PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
Nuclear LLC

MAR 27 2017

10 CFR 50.90

LR-N17-0063

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

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

Subject: Response to Request for Additional Information, Re: Permanently Extend Type A and Type C Leak Rate Test Frequencies (CAC No. MF8462)

Reference: NRC letter to PSEG, "Hope Creek Generating Station – Request for Additional Information Regarding License Amendment Request to Permanently Extend Type A and Type C Leak Rate Test Frequencies (CAC No. MF8462)," dated February 27, 2017 (ADAMS Accession No. ML17027A328)

In the referenced letter, the Nuclear Regulatory Commission (NRC) requested PSEG Nuclear LLC (PSEG) to provide additional information in order to complete the review of the license amendment request (LAR) to permanently extend Type A and Type C leak rate test frequencies. Attachment 1 provides a detailed response to the request for additional information.

PSEG has determined that the information provided in this submittal does not alter the conclusions reached in the 10 CFR 50.92 no significant hazards determination previously submitted. In addition, the information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no regulatory commitments contained in this letter.

Should you have any questions regarding this submittal, please contact Ms. Tanya Timberman at 856-339-1426.

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

Executed on 3/27/17
(Date)

Sincerely,

Eric S. Carr
Site Vice President
Hope Creek Generating Station

Attachments:
1. Response to Request for Additional Information

cc: Mr. D. Dorman, Administrator, Region I, NRC
Ms. C. Parker, Project Manager, NRC
NRC Senior Resident Inspector, Hope Creek
Mr. P. Mulligan, Chief, NJBNE
Hope Creek Commitment Tracking Coordinator
Corporate Commitment Tracking Coordinator

LR-N17-0063

Attachment 1

Response to Request for Additional Information

LR-N17-0063
Attachment 1

Response to Request for Additional Information
Regarding Amendment Request to Permanently Extend
Type A and Type C Leak Rate Test Frequencies
Hope Creek Generating Station
Docket No. 50-354

By letter dated October 7, 2016,[1] PSEG Nuclear LLC (PSEG or the licensee) submitted a license amendment request (LAR) to revise the Hope Creek Generating Station (Hope Creek or HCGS) Technical Specifications by incorporating Nuclear Energy Institute (NEI) topical report 94-01, Revision 3-A, and the conditions and limitations specified in NEI topical report 94-01, Revision 2-A, as the implementation document for the Hope Creek performance based containment leakage rate testing program. Based on guidance in NEI 94-01, Revision 3-A, the proposed change would allow the Hope Creek Type A Test (Integrated Leak Rate Test, or ILRT) frequency to be extended from 10 to 15 years, and the Type C Tests (Local Leak Rate Tests, or LLRTs) frequency to be extended from 60 to 75 months. In addition, the amendment would delete a one-time extension of the test frequencies previously granted in License Amendment No. 147 (dated April 16, 2003).[2]

The U.S. Nuclear Regulatory Commission (NRC or the Commission) staff has reviewed the application and, based upon this review, determined that the following additional information is needed to complete the review:

Request for Additional Information

RAI-1
LAR Attachment 3, Section 5.7.3, "Other External Events Discussion," states that other external hazards, including high winds and tornadoes, external floods, transportation accidents, and nearby facility accidents, were assumed to not impact the results and conclusion of the risk assessment. These hazards were determined to be negligible contributors to overall plant risk based on the individual plant examination for external events (IPEEE) analysis. Since the IPEEE studies are outdated (one-time review completed in 1997), discuss, in the context of the current plant and its environs, the applicability of the IPEEE conclusions for the current LAR. Also, discuss the impact of any updated risk studies, such as the reevaluated external hazards for Hope Creek arising from the Near-Term Task Force (NTTF)[3] recommendation.

PSEG Response to RAI-1
The IPEEE identified the following other external events that received a detailed plant specific assessment:

- High winds and tornadoes
- External floods
- Transportation and nearby facility accidents
- Release of on-site chemicals

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML16281A139.
[2] ADAMS Accession No. ML030660099.
[3] The NTTF was established in response to Commission direction to conduct a systematic and methodical review of NRC processes and regulations to determine whether the agency should make additional improvements to its regulatory system and to make recommendations to the Commission for its policy direction, in light of the accident at the Fukushima Dai-ichi Nuclear Power Plant (ADAMS Accession No. ML111861807).

- Detritus

High winds and tornadoes
The IPEEE used the Regulatory Guide 1.76 design basis tornado for Region I (which includes HCGS) as a maximum wind speed of 300 mph and a maximum translational speed of 60 mph. The IPEEE concluded that the seismic safety structures exposed to the design basis tornado wind, or missiles associated with this wind, are designed so that they are not affected by these conditions. Furthermore, structures not designed for tornado loads are checked to ensure that during a tornado they do not generate missiles that have more severe effects than the tornado missiles considered in the HCGS UFSAR.

Similar to the IPEEE, the design basis tornado as stated in the current UFSAR has a maximum wind speed of 360 mph, a maximum translational speed of 70 mph, and a maximum rotational speed of 290 mph.

The conclusions in the IPEEE are supported by the analysis done in support of an Early Site Permit for an adjacent site. In the PSEG Early Site Permit application for the site, Regulatory Guide 1.76, Revision 1 was used. The Design Basis Tornado characteristics include a maximum wind speed of 200 mph and a maximum translational speed of 40 mph (Reference 1). This less severe design basis tornado in a newer version of Regulatory Guide 1.76 indicates that the older IPEEE and UFSAR are likely to be conservative relative to newer information.

The comparison of the current UFSAR tornado parameters and those used in the IPEEE are nearly identical with the exception of the maximum translational speed. The IPEEE conclusions are valid for the current plant configuration and remain valid for this LAR. There is no impact from any updated risk studies.

External Floods
The IPEEE analyzed the probable maximum flood. This included walkdowns to identify any potential vulnerabilities and use of the Generic Letter 89-22, which provided the maximum precipitation criteria. The IPEEE concluded that the external flood was a negligible risk.

In response to the NRC request issued as part of implementing lessons learned from the accident at the Fukushima Dai-ichi nuclear power plant, PSEG analyzed failure of dams and onsite structures, storm surge, seiche, tsunami, ice-induced flooding, and channel migrations or diversions. The NRC staff confirmed PSEG's conclusion that the reevaluated hazard for flooding due to these events are bounded by the current design basis (CDB) flood hazard at the Hope Creek site. Therefore, the NRC staff determined that flooding from these events does not need to be analyzed in a focused evaluation or an additional assessment (Reference 2).

Only Local Intense Precipitation (LIP) was determined to be an applicable flood-causing mechanism at Hope Creek that could exceed the current design basis. Further analysis concluded HCGS does not consider LIP an event that can challenge key safety functions, and only considers LIP flooding elevations and associated effects in the protection of FLEX connections and equipment during storage. HCGS considers the requirement to address the reevaluated flooding hazards within its mitigating strategies as being satisfied with no further action required. (Reference 3)

The IPEEE discusses the maximum recorded snowfall of 30 inches for HCGS. Calculations show that safety structures can withstand about 289 inches of snow before their design limits are exceeded. Thus, there is ample margin in the design for snowfall.

Therefore, the IPEEE conclusions are valid for the current plant configuration and remain valid for this LAR. There is no impact from any updated risk studies.

Transportation and Nearby Facilities
The IPEEE states that all activities and facilities within five miles of the HCGS site are considered in the UFSAR. No significant activities or facilities are located in this area. This included manufacturing and chemical plants, oil refineries, storage facilities, military facilities, transportation routes, or gas and oil pipelines. No major highway or railroad is located within a five mile radius of the plant. Therefore the IPEEE concludes that the impact of any transportation type of accident on HCGS would be negligible.

The conclusions in the IPEEE are supported by the analysis done in support of an Early Site Permit. In the PSEG Early Site Permit application for the PSEG site north of the existing HCGS site, chemical hazards are analyzed at the following locations:

- Nearby transportation routes such as local roads in both New Jersey and Delaware
- River vessel traffic on the Delaware River
- Nearby chemical and fuel storage facilities (in Lower Alloways Creek (LAC) Township Buildings or Port Penn Sewage Treatment Plant)
- Chemical storage at Salem and Hope Creek (S/HC)

The analysis concluded there are no chemical hazards offsite from HCGS that are design-basis events and that the primary source of hazards to the new plant is chemical shipments on the Delaware River. Many of these chemicals are analyzed using a probabilistic analysis in order to show that the frequency of the hazard is sufficiently low so as not to pose a threat to the new plant at the site.

The frequency of core damage due to chemical explosions is shown to be over two orders of magnitude less than the acceptance criteria in NUREG-0800 (Reference 1).

Therefore, the IPEEE conclusions are valid for the current plant configuration and remain valid for this LAR. There is no impact from any updated risk studies.

On-site Chemicals
The IPEEE states that PSEG maintains an inventory of any hazardous chemicals stored at, delivered to, and used at HCGS. The IPEEE also states that quantitative analysis of water treatment chemicals that could affect the control room shows that the control room meets Regulatory Guide 1.78 criteria. The IPEEE 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 IPEEE also states that the evaluation of remaining hazardous chemicals stored on-site identified no other chemicals that could impact control room habitability.

The current UFSAR (Section 6.4) on control room habitability does not contain any updated information that would invalidate the IPEEE conclusions.

Therefore, the IPEEE conclusions are valid for the current plant configuration and remain valid for this LAR. There is no impact from any updated risk studies.

Detritus
The IPEEE states that the HCGS service water (SW) pumps have experienced problems due to mud and grass buildup in their travel screens. It also states that several plant modifications were incorporated to improve the SW system and its ability to cope with a detritus event. The IPEEE concludes that a detritus induced loss of all service water has been shown to have a frequency less than the IPEEE screening criteria and is therefore a negligible risk contributor.

The loss of service water initiating event is a plant-specific special initiator included in the HCGS PRA model used in the LAR. The calculation for this initiator considers the fact that no HCGS or industry complete loss of service water events have occurred; and NUREG/CR-6929 does not provide a prior frequency distribution for loss of station service water. It is a slow developing phenomenological event. The initiating frequency calculation acknowledges the complexity of this event and associated uncertainty. PRA Standard (Reference 4) Supporting Requirement (SR) IE-C6 allows exclusion of such a detritus event due to its slow evolution and absence of an “immediate” shutdown. Nonetheless, due to the experience of some precursor events to such an event, the loss of service water initiator is retained in the HCGS PRA model used for the LAR.

Therefore, the IPEEE conclusions are valid for the current plant configuration and remain valid for this LAR. Furthermore, the treatment of a detritus event is explicitly considered in the PRA model used in this LAR. There is no impact from any updated risk studies.

RAI-2
Based on the individual contributors to the total large early release frequency (LERF) reported in LAR Attachment 3, Table 5.7-7, "Impact of 15-YR ILRT Extension on LERF for HCGS," it appears that the total LERF would be 4.82E-6 per year, instead of the reported value of 8.17E-6 per year. In addition, the values for delta LERF in LAR Attachment 3, Table 5.6-1, "HCGS ILRT Cases: Base, 3 to 10, and 3 to 15 YR Extensions (Including Age Adjusted Steel Liner Corrosion Likelihood)," and Table 5.6-2, "HCGS ILRT Extension Results Comparison to Acceptance Criteria," are inconsistent. Address the cited discrepancies in total LERF and delta LERF.

PSEG Response to RAI-2
Table 5.6-1 is correct in showing a delta LERF of 3.91E-8. Table 5.6-2 had values that were incorrect. The Internal Events delta LERF due to ILRT (at 15 years) and External Events delta LERF due to ILRT (at 15 years) were revised in Table 5.6-2 to reflect the correct delta LERF of 3.91E-8/yr. The revised Table 5.6-2 is shown below:

Table 5.6-2

HCGS ILRT Extension results Comparison to Acceptance Criteria

FIGURE OF MERIT - >	ΔLERF	ΔPERSON-REM/YR	ΔCCFP
HCGS	3.91E-8/yr	5.15E-03/yr (0.01%)	0.93%
Acceptance Criteria	**<1.0E-7/yr (“very small”)**	**<1.0 person-rem/yr or <1.0%**	**<1.5%**

Table 5.7-7 had the incorrect value for Internal delta LERF due to ILRT and also propagated that incorrect value to External delta LERF and Total LERF. A revised Table 5.7-7 is shown below and is consistent with Tables 5.6-1 and 5.6-2:

Table 5.7-7

Impact of 15-yr ILRT Extension on LERF for HCGS

LERF CONTRIBUTOR	(1/YR)
Internal Events LERF	8.45E-07
Fire LERF	3.08E-06
Seismic LERF	5.63E-07
Internal Events ΔLERF due to ILRT (at 15 years)	3.91E-08
External Events ΔLERF due to ILRT (at 15 years)	2.29E-07 [Internal Events LERF due to ILRT * 5.9]
Total	**4.76E-06/yr**
Acceptance Criteria	**<1E-05/yr**

The conclusions of the LAR are not affected by the above described tabulation errors. The corrected version of Table 5.7-7 shows a slightly lower total impact of the ILRT extension.

RAI-3

In LAR Attachment 3, Appendix A, Section A.2.4, "Consistency with Applicable PRA Standards," the licensee provided a summary of peer reviews and self-assessments for the internal events probabilistic risk assessment (PRA) model.

a) Confirm that the 2009 peer review was a full-scope peer review of the internal events and internal flooding PRA model.

b) Describe all changes, including any new analyses or incorporation of new methodology, performed in the internal events and internal flooding PRA model after the peer review, and justify whether any of the changes fit the definition and criteria

of 2009 American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard (ASME/ANS RA-Sa-2009) for a PRA upgrade.

c) If a focused-scope peer review is deemed necessary based on the response to item b above, provide the results of such a review addressing the associated facts and observations (F&Os) and their disposition.

PSEG Response to RAI-3

a) The 2008 peer review performed for the HCGS internal events and internal flooding PRA model was a full-scope peer review. The F&Os for both internal events and internal flooding are summarized in the LAR and our recent self-assessment (HC-PRA-016) provides additional detail.

b) There have been two updates to the HCGS internal events PRA model since the performance of the 2008 peer review of the PRA model (HC108A). The core damage frequency (CDF) and LERF values from the peer reviewed model and model of record updates since the peer review are provided below.

Model	**CDF**	**LERF**
HC108A (2008 Peer Review Model)	7.60E-6	8.63E-7
HC108B	5.11E-6	4.76E-7
HC111A	4.20E-6	8.44E-7

A summary of changes made to the internal events PRA models of record are documented in the introduction sections of the respective full power, internal events (FPIE) PRA Summary Notebook (HC-PRA-013). Additionally, changes in the FPIE PRA models are tracked in the HCGS Updating Requirements Evaluation (URE) database which tracks PRA observations and open items identified in between scheduled FPIE PRA Update periods. URE items that involve model changes that were addressed in the updated models are also listed and described in the Summary Notebook. The summary of changes and table of UREs involving model changes were used to identify the changes made since the HC108A model was peer reviewed.

A summary of the identified changes for each of the PRA model updates since the 2008 peer review is discussed below. These changes, including any new analyses or incorporation of new methodologies performed in the internal events PRA model since the last full-scope peer review from 2008, were reviewed for this RAI response. No changes were identified that meet the definition and criteria of the ASME PRA Standard (Reference 4) for a PRA upgrade.

As noted previously, a PRA upgrade is defined as the incorporation into the PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Appendix 1-A of the PRA Standard (Reference 4) also notes that "consideration should be given to the scope or number of PRA maintenances

performed," and that "the integrated nature of several changes may make a peer review desirable." Although the changes to the HCGS PRA models since the 2008 Peer Review have not involved new methods, multiple model revisions have occurred over the intervening years. These revisions have resulted in changes to the risk metrics of CDF and LERF, as presented above. Since none of the model changes were identified as upgrades, the existing PRA technical adequacy of the PRA Model used for the LAR is judged to adequately support the ILRT application.

HC108B Model Changes

The HC108B model was developed as a result of addressing Peer Review Findings after the Peer Review in 2008. For each change, a discussion is also provided to identify whether the change item is *PRA Maintenance* or *PRA Upgrade*. A reference to the specific PRA Standard (Reference 4) Appendix 1-A "Example" to which the change item relates is provided when possible. Major changes incorporated into the model include the following data, plant, procedure, and analysis changes:

1. Incorporate procedural change to clarify crew actions to be taken if a single service water pump was available with only one SACS pump operating.
   - ***PRA Maintenance*** (Examples 22). Change due solely to plant procedure change. HEPs are updated using the same HRA methodology. No new methods are employed.

2. Update inverter room cooling logic.
   - ***PRA Maintenance*** (Example 7). Correction of logic model omission, no new methodology employed compared to the prior model.

3. Update manual shutdown effects on SRV challenges.
   - ***PRA Maintenance*** (Example 7). Correction of logic model omission, no new methodology employed compared to the prior model.

4. Fire protection system model change to include fire pumper truck.
   - ***PRA Maintenance*** (Example 6). Fire protection system was already in the previous model. Addition of the need for the fire pumper truck as a "booster pump" was added. Logic model enhancement, no new methodology employed compared to the prior model.

5. Correct data used for Loss of DC Bus Criticality Factor.
   - ***PRA Maintenance*** (Example 6). Logic model enhancement, no new methodology employed compared to the prior model.

6. Update HEP for EDG crosstie.
   - ***PRA Maintenance*** (Example 6 and 20). Logic model enhancement for completeness, no new methodology employed compared to the prior model.

7. Update SW pipe rupture frequency and TB flood frequency.
   - ***PRA Maintenance*** (Examples 3 and 6). Logic model correction, no new methodology employed compared to the prior model.

8. Correct errors in basic event database.
   - ***PRA Maintenance*** (Example 6). Basic event database correction, no new methodology employed compared to the prior model.

9. Revise SACS train unavailability.
   - ***PRA Maintenance*** (Example 6). Model correction, no new methodology employed compared to the prior model.

10. Add instrument air for torus vacuum breaker support.
    - ***PRA Maintenance*** (Example 7). Correcting dependency omission, no new methodology employed compared to the prior model.

11. Add drywell spray to Level 2 nodal logic.
    - ***PRA Maintenance*** (Example 10). Logic model change for completeness supported by the existing thermal hydraulic calculations, no new methodology employed compared to the prior model.

12. Update containment isolation fault tree logic.
    - ***PRA Maintenance*** (Example 15). Logic model enhancement to expand existing logic for completeness, no new methodology employed compared to the prior model.

13. Update main steam line isolation logic.
    - ***PRA Maintenance*** (Example 6). Logic model correction, no new methodology employed compared to the prior model.

14. Update failure probabilities for SACS pumps fail to start and fail to run, EDG fail to run, and HPCI and RCIC turbine drive pump failure to run.
    - ***PRA Maintenance*** (Example 6). HEP correction, no new methodology employed compared to the prior model.

HC111A Model Changes

The HC111A model was the result of a regularly scheduled update per PSEG Risk Management procedures. For each change, a discussion is also provided to identify whether the change item is *PRA Maintenance* or *PRA Upgrade*. A reference to the specific PRA Standard (Reference 4) Appendix 1-A "Example" to which the change item relates is provided when possible. Major changes incorporated into the model include the following data, plant, procedure, and analysis changes:

1. Update initiating event frequencies
   - ***PRA Maintenance*** (Example 2). Using new plant-specific data, no new methodology employed. This was not the first time Bayesian updating was performed.

2. Update component data frequencies with plant-specific data
   - ***PRA Maintenance*** (Example 2). Using new plant-specific data, no new methodology employed. This was not the first time Bayesian updating was performed.

3. Update common cause frequencies with plant-specific data
   - ***PRA Maintenance*** (Examples 3 and 26). Using new plant-specific data, no new methodology employed. This was not the first time Bayesian updating was performed.

4. Maintenance unavailability data updated based on the most recent plant-specific operating experience.
   - ***PRA Maintenance*** (Examples 2 and 19). Using new plant-specific data, no new methodology employed.

5. Migration to HRA Calculator version 4.1.1; update dependencies; add additional pre-initiators
   - ***PRA Maintenance*** (Example 20). HEP modeling enhancement, no new methodology employed. Pre-initiator HEPs were already included in the model. Additional pre-initiators were added to the model as a modeling enhancement, no new methodology employed. Dependencies were already included in the model.

6. Migration to MAAP4.0.6 from MAAP4.0.4; recalculation of all deterministic calculations; update of HEP timings.
   - ***PRA Maintenance*** (Example 8 and 10, 17 and 20). Refinement for success criteria based on thermal hydraulic calculations. MAAP was used in previous model.

7. Incorporation of B.5.b pump as an additional RPV injection source; delete fire protection system/fire pumper truck as injection source.
   - ***PRA Maintenance*** (Example 8). Logic model enhancement, no new methodology employed compared to the prior model.

8. Add SPC failure to lead to high drywell (DW) pressure signal and Safety and Turbine Auxiliaries Cooling System (STACS) isolation.
   - ***PRA Maintenance*** (Example 9). Logic model enhancement to correct an omission, no new methodology employed compared to the prior model.

9. Incorporation of dependent operator actions directly in the fault tree logic; eliminate need for post processor (i.e., QRecover).
   - ***PRA Maintenance*** (Example 20). HEP modeling update, new dependency groups were added directly to the model consistent with previous methods and other dependent groups treated as such, no new methodology employed.

10. Minor changes to revise accident sequence event tree nodal logic to include B.5.b pump.
    - ***PRA Maintenance*** (Example 8). Logic model enhancement for completeness, no new methodology employed compared to the prior model.

11. Minor changes to revise accident sequence event trees to correct inconsistencies with documented Success Criteria.
    - ***PRA Maintenance*** (Example 6). Logic model correction, no new methodology employed compared to the prior model.

12. Update internal flood HEPs.
    - ***PRA Maintenance*** (Example 20). HEP modeling update, no new methodology employed.

13. Incorporate potential hydrogen deflagration with a de-inerted containment during manual shutdown evolutions.
    - ***PRA Maintenance*** (Example 10). Logic model enhancement for completeness, no new methodology employed compared to the prior model.

14. Update select Level 2 end states to reflect the latest MAAP calculations. No Level 2 end states were modified to a more severe category.
    - ***PRA Maintenance*** (Example 8 and 10). Refinement for success criteria based on thermal hydraulic calculations.

15. Minor changes to revise fault tree logic.
    - ***PRA Maintenance*** (Example 6). Logic model corrections, no new methodology employed compared to the prior model.

c) Based on the response to item b above, no changes that fit the definition and criteria of the PRA Standard (Reference 4) for a PRA upgrade were identified. Therefore, a focused-scope peer-review is not necessary.

RAI-4

The following requests for information apply to the internal events F&Os and their corresponding resolutions as reported in Table A-1, "Resolution of Peer Review F&Os," of Appendix A, Attachment 3, to the LAR:

a) F&O LE-G1-01, Level 2 Analysis roadmap detail enhancement, states that the Hope Creek Level 2 analysis notebook was not "written in a manner conducive to demonstrating the requirements of the standard were met" and that it "limited the ability of the Peer-Review team to perform an adequate review." The corresponding resolution compares the Hope Creek Level 2 analysis notebook to those used and peer-reviewed elsewhere. However, the F&O statements appear to imply that the peer review team, due to the limitations cited in the F&O, did not review, or only partially reviewed, the Level 2 analysis against the LERF analysis (LE) supporting requirements of the 2005 ASME PRA standard. Since LERF is a key metric in the risk assessment supporting the LAR, justify, in the context of the cited F&O and the foregoing discussion, why a focused-scope peer review of the Level 2 PRA model is not required.

b) F&O QU-E4-01, Uncertainty analysis structured sensitivity evaluations, identified that the evaluation of uncertainties did not identify or address any plant specific sources of uncertainty. In the F&O resolution the licensee stated that this issue has not yet been resolved, because there was no published guidance on the treatment of uncertainties at the time the resolution was documented. This resolution appears to no longer be valid since NUREG-1855, Volume 1, "Guidance on the Treatment of

Uncertainties Associated with PRAs in Risk-Informed Decision Making,"[4] and supplemental Electric Power Research Institute (EPRI) guidance, such as EPRI Technical Report (TR)-1016737 and EPRI TR-1026511, on the treatment of uncertainties in PRA have been issued. Further, contrary to the F&O resolution stating that the PRA standard does not require evaluation of sources of model uncertainty, such an evaluation is required per the 2009 ASME/ANS PRA Standard (ASME/ANS RA-Sa-2009) endorsed by Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."[5] The F&O resolution contradicts the results of the self-assessment, which concluded that supporting requirement QU-E4 of ASME/ANS RA-Sa-2009 is met.

i. Describe how the uncertainty evaluation mentioned in F&O QU-E4-01 was performed, given that relevant guidance on the treatment of uncertainties is available. Discuss the impact of the results of the uncertainty evaluation on the LAR.

ii. If the uncertainty evaluation mentioned in the F&O has not been performed, justify the conclusion that the lack of the uncertainty analysis would have no impact on the current application.

c) F&O SY-B14-01, Missing Common Piping Failure Modeling, found that, "it cannot be demonstrated that components/failure modes which fail multiple systems have been included," as required, and provided an example that failure of common piping between the high pressure core injection (HPCI)/feedwater (FW)/core spray (CS) and reactor core isolation cooling (RCIC)/FW systems was not modeled. The resolution provides an explanation of how the example cited in the F&O was addressed. The resolution states that the model modification resulting from the F&O's concerns "has been evaluated and assessed as a negligible impact on PRA risk metrics." However, the resolution also states that, "it has been included as the highest priority model change in 2009."

i. Confirm that the failure of common piping between HPCI/FW/CS and RCIC/FW systems has been included in the internal events PRA model used for the LAR or, alternatively, if not included in the model, provide the basis for concluding that there is "negligible impact."

ii. Discuss how it was ensured that other than the example identified by the peer review team, no other possible instances of common piping that could fail multiple systems are left unaddressed in the internal events PRA model.

d) F&O AS-B2-01 identified that the operation of the automatic depressurization system (ADS), under conditions of a stuck open relief valve (SORV) with failure of the high pressure makeup, appears to be modeled as always successful. The resolution of this F&O states that this is a documentation issue only. However, it appears that the issue may point to a flaw in the PRA model that precludes a dependent failure that is logically possible. Explain whether or not the F&O identified a flaw in the PRA model

[4] ADAMS Accession No. ML090970525.
[5] ADAMS Accession No. ML090410014.

logic. Further, if a flaw exists, either correct it and reevaluate the model to determine the effect on the current submittal, or justify why it does not adversely impact the current submittal.

e) F&O SC-A6-01, Basis for the fire pump flow rate, questioned the basis for crediting the diesel-driven firewater pump as a low pressure source of makeup to the reactor vessel after depressurization, stating that the pump flow curve has not been rigorously analyzed. In the resolution to this F&O, the licensee performed a detailed deterministic calculation of the fire pump flow curve and identified that additional equipment, such as a "fire pumper truck" to boost pressure, is required for the success of the diesel-driven firewater pump. Confirm that given the changes to the PRA model described in the F&O resolution, human failure events associated with the changes, such as the need to stage and operate the "fire pumper truck," have been considered and incorporated into the PRA model.

PSEG Response to RAI-4

a) F&O LE-G1-01 states that the HCGS Level 2 Analysis Notebook (HC-PRA-015) was not “written in a manner conducive to demonstrating the requirements of the standard were met” and that it “limited the ability of the Peer-Review team to perform an adequate review.” The peer review team reviewed the HCGS Level 2 model against all LERF Analysis (LE) supporting requirements (SRs). There was no mention or discussion in the peer review report regarding any LE SRs that were not reviewed because of this or any other issue. F&O LE-G1-01 was the only finding for the LE technical element. All other SRs for the LE technical element were shown in the peer review report as “SR Met” with at least a Capability Category II. There was no mention or discussion in the peer review report regarding any LE SRs that were partially reviewed because of this or any other issue. Furthermore, the Possible Resolution identified in the peer review report for this F&O discussed revising the organization of the Level 2 Notebook and including references to the PRA Standard (Reference 4) in the Level 2 Notebook. Despite the inclusion of the statement by the peer review team in the Basis for Significance for this F&O which reads “[t]he lack of cross-reference, or organization according to, the PRA standard for the LERF analysis limited the ability of the Peer Review team to perform an adequate review,” there was no mention in the peer review report that the LERF review by the peer review team was incomplete in any way. Furthermore, the peer review report included no mention of any LE SRs that were not reviewed and did include other F&O Suggestions related to LE SRs recommending improving some documentation items similar to F&O LE-G1-01. Based on this evidence, PSEG concludes that there was an adequate review performed on the LE technical element and associated SRs by the peer review team and a focused-scope peer review of the Level 2 PRA model is not required.

b) F&O QU-E4-01
   i. The uncertainty evaluation mentioned in F&O QU-E4-01 was performed in the 2011 PRA periodic update consistent with NUREG-1855 and the complementary EPRI guidance. This is documented in Appendix B of the HCGS PRA Summary Notebook (HC-PRA-013). Table B-1 of this same notebook describes sources of model uncertainty for HCGS and, among other

things, the impact on the model. The parametric uncertainty associated with basic events for the quantification does not affect the mean CDF and LERF as calculated by the quantification process. Therefore, the treatment of uncertainty would not under predict the risk for the current LAR.

ii. As described in the response to item 4.b.i, the uncertainty evaluation mentioned in F&O QU-E4-01 was performed for the PRA model used in the LAR as described in HC-PRA-013.

c) F&O SY-B14-01

i. The resolution for F&O SY-B14-01 states that "[i]t has been included as the highest priority model change in 2009." A review of the modeling used in the LAR of failure of common piping between HPCI/FW/CS and RCIC/FW systems concludes that the modeling is adequate for the LAR. For HPCI evaluations, because of the multiple paths into the RPV from HPCI, breaks in the FW or CS pipe do not compromise the ability for HPCI to provide adequate makeup to meet the PRA success criteria. Unisolable breaks outside containment are treated to fail all RPV injection sources in the Reactor Building. Therefore, no additional dependency treatment is needed for those cases. All common valve failures (e.g., MOVs and CVs) between HPCI/CS A and HPCI/FW A are explicitly modeled to fail the common systems. The same holds true for RCIC/FW B.

ii. A review of system notebooks and relevant P&IDs was performed to identify any other instances of common piping that could fail multiple systems. No other instances were identified.

d) The plant-specific thermal hydraulic calculations for the PRA model used for the LAR show that for a SORV scenario with a failure of high pressure injection, operation of ADS is not required. These plant-specific thermal hydraulic calculations are documented in the Deterministic Calculations Notebook (HC-PRA-007). The SORV event tree and functional fault trees as modeled are consistent with this success criteria determined by the plant-specific thermal hydraulic calculations. The text description in the Success Criteria Notebook (HC-PRA-003) describing the SORV depressurization success criteria is not clearly written and could be interpreted incorrectly. Thus, the conclusion that F&O AS-B2-01 is a documentation issue only is accurate. A review of the plant-specific thermal hydraulic calculations, event trees, and fault trees showed that no flaw exists in the PRA model regarding this issue and the PRA model used in the LAR reflects the success criteria and event sequence correctly. No model changes were required as a result of this F&O.

e) In the 2011 periodic update, the fire protection system as an alternate injection system was removed from the PRA. Credit for the diesel-drive firewater pump and the fire pumper truck are no longer included in the PRA model. Therefore, given the changes to the PRA model described in F&O SC-A6-01 resolution, the human failure event associated with the changes, such as the need to stage and operate the fire pumper truck has been considered and is no longer incorporated into the PRA model.

The fire protections system was replaced with the B.5.b pump as an alternate injection source. The human failure event associated with using the B.5.b pump has been fully developed and documented for the PRA model used in the LAR. Analysis of the human failure event for operation of the B.5.b pump as an alternate injection source includes applicable plant procedures (e.g., EOPs, B.5.b pump operating procedures), use of plant-specific thermal hydraulic calculations to determine time frames for accomplishing associated operator actions, annual classroom training on the associated actions, and operator interviews to confirm diagnosis timings and establish manipulation timings. Therefore, the human failure event associated with using the B.5.b pump has been fully developed and documented for the PRA model used in the LAR.

RAI-5

The resolutions of the Hope Creek fire PRA (FPRA) peer review F&Os are presented in LAR Attachment 3, Appendix A, Table A-4, "Resolution of FPRA Peer Review F&Os." Some of the modeling choices made during the resulting FPRA model modifications appear to be non-conservative. Examples of such choices include, but are not limited to, the use of point estimates due to unquantified uncertainties (F&Os 1-11, 5-27, 5-40, 5-52, and 6-4), possible use of a probability of failure of alternate shutdown capability deemed to be non-conservative by the peer reviewers (F&Os 4-9 and 5-32), and the use of a potentially non-conservative value for the fraction of cable length participating in cable tray fires (F&Os 5-33 and 5-35). In addition, the resolution of F&O 4-14 appears to indicate the non-suppression probability in the Hope Creek FPRA are based on NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," instead of the updated values in NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database."[6] Identify all potential non-conservatisms in the Hope Creek FPRA and justify, preferably quantitatively, why the FPRA does not result in an underprediction of the fire risk for the current LAR.

PSEG Response to RAI-5

The RAI raises the concern that three particular aspects of the FPRA model may result in an under-prediction of the fire risk: the use of point estimates, the probability of failure of alternate shutdown procedures, and the fraction of cable assumed to participate in cable tray fires. To address this concern, it will be demonstrated that all of these treatments are either conservative or represent a best-estimate in accordance with industry methodology.

Use of Point Estimates due to Unquantified Uncertainties

The Fire PRA model uses point estimates for all basic events which are propagated through the quantification to result in the total CDF and LERF. The absence of uncertainty values on basic events will impact the parametric uncertainty calculations. However, the impacts on the mean CDF and LERF calculations are not large enough to affect the conclusions in this LAR. PSEG understands that NRC Regulatory Guides are based on mean, not point estimates and the discussion below shows that the numeric difference cannot change the risk insights and does not significantly change the numeric conclusions.

[6] ADAMS Accession No. ML 15016A069.

F&O 1-11 states that uncertainties related to ignition frequencies were not propagated through individual scenarios or into an uncertainty that is applied to the final CDF. F&O 5-27 states that ignition frequencies calculated on a Fire Area basis were estimated but were not propagated through the analysis further. The results of the FPRA are judged to be insensitive to the uncertainty of the fire ignition frequency (FIF) bin values due to the random nature of the uncertainty in the 40 bin types used. When considered together, the overall uncertainty in ignition frequency is negligible. Additionally, the apportionment of bin FIF over the hundreds of individual ignition sources disperses the impact of this uncertainty randomly throughout the model. The risk-significant cutsets would change only slightly, and the risk-significant components, Physical Analysis Units (PAUs), and accident sequences would not change at all. Finally, the treatment of plant partitioning and equipment selection, which excludes plant areas and ignition sources from analysis, tends to conservatively bias the scenario FIFs by "concentrating" the bin FIFs in a smaller number of ignition sources than actually exist in the plant. In addition to these conservatisms, it is noted that the point estimates used for FIFs were determined in accordance with all industry guidance and methodology and represent the accepted best prediction of these parameters. The uncertainty estimates for ignition frequencies are not a significant source of non-conservatism and would not affect the conclusions of the risk for the current LAR.

F&O 5-40 states that the uncertainty factors for the joint human error probability (JHEP) values were not included in the Fire PRA. The point estimate quantification results of the FPRA are not affected by the uncertainty of JHEPs. Human performance sources of uncertainty are discussed in the Fire HRA Notebook (HC-PRA-106) and the FPRA Summary and Quantification Notebook (HC-PRA-104). There are no identified sources of uncertainty related to JHEPs that would significantly affect the conclusions of the current LAR. Therefore, the uncertainty estimates for JHEPs are not a source of non-conservatism and would not affect the risk determination of the current LAR.

Consistent with the PRA Standard (Reference 4), quantitative parametric uncertainty analyses for CDF have been performed with EPRI's UNCERT software and are summarized below.

A sensitivity case of the FPRA model of record (MOR) used for the current LAR was created by assigning error factors of 10 to all dependent and joint human error probabilities (HEPs), as well as all fire ignition frequencies. Then both the original and sensitivity CDF probability distributions were approximated with a Monte Carlo simulation to propagate uncertainties from the basic events through the final distributions. The resulting uncertainty statistics are compared in the table below:

HC Model	Truncation (per yr)	Samples	CDF Point Estimate	Mean CDF	Deviation from Point Estimate	Mean CDF Confidence Range
HC114F0 (MOR One Top)	1E-11	50,000	1.803E-5	1.811E-5	8E-8 (0.4%)	[1.8E-05 , 1.8E-05]
HC114F0 (MOR One Top) with modified EFs	1E-11	50,000		1.840E-5	3.7E-7 (2.1%)	[1.8E-05 , 1.9E-05]

The change in deviation from the point estimate is consistent with previous BWR FPRA parametric uncertainty calculations and represents a small perturbation on estimated CDF.

Therefore, it is concluded that assigning error factors to the dependent HEPs, joint HEPs, and fire ignition frequencies would only slightly increase the CDF uncertainty interval of the Hope Creek FPRA. The sensitivity case shows that the conclusions of the model of record uncertainty evaluation remain valid with the additional error factors applied for the current LAR. As such, the point estimate CDF used for the current LAR accurately represents the mean FPRA CDF for Hope Creek.

F&O 5-52 states that the identified list of sources of uncertainty is neither complete nor fully discussed. Additionally, the limitations of the LERF analysis are not identified in the FPRA results. This list of sources of uncertainty was updated for the model used for the current LAR. A review of the list did not identify any sources of uncertainty that would be significantly non-conservative for the risk analysis for the current LAR.

F&O 6-4 states that no quantification of risk associated with multi-compartment fire scenarios was performed. Qualitative assessment (meeting Capability Category I) is also not complete for multi-compartment analysis (MCA). A qualitative assessment was performed. There are no significant non-conservatisms related to the MCA that would adversely affect the risk analysis of the current LAR.

<u>Probability of Failure of Alternate Shutdown</u>
Regarding the treatment of alternate shutdown procedures, there are two relevant F&Os addressed in the 2014 FPRA update (HC-PRA-104).

F&O 4-9 states that detailed calculations were not performed for failure of alternate shutdown capability for the scenarios resulting in main control room (MCR) abandonment. A conservative value of 0.1 was used in the peer-reviewed Fire PRA model. In order to more accurately model the risk significance of the Remote Shutdown Panel (RSP) following Main Control Room abandonment (MCRAB) scenarios, logic was inserted into the fault tree to explicitly model RSP failures. This logic took the form of accident sequences that consider the basic events relevant to RSP operation and route any resulting failure to the appropriate accident class. Cutset reviews and interviews with plant operators were also performed to determine or confirm fire scenarios where abandonment would be performed, as well as probability of failure to abandon. It was concluded based on these interviews that operators will fail to abandon the MCR with a 10% probability. It is noted that this model change falls under the definition of PRA maintenance set forth in the PRA Standard Appendix 1-A.2(c), as it enhances the completeness and realism of the existing accident sequences using previously established methodology. This change is documented in the Hope Creek Fire PRA Model Development Notebook (HC-PRA-102), Section 6.9.

F&O 5-32 states that no discussion of the factors affecting the important HEPs was discussed. For example, the top HEP (0.1 assumed control room abandonment) was not discussed regarding its uncertainty and impact on the results, nor were other factors affecting individual HEPs. All HEPs, including the control room abandonment HEP, were refined in the model used for the current LAR. Uncertainty for such HEPs was reviewed and no non-conservatisms were identified that would adversely affect the risk results for the current LAR.

<u>Fraction of Cable Length Participating in Cable Tray Fires</u>
Regarding the fraction of cable assumed to participate in cable tray fires, there are two relevant F&Os addressed in the 2014 FPRA update (HC-PRA-104).

F&O 5-33 states that the 0.08 and 0.01 severity factors used for transient fires were not well supported. F&O 5-35 states that the "area factors" applied to the transient fires do not appear to take into account the possible length of a cable tray or trays overhead. For example, a cable tray could wind around a room, increasing its exposure. In most cases, the scenario involved a number of cable trays, where it is not clear if the damage to a single tray is the main issue or if the damage to any of the trays can cause the event of concern.

In order to more realistically model transient fires, conservative transient ignition weighting factors were assigned. For general transients, these were calculated based on the fractional area participating in the fire; for cable tray fires (including those as a result of welding and cutting), these were calculated based on the fraction of cables participating. A transient source package area of 100 ft^2 was assumed, corresponding to a square region of 10 x 10 ft or a circular region approximately 11 ft in diameter. The transient ignition weighting factor assigned to transients in this PAU is this figure divided by the PAU's floor area. For example, transient fires in PAU CD83, which has a floor area of 2200 ft^2, are assigned a weighting factor of (100 ft^2) / (2200 ft^2) = 0.045 . The average value of this factor across all general transient scenarios is approximately 0.03, or 3%.

In the case of cable fires, it was assumed that the fraction of cables participating in the scenario was 10% of the PAU's total; therefore, a transient ignition weighting factor of 0.10 was assigned. Compared to general transient scenarios, this value is more than three times higher, conservatively biasing the scenario fire ignition frequency.

Additionally, it is noted that the treatment of plant partitioning, which excludes plant areas from analysis, tends to conservatively bias the transient scenario FIFs by "concentrating" the transient bin FIFs in a smaller area than actually exists in the plant.

<u>Potential Sources of Non-Conservatism</u>
The remaining F&Os not discussed above were reviewed for any other potential non-conservatisms that may result in under-prediction of fire risk for the current LAR. Four areas of potential non-conservatism were identified and are dispositioned as follows:

1. Cable data for operator instrumentation is unavailable (F&Os 1-1, 2-5, 4-1, and 4-2) – During a fire, plant operators will use procedure HC.OP-AB.FIRE-0001 to help identify potential spurious actuations of equipment. Also, Alarm Response Procedure HC.OP-AR.QK-0002 provides additional guidance for indication that may be lost or affected for fires in a given fire zone. Considering operators' training in this regard and the availability of redundant, diverse indication, it is unlikely that a single fire would compromise operators' ability to safely shutdown the plant. The external events total delta LERF, as shown in Table 5.7-6, shows margin to the delta LERF acceptance criteria. Delta LERF could increase by approximately a factor of four before reaching the acceptance criteria. Therefore, any potential non-conservatism in this regard would not alter the acceptability of the current LAR.

2. Some plant areas are assigned a cable loading of zero and/or have their ignition sources screened (F&O 1-5) – During cutset reviews, plant engineers confirmed that all of screened plant areas did not contain any PRA-significant equipment or cable trays.

Therefore, the exclusion of these areas from the FPRA does not affect the CDF or LERF risk metrics.

3. Transient heat release rate was inappropriate (F&O 3-12) – It was judged that the use of a 69 kW heat release rate for transient fires in the 2010 Hope Creek FPRA was potentially non-conservative. In the 2014 FPRA Update, transient fires were re-evaluated using the appropriate heat release rate distributions prescribed by NUREG-6850. Therefore, this treatment of transients is not non-conservative.

4. No review of plant-specific detection and suppression availability was performed for MCA and MCR scenarios (F&Os 4-14 and 5-49) – The applied detection and suppression availabilities was reviewed by plant engineers and judged to be representative of Hope Creek (i.e., not non-conservative). The applied suppression delay times of one minute and fifteen minutes (depending on detection success) were used in the model. Additionally, it is judged that these values are conservative for the MCR, which is continually manned.

As described above, these four areas of potential non-conservatism do not result in under-prediction of fire risk for the current LAR.

Conclusion
Considering the above, it is judged that the FPRA does not result in an under-prediction of fire risk for the current LAR.

References

1. PSEG Site ESP Application, Part 2, Site Safety Analysis Report (ML15169A282)

2. NRC Letter, “Hope Creek Generating Station-Staff Assessment Of Response to 10 CFR 50.54(f) Information Request – Flood Causing Mechanism Reevaluation (CAC No. MF3789)”, October 25, 2016 (ML16266A281)

3. PSEG Letter LR-N16-0112, “Hope Creek Generating Station's Flood Hazards Mitigating Strategies Assessment (MSA) Report Submittal”, dated December 29, 2016 (ML16364A217)

4. 2009 American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard (ASME/ANS RA-Sa-2009)

5. EM-HC-100-1000, Response to Beyond Design Basis External Events Program Document Hope Creek Generating Station

6. ER-AA-600-1015, FPIE PRA Model Update

7. HC-PRA-016, Self-Assessment of the Hope Creek PRA Against the ASME/ANS PRA Standard Requirements- HCGS PSA-016, Model HC111A

8. HC-PRA-013, PRA Summary Notebook

9. HC-PRA-015, Level 2 Evaluation - HC PSA-015, Model HC111A

10. HC-PRA-007, Deterministic Calculations Notebook