UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 27, 2017

Mr. Peter P. Sena, III
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: 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)

Dear Mr. Sena:

By letter dated October 7, 2016,[1] PSEG Nuclear LLC (PSEG or the licensee) submitted a license amendment request to revise the Hope Creek Generating Station (Hope Creek) 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).

Consistent with Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), an amendment to the license (including the technical specifications) must fully describe the changes requested, and following as far as applicable, the form prescribed for original applications. Section 50.34 of 10 CFR addresses the content of technical information required. This section stipulates that the submittal address the design and operating characteristics, unusual or novel design features, and principal safety considerations.

The U.S. Nuclear Regulatory Commission staff has reviewed the licensee's application and, based upon this review, determined that additional information is needed to complete our review. On January 25, 2017, a draft of these questions were sent to Mr. Paul Duke of your staff to ensure that the questions were understandable, the regulatory basis for the questions was clear, and to determine if the information was previously docketed. On February 14, 2017, and Feburary 21, 2017, teleconferences were held to clarify the questions. On Feburary 24, 2017, Mr. Duke indicated that PSEG will submit a response within 30 days of the date of this letter.

[1] Agencywide Documents Access and Management System Accession No. ML16281A139.

If you have any questions, please contact me at (301) 415-1603 or by e-mail at Carleen.Parker@nrc.gov.

Sincerely,

Carleen J. Parker, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosure:
Request for Additional Information

cc w/enclosure: Distribution via Listserv

SUBJECT: 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

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REQUEST FOR ADDITIONAL INFORMATION

REGARDING AMENDMENT REQUEST TO PERMANENTLY EXTEND

TYPE A AND TYPE C LEAK RATE TEST FREQUENCIES

PSEG NUCLEAR LLC

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 our review:

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.

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

[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).

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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.

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.

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.

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

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 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.

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 under-prediction of the fire risk for the current LAR.

[6] ADAMS Accession No. ML15016A069.