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PSEG
Nuclear LLC

10 CFR 50.90
10 CFR 50.69

LR-N19-0076
LAR H19-07

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

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

SUBJECT: Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors"

In accordance with the provisions of 10 CFR 50.69 and 10 CFR 50.90, PSEG Nuclear LLC (PSEG) is requesting an amendment to the license of Hope Creek Generating Station (HCGS).

The proposed amendment would modify the HCGS licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The enclosure to this letter provides the basis for the proposed change to the HCGS Operating License. The categorization process being implemented through this change is consistent with NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0 dated July 2005, which was endorsed by the NRC in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1 dated May 2006. Attachment 1 of the enclosure provides a list of categorization prerequisites. Use of the categorization process on a plant system will only occur after these prerequisites are met.

The Full Power Internal Events PRA model described within this LAR is the same as that described with the NRC's issuance of Amendment No. 215 for an inverter allowed outage time extension (ADAMS Accession Number ML19065A156) dated March 27, 2019. The fire PRA model has undergone a finding and observation (F&O) closure process, which included a

focused scope peer review and a periodic update since it was used for Amendment No. 215. The NRC observed implementation of the independent team's F&O closure process for the associated fire PRA model as described in NRC's memorandum dated October 12, 2018 (ADAMS Accession Number ML18269A252). PSEG requests that the NRC utilize the review of the PRA technical adequacy for that application when performing the review for this application.

PSEG requests approval of the proposed license amendment within one year of submittal acceptance. Once approved, the amendment shall be implemented within 60 days.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the State of New Jersey.

This letter contains no regulatory commitments.

If you should have any questions regarding this submittal, please contact Mr. Lee Marabella at (856) 339-1208.

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

Executed on 11-25-2019
(date)

Sincerely,

Jean Fleming

Jean A. Fleming
Director Site Regulatory Compliance

Enclosure:
1. Evaluation of the Proposed Change

cc: NRC Project Manager
NRC Region I Administrator
NRC Senior Resident Inspector, Hope Creek
Mr. P. Mulligan, NJBNE
Mr. L. Marabella, Corporate Commitment Tracking Coordinator
Mr. T. Cachaza, Station Commitment Coordinator

Enclosure
Evaluation of the Proposed Change

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1 SUMMARY DESCRIPTION

The proposed amendment modifies the licensing basis to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

2 DETAILED DESCRIPTION

2.1 CURRENT REGULATORY REQUIREMENTS

The Nuclear Regulatory Commission (NRC) has established a set of regulatory requirements for commercial nuclear reactors to ensure that a reactor facility does not impose an undue risk to the health and safety of the public, thereby providing reasonable assurance of adequate protection to public health and safety. The current body of NRC regulations and their implementation are largely based on a "deterministic" approach.

This deterministic approach establishes requirements for engineering margin and quality assurance in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design basis events (DBEs). The deterministic approach then requires that the facility include safety systems capable of preventing or mitigating the consequences of those DBEs to protect public health and safety. The Structures, Systems and Components (SSCs) necessary to defend against the DBEs are defined as "safety-related," and these SSCs are the subject of many regulatory requirements, herein referred to as "special treatments," designed to ensure that they are of high quality and high reliability, and have the capability to perform during postulated design basis conditions. Treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. The distinction between "treatment" and "special treatment" is the degree of NRC specification as to what must be implemented for particular SSCs or for particular conditions. Typically, the regulations establish the scope of SSCs that receive special treatment using one of three different terms: "safety-related," "important to safety," or "basic component." The terms "safety-related "and "basic component" are defined in the regulations, while "important to safety," used principally in the General Design Criteria (GDC) of Appendix A to 10 CFR Part 50, is not explicitly defined.

2.2 REASON FOR PROPOSED CHANGE

A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety significance, and allowing consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, Probabilistic Risk Assessments (PRAs) address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed,

including the potential for common cause failures. The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner.

To take advantage of the safety enhancements available through the use of PRA, in 2004 the NRC published a new regulation, 10 CFR 50.69. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The rule contains requirements on how a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, as described by NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline” [1], which uses both risk insights and traditional engineering insights. The safety functions include the design basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability, and is a function of the SSC categorization results and associated bases. Finally, periodic assessment activities are conducted to make adjustments to the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable requirements.

The rule does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, the rule enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. For SSCs that are categorized as high safety significant, existing treatment requirements are maintained or enhanced. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, the rule allows an alternative risk-informed approach to treatment that provides reasonable, though reduced, level of confidence that these SSCs will satisfy functional requirements.

Implementation of 10 CFR 50.69 will allow PSEG to improve focus on equipment that has safety significance resulting in improved plant safety.

2.3 DESCRIPTION OF THE PROPOSED CHANGE

PSEG proposes the addition of the following condition to the renewed facility operating license of Hope Creek Generating Station (HCGS) to document the NRC's approval of the use 10 CFR 50.69.

> PSEG is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for

Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the EPRI alternative approach described in EPRI 3002012988 for seismic risk for Tier 1 plants; as specified in License Amendment No. [XXX] dated [DATE].

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

3 TECHNICAL EVALUATION

10 CFR 50.69 specifies the information to be provided by a licensee requesting adoption of the regulation. This request conforms to the requirements of 10 CFR 50.69(b)(2), which states:

> A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:
>
> (i) A description of the process for categorization of RISC–1, RISC–2, RISC–3 and RISC–4 SSCs.
>
> (ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.
>
> (iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i).
>
> (iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).

Each of these submittal requirements are addressed in the following sections.

The Full Power Internal Events PRA model described within this LAR is the same that described with the NRC's issuance of Amendment No. 215 for an inverter allowed outage time extension (ADAMS Accession Number ML19065A156) dated March 27, 2019. The fire PRA model has undergone an F&O closure process, which included a focused scope peer review and a periodic update since it was used for Amendment No. 215. The NRC observed implementation of the independent team's F&O closure process for the associated fire PRA model as described in NRC's memorandum to Hope Creek dated October 12, 2018 (ADAMS Accession Number ML18269A252).

3.1 CATEGORIZATION PROCESS DESCRIPTION (10 CFR 50.69(b)(2)(i))

3.1.1 Overall Categorization Process

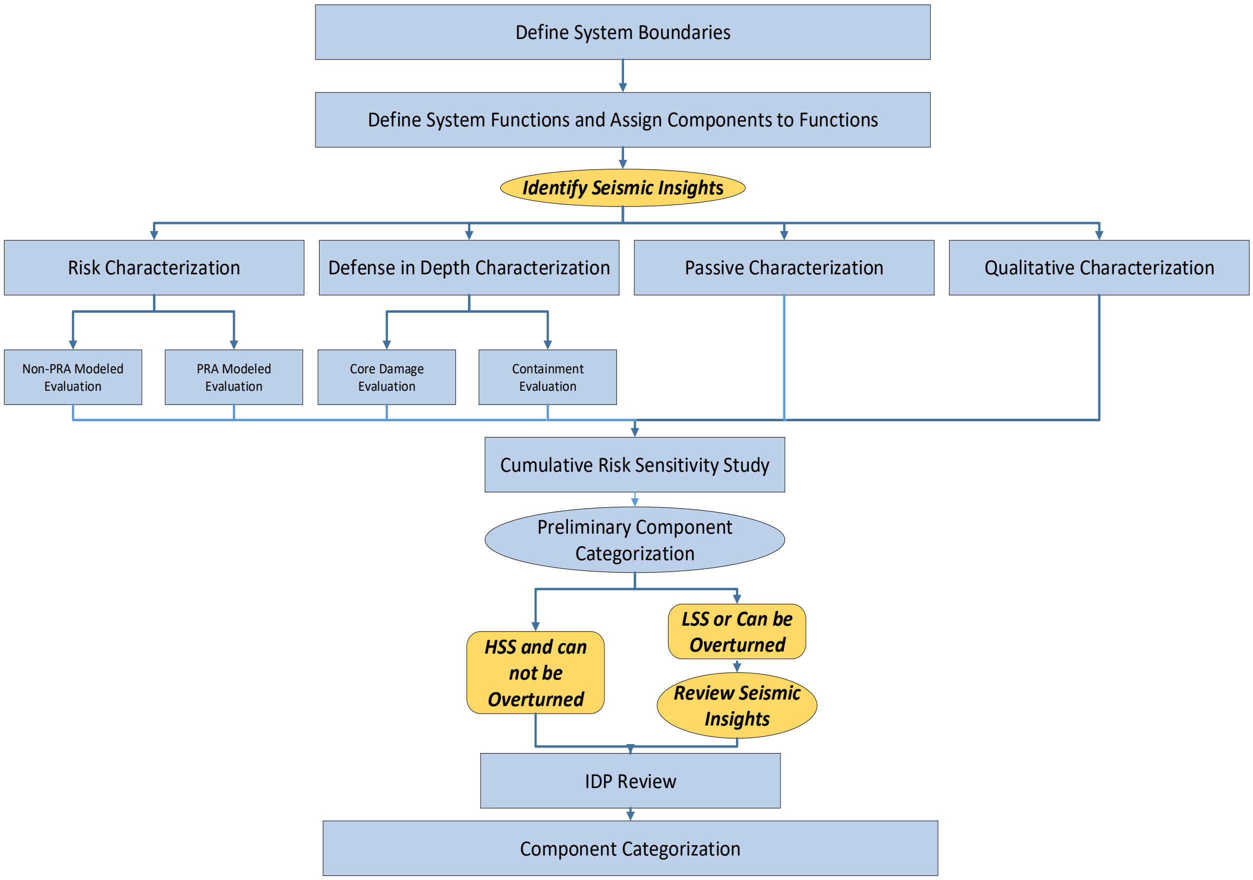
PSEG will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" [2]. NEI 00-04 Section 1.5 states "Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is assessed separately from each of five risk perspectives and used to identify SSCs that are potentially safety- significant." A separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The process to categorize each system will be consistent with the guidance in NEI 00-04,"10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201, with the exception of the evaluation of impact of the seismic hazard, which will use the EPRI 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10CFR50.69 Risk-Informed Categorization" [3] approach for seismic Tier 1 sites, which includes HCGS, to assess seismic hazard risk for 50.69. Inclusion of additional process steps discussed below to address seismic considerations will ensure that reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv) is achieved. RG 1.201 states that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(I)(iv)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. Therefore, the order in which each of the elements of the categorization process (listed below) is completed is flexible, and as long as they are all completed, they may even be performed in parallel. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as Low Safety Significant (LSS) by all other elements. Similarly, NEI 00-04 only requires Item 4 to be completed for safety-related active components/functions categorized as LSS by all other elements.

1. PRA-based evaluations (e.g., the internal events, internal flooding, and fire PRAs)
2. non-PRA approaches (e.g., other external events screening, and shutdown assessment)
3. Seven qualitative criteria in Section 9.2 of NEI 00-04
4. the defense-in-depth assessment
5. the passive categorization methodology

Figure 3-1 is an example of the major steps of the categorization process described in NEI 00-04; two steps (represented by four blocks on the figure) have been included to highlight review of seismic insights as pertains to this application, as explained further in Section 3.2.3:

Figure 3-1: Categorization Process Overview



Categorization of SSCs will be completed per the NEI 00-04 process, as endorsed by RG 1.201, which includes the determination of safety significance through the various elements identified above. The results of these elements are used as inputs to arrive at a preliminary component categorization (i.e., HSS or LSS that is presented to the Integrated Decision-Making Panel (IDP). Note: the term "preliminary HSS or LSS" is synonymous with the NEI 00-04 term "candidate HSS or LSS." A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 3-1 below. The safety significance determination of each element, identified above, is independent of each other and therefore the sequence of the elements does not impact the resulting preliminary categorization of each component or function. Consistent with NEI 00-04, the categorization of a component or function will only be "preliminary" until it has been confirmed by the IDP. Once the IDP confirms that the categorization process was followed appropriately, the final RISC category can be assigned.

The IDP may direct and approve detailed categorization of components in accordance with NEI 00-04 Section 10.2. The IDP may always elect to change a preliminary LSS component or function to HSS, however the ability to change component categorization from preliminary HSS to LSS is limited. This ability is only available to the IDP for select process steps as described in NEI 00-04 and endorsed by RG 1.201. Table 3-1 summarizes these IDP limitations in NEI 00-04. The steps of the process will be performed at either the function level, component level,

or both. This is also summarized in Table 3-1. A component is assigned its final RISC category upon approval by the IDP.

Table 3-1: Categorization Evaluation Summary

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
Risk (PRA Modeled)	Internal Events Base Case – Section 5.1	Component	Not Allowed	Yes
	Fire, Seismic and Other External Events Base Case		Allowable	No
	PRA Sensitivity Studies		Allowable	No
	Integral PRA Assessment – Section 5.6		Not Allowed	Yes
Risk (Non-modeled)	Fire and Other External Hazards –	Component	Not Allowed	No
	Seismic –	Function/Component	Allowed [2]	No
	Shutdown – Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage – Section 6.1	Function/Component	Not Allowed	Yes
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable[1]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

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

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

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

[2] IDP consideration of seismic insights can also result in an LSS to HSS determination.

The mapping of components to system functions will be used in some categorization process steps to facilitate preliminary categorization of components. Specifically, functions with mapped components that are determined to be HSS by the PRA-based assessment (i.e., Internal Events PRA or Integral PRA assessment) or defense-in-depth evaluation will be initially treated as HSS. However, NEI 00-04 Section 10.2 allows detailed categorization which can result in some components mapped to HSS functions being treated as LSS; and Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with a HSS function but which do not support the critical attributes of that HSS function. Note that certain steps of the categorization process will be performed at a component level (e.g. Passive, Non-PRA-modeled hazards – see Table 3-1). These components from the component level assessments will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped. Components having seismic functions may be HSS or LSS based on the IDP's consideration of the seismic insights applicable to the system being categorized. Therefore, if a HSS component is mapped to a LSS function, that component will remain HSS.

If an LSS component is mapped to an HSS function, that component may be driven HSS based on Table 3-1 above or may remain LSS. For the seismic hazard, given that HCGS is a seismic Tier 1 (low seismic hazard) plant (Reference [4]) as defined in Reference [3], seismic considerations are not required to drive an HSS determination at the component level, but the IDP will consider available seismic information pertinent to the components being categorized and can, at its discretion, determine that a component should be HSS based on that information.

The following are clarifications to be applied to the NEI 00-04 categorization process:

- The IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and probabilistic risk assessment. At least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in the modeling and updating of the plant-specific PRA.

- The IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address at a minimum the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the defense-in-depth philosophy and requirements to maintain this philosophy.

- The decision criteria for the IDP for categorizing SSCs as safety significant or low safety-significant pursuant to § 50.69(f)(1) will be documented in PSEG procedures. Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. However, a simple majority of the panel is sufficient for final decisions regarding high safety significant (HSS) and Low Safety Significant (LSS).

- Passive characterization will be performed using the processes described in Section 3.1.2. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

- An unreliability factor of 3 will be used for the sensitivity studies described in Section 8 of NEI 00-04. The factor of 3 was chosen as it is representative of the typical error factor of basic events used in the PRA model.

- NEI 00-04 Section 7 requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5 but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. This requirement is further clarified in the Vogtle SER [5] which states "…if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS."

- Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The IDP must intervene to assign any of these HSS Function components to LSS.

- With regard to the criteria that consider whether the active function is called out or relied upon in the plant Emergency/Abnormal Operating Procedures, PSEG will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

- PSEG proposes to apply an alternative seismic approach to those listed in NEI 00-04 Sections 1.5 and 5.3. This approach is specified in EPRI 3002012988 [3] for Tier 1 plants with the additional considerations discussed in Section of this LAR.

The risk analysis to be implemented for each hazard is described below.

- Internal Event Risks: Internal events including internal flooding PRA model version HC117A, January 2018.

- Fire Risks: Fire PRA model version HC119F0, June 2019.

- Seismic Risks: EPRI Alternative Approach in EPRI 3002012988 [3] for Tier 1 plants.

- Other External Risks (e.g., tornados, external floods): Using the IPEEE screening process as approved by NRC SE dated July 26, 1999 (TAC No. M83630) [6]. The other external hazards were determined to be insignificant contributors to plant risk.

- Low Power and Shutdown Risks: Qualitative defense-in-depth (DID) shutdown model for shutdown Configuration Risk Management (CRM) based on the framework for DID provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management" [7], which provides guidance for assessing and enhancing safety during shutdown operations.

A change to the categorization process that is outside the bounds specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach) will not be used without prior NRC approval. The SSC categorization process documentation will include the following elements:

1. Program procedures used in the categorization
2. System functions, identified and categorized with the associated bases
3. Mapping of components to support function(s)
4. PRA model results, including sensitivity studies
5. Hazards analyses, as applicable
6. Passive categorization results and bases
7. Categorization results including all associated bases and RISC classifications
8. Component critical attributes for HSS SSCs
9. Results of periodic reviews and SSC performance evaluations
10. IDP meeting minutes and qualification/training records for the IDP members

3.1.2 Passive Categorization Process

For the purposes of 10 CFR 50.69 categorization, passive components are those components that have a pressure retaining function. Passive components and the passive function of active components will be evaluated using the Arkansas Nuclear One (ANO) Risk-Informed Repair/Replacement Activities (RI-RRA) methodology contained in Reference [8] (ML090930246) consistent with the related Safety Evaluation (SE) issued by the Office of Nuclear Reactor Regulation.

The RI-RRA methodology is a risk-informed safety classification and treatment program for repair/replacement activities (RI-RRA methodology) for pressure retaining items and their associated supports. In this method, the component failure is assumed with a probability of 1.0 and only the consequence evaluation is performed. It additionally applies deterministic considerations (e.g., DID, safety margins) in determining safety significance. Component supports are assigned the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

The use of this method was previously approved to be used for a 10 CFR 50.69 application by NRC in the final Safety Evaluation for Vogtle dated December 17, 2014 [5]. The RI-RRA method as approved for use at Vogtle for 10 CFR 50.69 does not have any plant specific aspects and is generic. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, this RI-RRA process categorizes components solely based on consequence, which measures the safety significance of the passive component given that it ruptures. This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment. The passive categorization process is intended to apply the same risk-informed process accepted by the NRC in the ANO2-R&R-004 for the passive categorization of Class 2, 3, and non-class components. This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662 as published in Regulatory Guide 1.147, Revision 15. Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/ replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned high safety-significant, HSS, for passive categorization which will result in HSS for its risk-informed safety classification. and cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at HCGS for 10 CFR 50.69 SSC categorization.

3.2 TECHNICAL ADEQUACY EVALUATION (10 CFR 50.69(b)(2)(ii))

The following sections demonstrate that the quality and level of detail of the processes used in categorization of SSCs will be adequate. The PRA models described below have been peer reviewed and there are no PRA upgrades that have not been peer reviewed.

3.2.1 Internal Events and Internal Flooding

The HCGS categorization process for the internal events and flooding hazard will use the plant-specific PRA model. The PSEG risk management process ensures that the PRA model used in

this application reflects the as-built and as-operated plant. Attachment 2 of this enclosure identifies the applicable internal events and internal flooding PRA models.

3.2.2 Fire Hazards

The HCGS categorization process for fire hazards will use a peer reviewed plant-specific fire PRA model. The internal Fire PRA model was developed consistent with NUREG/CR-6850 and utilizes methods previously accepted by the NRC. The PSEG risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant. Attachment 2 at the end of this enclosure identifies the applicable Fire PRA model.

3.2.3 Seismic Hazards

10 CFR 50.69(c)(1) requires the use of PRA to assess risk from internal events. For other risk hazards such as seismic, 10 CFR 50.69(b)(2) allows, and NEI 00-04 summarizes, the use of other methods for determining SSC functional importance in the absence of a quantifiable PRA (such as Seismic Margin Analysis or IPEEE Screening) as part of an integrated, systematic process. For the HCGS seismic hazard assessment, PSEG proposes to use a risk informed graded approach that meets the requirements of 10 CFR 50.69(b)(2) as an alternative to those listed in NEI 00-04 sections 1.5 and 5.3. This approach is specified in Reference [3] and includes additional qualitative considerations that are discussed in this section.

HCGS meets the EPRI 3002012988 Tier 1 criteria for a “Low Seismic Hazard/High Seismic Margin” site. The Tier 1 criteria are as follows:

> Tier 1: Plants where the GMRS [Ground Motion Response Spectrum] peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE [Safe Shutdown Earthquake] between 1.0 Hz and 10 Hz. Examples are shown in Figures 2-1 and 2-2. At these sites, the GMRS is either very low or within the range of the SSE such that unique seismic categorization insights are not expected.
>
> > Note: EPRI 3002012988 applies to the Tier 1 sites in its entirety except for the sections 2.3 (Tier 2 sites), 2.4 (Tier 3 sites), Appendix A (seismic correlation), and Appendix B (criteria for capacity-based screening).

The Tier 1 criterion (i.e. basis) in EPRI 3002012988 is a comparison of the ground motion response spectrum (GMRS, derived from the seismic hazard) to the safe shutdown earthquake (SSE, i.e., seismic design basis capability). U.S. nuclear power plants that utilize the 50.69 Seismic Alternative (EPRI 3002012988) will continue to compare GMRS to SSE.

The trial studies in EPRI 3002012988 show that seismic categorization insights are overlaid by other risk insights even at plants where the GMRS is far beyond the seismic design basis. Therefore, the basis for the Tier 1 classification and resulting criteria is not that the design basis insights are adequate. Instead, it is that consideration of the full range of the seismic hazard produces limited unique insights to the categorization process. That is the basis for the following statements in Table 4-1 of the EPRI report.

> At Tier 1 sites, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low.

Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the FPIE PRA and other risk evaluations along with the required Defense-in-Depth and Integrated Decision-making Panel (IDP) qualitative considerations are expected to adequately identify the safety-significant functions and SSCs required for those functions and no additional seismic reviews are necessary for 50.69 categorization.

The proposed categorization approach for HCGS is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic PRA. For Tier 1 plants, this approach relies on the insights gained from the seismic PRAs examined in Reference [3] along with confirmation that the site GMRS is low (Reference [4]). Reference [3] demonstrates that seismic risk is adequately addressed for Tier 1 sites by the results of additional qualitative assessments discussed in this section and existing elements of the 50.69 categorization process specified in NEI 00-04.

For example, the 50.69 categorization process as defined in NEI 00-04 includes an Integral Assessment that weighs the hazard-specific relative importance of a component (e.g., internal events, internal fire, seismic) by the fraction of the total Core Damage Frequency (CDF) contributed by that hazard. The risk from an external hazard can be reduced from the default condition of HSS if the integral assessment meets the importance measure criteria for LSS. For Tier 1 sites, the seismic risk (CDF/LERF) will be low such that seismic hazard risk is unlikely to influence an HSS decision. In applying the EPRI 3002012988 process for Tier 1 sites to the HCGS 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the EPRI 3002012988 guidance and informed of plant SSC-specific seismic insights for their consideration in the HSS/LSS deliberations.

EPRI 3002012988 recommends a risk-informed graded approach for addressing the seismic hazard in the 50.69 categorization process. There are a number of seismic fragility fundamental concepts that support a graded approach and there are important characteristics about the comparison of the seismic design basis (represented by the SSE) to the site-specific seismic hazard (represented by the GMRS) that support the selected thresholds between the three evaluation Tiers in the EPRI report. The coupling of these concepts with the categorization process in NEI 00-04 is the key element of the approach defined in EPRI 3002012998 for identifying unique seismic insights.

The seismic fragility of a SSC is a function of the margin between an SSC's seismic capacity and the site-specific seismic demand. References such as EPRI NP-6041-SL [9] provide inherent seismic capacities for most SSCs that are not directly related to the site-specific seismic demand. This inherent seismic capacity is based on the non-seismic design loads (pressure, thermal, dead weight, etc.) and the required functions for the SSC. For example, a pump has a relatively high inherent seismic capacity based on its design and that same seismic capacity applies at a site with a very low demand and at a site with a very high demand. At sites with lower seismic demands such as HCGS, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference [9]. Low seismic demand sites have lower likelihood of seismically-induced failures and lesser challenges to plant systems. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazards at HCGS.

There are some plant features such as equipment anchorage that have seismic capacities more closely associated with the site-specific seismic demand since those specific features are specifically designed to meet that demand. However, even for these features, the design basis criteria have intended conservatisms that result in significant seismic margins within SSCs. These conservatisms are reflected in key aspects of the seismic design process. The SSCs used in nuclear power plants are intentionally designed using conservative methods and criteria to ensure that they have margins well above the required design bases. Experience has shown that design practices result in margins to realistic seismic capacities of 1.5 or more.

The following provides the basis for establishing Tier 1 criteria in EPRI 3002012988.

a. SSCs for which the inherent seismic capacities are applicable, or which are designed to the plant SSE will have low probabilities of failure at sites where the peak spectral acceleration of the GMRS < 0.2g or where the GMRS < SSE between 1 and 10 Hz.

b. The low probabilities of failure of individual components would also apply to components considered to have correlated seismic failures.

c. These low probabilities of failure lead to low seismic CDF and LERF estimates, from an absolute risk perspective.

d. The low seismic CDF and LERF estimates lead to reasonable confidence that seismic risk contributions would allow reducing a HSS to LSS due to the 50.69 Integral Assessment if the equipment is HSS only due to seismic considerations.

Test cases described in Section 3 of Reference [3] showed that it would be unusual even for moderate hazard plants to exhibit any unique seismic insights, including due to correlated failures. The plant specific Reference [3] test case information PSEG is using from the other licensees and being incorporated by reference into this application is described in Case Study A (Reference [10]), Case Study C (References [11] and [12], and Case Study D (References [13], [14], [15]). Hence, while it is prudent to perform additional evaluations to identify conditions where correlated failures may occur for Tier 2 sites, for Tier 1 sites such as HCGS, correlation studies would not lead to new seismic insights or affect the baseline seismic CDF in any significant way.

The Tier 1 to Tier 2 threshold as defined in EPRI 3002012988 provides a clear and traceable boundary that can be consistently applied plant site to plant site. Additionally, because the boundary is well defined, if new information is obtained on the site hazard, a site's location within a particular Tier can be readily confirmed. In the unlikely event that the HCGS seismic hazard changes to medium risk (i.e., Tier 2) at some future time, PSEG will follow its categorization review and adjustment process procedures to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e).

The following provides the basis for concluding that HCGS meets the Tier 1 site criteria.

In response to the NRC 50.54(f) letter associated with post-Fukushima recommendations (Reference [16], HCGS submitted a seismic hazard screening report (Reference [4]) to the NRC. HCGS meets the second of the Tier 1 definition criteria (GMRS < SSE in 1-10 Hz range). The HCGS SSE and GMRS curves from the seismic hazard and screening response in Reference [4] are shown in Figure 1 of Attachment 4.

The NRC's staff assessment of the HCGS seismic hazard and screening response is documented in Reference [17]. In section 3.4 of Reference [17] the NRC concluded that the methodology used by PSEG in determining the GMRS was acceptable and that the GMRS determined by PSEG adequately characterizes the reevaluated hazard for the HCGS site.

Section 1.1.3 of Reference [3] cites various post-Fukushima seismic reviews performed for the U.S. fleet of nuclear power plants. For HCGS, the specific seismic reviews prepared by the licensee and the NRC's staff assessments are provided here.

1. NTTF Recommendation 2.1 seismic hazard screening [4], [17].
2. NTTF Recommendation 2.3 seismic walkdowns [18], [19], [20].
3. NTTF Recommendation 4.2 seismic mitigation strategy assessment (S-MSA) [21], [22].

The following additional post-Fukushima seismic reviews were performed for HCGS.

4. NTTF Recommendation 2.1 seismic high frequency evaluation [23], [24].
5. NTTF Recommendation 2.1 screening and prioritization results [25]

The small percentage contribution of seismic to total plant risk makes it unlikely that an integral importance assessment for a component, as defined in NEI 00-04, would result in an overall HSS determination. Further, the low hazard relative to plant seismic capability makes it unlikely that any unique seismic condition would exist that would cause an SSC to be designated HSS for a Tier 1 site such as HCGS.

As an enhancement to the EPRI study results as they pertain to HCGS, the proposed HCGS categorization approach for seismic hazards will include qualitative consideration of the mitigation capabilities of SSCs during seismically-induced events and seismic failure modes, based on insights obtained from prior seismic evaluations performed for HCGS. For example, as part of the categorization team's preparation of the System Categorization Document (SCD) that is presented to the IDP, a section will be included in the SCD that summarizes the identified plant seismic insights pertinent to the system being categorized, and will also state the basis for applicability of the EPRI 3002012988 study and the bases for HCGS being a Tier 1 plant. The discussion of the Tier 1 bases will include such factors as:

- The low seismic hazard for the plant, which is subject to periodic reconsideration as new information becomes available through industry evaluations; and

- The definition of Tier 1 in the EPRI study.

At several steps of the categorization process (e.g., as noted in Figure 3-1 and Table 3-1) the categorization team will consider the available seismic insights relative to the system being categorized and document their conclusions in the SCD. Integrated importance measures over all modeled hazards (i.e., internal events, including internal flooding, and internal fire for HCGS) are calculated per Section 5.6 of NEI 00-04, and components for which these measures exceed the specified criteria are preliminary HSS which cannot be changed to LSS. For SSCs not uniquely identified as HSS by the HCGS PRA models but having design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events, these will be addressed using non-PRA based

qualitative assessments in conjunction with any available seismic insights provided by the seismic insights review described below.

For components that are HSS due to fire PRA but not HSS due to internal events PRA, the categorization team will review design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events and characterize these for presentation to the IDP as additional qualitative inputs, which will also be described in the SCD.

The categorization team will review available HCGS plant-specific seismic reviews and other resources such as those identified above. The objective is to identify plant-specific seismic insights derived from the above sources, relevant to the components in the system being categorized, that might include potentially important impacts such as:

- Impact of relay chatter
- Implications related to potential seismic interactions such as with block walls
- Seismic failures of passive SSCs such as tanks and heat exchangers
- Any known structural or anchorage issues with a particular SSC
- Components that are implicitly part of PRA-modeled functions (including relays)
- Components that may be subject to correlated failures

Such impacts would be compiled on an SSC basis. As each system is categorized, the system-specific seismic insights will be provided to the IDP for consideration as part of the IDP review process, as noted in Figure 3-1. As such, the IDP can challenge, from a seismic perspective, any candidate LSS recommendation for any SSC if they believe there is basis for doing so. Any decision by the IDP to downgrade preliminary HSS components to LSS will also consider the applicable seismic insights in that decision. These insights will provide the IDP a means to consider potential impacts of seismic events in the categorization process.

Use of the EPRI approach to assess seismic hazard risk for 50.69 with the additional reviews discussed above will ensure that reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv) is achieved.

Based on the above, the Summary/Conclusion/Recommendation from Section 2.2.3 of Reference [3] applies to HCGS, i.e., HCGS is a Tier 1 plant for which the GMRS is very low such that unique seismic categorization insights are expected to be minimal. As discussed in Reference [3], the likelihood of identifying a unique seismic insight that would cause an SSC to be designated HSS is very low. Therefore, with little to no anticipated unique seismic insights, the 50.69 categorization process using the Full Power Internal Events (FPIE) PRA and other risk evaluations along with the defense in-depth and qualitative assessment by the IDP adequately identifies the safety significant functions and SSCs.

3.2.4 Other External Hazards

The HCGS categorization process will use screening results from the IPEEE in response to Generic Letter (GL) 88-20 [47] for evaluation of safety significance related to the Extreme Wind or Tornado hazard. Figure 5-6 in Section 5.4 of NEI 00-04 illustrates the process that will be used to determine safety significance related to the Extreme Wind or Tornado hazard.

All other external hazards were screened for applicability to HCGS per a plant-specific evaluation in accordance with GL 88-20 [47] and updated to use the criteria in ASME PRA

Standard RA-Sa-2009. Attachment 1 provides a categorization prerequisite for screening of the extreme wind or tornado hazard. Attachment 4 provides a summary of the other external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

3.2.5 Low Power & Shutdown

Consistent with NEI 00-04, the HCGS categorization process will use the shutdown safety management plan described in NUMARC 91-06 for evaluation of safety significance related to low power and shutdown conditions. The overall process for addressing shutdown risk is illustrated in Figure 5-7 of NEI 00-04.

NUMARC 91-06 specifies that a defense-in-depth approach should be used with respect to each defined shutdown key safety function. The key safety functions defined in NUMARC 91-06 are evaluated for categorization of SSCs.

SSCs that meet either of the two criteria (i.e., considered part of a “primary shutdown safety system” or a failure would initiate an event during shutdown conditions) described in Section 5.5 NEI 00-04 will be considered preliminary HSS.

3.2.6 PRA Maintenance and Updates

The PSEG risk management process ensures that the applicable PRA model(s) used in this application continues to reflect the as-built and as-operated plant for HCGS. The process delineates the responsibilities and guidelines for updating the PRA models, and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, and industry operational experience) for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process will assess the impact of these changes on the plant PRA model in a timely manner but no longer than once every two refueling outages. If there is a significant impact on the PRA model, the SSC categorization will be re-evaluated.

In addition, PSEG will implement a process that addresses the requirements in NEI 00-04, Section 11, “Program Documentation and Change Control.” The process will review the results of periodic and interim updates of the plant PRA that may affect the results of the categorization process. If the results are affected, adjustments will be made as necessary to the categorization or treatment processes to maintain the validity of the processes. In addition, any PRA model upgrades will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.2.7 PRA Uncertainty Evaluations

Uncertainty evaluations associated with any applicable baseline PRA model(s) used in this application were evaluated during the assessment of PRA technical adequacy and confirmed through the self-assessment and peer review processes as discussed in Section 3.3 of this enclosure.

Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in Section 8 of NEI 00-04 and in the prescribed sensitivity studies discussed in Section 5.

In the overall risk sensitivity studies, PSEG will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference [5]. Consistent with the NEI 00-04 guidance, PSEG will perform both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in all identified PRA models for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study.

The detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 [26] and Section 3.1.1 of EPRI TR-1016737 [27]. The process in these references was mostly developed to evaluate the uncertainties associated with the internal events PRA model; however, the approach can be applied to other types of hazard groups.

The list of assumptions and sources of uncertainty were reviewed to identify those which would be significant for the evaluation of this application. If the HCGS PRA model used a non-conservative treatment, or methods that are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine its impact on this application. Only those assumptions or sources of uncertainty that could significantly impact the risk calculations were considered key for this application.

Key HCGS PRA model specific assumptions and sources of uncertainty for this application were identified and dispositioned in Attachment 6. The conclusion of this review is that no additional sensitivity analyses are required to address HCGS PRA model specific assumptions or sources of uncertainty.

3.3 PRA REVIEW PROCESS RESULTS (10 CFR 50.69(b)(2)(iii))

The PRA models described in Section 3.2 have been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 [28] consistent with NRC RIS 2007-06.

The internal events PRA model was subject to a self-assessment and a full-scope peer review conducted in March 2009. Subsequent to the peer review, all open F&Os were addressed in the internal events PRA model (HC111A). The internal events PRA model (HC117A) was subject to a F&O closure review conducted in early 2017 and all open findings were closed. There are no remaining findings or open items in the internal events PRA model.

The Fire PRA model (HC108BF0) was subject to a self-assessment and a full-scope peer review conducted in October 2010. The fire PRA model (HC114F0) was updated to address the peer review F&Os in 2015 and again in 2018 (HC118F0). An F&O Closure review was conducted on the identified Fire PRA model (HC118F0) in September 2018 that was observed by the NRC [29]. Findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "Close-out of Facts and Observations" (F&Os) [30] as accepted by NRC in the letter dated May 3, 2017 (ML17079A427) [31]. The results of this review have been documented and are available for NRC audit. All except eight of the Findings were considered closed and met at least CC II as part of the F&O Closure. A Focused Scope Peer Review was also performed for selected elements from HRA, FSS, and PRM in September 2018 on the HC118F0 model. Twenty-one Findings were generated from this Focused Scope Peer Review. A Subsequent F&O Closure was performed in 2019 on HC119F0 Fire PRA Model and all twenty-one Findings were closed. There are no remaining findings or open items in the Fire PRA model.

All the remaining findings and open items were closed, demonstrating that the PRA is of sufficient quality and level of detail to support the categorization process, and has been subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC as required 10 CFR 50.69(c)(1)(i).

3.4 RISK EVALUATIONS (10 CFR 50.69(b)(2)(iv))

The HCGS 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04. The overall risk evaluation process described in the NEI guidance addresses both known degradation mechanisms and common cause interactions, and meets the requirements of §50.69(b)(2)(iv). Sensitivity studies described in NEI 00-04 Section 8 will be used to confirm that the categorization process results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF). The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, and human errors). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data, and provide timely insights into the need to account for any important new degradation mechanisms.

3.5 FEEDBACK AND ADJUSTMENT PROCESS

If significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can (or actually did) prevent a safety significant function from being satisfied, an immediate evaluation and review will be performed prior to the normally scheduled periodic review. Otherwise, the assessment of potential equipment performance changes and new technical information will be performed during the normally scheduled periodic review cycle.

To more specifically address the feedback and adjustment (i.e., performance monitoring) process as it pertains to the proposed HCGS Tier 1 approach discussed in section 3.2.3, implementation of the PSEG design control and corrective action programs will ensure the inputs for the qualitative determinations for seismic continue to remain valid to maintain compliance with the requirements of 10 CFR 50.69(e).

The performance monitoring process will be described in PSEG's 10 CFR 50.69 program documents. The program will require that the periodic review assess changes that could impact the categorization results and will provide the Integrated Decision-making Panel (IDP) with an opportunity to recommend categorization and treatment adjustments. Station personnel from engineering, operations, risk management, regulatory affairs, and others will have responsibilities for preparing and conducting various performance monitoring tasks that feed into this process. The intent of the performance monitoring reviews is to discover trends in component reliability; to help catch and reverse negative performance trends and take corrective action if necessary.

The HCGS configuration control process ensures that changes to the plant, including a physical change to the plant and changes to documents are evaluated to determine the impact to drawings, design bases, licensing documents, programs, procedures, and training.

HCGS has a comprehensive problem identification and corrective action program that ensures that issues are identified and resolved. Any issue that may impact the 10 CFR 50.69 categorization process will be identified and addressed through the problem identification and corrective action program, including seismic-related issues.

The HCGS 10 CFR 50.69 program will require that SCDs cannot be approved by the IDP until the panel's comments have been resolved to the satisfaction of the IDP. This includes issues related to system-specific seismic insights considered by the IDP during categorization.

Scheduled periodic reviews no longer than once every two refueling outages will evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process will be updated. This review will include:

- A review of plant modifications since the last review that could impact the SSC categorization.
- A review of plant specific operating experience that could impact the SSC categorization.
- A review of the impact of the updated risk information on the categorization process results.
- A review of the importance measures used for screening in the categorization process.
- An update of the risk sensitivity study performed for the categorization.

In addition to the normally scheduled periodic reviews, if a PRA model or other risk information is upgraded, a review of the SSC categorization will be performed.

The periodic monitoring requirements of the 10 CFR 50.69 process will ensure that these issues are captured and addressed at a frequency commensurate with the issue severity. The 10 CFR 50.69 periodic monitoring program will include immediate and periodic reviews, that include the requirements of the regulation, to ensure that all issues that could affect 10 CFR 50.69 categorization are addressed. The periodic monitoring process will also monitor the performance and condition of categorized SSCs to ensure that the assumptions for reliability in the categorization process are maintained.

4 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The following NRC requirements and guidance documents are applicable to the proposed change.

- The regulations in Title 10 of the Code of Federal Regulations (10 CFR) Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors."
- NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
- Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018.
- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

The proposed change is consistent with the applicable regulations and regulatory guidance.

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

PSEG proposes to modify the licensing basis to allow for the voluntary implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors.” The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

PSEG has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of Structures, Systems and Components (SSCs) subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The process used to evaluate SSCs for changes to NRC special treatment requirements and the use of alternative requirements ensures the ability of the SSCs to perform their design function. The potential change to special treatment requirements does not change the design and operation of the SSCs. As a result, the proposed change does not significantly affect any initiators to accidents previously evaluated or the ability to mitigate any accidents previously evaluated. The consequences of the accidents previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition following an accident will continue to perform their design functions.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not change the functional requirements, configuration, or method of operation of any SSC. Under the proposed change, no additional plant equipment will be installed.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not affect any Safety Limits or operating parameters used to establish the safety margin. The safety margins included in analyses of accidents are not affected by the proposed change. The regulation requires that there be no significant effect on plant risk due to any change to the special treatment requirements for SSCs and that the SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, PSEG concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 CONCLUSIONS

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6 REFERENCES

[1] NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute, July 2005.

[2] NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.

[3] Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10CFR50.69 Risk-Informed Categorization," July 2018.

[4] PSEG Nuclear LLC's Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident - Hope Creek Generating Station, March 28, 2014 (ML14087A436).

[5] NRC letter to Southern Nuclear Operating Company, "Issuance of Amendments Re: Use of 10 CFR 50.69 (TAC Nos. ME9472 and ME94473)," December 17, 2014 (ADAMS Accession No ML14237A034).

[6] Review of Individual Plant Examination of External Events (IPEEE) Submittal for Hope Creek Generating Station (TAC NO. M83630), July 26, 1999.

[7] NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.

[8] ANO SER Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative AN02-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems, (TAC NO. MD5250) (ML090930246), April 22, 2009.

[9] Electric Power Research Institute (EPRI) NP-6041-SL, "A Methodology for Assessment of Nuclear Plant Seismic Margin, Revision 1," August 1991.

[10] Peach Bottom Atomic Power Station Seismic Probabilistic Risk Assessment Report, "Response to NRC Request Regarding Recommendation 2.1 of the Near Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," August 28, 2018 (ML18240A065).

[11] Plant C, License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process,, June 22, 2017 (ML17173A875).

[12] Plant C, "Issuance of Amendments Regarding Application of Seismic Probabilistic Risk Assessment Into the Previously Approved 10 CFR 50.69 Categorization Process (EPID L-2017-LLA-0248)", August 10, 2018 (ML18180A062).

[13] Seismic Probabilistic Risk Assessment for Plant D Nuclear Plant, Units 1 and 2, "Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the NTTF Review of Insights from the Fukushima Dai-ichi Accident," June 30, 2017 (ML17181A485).

[14] Plant D Nuclear Plant Seismic Probabilistic Risk Assessment Supplemental Information, April 10, 2018 (ML18100A966).

[15] Plant D Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," November 29, 2018 (ML18334A363).

[16] U.S. Nuclear Regulatory Commission, Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(F) Regarding Recommendations 2.1,2.3, And 9.3, of

the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident, March 12, 2012 (ML12053A340).

[17] U.S. Nuclear Regulatory Commission (NRC) Staff Assessment of Seismic Hazard Evaluations for NTTF 2.1, "Seismic" (CAC NO. MF3924), February 29, 2016 (ML16049A609).

[18] PSEG Nuclear LLC Seismic Walkdown Report SL-011553, January 29, 2014 (ML14056A345).

[19] Seismic Walkdown Report Update - Hope Creek Generating Station Response to Recommendation 2.3: Seismic Walkdown of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident, February 7, 2014 (ML14056A344).

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

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[49] Electric Power Research Institute (EPRI)Technical Report TR-1026511, "Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," December 2012.

Attachment 1: List of Categorization Prerequisites

1. PSEG will establish procedure(s) prior to the use of the categorization process on a plant system. The procedure(s) will contain the elements/steps listed below.

   - Integrated Decision-Making Panel (IDP) member qualification requirements

   - Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary High Safety Significant (HSS) or Low Safety Significant (LSS) based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.2). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting, an LSS function are categorized as preliminary LSS.

   - Component safety significance assessment. Safety significance of active components is assessed through a combination of Probabilistic Risk Assessment (PRA) and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.

   - Assessment of defense-in-depth (DID) and safety margin. Safety-related components that are categorized· as preliminary LSS are evaluated for their role in providing DID and safety margin and, if appropriate, upgraded to HSS.

   - Review by the IDP. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.

   - Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of Regulatory Guide 1.174.

   - Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized.

   - Documentation requirements per Section 3.1.1 of the enclosure.

2. Regarding external hazard tornado missile risk, in order to ensure current efforts to respond to RIS 2015-06 are reflected in the 10 CFR 50.69 categorization process, all necessary actions (e.g., analyses, modifications, etc.) will be completed to allow the hazard to be screened according to the screening process described in Section 5.4 and Figure 5-6 of NEI 00-04. All SSCs credited for screening of extreme wind and tornados will be categorized HSS, and the basis for that conclusion will be identified.

Attachment 2: Description of PRA Models Used in Categorization

Full Power Internal Events / Internal Flooding PRA Model				
Unit	**Model**	**Baseline CDF**	**Baseline LERF**	**Comments**
1	HC117A January 2018 Peer Reviewed against RG 1.200 R1 in March 2009	5.9E-06 per year	1.8E-07 per year	Gap Assessment to RG 1.200 R2 Documented (Attachment 7)
Fire PRA Model				
Unit	**Model**	**Baseline CDF**	**Baseline LERF**	**Comments**
1	HC119F0 June 2019 Peer Reviewed against RG 1.200 R2 in October 2010	3.7E-05 per year	6.7E-06 per year	

Attachment 3: Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
This attachment is intentionally blank. There are no open peer review findings and self-assessment open items for the internal events / internal flooding or fire PRA models.				

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Aircraft Impact	Y	PS2 PS4	An updated evaluation of aircraft hazards is discussed in Section 3.5.1.6 of the Early Site Permit (ESP) Site Safety Analysis Report (SSAR) [32]. The SSAR evaluates the suitability of an adjacent site to HCGS for future construction and operation of a nuclear power plant and is appropriate to assess the hazard impact to HCGS. There are seven airports and a helipad between 8 and 16.1 km (5 and 10 mi) of the location of the proposed plant at the PSEG Site. The airports have a very small infrequent number (sporadic) of flights annually. The UFSAR [33] and SSAR concluded that aircraft hazard impacts would not contribute to exceeding the acceptable aircraft hazards frequency of 10-7 per year **(PS2, PS4)**, and therefore are not considered a negligible safety hazard.
Avalanche	Y	C3	Not applicable to the site because of climate and topography **(C3)**.
Biological Event	Y	C5	Slow developing hazard, environmental programs and procedures are in place to detect and manage the hazard and to periodically inspect and clean affected systems **(C5)**.
Coastal Erosion	Y	C3 C5	Hazard is slow to develop **(C5)**. Per UFSAR Section 2.4.10 [33], shore protection extends 100 feet north and south of the intake structure to assure that no blockage to the water intake will occur and that erosion will not impede the operation of the service water pipes **(C3)**.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Drought	Y	C5	Per the IPEEE [34], drought is a slowly developing hazard and is not applicable to the site **(C5)**.
External Flooding	Y	C1	By letter dated June 27, 2017 (ML17178A307), HCGS submitted its flooding focused evaluation (FE) for Hope Creek. In a related follow-up letter dated July 13, 2017 (ML17194A460), HCGS stated that that the re-evaluated external flooding hazard demonstrated that flooding above plant grade can occur only for the Local Intense Precipitation (LIP), storm surge, and probable maximum flood in conjunction with a storm surge. In its technical evaluation, the NRC found that only the LIP flooding mechanism was found to exceed the plant's current design basis. [35]. However, for both LIP and storm surge events the mitigation response is to close watertight doors prior to water reaching the door sill. Watertight doors are the only active flood protection features. Therefore, since certain watertight doors are credited with screening of this hazard, these SSCs will be considered HSS during categorization of systems containing these doors **(C1)**.
Extreme Wind or Tornado	Y	C1 PS4	Per UFSAR 2.3.1.2.3 [33], the design basis tornado has a maximum wind speed of 360 mph. Per Table 6-1 of NUREG/CR 4461 [36], the 1E-6 probability tornado wind speed is 205 mph, based on the F-scale, and 166 mph, based on the more recent EF-scale **(C1)**. Since tornado winds dominate the extreme wind hazard, and the 1E-6/yr

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			tornado wind speeds are much less than the design value for the tornado maximum wind speed of 360 mph, damage due to the forces associated with high winds and tornadoes can be screened **(PS4)**. Regarding tornado missile hazards, in order to ensure current efforts to respond to RIS 2015-06 are reflected in the 10 CFR 50.69 categorization process, a categorization prerequisite has been added to Attachment 1 of this application to complete necessary actions (e.g., analyses, modifications, etc.) to screen tornado missile hazards prior to the adoption of 10 CFR 50.69. If tornado missile protection vulnerabilities are discovered as part of assessing tornado missile protection in response to RIS 2015-06, then this information will be used to update the screening process. Per the screening process described in Section 5.4 and Figure 5-6 of NEI 00-04, all SSCs credited for screening of extreme wind and tornados will be categorized HSS, and the basis for that conclusion will be identified.
Fog	Y	C1 C4	The plant design considers this hazard **(C1)**. The principal effects of fog on the plant would be to cause a loss of off-site power which is addressed in the weather-related Loss of Offsite Power initiating events in the internal events PRA model **(C4)**.
Forest or Range Fire	Y	C4	Per ESP SSAR Section 2.3.1.1 [32], within a distance of approximately 8 km (5 mi) surrounding the PSEG site, the ground surface is primarily marsh. The most significant consequence of forest or range fire would be a loss of offsite power (LOOP) which is evaluated in the internal events PRA model **(C4)**.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Frost	Y	C1 C4	The plant design considers this hazard **(C1)**. There is negligible impact on the plant due to frost. The worst-case impact is frost induced freezing leading to a loss of off- site power event which is evaluated in the internal events PRA model **(C4)**.
Hail	Y	C1 C4	The plant design considers this hazard **(C1)**. Hail can accompany severe thunderstorms and can be a major weather hazard. The principal effects of such events would be occurrence of weather-related Loss of Offsite Power initiating events which are addressed in the internal events PRA model **(C4)**.
High Summer Temperature	Y	C1 C4	The plant design considers this hazard **(C1)**. Associated plant trips are rare and are covered in the weather-related Loss of Offsite Power initiating events which are addressed in the internal events PRA model **(C4)**.
High Tide, Lake Level, or River Stage	Y	C1	Per UFSAR Section 2.4.1.1 [33], HCGS is located within the tidally-affected portion of the Delaware Estuary System. Historical extremes in water-level occurred as a result of wind related tide level variations, and not as a result of fluvial discharges. In addition, Per UFSAR Section 2.4.1.1, HCGS is not susceptible to flooding by rivers, dam failures, ice flooding or channel migration. Per UFSAR 2.4.6.7, The estimated maximum wave height coincident with 10 percent exceedance high tide and 2-year extreme wind condition at the HCGS site location is 9.6 feet above the maximum stillwater level, which is at 6.0 feet MSL. The plant grade is at 12.5 feet MSL. Therefore, the effect of the maximum

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			wave's height on safety-related facilities above the plant grade is insignificant. Given the disproportionally high tidal flow conditions at the PSEG site, riverine-based flooding scenarios have been previously viewed to be inconsequential relative to Marine-derived ones. Based on these facts, HCGS is not susceptible to flooding by rivers, dam failures, ice flooding, or channel migration **(C1)**.
Hurricane	Y	C4	Per UFSAR Section 2.4.5.1 [33], the maximum probable hurricane wind speed is 132 mph. Tornado winds bound this hazard which is discussed in the Extreme Wind or Tornado hazard **(C4)**, and in the External Flooding Hazard **(C4)**.
Ice Cover	Y	C1 C4	Plant is designed for freezing temperatures **(C1)** and the principal effects of such events would be to cause a loss of off-site power, which is addressed in the weather-related Loss of Offsite Power (LOOP) initiating events in the internal events PRA model **(C4)**.
Industrial or Military Facility Accident	Y	C1 C3	There are no manufacturing, industrial chemical plants, or storage facilities or military facilities within 5 miles of the site **(C1)**. With the exception of Salem Generating Station (SGS), much of the area within 5 miles of the site is wetland area within which industrial development is prohibited. **(C3)**. There are no credible events at SGS that could impact HCGS.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Internal Flooding	N	N/A	The HCGS Internal Events PRA includes evaluation of risk from internal flooding events.
Internal Fire	N	N/A	The HCGS Internal Fire PRA includes evaluation of risk from internal fire events.
Landslide	Y	C3	Not applicable to the site because of topography **(C3)**.
Lightning	Y	C4	The principal effects of such events would be to cause a loss of off-site power or turbine trip and are addressed in the weather-related Loss of Offsite Power initiating events in the internal events PRA model for HCGS **(C4)**.
Low Lake Level or River Stage	Y	C3	This hazard is of negligible impact on the plant due to the large volume and tidal characteristics of the Delaware River **(C3)**. The plant location in the Delaware Estuary system precludes impact on the plant due to this hazard.
Low Winter Temperature	Y	C1 C5	The plant is designed for extended freezing events, and their impacts are slow to develop **(C1)**. The phenomenon provides large amount of time for preparation (weather forecast) with time for implementation of appropriate mitigation actions (e.g., plant power reduction or shutdown) **(C5)**.
Meteorite or Satellite Impact	Y	PS4	Likelihood of a large meteorite or satellite, large enough to cause significant plant damage, is very low **(PS4)**.
Pipeline Accident	Y	C3	Per UFSAR 2.2.3.1 [33], there are no major manufacturing or chemical plants

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			within 5 miles of the site and there are no pipelines within 10 miles of the site **(C3)**.
Release of Chemicals in Onsite Storage	Y	C4 PS1	Hazardous chemical evaluations have been performed for all of the chemicals stored onsite. The evaluation of bulk gases stored on-site concluded control room habitability would not be impacted during postulated releases due to relatively small storage containers, locations, high threshold values, and their ability to disperse rapidly in air. The impact of releases of chemicals in onsite storage do not pose a risk to the site **(PS1)**. See also “Transportation Accidents” **(C4)**.
River Diversion	Y	C3	Per UFSAR Section 2.4.9 [33], there is no historical or topographical evidence of channel diversions of significance in the Delaware River Basin **(C3)**.
Sand or Dust Storm	Y	C3	Plant site not located near sand dunes or other large sources of small airborne particles **(C3)**.
Seiche	Y	C1	Per the FHRR [37], flooding at HCGS due to seiche is not likely to occur. Per UFSAR Section 2.4.5 [33], large amplitude oscillations from seiche or resonance flooding are not possible, because the most probable forcing mechanisms identified lack either a period of oscillation close enough to the fundamental period of the Delaware Estuary to be of concern, or a magnitude and duration great enough to supply a significant amount of energy into the basin. In addition, energy dissipation of any water level oscillation occurs by

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
			frictional damping and reflection along the banks of the estuary **(C1)**.
Seismic Activity	N/A	None	See Section 3.2.3 and Figure 1 in this Attachment.
Snow	Y	C4	The principal effect of snow would be to cause a loss of off-site power which is addressed in the weather-related Loss of Offsite Power initiating events in the internal events PRA model for HCGS **(C4)**. See also External Flooding.
Soil Shrink-Swell Consolidation	Y	C5	Per UFSAR 2.5.1.2.5 [33], at the site no soils were detected that might be unstable because of their mineralogy or unstable physical and chemical properties **(C5)**.
Storm Surge	Y	C1 C4	Flooding from the Probable Maximum Storm Surge was reevaluated in the FHRR [37]. Storm Surge was considered bounded by the design basis of the plant **(C1)** and is also addressed under External Flooding **(C4)**.
Toxic Gas	Y	C4	Toxic gas covered under release of chemicals in onsite storage, industrial or military facility accident, and transportation accident **(C4)**.
Transportation Accident	Y	C1 C3	Water Transportation: The Delaware River is a major route for barge and freight traffic between the Philadelphia area ports and the Atlantic Ocean. Per UFSAR 2.2.3 [33], detailed studies of hazards of ship transportation have been performed. Collisions to the intake structure or explosions on the river are extremely unlikely and therefore are not

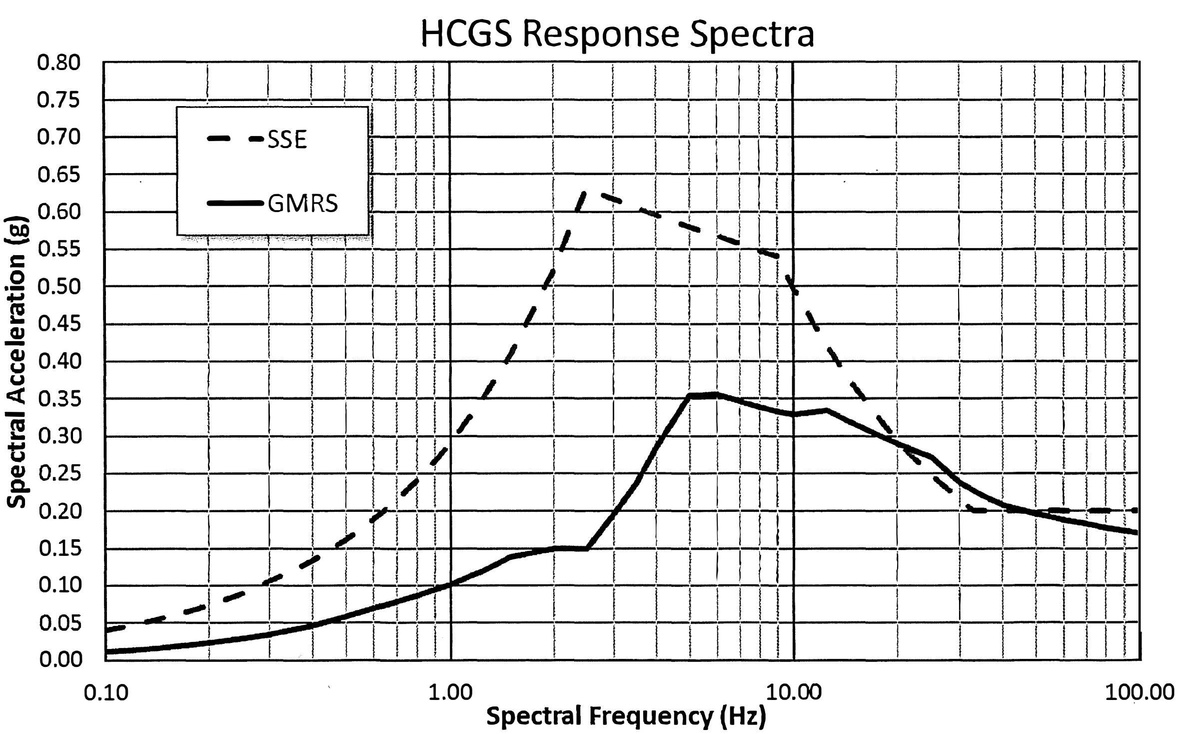
External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			of concern **(C1)**. Land and Rail Transportation: There is no major highway or railroad located within 5 miles of the plant [34] **(C3)**. Chemical hazards stored and transported in the vicinity of the plant were analyzed. The analysis concluded that toxic chemicals transported (or stored) within the vicinity of the plant, do not pose a threat [34] **(C1)**.
Tsunami	Y	C3	Per Reference [37] Section 2.6, the probable maximum tsunami (PMT) event does not lead to flooding anywhere on the PSEG Site. The site is not subject to drawdown effects or velocity effects due to PMT that require further analysis. Therefore, the tsunami event is not a viable flood causing mechanism at the PSEG Site **(C3)**.
Turbine-Generated Missiles	Y	PS4	Per UFSAR Section 3.5.1.3 [33], the original LP rotors on the HCGS turbine generator set were replaced with monoblock rotor forgings. If the unit trips, valves fail to operate and full flow steam remains, the maximum possible speed the rotors can obtain is about 220% running speed. The rotor overspeed capability, with the assumption that all buckets remain in place, is 225% for typical rotor strengths. If the unit trips, valves fail to operate and full flow steam remains, the maximum possible speed the rotors can obtain is about 220% Therefore, rotor missiles will not be generated. A complete failure of the control and safety systems is required for rotor missiles to be generated. The probability of a control failure of this nature is approximately 10^{-8} per year **(PS4)**.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Volcanic Activity	Y	C3	Not applicable to the site because of location **(C3)**.
Waves	Y	C1	Per UFSAR 2.4.3.6 [33], the maximum wave run-up height estimated is 20.8 feet MSL. Per UFSAR 1.2.1.6, all Seismic Category I structures are flood protected and structurally designed to withstand the static and dynamic effects of a flood with coincident waves up to Elevation 31.4 feet MSL **(C1)**.
Note a: See Attachment 5 for descriptions of the screening criteria.			

Figure 1: GMRS and SSE Response Spectra for HCGS

(From Reference [4])



Attachment 5: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is < events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
	PS3. Design basis event mean frequency is < 1E-5/y and the mean conditional core damage probability is < 0.1.	NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009	
	PS4. Bounding mean CDF is < 1E-6/y.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Event Analysis	Criterion	Source	Comments
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty

The HCGS internal events and fire PRA models and documentation were reviewed for plant-specific modeling assumptions and related sources of uncertainty. Reference [38] and Reference [39] document sources of PRA modeling uncertainty. They identify assumptions and determine if those assumptions are related to sources of model uncertainty and characterize that uncertainty, as necessary. The identified uncertainties were reviewed for this application.

Each PRA model includes an evaluation of the potential sources of uncertainty for the base case models using the approach that is consistent with the ASME/ANS RA-Sa-2009 [40] requirements for identification and characterization of uncertainties and assumptions. This evaluation identifies those sources of uncertainty that are important to the PRA results and may be important to PRA applications which meets the intent of steps C-1 and E-1 of NUREG-1855,Revision 1 [26].

The results of the base PRA evaluations were reviewed to determine which potential uncertainties could impact the 50.69 categorization process results. This evaluation meets the intent of the screening portion of steps C-2 and E-2 of NUREG-1855, Revision 1.

For the 50.69 Program, the guidance in NEI 00-04 [1] specifies sensitivity studies to be conducted for each PRA model to address key sources of uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask the SSC(s) importance. Regulatory Guide 1.174, Revision 3 [41] cites NUREG-1855, Revision 1, 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. The results of the evaluation of PRA model sources of uncertainty as described above are evaluated relative to the 50.69 application in Attachment 6 to determine if additional sensitivity evaluations are needed.

> Note: As part of the required 50.69 PRA categorization sensitivity cases directed by NEI 00-04, internal events / internal flood and fire PRA models’ human error and common cause basic events are increased to their 95th percentile and also decreased to their 5th percentile values. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs and CCFs are accounted for in the 50.69 application.

The table below describes the internal events / internal flooding (IE / IF) PRA sources of model uncertainty and their impact.

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
Digital Feedwater Control Failure Probabilities.	Impacts Feedwater System Logic Model. The reliability analysis for causing plant trips performed by similar FW vendor studies is assumed to be equally applicable to the reliability of the system post plant trips that are caused by other means that do not directly affect the feedwater availability.	Basic events representing the reliability values for the auto level controller, the field buses, false signal from the redundant reactivity control system, and false signal from the Level 8 trip system are included in the system logic model. The difference between digital and analog equipment unreliability's is not significant. The functional reliability is driven by other components.
RPV Water Level Indication	Impacts level indication needed in many scenarios. The modeling evaluates that the Control Room indication is available and corrected by the crew as part of the HEP for RPV injection and emergency depressurization. The model includes an operator action to monitor the RPV water level using the Fuel Zone (FZ) indicators.	Failure to obtain these readouts would force an emergency depressurization approximately 1 hour earlier than if the readouts were available. This affects the level of offsite AC recovery that can be credited. Control Room crew has all of the water level indication needed to implement EOPs effectively. In addition, because of the alternate readouts for the FZ not dependent on AC power and the small impact on offsite AC recovery time, the impact is minimal.
FLEX equipment failure probability	The model includes the FLEX equipment important to SBO scenarios. The FLEX equipment provides power to A and B channels, air to the SRVs, and an independent RPV injection method. The failure probabilities for the FLEX equipment are doubled based on the latest generic failure data.	Failure to use the FLEX equipment would primarily affect SBO sequences where ELAP should be declared. There are alternate SBO sequences where, if ELAP is not successfully declared, a success path without FLEX equipment may be available. The modeling uses equipment unavailability's and HEPs that

IE / IF PRA Sources of Assumption/ Uncertainty	IE / IF PRA 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (50.69)
		account for the fact that the site may not control the FLEX equipment properly or perform periodic inspections or maintenance on the equipment in a similar manner to other plant equipment. PSEG believes that FLEX equipment is well maintained, operators are well trained, thus this treatment is conservative. Quantifying the Hope Creek PRA with FLEX equipment in the model provides the best representation of the as-built, as operated plant. The PSEG PRA maintenance process requires looking for updated equipment reliability data during each PRA update.
FLEX Human Error Probabilities	NRC and industry PRA practitioners questions whether the HRA approach used in the Hope Creek PRA is adequate. No widely accepted state of the practice method exists.	Current HRA practices have not been validated for the activities needed to install and operate portable equipment. PSEG believes that their approach to estimate HEPs provides reasonably conservative probability estimates. Quantifying the Hope Creek PRA with FLEX HEPs in the model provides the best representation of the as-built, as operated plant. The PSEG PRA maintenance process requires looking for updated and improved HRA practices during each PRA update. If an updated or improved HRA practice is determined to be an upgrade, a focused-scope peer review may be required.
Interfacing System Loss Of Coolant Accident (ISLOCA) IE Frequency Determination	Impacts ISLOCA initiating event sequences Detailed ISLOCA analysis includes the relevant considerations listed in IE-C12	One ISLOCA initiating event frequency is implemented in the model representing the sum of all the individual flow paths analyzed for rupture initiating event frequency.

IE / IF PRA Sources of Assumption/ Uncertainty	**IE / IF PRA 50.69 Impact**	**IE / IF PRA Model Sensitivity and Disposition (50.69)**
	of the ASME/ANS PRA Standard and accounts for common cause failures and captures likelihood of different piping failure modes.	Unique contributions from each flow path included in the model via a multiplier on the total ISLOCA initiating event frequency to delineate that fraction of system unavailability from the initiating event. In addition, as part of the 50.69 categorization process, components associated with either initiating or providing a significant level of mitigation for an ISLOCA event are required to be evaluated for their risk significance as part of the containment defense-in-depth assessment.

The table below describes the fire PRA sources of model uncertainty and their impact.

Fire PRA Description	**Fire PRA Sources of Uncertainty**	**Fire PRA Disposition**
Fire PRA Component Selection	This task is associated with the development of the linkage between safe shutdown analysis component / cable data to fault tree failure modes. Also included in this task is the development and incorporation of Multiple Spurious Operation (MSO) scenarios not addressed in the Internal Events model fault tree.	The uncertainty associated with this task is related to the identification of all components that should be credited/linked in the FPRA. This source of uncertainty is reduced as a result of multiple overlapping tasks including the MSO expert panel, reviews of FPIE screened initiating events, screened containment penetrations, and screened ISLOCA scenarios. Additional internal reviews of analysis results further reduce the uncertainty associated with this task. MSOs typically appear in cutsets (scenarios) with little or no mitigating equipment, but with important human errors. Therefore, uncertainties in the application and quantification of MSOs will not affect the parameters used for 50.69 categorization. See the discussion on Post Fire Human Reliability Analysis.
Fire PRA Cable Selection	No treatment of uncertainty is typically required for this task beyond the understanding of the cable selection approach (i.e., mapping an active basic event to a passive component for which power cables were not selected).	Assumed routing was not performed for the fire PRA except as detailed and justified in section 3.3.2 of the Cable Selection Notebook [42]. Previous assumed cable routing was removed from the model and replaced with a UNL (unknown location) approach. Thus, if any component did not have known cable routing, it was assumed to be failed in all scenarios until detailed routing information was obtained. In some cases, a cutset review identified a partial refinement to unknown cable locations that would qualify as partial assumed routing (e.g. cables between components in the

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
		Reactor Building were assumed not to route through buildings in the Yard). Such cases were identified as manual exclusions. A detailed list of manual exclusion and inclusions is contained in Appendix F of Reference [42]. The manual exclusions list in Appendix F also contains justifications for assumed cable routing for UNLs (this list also contains exclusions for other reasons that are described in that table). Therefore, it is not feasible for cables to be routed through buildings in the Yard." A sensitivity on the UNL cables was performed to examine the effect of all UNL cables excluded from the quantification (i.e., all UNL cables are NOT affected by fire). The reduction in CDF and LERF was less than 18%. This is a limiting case demonstrating the maximum theoretical benefit from the UNL cables, which cannot be fully realized if cable routing was actually implemented into the Fire PRA.
Fire-Induced Risk Model	The construction of the FPRA model itself is a source of uncertainty. The FPIE model readiness as a starting point for the FPRA is a source of uncertainty. At a minimum, a turbine trip is assumed for each fire scenario. This is conservative since not all fires postulated will result in a plant trip.	FPIE and FPRA peer reviews, internal assessments, and the PRA cutset reviews are useful for minimizing uncertainty associated with modeling errors. Implementation of the Fire Induced Initiating Event logic ensures that the correct Internal Events Initiating Event is selected for each fire scenario. The logic uses the fire-induced equipment failures for each scenario to trigger the appropriate initiator. This approach has reduced the uncertainty associated with this task.
Quantitative	Other than screening out	Quantitative screening was not

Fire PRA Description	**Fire PRA Sources of Uncertainty**	**Fire PRA Disposition**
Screening	potentially risk-significant scenarios (ignition sources), there is no uncertainty from this task on the Fire PRA results.	performed for the fire PRA.
Scoping Fire Modeling	This task was accomplished with the use of fire modeling treatments in lieu of a conservative scoping analysis technique. The primary conservatism introduced by this task is associated with the heat release rates specified in NUREG/CR 6850 [43] for non-electrical enclosure ignition sources.	The employment of fire modeling solutions (as opposed to more generic scoping fire modeling) has the potential to reduce conservatisms. Conservatisms, however, remain. For example, inclusion of secondary combustibles has the potential to increase the likelihood of forming a hot gas layer (HGL). The conservatism associated with the fire modeling is generally judged to be reasonable and not overly conservative.
Detailed Circuit Failure Analysis	Uncertainty considerations for the circuit failure analysis task are addressed via the use of circuit failure mode probability factors in the Circuit Failure Model Likelihood Analysis (CFMLA) Task. No specific uncertainty is associated with the performance of the circuit analysis.	No specific uncertainty is associated with the performance of the circuit analysis, though the CFMLA notebook [44] identifies several assumptions in Section 2.2.
Circuit Failure Model Likelihood Analysis	The uncertainty associated with the applied conditional failure probabilities poses competing considerations is primarily due to the assumption that all spurious operations occur at the same time. The hot short probability and the hot short duration factors defined in NUREG/CR-7150 [45] are considered best available data.	Circuit failure mode likelihood analysis was generally limited to those components where spurious operation was expected to be a large contributor to total risk. The assumption that all spurious operations (hot shorts) occur at the same time results in a significant conservatism in the analysis but is not easily assessed with respect to the impact on the overall results.
Detailed Fire Modeling	The primary uncertainty in this task is in the area of target failure probabilities. Conservative heat release rates may result in additional target damage. Non-conservative heat release rates would have an opposite effect.	Fire modeling was used to evaluate the time to abandonment for control room fire scenarios for a range of fire heat release rates.

Fire PRA Description	Fire PRA Sources of Uncertainty	Fire PRA Disposition
	Credit for fire brigade response and detection are limited to the Multi-Compartment Analysis and the Hot Gas Layer evaluation.	
Post-Fire Human Reliability Analysis	Human error probabilities (HEPs) represent a potentially large uncertainty for the Fire PRA given the importance of human actions in the base model. Since many of the HEP values were adjusted for fire, the joint dependency multipliers developed for the FPIE model also represent a potential for introducing a degree of conservatism.	Generally conservative HEP adjustments were made to the nominal HEP values used in the FPIE model then revisited to address unique fire considerations. A detailed analysis was performed for all fire specific HFEs. A floor value of 1E-06 or 5E-7 was applied for identified dependent combinations. Uncertainty in HEP values is propagated through the parametric uncertainty analysis documented in Section 4.4.1 of Reference [39] and is further characterized by the HEP sensitivity documented in Section 4.4.2.6 of Reference [39]. Further, as directed by NEI 00-04, fire model human error basic events are increased to their 95th percentile and also decreased to their 5th percentile values as part of the required 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and respective functions HSS and therefore the uncertainty of the PRA modeled HEPs are accounted for in the 50.69 application.
Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit.	The Fire PRA solves for CDF and LERF at a truncation limit of 1E-11/yr. The quantification achieves a convergence of <5% over the final decade for CDF, there should not be a significant truncation contribution. This truncation limit is many orders of magnitude below the typical CDF values calculated

Fire PRA Description	**Fire PRA Sources of Uncertainty**	**Fire PRA Disposition**
		for each fire scenario. A convergence evaluation of the truncation limit used in the analysis is provided in Section 4.1 of Reference [39]. The LERF convergence is slightly above 5%, however this is judged to be adequate. The CDF and LERF truncations at 1.00E-11/yr are judged to be appropriate for assessing the Fire CDF for risk-informed applications. This is consistent with the NEI PRA Peer Review Guidelines which indicate that a truncation of four orders of magnitude below the CDF is adequate for a high quality PRA. Additionally, per the PRA Standard, truncation convergence may be achieved once a less than 5% change occurs between two decade truncations. The Fire truncation level is approximately six orders of magnitude below the CDF and LERF values. CDF values meet the “less than 5%” criterion at the 1E-11 level and LERF values are only slightly above 5%. The quantification time at 1E-12 is significantly longer, and the LERF results with a truncation of 1E-12 would not significantly alter the risk results and risk insights compared to a truncation of 1E-11

Attachment 7: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III

Since the peer review of the Internal Events PRA model was performed prior to the publication of RG 1.200 Rev 2, a self-assessment was conducted to assess the differences between RG 1.200 Rev 2 and RG 1.200 Rev 1. That assessment confirmed that the PRA model meets the requirements of RG 1.200 Rev 2. Results from that assessment are documented below.

Note: All supporting requirements affected by "key" assumptions and uncertainty have been gap assessed (per NEI 05-04) as described with the NRC's issuance of Amendment No. 215 for an inverter allowed outage time extension (ADAMS Accession Number ML19065A156) dated March 27, 2019. Each peer review performed on the HC PRA considers the appropriate version of the PRA standard and the current RG 1.200 clarifications.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
IE-C10:CC-I/II/III: **...** **An example of an acceptable generic data sources is NUREG/CR-5750 [Note 1]..**	IE-C12: CC-I/IIIIII: ... An example of an acceptable generic data sources is NUREG/CR-6928 Note 1.	The sentences were clarifications provided in RG 1.200 Revision 1 and Revision 2, respectively. The updated SR cites a more recent example of an acceptable generic data source.	NUREG/CR-6928 was used as a generic data source for the Hope Creek FPIE and Fire PRA models.
SY-B15: CC-I/II/III: **...** **(h) harsh environments induced by containment venting, or failure that may occur prior to the onset of core damage.**	SY-B14: CC-I/II/III: ... (h) harsh environments induced by containment venting, failure of the containment venting ducts, or failure of the containment boundary that may occur prior to the onset of core damage	The sentences were clarifications provided in RG 1.200 Revision 1 and Revision 2, respectively. The updated SR explicitly requires consideration of containment venting ducts and failure of the containment boundary prior to core damage.	Degraded environments including those caused by venting, containment failure, internal flooding, etc. are explicitly treated in system models or accident sequence dependent failures. For example, the Hope Creek PRA includes impacts of degraded environment on equipment such as: plugging of ECCS and SW strainers, loss of NPSH for ECCS pumps, etc.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
			In addition, the PRA includes impact of degraded environmental conditions on equipment operability beyond their environmental qualifications. These effects are properly treated in both the Level 1 and Level 2 PRA.
HR-D6: CC-I/II/III **PROVIDE an assessment of the uncertainty in the HEPs consistent with the quantification approach. USE mean values when providing point estimates of HEPs**	HR-D6: CC-I/II/III CHARACTERIZE the uncertainty in the estimates of the HEPs consistent with the quantification approach, and PROVIDE mean values for use in the quantification of the PRA results.	RG 1.200, Revision, 2 provides clarification that should be evaluated.	The HCGS HRA models characterize the uncertainty in the estimates of the human error probabilities (HEPs) consistent with the quantification approach and use mean values in the quantification of the PRA results. Uncertainty cases are also provided using the 50th and 95th percentiles of the HEPs.
HR-G3: CC-II/III **<u>CC-I:</u>** **USE an approach that takes the following into account:** **(a) the complexity of the response** **…** **The ASEP Approach is an acceptable approach.** **<u>CC-II/III:</u>** **When estimating HEPs**	HR-G3: <u>CC-I:</u> USE an approach that takes the following into account: (a) the complexity of detection, diagnosis, decision-making and executing the required response The ASEP Approach [2-6] is an acceptable approach.	RG 1.200, Revision 2, provided clarification to items (d) and (g) of the SR. Some of the RG 1.200, Revision 1 wording remains, while some additional clarification is provided.	The HCGS HRA models use the EPRI HRA calculator, which includes a discussion of the specific scenario to evaluate; the (d) degree of clarity of the cues/indications in supporting the detection, diagnosis, and decision-making give the plant clarification and scenario-specific context of the event,

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
EVALUATE the impact of the following plant-specific and scenario-specific performance shaping factors: **…** ***(d)* degree of clarity of the meaning of the cues/indications.**	CC-II/III: When estimating HEPs EVALUATE the impact of the following plant-specific and scenario-specific performance shaping factors: (d) degree of clarity of cues/indications in supporting the detection, diagnosis, and decision-making give the plan specific and scenario-specific context of the event.		and (g) complexity of detection, diagnosis, and decision-making and executing the required response.
DA-C1: CC-I/II/III: **...** **Examples of parameter estimates and associated sources include:** **(a) component failure rates and probabilities: NUREG/CR-4639 Note (1), NUREG/CR-4550 Note (2), NUREG-1715 Note 7**	DA-C1: CC-I/II/III: ... Examples of parameter estimates and associated sources include: (a) component failure rates and probabilities: NUREG/CR-4639 [2-7], NUREG/CR-4550 [2-3], NUREG-1715 [2-21], NUREG/CR-6928 [2-20]	Reference NUREG-1715 was added by RG 1.200 Revision 1; References NUREG-1715 and NUREG/CR-6928 were included in the 2009 version of the PRA Standard. The updated SR cites more recent examples of acceptable generic data sources.	Though additional examples of generic data were identified, they don't super cede the previous data source and will not impact the technical adequacy of the PRA.
DA-D8 (new SR) **CC-I/II/III:** **For each SSC for which repair is to be modeled, ESTIMATE, based on the data collected in DA-C14, the probability of failure to repair the SSC in time to prevent**	DA-D9 CC-I/II/III: For each SSC for which repair is to be modeled, ESTIMATE, based on the data collected in DA-C15 the probability of failure to repair the SSC in time to prevent core damage as a function of	RG 1.200, Revision 1, included a new SR -- DA-D8. The recommended new SR is included in RG 1.200, Revision 2, as DA-D9 (with the renumbering).	The HCGS PRA models only take credit for repairing the emergency diesel generators (EDGs) in the electric power recovery (EPR) model. This EPR model uses a convolution methodology to

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
core damage as a function of the accident sequence in which the SSC failure appears.	the accident sequence in which the SSC failure appears.		calculate the probability of recovering offsite power or repairing an EDG in time to prevent core damage as a function of the accident sequence in which the SSE failure appears.
QU-A2a: CC-I/II/III: **PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF ...**	QU-A2: CC-I/II/III: PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF (and LERF) ...	The LERF requirement was added by RG 1.200 Revision 2. The updated SR explicitly requires consideration of LERF for sequence quantification.	Sequence quantification for LERF may identify enhancements to be made in the LERF model for a more realistic estimate of LERF. However, as the sequence quantification is not used in the NEI 00-04 Risk Ranking methodology along with Defense-in-Depth considerations, not having LERF quantified at the sequence level will not impact the categorization results. The HCGS PRA models provide estimates of the individual sequences in a manner consistent with the estimation of CDF and LERF to identify significant accident sequences and confirm that the logic is appropriately reflected. These estimates are accomplished by quantifying the

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
			individual accident sequences.
QU-A2b: **CC-I:** **ESTIMATE the point estimate CDF from internal events.** **CC-II:** **ESTIMATE the mean CDF from internal events, accounting for the "state-of-knowledge" correlation between event probabilities Note (1).** **CC-III:** **CALCULATE the mean CDF from internal events by propagating the uncertainty distributions, ensuring that the "state-of-knowledge" correlation between event probabilities is taken into account.**	QU-A3: CC-I: ESTIMATE the point estimate CDF (and LERF). CC-II: ESTIMATE the mean CDF (and LERF) accounting for the "state-of-knowledge" correlation between event probabilities Note (1). CC-III: CALCULATE the mean CDF (and LERF) by propagating the uncertainty distributions, ensuring that the "state-of-knowledge" correlation between event probabilities is taken into account.	The phrase, "from internal events", was deleted from the 2009 version of the PRA Standard. The LERF requirement was added by RG 1 .200 Revision 2. The SR explicitly requires consideration of LERF.	Per the note in 2007 SR LE-E4 and LE-F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard was addressed these LERF requirements. The HCGS PRA models are quantified using PRAQuant. UNCERT is used to determine the mean CDF and LERF to be estimated by correlating event probabilities. When propagating uncertainty distributions, the CDF and LERF are estimated.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
QU-B5: **<u>CC-I/II/III:</u>** **Fault tree linking and some other modeling approaches may result in circular logic that must be broken before the model is solved. BREAK the circular logic appropriately. Guidance for breaking logic loops is provided in NUREG/CR-2728 [Note (1)]. When resolving circular logic, AVOID introducing unnecessary conservatisms or non-conservatisms.**	QU-B5: <u>CCI/II/III</u> Fault tree linking and some other modeling approaches may result in circular logic that must be broken before the model is solved. BREAK the circular logic appropriately. Guidance forbreaking logic loops is provided in NUREG/CR-2728 [2-13]. When resolving circular logic, DONOT introduce unnecessary conservatisms or nonconservatisms	RG 1.200, Revision 2, provides clarification that should be evaluated. Need to verify breaking logic loops does not result in undue conservatism.	Both RG 1.200, Rev. 1, Table A-1. "Staff Position on ASME RA-S-2002, ASME RA-Sa-2003, and ASME RA-Sb-2005," and RG 1.200, Rev. 2, Table A-2. "Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events" have "No objection" to SR QUB5. Furthermore, the HCGS PRA model logical loops are broken in a manner that still permits each dependency to be accounted for when quantified using event trees with conditional split fractions.
QU-B6: **<u>CC I/II/III:</u>** **ACCOUNT for system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF. This accounting may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation and includes the treatment of transfers among event trees where the**	QU-B6: <u>CC I/II/III:</u> ACCOUNT for system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF or LERF. This accounting may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation and includes the treatment of transfers among event trees where the	The LERF requirement was added by RG 1.200 Revision 2.	The SR explicitly requires consideration of LERF. However, per the note in 2007 SR LE E-4 and LE F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard was addressed these LERF requirements. The CAFTA event tree linking quantification process that is used by the HCGS PRA models account for

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
successes may not be transferred between event trees.	successes may not be transferred between event trees.		system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF and LERF. This accounting is accomplished by using numerical quantification of success probability. Since the event trees are linked, all "successes" are transferred between event trees.
QU-E3: **<u>CC-I:</u>** **ESTIMATE the uncertainty interval of CDF results. Provide a basis for the estimate consistent with the characterization parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15).** **<u>CC-II:</u>** **ESTIMATE the uncertainty interval of the CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), taking into account the state-of-knowledge correlation.** **<u>CC-III:</u>**	QU-E3: <u>CC-I:</u> ESTIMATE the uncertainty interval of CDF (and LERF) results. Provide a basis for the estimate consistent with the characterization parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15). <u>CC-II:</u> ESTIMATE the uncertainty interval of the CDF (and LERF) results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), taking into account the state-of-knowledge correlation. <u>CC-III:</u> Propagate parameter	The LERF requirement was added by RG 1.200 Revision 2.	The SR explicitly requires consideration of LERF. However, per the note in 2007 SR LE E-4 and LE F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard was addressed these LERF requirements. The HCGS PRA models take into account the "state of knowledge" correlation between selected parameter distributions, propagate these uncertainties through a Monte Carlo quantification using UNCERT, and

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
Propagate parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15)….(no change)	uncertainties (DA-D3, HR-D6, HR-G8, IE-C15)….(no change)		calculate the estimated CDF and LERF distributions.
<u>QU-E4:</u> **<u>CC-I:</u>** **PROVIDE an assessment of the impact of the model uncertainties and assumptions on the results of the PRA.** **<u>CC-II:</u>** **EVALUATE the sensitivity of the results to model uncertainties and key assumptions using sensitivity analyses Note (1).** **<u>CC-III:</u>** **EVALUATE the sensitivity of the results to uncertain model boundary conditions and other assumptions using sensitivity analyses except where such sources of uncertainty have been adequately treated in the quantitative uncertainty analysis**	<u>QU-E4:</u> <u>CC-I/II/III:</u> For each source of model uncertainty and related assumption identified in QU-E1 and QU-E2, respectively, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event).	Separate requirements for CC-I, II and III were collapsed into a single requirement for CC-I/II/III in the 2009 version of the PRA Standard. The reference to Note 1 was deleted by RG 1.200 Revision 2.	The updated SR assigns the same requirement to all three CCs. Meeting CC-II: in the 2007 version of the PRA Standard assures that the new SR is met. The HCGS PRA models identify sources of model uncertainty and their related assumptions, as well as how the PRA model is affected by these.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
Note (1).			
LE-F2: **CC-I:** **PROVIDE a qualitative assessment of the key sources of uncertainty. Examples:** **(a) Identify bounding assumptions.** **(b) Identify conservative treatment of phenomena.** **CC-II:** **PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies for the significant contributors to LERF.** **CC-III:** **PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies.**	LE-F3: CC-I/II/III: IDENTIFY and CHARACTERIZE the LERF sources of model uncertainty and related assumptions, in a manner consistent with the applicable requirements of Tables 2-2.7-2(d) and 2-2.7-2(e).	Separate requirements for CC-I, II, and III were collapsed into a single requirement for CC-I/II/III in the 2009 version of the PRA Standard.	The updated SR assigns the same requirement to all three CCs. Meeting CC-II: in the 2007 version of the PRA Standard assures that the new SR is met.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
	Flooding SRs: IFPP-B1, B2, B3, IFSO-B1, B2, B3, IFSN-B1, B2, B3, IFEV-B1, B2, B3, and IFQU-B1, B2, B3.	These are new requirements for flooding that expand on the original SRs in the ASME/ANS PRA Standard Generally described here – more detail below.	The HCGS Internal Flooding PRA model documentation is consistent with the SRs. All technical determinations of the internal flooding analysis, as well as the methodologies and sources of uncertainty involved, are documented. Additionally, the uncertainty in CDF and LERF is addressed by using UNCERT.
IF-F2: **<u>CC-I/II/III:</u>** **DOCUMENT the process used to identify ... flood areas... , For example, this documentation typically includes** **...** **(b) flood areas used in the analysis and the reason for eliminating areas from further analysis**	IFPP-B2: <u>CC-I/II/III:</u> DOCUMENT the process used to identify flood areas. For example, this documentation typically includes (a) flood areas used in the analysis and the reason for eliminating areas from further analysis (b) any walkdowns performed in support of the plant partitioning	The requirement to document walkdowns performed in support of plant partitioning was added to the 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood sources.	Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard. A self-assessment against the 2009 version of the standard was performed and it was determined that the documentation of flood walkdowns meets the requirement of the 2009 standard.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
IF-B1: **CC-I/II/III:** **For each flood area, IDENTIFY the potential sources of flooding Note (1). INCLUDE:** **(a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, fire protection system, component cooling water system, feedwater system, condensate and steam systems)**	IFSO-A1 : CC-I/II/III: For each flood area, IDENTIFY the potential sources of flooding Note (1). INCLUDE: (a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, fire protection system, component cooling water system, feedwater system, condensate and steam systems, and reactor coolant system) ...	The requirement to include the fire protection system in Item (a) as a potential flooding source was added by RG 1.200 Revision 1. The requirement to include the reactor coolant system in Item (a) as a potential flooding source was added to the 2009 version of the PRA Standard.	The flood model was reviewed and it was confirmed that all significant flood sources, including the fire protection and RCS systems, are included for evaluation in the flood model.
IF-F2 **CC-I/II/III:** **DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes: flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined** **…** **(f) screening criteria used in the analysis** **….** **(j) calculations or other analyses used to support or refine the**	IFSO-F2 CC-I/II/III: DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes: Flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined Screening analysis used in the analysis calculations or other analyses used to support or refine the flooding evaluation any walkdowns performed in support of identification or screening of flood	The requirement to document walkdowns performed in support of the identification or screening of flood sources as added to the 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood sources.	The internal flood PRA documents the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas that are considered in flood sequence definition.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
flooding evaluation	sources		
IF-F2 **<u>CC-I/II/III:</u>** **DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes:** **...** **(c) propagation pathways...** **...** **(d) accident mitigating features and barriers credited...** **...** **(e) assumptions or calculations used in the determination of ...flood-induced effects on equipment operability** **...** **(f) screening criteria used in the analysis** **...** **(g) flood scenarios considered, screened, and retained** **...** **(h) description of how the internal events**	IF-F2 <u>CC-I/II/III:</u> DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes: ... (a) propagation pathways... ... (b) accident mitigating features and barriers credited... ... (c) assumptions or calculations used in the determination of ...flood-induced effects on equipment operability ... (d) screening criteria used in the analysis ... (e) flood scenarios considered, screened, and retained ... (f) description of how the internal events analysis models were modified...	The requirement to document walkdowns performed in support of the identification or screening of flood sources as added to the 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood sources. Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard.	The internal flood PRA documents the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas that are considered in flood sequence definition.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
analysis models were modified... **....** **(j) calculations or other analyses used to support or refine the flooding evaluation** **...**	(g) calculations or other analyses used to support or refine the flooding evaluation ... (h) any walkdowns performed in support of identification or screening of flood scenarios		
IF-F2 **<u>CC-I/II/III:</u>** **DOCUMENT the process used to define the applicable internal flood accident sequences and their associated quantification. For example, this documentation typically includes:** **...** **(j) calculations or other analyses used to support or refine the flooding evaluation** **...** **(f) screening criteria used in the analysis** **...** **(i) flooding scenarios considered screened, and retained** **...** **(k) results of the internal flood analysis, consistent with the quantification requirements provided in HLR-QU-D**	IF-F2 <u>CC-I/II/III:</u> DOCUMENT the process used to define the applicable internal flood accident sequences and their associated quantification. For example, this documentation typically includes: ... (j) calculations or other analyses used to support or refine the flooding evaluation ... (f) screening criteria used in the analysis ... (i) flooding scenarios considered screened, and retained ... (k) results of the internal flood analysis, consistent with the quantification requirements provided in HLR-QU-D	The requirement to document walkdowns performed in support of the identification or screening of flood sources as added to the 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood related features considered in flood sequence quantification.	Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard. The internal flood PRA documents the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas that are considered in flood sequence definition.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
	... (e) any walkdowns performed in support of internal flood accident sequence quantification		
IF-C3 **<u>CC-I</u>:** **For the SSCs identified in IF-C2c, ...(no change)** **<u>CC- II</u>:** **INCLUDE failure by submergence and spray in the identification process. ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions.** **<u>CC-III</u>:** **For the SSCs identified in IF-C2c, ...(no change)**	IFSN-A6 <u>CC-I:</u> For the SSCs identified in IFSN- AS, ...(no change) <u>CC-II:</u> For the SSCs identified in IFSN- AS, IDENTIFY the susceptibility of each SSC in a flood area to flood- induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. ASSESS qualitatively the impact Df flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions. <u>CC-III:</u> For the SSCs identified in IFSN-A5, ...(no	RG 1.200, Revision 2, provides clarification that should be evaluated	The HCGS Internal Flooding PRA model included investigation into component failure due to flooding, induced jet impingement, humidity, condensation, temperature, etc.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
	change)		

Note: All supporting requirements affected by “key” assumptions and uncertainty have been gap assessed (per NEI 05-04) as reported in the inverter LAR (ADAMS Accession No. ML18103A218). Each peer review performed on the HC PRA considers the appropriate version of the PRA standard and the current RG 1.200 clarifications.