the modeled SSCs is considered to be bounded given the limited number of UNL components and their plant function, and there will be minimal impact on the 10 CFR 50.69 categorization process.

OR

ii. If the uncertainty addressed in part (i) above cannot be justified to have no adverse impact on the 10 CFR 50.69 categorization process, then propose a mechanism that ensures that a sensitivity study is performed during the 10 CFR 50.69 categorization process. Include a brief discussion of the sensitivity study proposed and how it addresses the uncertainty associated with untraced secondary-side cables. Include in the justification the following: (1) a description of the sensitivity study that is proposed, (2) an explanation of how it considers the potential to mask or skew the importance of certain SSCs and (3) an explanation of how the sensitivity study bounds the uncertainty associated with untraced secondary-side cables. An example would be a table of listed implementation items referenced in a license condition.

SNC Response:

This is not applicable due to the response in 5(b)(i).

OR

iii. Propose a mechanism that eliminates the uncertainty associated with the untraced secondary-side cables. This mechanism should also provide an explicit description of changes that will be made to the PRA model(s) and/or documentation to resolve this issue. If these changes are determined to involve a PRA upgrade, the mechanism should include a focused-scope peer review and require resolution of all generated finding-level F&Os from the peer review prior to implementation of the 10 CFR 50.69 categorization process. An example would be a table of listed implementation items referenced in a license condition.

SNC Response:

This is not applicable due to the response in 5(b)(i).

RAI 06 (APLA/APLB) - Addition of FLEX to the PRA Model

The NRC memorandum dated May 30, 2017, "Assessment of the NEI 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 differences between NEI 16-06 guidance and the guidance in RG 1.200 Revision 2 for incorporating diverse and flexible (FLEX) coping strategies and equipment into a PRA model in support of risk-informed decision making. LAR Attachment 6, LAR Table 6-2 indicates that FLEX is credited in the SPRA and does not address whether it is credited in the IEPRA or FPRA. For the NRC staff to determine the acceptability of incorporation of FLEX equipment into the PRA model(s) provide the following:

a) Confirm whether FLEX equipment and associated operator actions have been credited in the IEPRA, FPRA, and/or the SPRA.

SNC Response:

For the purposes of the 50.69 application, portable FLEX equipment stored in the FLEX storage dome and the associated operator actions will not be credited in the Hatch IEPRA, FPRA, and/or SPRA models. However, as part of the plant modifications associated with FLEX implementation, some additional equipment has been installed as permanent plant equipment. In addition, the plant procedures have been revised to include these plant modifications. The following responses are provided with the consideration that portable FLEX equipment stored in the FLEX storage dome and the associated operator actions will not be credited in the Hatch IEPRA, FPRA, and/or SPRA models used for 10 CFR 50.69 categorization.

b) *If FLEX equipment or operator actions have been credited in the PRA, address the following, separately for IEPRA (includes internal flooding), FPRA, SPRA, and external hazards screening as appropriate:*

i. *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. Include discussion of whether the credited FLEX equipment is portable or permanently installed equipment.*

SNC Response:

The modifications to permanent plant equipment that are being credited in the Plant Hatch IEPRA, FPRA, and/or the SPRA models are associated with new panels providing power to critical instrumentation cabinets, manual transfer switches, and modifications to add a backup air supply to the hardened containment vent system (HCVS), which required installing air accumulators, manual valves, and check valves.

ii. *Discuss whether the credited equipment (regardless of whether it is portable or permanently-installed) are similar to other plant equipment (i.e. SSCs with sufficient plant-specific or generic industry data) and whether the credited operators actions are similar to other operator actions evaluated using approaches consistent with the endorsed ASME/ANS RA-Sa-2009 PRA standard.*

SNC Response:

This equipment (that is, instrument panels, inverters, air accumulators, manual valves, check valves, etc.) is very similar to other plant equipment and has sufficient plant-specific or industry generic data. Uncertainties associated with the parameter values are in accordance with the ASME/ANS PRA Standard as endorsed by RG 1.200 Revision 2.

While the station blackout procedure has been modified to provide a link to the FLEX procedures, this link is not being credited in the current Plant Hatch IEPRA, FPRA, and/or SPRA models. No FLEX procedures requiring actions outside of the main reactor buildings or control building are credited in these PRA models. The credited operator actions have been peer reviewed and are similar to other operator actions evaluated using approaches consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200 Revision 2.

iii. *If any credited FLEX equipment is dissimilar to other plant equipment credited in the PRA (i.e. SSCs with sufficient plant-specific or generic industry data), discuss the data and failure probabilities used to support the modeling and provide the rationale for using the chosen data. Discuss whether the uncertainties associated with the parameter values are in accordance with the ASME/ANS PRA Standard as endorsed by RG 1.200 Revision 2.*

SNC Response:

This is not applicable as the installed equipment has been judged to be very similar to other plant equipment.

iv. *If any operator actions related to FLEX equipment are evaluated using approaches that are not consistent with the endorsed ASME/ANS RA-Sa-2009 PRA Standard (e.g., using surrogates), discuss the methodology used to assess human error probabilities for these operator actions. The discussion should include:*

1. *A summary of how 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 the ASME/ANS RA-Sa-2009 PRA Standard were evaluated.*

2. *Whether maintenance procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and if the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of the ASME/ANS RA-Sa-2009 PRA Standard.*

3. *If the procedures governing the initiation or entry into mitigating strategies are ambiguous, vague, or not explicit, a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.*

SNC Response:

None of the credited operator actions related to this added equipment are evaluated using approaches that are not consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200 Revision 2.

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

Provide 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, and (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences;

SNC Response:

For the SPRA model, the model changes incorporating permanent plant modifications associated with FLEX implementation have been peer reviewed as part of the SPRA model development project.

The implementation of permanent plant modifications associated with FLEX into the Internal Events PRA and Fire PRA models was performed consistent with the PRA methods and modeling that had already been peer reviewed within the existing PRA model maintenance framework. The ASME/ANS PRA Standard defines a PRA upgrade as a new methodology, or a change in scope or change in capability that impacts the significant accident sequences or the significant accident progression sequences. PRA maintenance is defined as changes within the framework of an existing model structure. The modeling of the FLEX modifications is not a new methodology since the methods used to model FLEX modifications are no different than the methods used to model non-FLEX modifications. It is also not a change of scope of the model, i.e., the equipment, dependencies, and types of accident sequences remain the same. It is also not a change in PRA model capability, i.e., the peer reviewed PRA model can still evaluate the risk associated with LOSP and station blackout. Thus, the modeling of FLEX modifications is a change implemented within the framework of the existing peer reviewed PRA model structure. The framework of the model remains essentially the same, and the High Level and Supporting Requirements in the PRA Standard for the Technical Elements associated with modeling of FLEX modifications (e.g., those within the Accident Sequence Analysis, Data Analysis, Human Reliability Analysis, and Quantification technical elements) will continue to be Met or Not Met regardless of implementation of the change from FLEX modifications. Although the implementation of permanent plant modifications associated with FLEX will affect the ordering of the accident sequences, the overall accident sequences are not significantly changed and does not result in significantly different risk insights. Given the above reasonings, the implementation of permanent plant modifications associated with FLEX in the Internal Events PRA and Fire PRA models does not constitute a PRA upgrade and does not require a focused-scope peer review.

d) *LAR Attachment 6 Table 6-2 states that the uncertainty associated with credit taken for FLEX implementation in the SPRA will be addressed by the human reliability analysis (HRA) sensitivity study performed as part of 10 CFR 50.69 categorization. The HRA sensitivity study performed as part of the categorization process increases all human error probabilities (HEPs) to their 95th percentile value. The disposition also states that the results of a sensitivity study, in which FLEX was assumed to completely fail, showed a significant increase in seismic CDF (41%) and LERF (57%) demonstrating that credit for FLEX is important to seismic risk. The NRC staff notes that the HRA methodologies used to calculate HEP values for FLEX actions, which may occur outside the main reactor buildings and may not be part of a normal or emergency operating procedure, should be consistent with technical elements of the NRC endorsed ASME/ANS PRA Standard (e.g., consideration of environmental conditions). Accordingly, it is not clear to the NRC staff that a sensitivity study using the 95th percentile value for the failure probability of FLEX operator actions is sufficient to address the uncertainty associated with FLEX actions and the lack of industry failure rate information for FLEX equipment.*

The disposition in Attachment 6 of the LAR for the above key source of uncertainty also states in part, the treatment of this uncertainty has minimal impact on the application because the seismic risk is small compared to the overall risk. The NRC staff notes that for 10 CFR

50.69 categorization the uncertainty associated with crediting FLEX equipment and actions impacting the seismic importance of SSCs and could skew the integrated importance of certain SSCs.

Considering these observations:

i. *Provide justifications that the HEP 95th percentile value sensitivity study is sufficient to address the uncertainty associated with crediting FLEX equipment and operator actions in the SPRA.*

SNC Response:

As discussed above, FLEX operator actions outside of the main reactor buildings or control building are not being credited in the SPRA for this LAR. Therefore, the HEP 95th percentile value sensitivity study is sufficient to address the uncertainty in operator actions.

ii. *If the HEP 95th percentile value sensitivity study cannot be justified to be sufficient to address the uncertainty associated with crediting FLEX equipment and operator actions in the SPRA in response to part (iii) above, then propose a mechanism to ensure that a separate acceptable sensitivity study is performed as part of 10 CFR 50.69 categorization to address the use of FLEX equipment and operator actions in the SPRA.*

SNC Response:

This is not applicable due to the response in 6.d.i.

RAI 07 (APLB) – Seismic PRA Screened SSCs

Section 5.1 of NEI 00-04 provides guidance on the use of importance measures for identifying the "candidate safety significance" of components during the categorization process. Based on the information provided in the LAR, the NRC staff was unable to determine whether the potential use of capacity-based screening level in the licensee's SPRA is consistent with the guidance for developing importance measure to identify candidate safety significance.

Describe whether a capacity-based screening level is used in the SPRA and how the potential use of the screening level maintains consistency with the importance measure criteria in NEI 00-04 or justify any deviations from the guidance by using the selected screening level. This justification may include demonstration of the impact of the selected screening level in the SPRA on the importance measure criteria and the categorization of SSCs.

SNC Response:

SNC uses a screening level in the SPRA and the usage of such a level does not impact the relative importance. Therefore, use of the screening level maintains consistency with the importance measure criteria in NEI 00-04. The justification for the conclusion is:

- Although an SSC may have a fragility above the screening value, these fragilities are usually conservative. Once the SSC fragility is evaluated to be above the screening value, no further refinement of the fragility is performed although conservatisms may still be present.

- Many of the screened components would not lead directly to core damage or large early release; other failures would also be needed to result in core damage or large early release; the combination of failures needed to result in core damage or large early release would result in a contribution lower than the screening level.

- When SSCs were judged to be important to CDF and LERF, they were included in the logic model even though their fragility was greater than the screening level. For instance, structures were included, as were components associated with the diesel generators, switchgear, and the instrumentation and control power.

In addition, a review of SSCs with fragilities greater than the screening level would be performed to identify any seismic “singletons”, similar to the Vogtle 50.69 process. That is, any SSC (or correlated group of SSCs) that could lead directly to core damage or large early release would be identified during the categorization process, to be consistent with NEI 00-04.

RAI 08 (APLB) – Use of Addendum B of the PRA Standard (2013)

Paragraph 50.69(c)(1)(i) of 10 CFR requires the PRA to be of sufficient quality and be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

Section 3.2.3 of the LAR indicates that the SPRA model was peer reviewed using the requirements in Addendum B of the PRA Standard (ASME/ANS RA-Sb–2013, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications”), which has not been endorsed by the NRC. LAR Section 3.2.3 also references discussion in the Vogtle 10 CFR 50.69 LAR (ADAMS Accession No. ML17173A875), as supplemented, justifying use of Addendum B based on an assessment of the differences between Addendum A and B Supporting Requirements (SRs). That assessment included evaluation of the Vogtle SPRA to Addendum A for SRs identified to be different from or not encompassed by the requirements in the Addendum B SRs. The licensee indicates that the SPRA peer review based on Addendum B can be justified using the Vogtle experience.

Section 3.3.1.1 of the Vogtle 10 CFR 50.69 Safety Evaluation (SE) (ADAMS Accession No ML18180A062) accepted the use of the 2013 PRA Standard based on (1) the discussion provided in the Vogtle LAR, (2) Vogtle’s comparison of Addendum B to Addendum A SRs in a report titled “Response to Supplemental Information Needed for Acceptance of Systematic Risk-Informed Assessment of Debris Technical Report (ADAMS Accession No. ML17192A245); and (3) a response to a request for additional information clarifying the acceptability of a practice used in the Vogtle SPRA associated with SR SFR-C6. The assessment of debris report defined four comparison categories: (1) “Addendum B Equates to Addendum A”, (2) Addendum B Envelopes Addendum A,” (3) “Vogtle Conforms to Addendum A,” and (4) “Vogtle Conforms to Accepted Current Practices.” The NRC staff notes that the first two comparison categories concern generic resolutions and, therefore, are expected to apply to the Hatch SPRA, but the remaining two comparison categories (i.e., “Vogtle Conforms to Addendum A, and “Vogtle Conforms to Accepted Current Practices”) involve plant-specific resolutions. The NRC staff also notes that these two later categories were only applied to a limited set of SRs (i.e., SHA-B3, SHA-C3, SFR-C3, SFR-G3, SPR-B1, and SFR-C6).

In light of the observations above, confirm that the generic resolutions are applicable to the SPRA and provide plant-specific justification that the SPRA is in conformance with Addendum A

SRs SHA-B3, SHA-C3, SFR-C3, SFR-G3, SPR-B1 and SFR-CR (C-II where it applies); or that the SPRA conforms to an industry practice considered more current than the practice required by Addendum A (e.g., like Vogtle did for SR SFR-C6).

SNC Response:

The categorization of those ASME/ANS PRA Standard RA-S-2008 Addendum B (RA-Sb-2013) Part 5 (Seismic) SRs as meeting one of the two generic comparison categories (1) “Addendum B Assessment Equates to Addendum A” and (2) “Addendum B Envelopes Addendum A” as documented in the report titled “Response to Supplemental Information Needed for Acceptance of Systematic Risk-Informed Assessment of Debris Technical Report (ADAMS Accession No. ML17192A245)”, is applicable to the Hatch SPRA.

For the following Addendum B Part 5 SRs to which neither of the above categories applies, i.e., SHA-B3, SHA-C3, SFR-C3, SFR-C6, SFR-G3, and SPR-B1, the following table provides plant specific justification that either the Hatch SPRA is in conformance with the corresponding Addendum A Part 5 SRs or the Hatch SPRA conforms to accepted current industry practices.

Table 8.1: Comparison of Supporting Requirements of Addendum A and Addendum B

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SHA-B3	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	As a part of data collection, COMPILE a catalog of historically reported, geologically identified, and instrumentally recorded earthquakes. USE reference [5-30] requirements or equivalent.		**Hatch Conforms to Addendum A** A catalog of historically reported, geologically identified, and instrumentally recorded earthquakes for the entire CEUS was compiled by the 2012 CEUS SSC report. Following a SSHAC Level 3 process, the CEUS SSC report is a robust evaluation of available information on historical seismicity, paleoseismic data on large-magnitude recurrence rates, and state-of-the-knowledge of earthquake seismic sources as considered in the informed technical community. The 2012 CEUS SSC catalog followed a SSHAC Level 3 process and is applicable for risk informed applications. Compiling a new catalog will not be as rigorous as the SSHAC Level 3 process. The Addenda B SR requirement is appropriate for CC-II. The 2012 CEUS SSC report used an earthquake catalog which extended through 2008. Recent earthquake activity in the vicinity of the Hatch site was assessed for its impact on hazard. The study was based on a temporal update of the earthquake catalog from 2009 through February 2016. The assessment concluded that the 2012 CEUS SSC report seismicity parameters are appropriate for evaluation of seismic hazard at Hatch. Based on this, the Hatch PSHA that was performed conforms to Addendum A.
	ASME/ANS RA-Sb–2013	INCLUDE an appropriate existing catalog of historically reported earthquakes, instrumentally recorded earthquakes, and earthquakes reported through geological investigations. USE reference [5-30] requirements or equivalent.		<not printed here; not focus of this assessment>	

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SHA-C3	ASME/ANS RA-Sa–2009	<not printed here; not focus of this assessment>	The seismic sources are characterized by **source location** and geometry, maximum earthquake magnitude, and earthquake recurrence. INCLUDE the aleatory and epistemic uncertainties explicitly in these characterizations.		**Hatch Conforms to Addendum A** Addenda B added additional clarification into the text of this SR, and also added a clause "where significant" at the end. The Addenda B SR requirement is appropriate for CC-II. Under the SSHAC Level 3 process the aleatory and epistemic uncertainties in seismic sources are characterized for source location and geometry, magnitude, and activity rate. Logic trees to account for the epistemic uncertainty were developed as part of the SSHAC Level 3 methodology implemented in the CEUS SSC report. The aleatory uncertainty was also accounted for in the PSHA framework of the Hatch PSHA. For seismic sources representing repeated large magnitude earthquakes (RLMEs), uncertainties in location and geometry, magnitude model, activity rate, and maximum magnitude were explicitly included in the characterization. For background sources, uncertainty in geometry was represented with alternative sets of area sources, uncertainties in recurrence rates were represented with alternative rates, and uncertainties in maximum magnitude were represented with distributions of values. These uncertainties were documented in the 2012 CEUS SSC report and were included in the Hatch PSHA. Based on this, the Hatch PSHA that was performed conforms to Addendum A.
	ASME/ANS RA-Sb–2013	<not printed here; not focus of this assessment>	The seismic sources are characterized by **alternative source representation** and source geometry, maximum earthquake magnitude, and earthquake recurrence. INCLUDE the aleatory and epistemic uncertainties explicitly in these characterizations, **where significant.**		

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
SFR-C3	ASME/ANS RA-Sa–2009	If scaling of existing design response analysis is used, JUSTIFY it based on the adequacy of structural models, foundation characteristics, and similarity of input ground motion.		<not printed here; not focus of this assessment>	**Hatch Conforms to Addendum A** The change from Addendum A to Addendum B involved the deletion of the word "design" from "existing design response analysis." However, Plant Hatch did not perform scaling of any existing response analysis. Therefore, the change is irrelevant, and Hatch conforms to Addendum A.
	ASME/ANS RA-Sb–2013	If scaling of existing response analysis is used, JUSTIFY it based on the adequacy of structural models, foundation characteristics, and similarity of input ground motion.		<not printed here; not focus of this assessment>	
SFR-C6	ASME/ANS RA-Sa–2009	When soil-structure interaction (SSI) analysis is conducted, ENSURE that it is median centered using median properties, at soil strain levels corresponding to the input ground motions that **dominate** the seismically induced core damage frequency. ACCOUNT for the uncertainties in the SSI analysis **by varying the low strain soil shear modulus between the median value times (1 + Cv) and the median value divided by (1 + Cv), where Cv is a factor that accounts for uncertainties in the SSI analysis and soil properties. If adequate soil investigation data are available, ESTABLISH the mean and standard deviation of the low strain shear modulus for every soil layer. Then ESTABLISH the value of Cv so that it will cover the mean plus or minus one standard deviation for every layer. The minimum value of Cv is 0-5. When insufficient data are available to address uncertainties in soil properties, ENSURE that Cv is taken as no less than 1.0.**		<not printed here; not focus of this assessment>	**Hatch Conforms to Accepted Current Practices** The changes in SFR-C6 involved the replacement of "ACCOUNT for" with the more precise action verb "INCLUDE", the non-substantive replacement of "dominate" with "contribute most" for PRA standard consistency, and the removal of how to perform SSI uncertainty analysis. The SSI uncertainty analysis method presented in Addendum A is derived from ASCE 4-98 (as indicated by the non-mandatory Note 5). Section 3.3.1.7 of ASCE 4-98 states that the use of (1 + Cv) to vary low strain soil shear moduli is an acceptable method *in lieu of probabilistic evaluation,* which Section C.3.3.1.7 further states is the preferred approach. Plant Hatch accounted for uncertainties in the SSI analysis by applying strain-compatible

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANS RA-Sb–2013	When soil-structure interaction (SSI) analysis is conducted, ENSURE that it is median centered using median properties, at soil strain levels corresponding to the input ground motions that **contribute most** to the seismically induced core damage frequency. INCLUDE the uncertainties in the SSI analysis.		<not printed here; not focus of this assessment>	soil properties derived from probabilistic evaluation via the local site response analysis which includes epistemic and aleatory uncertainties. Therefore, the Addendum B assessment is considered appropriate.
SFR-G3	ASME/ANS RA-Sa–2009	**DOCUMENT the sources of model uncertainty and related assumptions associated with the seismic fragility analysis.**			**Hatch Conforms to Addendum A** Addendum B deleted this SR. However, the Plant Hatch SPRA documentation describes in detail the sources of model uncertainty and related assumptions associated with the seismic fragility analysis. Therefore, the Hatch SPRA conforms to Addendum A.
	ASME/ANS RA-Sb–2013	**Deleted.**			
SPR-B1	ASME/ANS RA-Sa–2009	In each of the following aspects of the seismic-PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. **DEVELOP a defined** basis to support the claimed nonapplicability of any exceptions. The aspects governed by this requirement are (a) initiating-event analysis (b) accident-sequence analysis (c) success-criteria analysis (d) systems analysis (e) data analysis (f) human-reliability analysis (g) use of expert judgment **When the Part 2 requirements are used, FOLLOW the Capability Category designations in Part 2, and for consistency USE the same Capability Category in this analysis.**			**Hatch Conforms to Addendum A** Addendum B removed the last sentence of this SR in response to an EPRI 2011 comment on the Addendum B ballot. The last sentence was removed in Addendum B because it was determined to be confusing as wells as inappropriate specificity to require all new aspects in the SPRA to meet the exact same CCs of Part 2 SRs. In addition, Addendum B changed the action verb to be consistent with accepted verb usage across SRs. The Addendum B SR requirement clarifications are appropriate. Regardless, the Plant Hatch 1&2 SPRA builds upon the internal events PRA and uses the same

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment (CC-II Focus)
	ASME/ANS RA-Sb–2013	In each of the following aspects of the seismic-PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. **SPECIFY** a basis to support the claimed nonapplicability of any exceptions. The aspects governed by this requirement are (a) initiating-event analysis (b) accident-sequence analysis (c) success-criteria analysis (d) systems analysis (e) data analysis (f) human-reliability analysis (g) use of expert judgment			general methodologies as used for Part 2 where applicable; therefore, the Hatch SPRA conforms to Addendum A.

<u>RAI 09 (APLB) - Seismic PRA Peer Review Criteria</u>

Paragraph 50.69(c)(1)(i) of 10 CFR requires the PRA must be of sufficient quality and be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. LAR Section 3.3 states that the PRA models have been assessed against RG 1.200, Revision 1.

Section 2.2 of RG 1.200 provides regulatory guidance regarding peer reviews and the staff regulatory position on NEI 00-02, 05-04, and 07-12. NRC letter, 'U.S. Nuclear Regulatory Commission Acceptance of NEI Guidance NEI 12-13, "External Hazards PRA Peer Review Process Guidelines' (August 2012)," dated March 7, 2018 (ADAMS Accession No. ML18025C025), provides the NRC staff's comments on this guidance for seismic and external hazard PRA peer reviews. Based on the information provided in the LAR the NRC staff was unable to determine if the SPRA peer review and focused-scope peer review considered the NRC staff's comments in the March 7, 2018 letter.

Discuss how SPRA peer review and focused-scope peer review considered the NRC staff's comments in the March 7, 2018 NRC acceptance letter. In addition, provide justifications for not considering specific comments in the acceptance letter in the context of this application if applicable.

<u>SNC Response:</u>

The Plant Hatch full scope Seismic PRA peer review was performed October 17 through October 21, 2016, at Plant Hatch using the NEI 12-13 process and the PRA Standard (ASME/ANS RA-Sb-2013). The SPRA peer review team did not identify any unreviewed analysis methods in the SPRA.

The Plant Hatch Seismic PRA Finding Level Fact and Observation (F&O) Appendix X closure was performed in June 2017. A concurrent Focused-scope Peer Review was also performed in June 2017 using NEI 12-13 and the PRA Standard. The SPRA focused-scope peer review team did not identify any unreviewed analysis methods in the SPRA.

Although the NRC staff's comments in the NRC acceptance letter were issued on the March 7, 2018, the SPRA peer review and focused-scope peer review had already incorporated them. SNC has reviewed all 32 comments (ML18025C024) and has determined that they are incorporated. SNC provides additional information on comments that were deemed not to be generic, as outlined in the following table.

Note: The first four columns are copied directly from ML18025C024 with the original formatting. Some of this formatting incudes strikethroughs, boldface and deletions.

Table 9.1 SNC Dispositions to Specific Comments

ID	Index	Issue	Proposed Staff Resolution	SNC Disposition
1	Section 2.1	It is recognized that because of the unique aspect of a seismic PRA, a form of sequencing the peer review may be needed. However, the way the guidance is written, it can be interpreted (e.g., “one week onsite”) as not supporting an “in-process” approach. The guidance need to distinguish between an “in-process” and “all at once” approach. Regardless, each approach has to meet (1) the requirements of an independent peer review as stated in the PRA standard as endorsed in RG 1.200, and (2) the process described in NEI 12-13.	To follow the third paragraph of Section 2.1: **The peer reviews may be performed in various phases of the development of the PRA. It is recognized that the unique and discrete aspects of seismic PRA (i.e., hazard analysis, fragility analysis, and event and fault tree modeling) lends itself to some form of sequenced peer reviews that may occur during the development of the PRA (i.e., an in-process PRA peer review). However, regardless of whether the peer review being performed is an in-process peer review or a final peer review after the PRA is completed, either approach needs to meet:** **1. the requirements for an independent peer review as stated in the ASME/ANS PRA standard and as endorsed in RG 1.200, and** **2. the process described in NEI 12-13.** **Peer review findings from an in-process review may be formalized as part of that in-process peer review or deferred as a draft finding to the final peer review following the completion of the PRA. An in-process peer review is not considered to be final until the final peer review is performed following the completion of the PRA. In addition to creation of any new findings, the final peer review would assess any draft findings from in-process PRA peer reviews, which may require a re-review of the related PRA**	An in-process peer review of the Hatch SPRA was not performed. A final full scope peer review was performed to judge the technical adequacy of the SPRA model.

ID	Index	Issue	Proposed Staff Resolution	SNC Disposition
			aspects. Licensees that use an in-process peer review must assure that the independence of the members of the peer review team is maintained given that those members will also participate in the final peer review. …as expansive as a peer review of the entire External Hazards PRA. **The F&O independent assessment process is not a substitute for the Follow-on Peer Review.**	
4	Page 2, Section 1.1, 3rd and 4th paragraph	Internal events F&Os that were not appropriately "addressed" prior to the External Hazards PRA Peer Review may have a significant detrimental effect on the external hazard PRA.	• …F&Os that were not addressed **closed by a focused-scope peer review or independent assessment** prior to the External… The review of Internal Events PRA model issues pertinent to the External Hazards PRA undergoing the peer review **is required to be** addressed in the self-assessment, as discuss**ed**~~ion~~ in Section 1.4 below. **The External Hazards PRA Peer Review is required to review all of the findings level F&Os from the internal events PRA peer review and determine whether the resolution was appropriate and in accordance with the endorsed or accepted ASME/ANS PRA standard**.	Prior to the conduct of the Hatch SPRA peer review, the finding-level F&Os from the Hatch internal events (including internal flooding) PRA peer review were dispositioned and incorporated into the PRA model as appropriate prior to use of the internal events PRA as the basis for the SPRA. There were twenty-five findings for the Internal Events (including Internal Flooding) PRA model. Since the peer review, the findings were closed per NRC endorsed process outlined in Appendix X of NEI 12-13. Further, there were no PRA upgrades required for the internal events PRA to incorporate the finding resolutions prior to applying the model to the SPRA. Therefore, there was no

ID	Index	Issue	Proposed Staff Resolution	SNC Disposition
				need for a focused-scope peer review. An assessment of the resolution of each internal events PRA peer review finding was made to determine if the resolution was appropriate for the SPRA. The SPRA peer review team was provided with the internal events peer review report and the dispositions of the findings to facilitate their assessment of adequacy of the internal events PRA model as the basis for the SPRA as part of their review of supporting requirement SPR-B1.
6	Page 5, Section 1.4, External Hazards PRA Peer Review Preparatory Review and Self Assessment	A high-quality self-assessment is an important part of ensuring a successful External Hazard PRA peer review. However, it is unclear whether the self-assessment is required in whole or in part (throughout the document, including page A-12 which indicates it is “optional but recommended”).	The self-assessment is key to ensuring that the overall Peer Review process is completed within the scheduled time and that all of the required review is completed. **The self-assessment is required to be performed prior to the peer review and must include a self-assessment of:** • **The referenced Internal Events PRA against the SRs listed in Table D-1** • **The seismic, high winds, or external flooding PRA against the respective SRs listed in Part 5, Part 7, or Part 8.**	A self-assessment was provided to the SPRA peer review team prior to the peer review. The self-assessment is assessed against the respective SRs listed in the Part 5 of the ASME/ANS Standard.
11	Page 18, Step 11	New information should not be provided subsequent to the peer review team’s departure	~~New information provided… …this new information.~~	No information was provided to the full scope peer review team subsequent to the

ID	Index	Issue	Proposed Staff Resolution	SNC Disposition
		from the peer-review location. The peer-review is intended to capture the "snapshot" of the model. New information subsequent to the departure of the peer review team is outside of the scope of the peer review and should be part of the resolution of the F&O/open item. Providing information after the peer review team has left the site is also inconsistent with the performance of actual peer reviews.		departure of the peer review team.
13	Page 20 2.2, footnote 8	The External Hazards PRA Peer Review Team should meet the requirements in Sections 1-6.2 and the peer review section in each applicable external hazard Part of the ASME/ANS PRA Standard.	In addition to the requirements in Section 1-6, each Part of the PRA Standard includes requirements….. the review team should be is assembled to meet those requirements.	The qualifications of the Hatch SPRA peer review team were reviewed by SNC against the requirements in ASME/ANS PRA Standard RA-Sa-2009 to establish that the peer review team met these requirements. Section 1-6.2 of this standard defines requirements for a PRA peer review team as a whole and for individual reviewers. Section 5-3.2 further establishes seismic PRA-specific reviewer requirements. Prior to the peer review, SNC was responsible for accepting each of the proposed peer review team members relative to the

ID	Index	Issue	Proposed Staff Resolution	SNC Disposition
				requirements in the PRA standard. Team requirements in Section 1-6.2 for a seismic PRA peer review include the ability to assess all the applicable PRA Elements of the Technical Requirements section in Part 5 of the Standard, and collectively having knowledge of the plant NSSS design, containment design, and plant operation. Team requirements in Section 5-3.2 include having combined experience in the areas of systems engineering, seismic hazard, seismic capability engineering, and seismic PRAs or seismic margin methodologies. Individual peer reviewer requirements in Section 1-6.2 include having knowledge of the requirements in the Standard for their area of review, having experience in performing the activities related to the PRA Elements for which the reviewer is assigned, and having neither performed nor directly supervised any work on the

ID	Index	Issue	Proposed Staff Resolution	SNC Disposition
				portions of the PRA being reviewed. The peer reviewers must also have direct experience with the specific methodology, code, tool, or approach that was used in the PRA Element assigned for review. Section 5-3.2 further requires that reviewers focusing on the seismic-fragility work have successfully completed the SQUG Walkdown Screening and Seismic Evaluation Training Course or equivalent or have demonstrated equivalent experience in seismic walkdowns.
14	Page 20, Section 2.2, 2nd paragraph	There have been some recent questions and concerns regarding the independence of peer review team or independent assessment team members.	…With the exception of individuals who have worked on or directly supervised the subject PRA, there are no automatic exclusion criteria; however, the host utility may question the independence of any proposed Peer Review Team member. **The term "worked on" is intended to include any utility staff or contractors that had any association with the portion of the External Hazard PRA that they are reviewing. Similarly, an external hazard PRA team member who had an association with the basis internal events PRA model would not meet the independence requirement for reviewing the closure of the associated internal events findings.**	The SPRA peer reviewers had no previous involvement in the Hatch Seismic PRA. This is certified by the reviewers' signatures on the cover of the peer review report. This satisfies the independence requirements of Section 1-6.2.2 of the ASME/ANS PRA Standard.

ID	Index	Issue	Proposed Staff Resolution	SNC Disposition
17	Page 22 2.2, bullet titles	The External Hazards PRA Peer Review Team should meet the requirements in Sections 1-6.2 and the peer review section in each applicable external hazard Part of the ASME/ANS PRA Standard.	• Experience Expectations Needs for Peer Review Team Lead:….. • Experience Expectations Needs for Individual Peer Review Team Members:….. • Additional Experience Expectations Needs for the Team as a Whole…..	The qualifications of the Hatch SPRA peer review team were reviewed by SNC against the requirements in ASME/ANS PRA Standard RA-Sa-2009 to establish that the peer review team met these requirements. Section 1-6.2 of this standard defines requirements for a PRA peer review team as a whole and for individual reviewers.
18	Page 22 2.2, 7th paragraph, last sub-bullet	The External Hazards PRA Peer Review Team should meet the requirements in Sections 1-6.2 and the peer review section in each applicable external hazard Part of the ASME/ANS PRA Standard.	Specialized expertise in seismic, high winds, external flood or other External Hazards PRAs ~~should be strongly considered~~ **is needed** if these hazards are being reviewed.	Section 1-6.2 of this standard defines requirements for a PRA peer review team as a whole and for individual reviewers. Section 5-3.2 further establishes seismic PRA-specific reviewer requirements. Prior to the peer review, SNC was responsible for accepting each of the proposed peer review team members relative to the requirements in the PRA standard. Team requirements in Section 1-6.2 for a seismic PRA peer review include the ability to assess all the applicable PRA Elements of the Technical Requirements section in Part 5 of the Standard, and collectively

ID	Index	Issue	Proposed Staff Resolution	SNC Disposition
				having knowledge of the plant NSSS design, containment design, and plant operation. Team requirements in Section 5-3.2 include having combined experience in the areas of systems engineering, seismic hazard, seismic capability engineering, and seismic PRAs or seismic margin methodologies.
21	Page 25, Section 3.2, 2nd paragraph	The requirement to review the changes to the internal events model against appropriate Part 2 SRs is included, but the requirement to review all findings level internal events PRA F&Os and their dispositions is not included.	Add a paragraph to discuss the requirements associated with reviewing the internal events PRA F&Os and their disposition.	An assessment of the resolution of each internal events PRA peer review finding was made to determine if the resolution was appropriate for the SPRA. The SPRA peer review team was provided with the internal events peer review report and the dispositions of the findings to facilitate their assessment of adequacy of the internal events PRA model as the basis for the SPRA as part of their review of supporting requirement SPR-B1.
22	Page 26, Section 3.2, 1st paragraph	Any resolved Inquiries that are used in the interpretation of SR(s) for the peer review need to be documented explicitly.	…assignment of a Capability Category for the SR. **All such instances will be documented in the peer review report along with the specific SRs that were interpreted using each Inquiry.**	There were no SRs that required interpretation for the peer review.

ID	Index	Issue	Proposed Staff Resolution	SNC Disposition
23	Page 26, Section 3.2, last paragraph	Based on lessons learned in the implementation of NEI 0712, the definition of a UAM should be revised to include all new methods or changes to existing methods which have not been vetted by a broad technical community, even if they were reviewed by the peer reviewer. Such methods should be flagged as UAMs and documented in the peer-review report. A definition of what constitutes a new method is necessary which is consistent with established staff position.	Unreviewed Analysis Method – an observation regarding the use of methods that are new or beyond the expected expertise of the review team or, and for which the review would exceed the time and capability of the External Hazards PRA Peer Review team. When a**n F&O** is written with this classification, would need the method would need to be reviewed by a separate body of experts. **New Method – an observation regarding the use of methods that are new. An F&O written with this classification will be reviewed during the peer review, the peer review report will identify it explicitly as a new method along with the aspect(s) that makes it novel, and the reason(s) why the method was found to be acceptable or unacceptable (in whole or part) to the peer review team. [F&Os with this classification cannot be closed out via the F&O closure process described in NEI 05-04/07-12/12-13 Appendix X process (ML17086A431) or a follow-on peer review unless the method no longer meets the current definition of "new method".]**	There were no unreviewed analysis methods used in the Hatch SPRA. The SPRA peer review team did not identify any unreviewed analysis methods in the SPRA

RAI 10 (APLA/APLB) - SSC Categorization Based on Other External Hazards

Paragraph 50.69(b)(2)(ii) of 10 CFR requires that the quality and level of detail of the systematic processes that evaluate the plant for external events during operation are adequate for the categorization of SSCs.

LAR Section 3.2.4 states that "[a]ll other hazards were screened from applicability to Hatch Units 1 and 2 per a plant-specific evaluation in accordance with the criteria in Section 6 of ASME PRA Standard RA-Sb-2013." This statement appears to indicate that Hatch proposes to treat all SSCs as low-safety-significant (LSS) with respect to other external events risk. The LAR provides no further explanation of how the risk for other external hazards will be considered in 10 CFR 50.69 categorization (i.e., components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario). LAR Attachments 4 and 5 provide a summary of the other external hazards screening results, but do not appear to address any considerations related to applying Figure 5-6 of NEI 00-04 guidance to those hazards. Considering these observations, address the following:

a) LAR Section 3.2.4 states that external hazards were screened using the criteria in Section 6 of the 2013 PRA Standard ASME/ANS RA-Sb-2013. LAR Attachment 5 however appears to list the criteria from the 2009 PRA Standard ASME/ANS RA-Sa-2009. Clarify and justify the criteria used for the screening of external hazards.

SNC Response:

Section 3.2.4 of the LAR states that, *"All other external hazards were screened from applicability to Hatch Units 1 and 2 per a plant specific evaluation in accordance with the criteria in Section 6 of ASME PRA Standard RA-Sb-2013. RG 1.200 Revision 2 endorses the RA-Sa-2009 version of the standard, the 2013 version of the standard contains the same technical requirements as the 2009 version, however editorial changes to the layouts of the tables and attachments were performed."*

While LAR Attachment 5 lists the 2009 version of the standard and associated criteria, this was a mistake and should have listed the 2013 version and associated criteria, as the 2013 version was used as stated in section 3.2.4 of the LAR.

Use of ASME PRA Standard RA-Sb-2013 for screening other external hazards meets all the requirements in the RA-Sa-2009 version of the standard. The set of screened hazards is the same using either version of the standard. Additionally, the differences in the screening criteria between the 2009 and 2013 versions of the standard are minor editorial changes. There is an exception regarding Criterion PS1 (EXT-C1, Criterion A in the 2009 version of the standard) which is discussed below.

During the March 2019 NRC LAR Audit, the NRC noted that Criterion PS1, did not transition to the 2013 version of the standard. However, as HNP utilized the 2013 version of the standard for the external hazard screening analysis, PS1 was not applied.

There are minor editorial differences in the "fundamental screening criteria" listed in section 6-2.3 of the standard, listed below (Table 10.a.1). Additionally, the Progressive Screening Approach shown in Attachment 5 of the HNP LAR is reproduced in part below

(Table 10.a.2) with an additional source column and additional comments focusing on the differences in the PRA Supporting Requirements used for screening.

Table 10.a.1: Comparison of Fundamental Screening Criteria

2009 version	2013 version
There are three fundamental screening criteria embedded in the requirements here, as follows. An event can be screened out either	There are three fundamental screening criteria embedded in the requirements here, as follows. A hazard can be screened out if
(a) if it meets the criteria in the NRC's 1975 Standard Review Plan (SRP) [6-2] or a later revision; or	(a) it meets the criteria in the U.S. Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP) [6-2] or a later revision; or
(b) if it can be shown using a demonstrably conservative analysis that the mean value of the frequency of the design-basis hazard used in the plant design is less than~10-5/yr and that the conditional core damage probability is <10-1, given the occurrence of the design-basis hazard event; or	(b) it can be shown, by using a demonstrably conservative analysis, that the mean value of the frequency of the design-basis hazard event used in the plant design is less than~10-5/yr and that the conditional core damage probability is <10–1, given the occurrence of the design basis-hazard event; or
(c) if it can be shown using a demonstrably conservative analysis that the CDF is <10-6/yr.	(c) it can be shown, by using a demonstrably conservative analysis, that the CDF is <10–6/yr.

Table 10.a.2: HNP LAR Attachment 5 Clarification

Progressive Screening Approach for Addressing External Hazards				
Event Analysis	**Criterion**	**Source**		**Comments**
		2009 Version	**2013 Version**	
Initial Preliminary Screening	C1. Event damage potential is < events for which plant is designed.	ASME/ANS Standard RA-Sa-2009 EXT-B1 Criterion 1	ASME/ANS Standard RA-Sa-2013 EXT-B1 Criterion 1	The Criterion is the same except for minor editorial language (e.g., use of “event” in 2009 version vs. “hazard” in 2013 version.
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	ASME/ANS Standard RA-Sa-2009 EXT-B1 Criterion 2	ASME/ANS Standard RA-Sa-2013 EXT-B1 Criterion 2	The Criterion is the same except for minor editorial language (e.g., use of “event” in 2009 version vs. “hazard” in 2013 version.
	C3. Event cannot occur close enough to the plant to affect it.	ASME/ANS Standard RA-Sa-2009 EXT-B1 Criterion 3	ASME/ANS Standard RA-Sa-2013 EXT-B1 Criterion 3	The Criterion is the same except for minor editorial language (e.g., use of “event” in 2009 version vs. “hazard” in 2013 version.
	C4. Event is included in the definition of another event.	ASME/ANS Standard RA-Sa-2009 EXT-B1 Criterion 4	ASME/ANS Standard RA-Sa-2013 EXT-B1 Criterion 4	The Criterion is the same except for minor editorial language (e.g., use of “event” in 2009 version vs. “hazard” in 2013 version.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009 EXT-B1 Criterion 5	ASME/ANS Standard RA-Sa-2013 EXT-B1 Criterion 5	The Criterion is the same except for minor editorial language (e.g., use of “event” in 2009 version vs. “hazard” in 2013 version.

Progressive Screening Approach for Addressing External Hazards				
Event Analysis	**Criterion**	**Source**		**Comments**
		2009 Version	**2013 Version**	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009 EXT-C1 Criterion A	Deleted	Criterion A in the 2009 version of the Standard did not transition to the 2013 version of the Standard. No hazards in the HNP LAR were screened using PS1.
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	ASME/ANS Standard RA-Sa-2009 EXT-B2	ASME/ANS Standard RA-Sa-2013 EXT-B2	The Criterion is the same except the 2013 version of the Standard states to "JUSTIFY any screening out of an external hazard based *solely* on conformance to SRP." No hazards in the HNP LAR were screened solely using PS2. Aircraft Impacts was the only hazard using PS2 which also used PS4.
	PS3. Design basis event mean frequency is < 1E-5/y and the mean conditional core damage probability is < 0.1.	ASME/ANS Standard RA-Sa-2009 EXT-C1 Criterion B	ASME/ANS Standard RA-Sa-2013 EXT-C1 Criterion A	The Criterion is the same except the 2009 version uses Criterion B and has been renumbered to Criterion A in the 2013 version.
	PS4. Bounding mean CDF is < 1E-6/y.	ASME/ANS Standard RA-Sa-2009 EXT-C1 Criterion C	ASME/ANS Standard RA-Sa-2013 EXT-C1 Criterion B	The Criterion is the same except the 2009 version uses Criterion C and has been renumbered to Criterion B in the 2013 version.

Progressive Screening Approach for Addressing External Hazards				
Event Analysis	Criterion	Source		Comments
		2009 Version	2013 Version	
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	ASME/ ANS PRA Standard.	ASME/ ANS PRA Standard.	Detailed PRAs were developed for internal flooding, fire, and seismic.

b) *Identify the external hazards that will be evaluated according to the flow chart in NEI 00-04, Section 5.4, Figure 5-6. Provide detailed justification for screening external hazards (i.e., external flood, high winds, and tornados) using the criteria described in part a above. As applicable, the justification should include consideration of uncertainties in the determination of demonstrably conservative mean values, as discussed in Section 6.2-3 of the ASME/ANS RA-Sa-2009 PRA Standard.*

SNC Response:

The external hazards that will be evaluated according to the flow chart in NEI 00-04, Section 5.4, Figure 5-6 are all the hazards shown in Attachment 4 of the LAR except internal flood, internal fire, and seismic. Specifically, they are:

- Aircraft Impact
- Avalanche
- Biological Event
- Coastal Erosion
- Drought
- External Flooding
- Extreme Wind/Tornado
- Fog
- Forest/Range Fire
- Frost
- Hail
- High Summer Temperature
- High Tide/Lake Level/River Stage
- Hurricane
- Ice Cover
- Industrial/Military Facility Accident
- Landslide
- Lightning
- Low Lake Level/River Stage
- Low Winter Temperature
- Meteorite/Satellite Impact
- Pipeline Accident
- Release of Chemicals in Onsite Storage
- River Diversion
- Sand/Dust Storm
- Seiche
- Snow
- Soil Shrink-Swell Consolidation
- Storm Surge
- Toxic Gas
- Transportation Accident
- Tsunami
- Turbine-Generated Missiles
- Volcanic Activity
- Waves

LAR attachment 4 provides justification for the use of the chosen screening criterion for a given hazard. The following discussion provides additional details and justification for the screening of external floods, extreme winds and tornados.

<u>External Flooding</u>

The external flooding hazard was screened at HNP using the C1 criterion, *"The hazard is of equal or lesser damage potential than the hazards for which the plant has been designed."*

The justification for use of C1 is based on the results of a series of calculations and analyses performed in response to Near Term Task Force recommendation 2.1, which were developed in accordance with NEI 12-06, NEI 16-05, and NUREG/CR-7046. The HNP Flood Hazard Reevaluation Report (FHRR) (ML14069A054, as supplemented by ML14219A570, ML15154B601, and ML16069A088) examined the external flooding hazard in detail, including the following mechanisms which were not bounded by the existing design basis flood height:

- Local Intense Precipitation (LIP)
- Combined effects flooding (Probable Maximum Flood (PMF) with upstream overtopping dam failure with wind-induced waves)
- Flooding in rivers and streams (all season Probable Maximum Flood (PMF) with a 1/2 PMF antecedent storm)
- Seismic upstream dam failure
- PMF with upstream overtopping dam failure

The final NRC staff assessment of the FHRR was issued in ML16237A095.

In the follow-up evaluation documented in the HNP Mitigating Strategies Assessment (MSA, ML16351A087) and NRC's assessment letter (ML17069A234), HNP determined that three (listed below) of the five previously identified flooding mechanisms have maximum flooding elevations that remain below the 111 ft. Geodetic Vertical Datum of 1929 (NGVD 29) elevation and did not require further analyses.

- Flooding in rivers and streams (all season Probable Maximum Flood (PMF) with a 1/2 PMF antecedent storm)
- Seismic upstream dam failure
- PMF with upstream overtopping dam failure

The elevation of 111 ft. NGVD 29 is the elevation of the lowest floor of the intake structure; this is also lower than the power block, which sits at approximately 130 ft. NGVD 29. Therefore, these mechanisms cannot cause damage to key SSCs and are therefore screened under C1.

Only LIP and the combined effects of flooding required further evaluation due to potential to exceed the 111 ft. NGVD 29 elevation. The flooding focused evaluation (SCNH-16-007, ML17173A777) was developed to assess LIP and combined effects flooding and the NRC response has been documented (CAC NOS. MF9687 and MF9868). The underlying SNC calculation was audited by the NRC (ML17192A452). The following discussion summarizes the LIP and combined effects conclusions and applicability to the screening evaluation.

<u>Local Intense Precipitation (LIP)</u>

The LIP analysis utilized a 1-hour/1-square mile Probable Maximum Precipitation (PMP) approach to determine local flood levels across a grid. Some doors were identified where the maximum LIP exterior water surface elevation would be greater than the finished floor elevation (both based on NGVD 29) for a given duration.

Calculations show that the water ingress from the LIP event is insufficient to damage key SSCs. SNC used conservative assumptions in the calculation of water ingress and available physical margin. For example, it was assumed that there was no water leakage during the time of inundation; this means the maximum exterior surface water elevation, and therefore maximum head, was used in calculations of ingress under doors. Additionally, storm drains were not credited in the computation of water heights at the exterior doors.

The NRC has previously concluded that there is a reasonable assurance that areas containing key SSCs will not be adversely impacted by water ingress from the LIP reevaluated hazard, and that there are adequate flood protection features associated with key SSCs that will ensure their continued function in the event of LIP (CAC NOS. MF9687 and MF9868). Therefore, the LIP flooding mechanism screens under C1, as the LIP hazard is not capable of causing plant transients or impacting mitigation equipment that provides key safety functions due to site topography and design.

Combined Effects Flooding

Analysis was documented in the focused flooding evaluation (SCNH-16-007, ML17173A777) and in the NRC response (CAC NOS. MF9687 and MF9868) for Combined Effects flooding. The Combined Effects flood is a beyond design basis event, requiring a combination of probable maximum flooding, dam overtopping failure, and wind-driven wave. Combined, the surface water elevation could temporarily reach 118.6 ft. NGVD 29.

Due to the site topography, the main power block buildings are located at an elevation (approximately 130 ft.) that provides enough available physical margin to prevent damage to SSCs within those structures.

At the intake structure, the fixed floor elevation of 111 ft. NGVD 29 would not be reached by PMF with dam overtopping failure only; however, with the addition of the wind-driven wave the fixed floor elevation is exceeded. The calculations show that the intake structure walls would withstand both hydrostatic and hydrodynamic loads, and that the doors would withstand hydrodynamic loads (SCNH-16-007). The doors would not be subject to hydrostatic loading.

Due to the bathymetry of the site, the intake structure would not be directly impacted by waves because the waves would break prior to impact. The hydrostatic loading at the doors was computed as if waves were normally incident to the doors; however, this is conservative because the doors have a labyrinth geometry and the waves could not be normally incident; also, the doors have weather stripping which aids in prevention of water ingress. Additionally, grating south of the intake structure would drain waves to the valve pit below; this was not credited in the calculations. Furthermore, there are grates within the structure that would allow drainage of water ingress from a wave, were it to occur, prior to reaching key SSCs.

The NRC staff has previously concluded that there is a reasonable assurance that the intake structure has adequate flood protection such that its functions will not be adversely impacted by the combined flooding effects reevaluated hazard (CAC NOS. MF9687 and MF9868). Therefore, the combined effects flooding mechanism screens under C1, as the hazard is not capable of causing plant transients or impacting mitigation equipment that provides key safety functions due to site topography and design.

Plant Response to External Flooding

HNP does not rely on any personnel actions to respond to the design basis or beyond design basis flooding mechanisms. The NRC staff previously concluded that HNP has demonstrated that effective flood protection exists and that an "Integrated Assessment" based on the NRC's JLD-ISG-2016-1 document was not necessary (CAC NOS. MF9687 and MF9868). Therefore, only passive features (further discussed in response to 10.d) are credited in the screening of the hazard.

While the site does not rely on any personnel actions for external hazards, HNP does have a "Naturally-Occurring Phenomena" procedure (34AB-Y22-002-0) that directs the site to take actions based on Altamaha River elevation or expected elevation. Based on the time required for the maximum probable flood to develop, and the time required for flooding water from a dam failure to reach the site, there is adequate time to perform these proceduralized actions.

- The procedure directs an inspection of the intake building sumps, monitoring of the intake valve pit for leakage, and monitoring of the plant service water flows for degraded performance. The procedure also directs alignment and operation of traveling water screens and trash rakes when necessary.

- The procedure further directs additional actions to be taken if flood levels are expected to exceed the design basis water elevation. This includes an orderly shutdown of both units' reactors and operation of sump pumps to maintain valve pit water level as low as possible. If necessary, the procedure directs deployment of FLEX equipment to protect plant service water strainers and backwash MOVs.

- If the elevation is expected to exceed 111 ft., then additional actions are directed to protect plant equipment and pre-stage/deploy FLEX equipment.

Extreme Winds and Tornados

HNP extreme wind or tornado hazard is screened based on the criterion C1, "*event damage potential is less than for which the plant was designed."* The criterion originally provided in the LAR was PS3 *"design basis mean event frequency is < 1E-5/yr and the mean conditional core damage probability damage is <0.1"*. Given the information below, criterion C1 is the appropriate screening criterion for this hazard.

NUREG/CR-4461 provides tornado strike probabilities and maximum wind speeds for use in nuclear power plants. Table 6-1, "Tornado Wind Speed Estimates for United States Nuclear Power Plant Sites," lists the 1E-6 probability using the Fujita Scale for Hatch as 228 mph and the 1E-7 wind speed as 278 mph. Using the more recent Enhanced Fujita Scale, the 1E-6 probability is 181 mph and the 1E-7 wind speed is 213 mph.

Per UFSAR Rev. 37, Section 3.3, all Seismic Category I structures are designed for tornado loadings are based on a 300-mph wind speed. Non-Category I structures have been designed to comply with Seismic II/I (two over one) requirements. Thus, the frequency of a tornado impacting the site that exceeds the plant design basis is less than 1E-7/yr. Since the design wind speed of the plant structures has a frequency of occurrence less than 1E-7, extreme wind hazards are screened out of PRA modeling.

In addition, as stated in the LAR, in response to Regulatory Issue Summary 2015-06, a Tornado Missile Vulnerability Evaluation was performed (RER SNC826314) to verify compliance to the existing license basis requirements. The walkdowns identified two non-conformances which were corrected (one by adding physical protection (CCE SNC799744) and one by engineering calculations showing non-vulnerability due to inherent robustness (SCNH-16-042)), and the site now fully complies with the license basis.

Since safety-related systems and components necessary for safe shutdown are located within Seismic Category I structures or otherwise have been evaluated in engineering calculations as being able to withstand the design hazard, and as the Seismic Category I structures provide adequate physical protection from extreme winds, tornados and associated missiles at greater than the 1E-7 wind speed, this hazard screens from PRA modeling based on Criterion C1.

HNP does not rely on any personnel actions to the plant in order to respond to tornados. The passive protection of structures and inherent robustness of a limited number of components is adequate protection from a tornado that the site might encounter.

<u>Plant Response to Extreme Winds/Tornados</u>

While the site does not rely on any personnel actions for external hazards, the site does have a "Naturally-Occurring Phenomena" procedure (34AB-Y22-002-0) that directs the site to take actions based on direct observation of a tornado within 5 miles of the site or the issuance of a tornado warning in the surrounding area. The site will suspend core alterations and movement of fuel. If an automatic scram does not occur, the site will reduce reactor power to 40-50% until inspections for damage have taken place. Maintenance and facilities will take actions to secure equipment in both the protected area and outside the protected area. Contingency plans will be discussed to facilitate offsite power restoration in the event power is lost. Critical systems will be restored to operable status and all surveillance testing will be suspended. Operators will be pre-dispatched to the Diesel Building to facilitate local recovery of diesels that fail to start automatically. Again, while these actions are not necessary to screen the hazard as the design is sufficiently robust to allow safe shutdown, they demonstrate a plant response that is proactive in mitigating the hazard. A similar set of actions are required by this procedure for sustained high winds observed or projected to exceed 35 mph.

c) *LAR Attachment 4 states, regarding the extreme wind and tornado hazard, that "[c]alculations show that the initiator probability is 3.3E-06 and the CCDP is 1E-03." Provide detailed justification for concluding that for the high winds and tornados hazard, the screening criterion PS3 applies, i.e., the mean frequency is less than 1×10-5 per reactor-year and the mean conditional core damage probability is less than 0.1.*

SNC Response:

HNP has been designed to withstand substantial extreme wind and tornado loadings. The preferred screening basis for the Extreme Wind or Tornado hazard is C1 (Event damage potential is less than events for which the plant is designed.) A fuller justification for use of this screening criterion is shown in the response to 10b.

The statement from LAR Attachment 4 was part of the justification for screening the hazard under PS3; this is no longer the screening criterion that SNC intends to use. The values referred to for initiator frequency and CCDP were from an NRC Significance Determination Process (SDP) evaluation related to the failure of several LOCA/LOSP timer cards in 2009.

During the SDP evaluation (HAT0905), the NRC calculated the Hatch area tornado probability using data from NUREG/CR-4461, updated with Hatch area tornado data. This evaluation concluded that the tornado initiator probability was 3.3E-06. This is significantly lower than the values in the FSAR or IPEEE (SNC Calculation H-RIE-OEE-U00). The CCDP associated with an unrecoverable LOSP event was estimated to be less than 1.2E-03. This would make the contribution to CDF of tornados approximately 4E-9, which is less than the 1E-6/yr bounding value used in PS4 (HLR-C1, Criterion B) and, taken individually, less than the initiator and CCDP bounding values in PS3 (HLR-C1, Criterion A).

d) *Figure 5-6 of NEI 00-04 shows that if an SSC is included in a screened scenario, then in order for that SSC to be considered a candidate LSS, the licensee has to show that if the component was removed, the screened scenario would not become unscreened.*

   i. *Identify and justify what type of SSCs, if any, are credited in the screening of the external hazard(s), including both passive, active, and temporary features.*

SNC Response:

Regarding Question 10.d.i, the following SSCs were identified as being credited in screened scenarios:

Seismic Category I structures are credited for defense against extreme winds and tornado missile protection. These structures are described in the UFSAR Rev. 37, Section 3.5.1.3 as having at least 1 ft. 6 in. thick concrete exterior walls. Calculations show that the deepest missile penetration of the concrete barriers would be 10 in. Additionally, they were designed for 300 mph winds. Therefore, the structures provide adequate protection. The Seismic Category I structures are also credited for protection from external flooding. Any structure providing tornado missile protection is considered a credited SSC, including those structures added in response to RIS 2015-06 to provide protection to the diesel fuel oil storage tank vents (as documented in RER SNC826314). Components which have engineering evaluations demonstrating inherent robustness to extreme winds and tornado missiles absent physical protection from Seismic Category I structures are also considered credited SSCs for the extreme winds/tornado missile screening.

The credited doors are those shown in SNC calculation SCNH-16-007 as closed with a small gap between door and floor (specifically, D-130, D-131, D-166, D-167, R-30A, R-23A, T-15, T-16, 2T-17, 2T-18, Truck Bay Door, and Freight Elevator). The ability of doors to limit floodwater and protect key SSCs during LIP conditions was evaluated through engineering evaluations. The exterior water depth estimates are relatively small; therefore, the design capacity of the doors should not be compromised by pressure from floodwater. Also, these doors are checked as part of normal operator rounds and are subject to an appropriate maintenance and inspection regime. D-130 and D-131 are also credited for limiting ingress from combined effects flooding; these doors will not be subject to hydrostatic loading and calculations show they will withstand postulated hydrodynamic loading.

Components associated with lightning protection systems were credited for screening of the lightning hazard.

ii. *If there are any SSCs credited for screening of the external hazard(s), then explain and justify how the guidance in Figure 5-6 of NEI 00-04 will be applied for each of the external hazard(s).*

SNC Response:

Regarding Question 10.d.ii, SSCs credited for screening of the external hazards will follow the guidance in Figure 5-6 of NEI 00-04. During categorization, SSCs will be assessed for their participation in screened scenarios and the impact of their removal will be considered by the IDP; the component will then follow the LSS candidate or HSS path accordingly.

e) *If the external hazards (i.e., high winds and tornados) cannot be screened out in item (a) above, discuss, using quantitative or qualitative assessments, how the risk from those hazards will be considered in the categorization program. The discussion should include consideration of and, as applicable, the basis for the following factors:*

- *The frequency of the external hazard(s),*
- *The impact of the external hazard(s) on plant SSCs and plant's operation including the ability to respond to the external hazard initiating event,*
- *The operating experience associated with reliability of the external hazard(s) protection measures, and*
- *The reliability of operator actions.*

SNC Response:

As indicated in the response to Item 10.b. above, other external hazards, including the extreme winds and tornados hazards, were screened out. Therefore, additional discussion is not provided in response to this RAI.

RAI 11 (APLB) – Seismic importance measures

NEI 00-04, Section 5.6, "Integral Assessment," discusses the need for an integrated computation using the available importance measures. It further states, in part, that the "integrated importance measure essentially weights the importance from each risk contributor (e.g., IE, FPRA, and SPRAs) by the fraction of the total CDF [or LERF] contributed by that contributor." The guidance provides formulas to compute the integrated Fussell-Vesely (FV), and integrated Risk Achievement Worth (RAW).

To confirm that the importance measures generated for use in the 10 CFR 50.69 process are consistent with the NEI guidance and do not inadvertently introduce a deviation from the computations for FV and RAW provided in the NEI 00-04 guidance, as endorsed by RG 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" (ADAMS Accession No. ML061090627), address the following:

Discuss how the individual importance measures (i.e., FV and RAW) for the PRA model are derived considering the different hazards, specifically for those hazards that discretized SSC functions into 'bins'. The discussion should include justification of why the importance measures generated do not deviate from the NEI guidance. If the practice or method used to generate the integrated importance measures is determined to deviate from the NEI guidance, justify why the integrated importance measures computed are appropriate for use in the categorization process.

SNC Response:

Currently the only hazard for that derives individual importance measures using a binning approach for Plant Hatch is the SPRA. This approach calculates the FV and RAW measures for a component for each seismic acceleration interval, and then develops overall seismic importance values (for FV and RAW) using the following weighted process to combine the importance values over all seismic acceleration intervals.

a. For a component/basic event, the FV and RAW are calculated by ACUBE 2.0 for each of the 14 seismic acceleration intervals, resulting in 14 FV and RAW importance values by interval.

b. The interval FV values are weighted based on the seismic acceleration interval CDF divided by the total seismic CDF, and summed together for each seismically failed fragility group to obtain the total FV from the seismic failure. This is essentially using the integrated FV formula given in Section 5.6 of NEI 00-04. (Note that the seismic LOSP is removed from the importance analysis since it is virtually assured for all seismic sequences and cutsets, and does not correspond to an explicit component.)

c. The RAW values are weighted and summed similarly to the FV importance values, using the integrated RAW formula given in Section 5.6 of NEI 00-04.

d. The FV of the seismic failure is then combined with the FV of the random failures for that component to get a complete picture of the SPRA FV importance measure for that component.

e. The maximum of the RAW for seismically induced failure and RAWs of random failures for that component is used to get a complete picture of the SPRA RAW importance measure.

A similar process and weighting are used for LERF importance measures. Thus, the formulae in NEI 00-04 for performing an integral assessment, while not specifically identified for calculation of the SPRA importance values, can be used for the SPRA.

RAI 12 (APLA) – Implementation Items

10 CFR 50.69(b)(2)(ii) 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 evaluates 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 any of the 50.69 RAI 01 through RAI 11 above or the responses to any of the requests for additional information related to the NFPA-805 application 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 modeling 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. An example would be a table of listed implementation items referenced in a license condition.

As an alternative to providing an implementation item for an F&O, demonstrate that the F&O(s) will have no adverse impact and/or insignificant impact on the 10 CFR 50.69 categorization process.

SNC Response:

SNC is unable to provide this response until after SNC develops the response to remaining open items described in the May 28, 2019 NFPA-805 RAI response letter. The response to this RAI will be provided after SNC responds to remaining NFPA-805 RAIs.