June 24, 2010

MEMORANDUM TO: Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

FROM: Richard B. Ennis, Senior Project Manager ***/ra/***
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

SUBJECT: HOPE CREEK GENERATING STATION, DRAFT REQUEST FOR ADDITIONAL INFORMATION (TAC NO. ME2949)

The attached draft request for additional information (RAI) was transmitted on June 24, 2010, to Mr. Jeff Keenan of PSEG Nuclear LLC (the licensee). This information was transmitted to facilitate an upcoming conference call in order to clarify the licensee's amendment request for Hope Creek Generating Station (HCGS) dated December 21, 2009. The proposed amendment would allow the production of Cobalt-60 (Co-60) by irradiating Cobalt-59 targets located in modified fuel assemblies called Isotope Test Assemblies (ITAs). The amendment would allow the licensee to load up to twelve ITAs into the HCGS reactor core beginning with the fall 2010 refueling outage. The modified fuel assemblies, also referred to as GE14i ITAs, are planned to be in operation as part of a joint pilot program with Global Nuclear Fuel – Americas, LLC and GE – Hitachi Nuclear Energy Americas, LLC. The purpose of the pilot program is to obtain data to verify that the modified fuel assemblies perform satisfactorily in service prior to use on a production basis. The Co-60 is ultimately intended for use in the medical industry for use in cancer treatments, and blood and instrument sterilization; in the radiography and security industries for imaging; and in the food industry for cold pasteurization and irradiation sterilization.

This memorandum and the attachment do not convey or represent an NRC staff position regarding the licensee's request.

Docket No. 50-354

Attachment: Draft RAI

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Attachment: Draft RAI

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DRAFT REQUEST FOR ADDITIONAL INFORMATION

REGARDING PROPOSED LICENSE AMENDMENT

USE OF ISOTOPE TEST ASEMBLIES FOR COBALT-60 PRODUCTION

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

By application dated December 21, 2009, as supplemented by letters dated May 11, June 10, and June 24, 2010, PSEG Nuclear LLC (PSEG or the licensee) submitted a license amendment request for the Hope Creek Generating Station (HCGS). The proposed amendment would allow the production of Cobalt-60 (Co-60) by irradiating Cobalt-59 targets located in modified fuel assemblies called Isotope Test Assemblies (ITAs). The amendment would allow the licensee to load up to twelve ITAs into the HCGS reactor core beginning with the fall 2010 refueling outage. The modified fuel assemblies, also referred to as GE14i ITAs, are planned to be in operation as part of a joint pilot program with Global Nuclear Fuel - Americas, LLC and GE - Hitachi Nuclear Energy Americas, LLC. The purpose of the pilot program is to obtain data to verify that the modified fuel assemblies perform satisfactorily in service prior to use on a production basis. The Co-60 is ultimately intended for use in the medical industry for use in cancer treatments, and blood and instrument sterilization; in the radiography and security industries for imaging; and in the food industry for cold pasteurization and irradiation sterilization.

The NRC staff has reviewed the information the licensee provided that supports the proposed amendment and would like to discuss the following issues to clarify the submittal.

Question 1

On pages 18-20 of calculation H-1-ZZ-MDC-1880 (Revision 3), “Post-LOCA EAB, LPZ and CR Doses,” (Reference 3), the licensee provided an assessment of the reactor coolant system activity release via open primary containment isolation valves. Table 25 of Reference 3 provides a list of 90 primary containment isolation valves (PCIVs) expected to remain open for 120 seconds following a loss of coolant accident (LOCA). Table 25 also lists the “existing maximum isolation time” and “proposed maximum isolation time” for each PCIV. The “existing maximum isolation times” ranged from 5 to 80 seconds. The “proposed maximum isolation time” is 120 seconds for each valve.

Hope Creek Technical Evaluation DCR 80096650-0210, Revision 0, “Technical Evaluation to Determine post-LOCA Design Functional Impact on Systems & Components Located Downstream of Outboard Containment Isolation Valves which are Expected to Remain Open for 120 seconds at the Hope Creek Generating Station (HCGS),” (Reference 4) is used to support Reference 3. In Reference 4, the licensee uses a screening criteria to “screen out” certain PCIVs from further evaluation of increasing PCIV closure time. Screening Design Criterion 2 in Reference 4, states: “Exemption of Non-ESF systems (non-safety related systems), because they are not needed for a post-accident mitigation function.”

Per Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792) (Reference 5), Regulatory Position C.5.1.2, "Credit of Engineered Safeguard Features," states, in part, that:

"Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures." [emphasis added]

Per Regulatory Position C.5.1.2, non-engineered safety features (ESF) piping outboard of the PCIVs should not be credited. Contrary to Regulatory Position C.5.1.2, the licensee uses Screening Design Criterion 2 to screen out PCIVs with non-ESF piping from further consideration. By not considering these PCIVs, the licensee is implicitly assuming that the non-ESF piping does not contribute to the LOCA dose. Physically this could be because the piping is assumed to remain intact following a design basis LOCA. Therefore, this release pathway does not contribute to offsite or control room doses, nor does it provide a source of energy to secondary containment that could impact its integrity including drawdown times after a LOCA. Using Screening Design Criterion 2, the licensee now appears to have swapped credit for ESF PCIVs with non-ESF piping, to maintain the integrity of primary containment while the PCIVs close.

In HCGS Updated Final Safety Analysis Report (UFSAR) Section 1.8.1.183 (Reference 7), "Conformance to Regulatory Guide 1.183, Revision 0, July 2000: Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Plants," the licensee states: "HCGS complies with Regulatory Guide 1.183." Reference 3, page 25 states that credit is only taken for accident mitigation features that are classified as safety-related. The use of Screening Design Criterion 2 in Reference 4 does not appear to comply with Regulatory Position C.5.1.2 or page 25 of Reference 3. Please justify the use of Screening Design Criterion 2 or perform an analysis that complies with Regulatory Position C.5.1.2 or justify why this is not necessary. Any analysis provided should address compliance with 10 CFR 50.67 requirements, as well as reactor building drawdown times, whether secondary containment design pressure is exceeded, the impact on wetting of Filtration, Recirculation, and Ventilation System (FRVS) filters/absorbers and the impact on housed safety equipment/systems. In addition, please submit Reference 4 for formal docketing (document was reviewed as part of NRC audit activities).

Question 2

The screening form in the licensee's 10 CFR 50.59 Evaluation HC 2008-215, Revision 0 (Reference 6), which supports Revision 3 of Calculation H-1-ZZ-MDC-1880, states: "The design pressures and temperatures of all systems downstream of the open PCIVs are less than the post-LOCA containment pressure and temperature, except for the primary containment instrument gas system (PCIGS) (Ref. II.2, Table 5), which has a design temperature that is less than the post-LOCA containment peak temperature."

As written, the statement would indicate that the integrity of the systems downstream of the PCIVs would not be reasonably assured of being maintained. During post-LOCA conditions, pressures and temperatures in the systems downstream of the open PCIVs would exceed the design pressures of these systems and they would fail. Failure of these systems would provide a potential release pathway to the environment. Since failure of these downstream system

appears to not have been evaluated, please clarify the statement above, or justify why this is acceptable.

Question 3
An assessment entitled “Reactor Coolant System (RCS) Activity Release Via Open PCIV” is provided on page 18 of Reference 3. The assessment provides a calculation of the radiological consequences of PCIVs that establish a direct release pathway to the environment by bypassing the reactor building. The licensee assumed that the release rate to the environment is equivalent to the maximum purge flow rate of 9000 cfm.

Given that the conditions in these systems will be much different during a LOCA than during normal operations, it is unclear how the maximum purge flow rate is relevant for modeling the flow in these systems during a LOCA. During a design-basis LOCA, the containment will be at much higher temperatures and pressures and the releases will contain much more water than during normal operations. The flow could possibly be critical flow, which would likely be larger than the maximum purge rate. In light of these considerations, justify why the use of the maximum purge flow rate of 9000 cfm is appropriate or reevaluate the direct release pathway to the environment to consider the conditions of the containment during a design-basis LOCA.

Question 4
UFSAR Section, 6.2.3.2.3, “Containment Bypass Leakage,” (Reference 7) provides an evaluation of potential reactor containment bypass leakage pathways. One method of containing bypass leakage is via a water seal. Section 6.2.3.2.3 states:

“Those penetrations for which credit is taken for water seals as a means of eliminating bypass leakage, as outlined in Table 6.2-15, are preoperationally leak tested with air or water. For these water seals, either a loop seal is present, or the water for the seal is replenished from a large reservoir. For those valves maintaining a water seal, calculations have been done to verify that there is a sufficient water inventory for 30 days assuming leakage rates of 10 ml/hr of nominal valve diameter unless indicated otherwise below.”

UFSAR Section 6.2.3.2.3 also states:

“Following a LOCA, the feedwater line fill network is manually aligned from the main control room by opening the HPCI and RCIC injection valves to provide sealing water to the feedwater lines. In the unlikely event that either the HPCI or the RCIC injection line cannot be used as a flow path to the feedwater piping, the motor operated valve in the crosstie would be manually opened from the main control room. Manual operator action to align the fill network is not required sooner than 20 minutes following detection of a LOCA. This is due to the fact that during the time period required to refill the feedwater lines, no radioactive contaminants would be expected to leak through the feedwater isolation valves out to the environment as discussed below.”

While the feedwater lines typically have check valves that do not have closure times defined in the technical requirements manual, other systems may have water seals that credit manual operator actions to fill the line and create a seal. As discussed in question 1 above, the isolation time for 90 valves was increased to 120 seconds. The impact of the 120 second blowdown on the water seals is not provided in References 3 or 4.

Given the longer closure time and impact of higher DBA pressures and temperatures during the 120 second blowdown, please justify:

a) that there is sufficient water inventory for 30 days to maintain the water seals;

b) that operator actions to maintain the seals can still be accomplished in 20 minutes or more, and

c) that the seals will be maintained throughout accident.

Question 5
In Table 4 of Attachment 3 in Reference 1, the licensee indicates that aerosol deposition is credited within the volume between the reactor pressure vessel (RPV) and the inboard main steam isolation valve for the intact line. The licensee states on page 25 of Attachment 3 in Reference 1, that:

"The comparisons provided in Table 4 demonstrate that the aerosol deposition model used in H-1-ZZ-MDC-1880, Revision 3 for the MSIV leakage paths conservatively complies with the AEB 98-03 guidance."

The NRC staff reviewed AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," (Reference 8) and the model does not appear to credit deposition from the RPV to the inboard main steam isolation valve for either the intact or the failed line. This is appropriate since this line is part of the drywell boundary.

Given this part of the licensees model appears inconsistent with the NRC AEB 98-03 model, justify the following statement provided on Attachment 3, page 26 of Reference 1:

"In conclusion, the parameters, assumptions and methodologies used in the current licensing basis analysis are consistent with plant specific design inputs, NRC guidance, and industry applications and prior NRC approvals."

Question 6
UFSAR Section 6.2.4.2, "System Design" states:

"The closure times of containment isolation valves are selected to ensure rapid isolation of the primary containment following postulated accidents. The isolation valves in lines that provide an open path from the primary containment to the environs have closure times that minimize the release of containment atmosphere to the environs and mitigate the offsite radiological consequences. The isolation valves for lines outside the containment, in which high energy line breaks can occur, have closure times that minimize the resultant pressure and temperature transients as well as the radiological consequences."

UFSAR Table 3.6-4, "Blowdown Time History for High Energy Pipe Breaks Outside Primary Containment," contains the assumed isolation valve closure times for high energy lines. For those valves that were changed to 120 second closure times, provide an updated blowdown and an assessment and the impact on peak temperatures in rooms with high energy lines. Note the evaluation on page 4 of 22 of Reference 4 states that "The above sets of assumption provide conservative qualification requirements; and, therefore, long-term profile are not required."

It is not clear how the proposed increase in closure time for PCIVs is factored into the peak temperature and pressure assessment. For lines with changes to the closure time of PCIVs, please provide a confirmatory analysis of the pressure and temperature response of the secondary containment for high energy line ruptures occurring within the secondary containment (reference Standard Review Plan 6.2.3, "Secondary Containment Functional Design," Revision 3, Section III.3 (Reference 9).

Question 7
Standard Review Plan (SRP) 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," (Reference 10) provides guidance for reviewing the impact of high and moderate energy fluid system piping located outside of containment. This SRP also provides guidance for reviewing the impact of postulated failures on habitability of the control room and access to areas important to safe control of post-accident operations. If these review areas are part of the licensing basis for your facility please provide the impact of the increased PCIV closure time on these analyses or justify why these analyses are not needed.

Question 8
Per Regulatory Guide 1.183, Section 1.3.2, "Re-Analysis Guidance:"

"For selective implementations based on the timing characteristic of the AST, e.g., change in the closure timing of a containment isolation valve, re-analysis of radiological calculations may not be necessary if the modified elapsed time remains a fraction (e.g., 25%) of the time between accident initiation and the onset of the gap release phase. Longer time delays may be considered on an individual basis. For longer time delays, evaluation of the radiological consequences and other impacts of the delay, such as blockage by debris in sump water, may be necessary." [emphasis added]

Please justify that debris (reactor core debris, LOCA induced debris, or debris that caused the LOCA) will not block the PCIVs with a closure time of 120 seconds or justify why this analysis is not necessary.

Question 9
Attachment 1, page 15 of Reference 1 provides the licensee's response to the NRC staff's request for additional information (RAI) question 16. As stated in question 16, the review considers the possible case variations of anticipated operational occurrences and postulated accidents to verify that the licensee has identified the limiting cases.

The NRC staff's review of the change in PCIV closure times did not find any evaluation of the impact of these changes on UFSAR 15.6.2, "Instrument Line Pipe Break," or UFSAR 15.6.6, "Feedwater Line Break - Outside Primary Containment." Please provide an evaluation of the impact of the PCIV changes on all accidents in the design bases or include a justification why an evaluation is not needed. For those accidents analyzed, please provide the regulatory bases for the acceptance criteria and the regulatory guidance used to make this determination or the alternative methodology used.

Question 10
Attachment 1, page 15 of Reference 1 provides the licensee's response to the NRC staff's RAI, question 16. The licensee states for the Reactor Recirculation Pump Shaft Seizure event (UFSAR Section 15.3.3), "If purging of the containment is chosen, the release will be in

accordance with established technical specification limits." It appears that the only purge specification in the HCGS Technical Specifications is LCO 3.6.1.8, "Drywell and Suppression Chamber Purge System." Please clarify what the statement "will be in accordance with established technical specification limits," means and how it relates to the Reactor Recirculation Pump Shaft Seizure accident.

Question 11
Attachment 1, page 17 of Reference 1 provides the licensee's response to the NRC staff's RAI, question 17. The response includes updates of several sections of GEH report NEDC-33529P, from the original version of the report which was included in the application dated December 21, 2009. Section 4.3.1, Control Rod Drop Accident," of the updated GEH report contains a revised assumption regarding the number of Cobalt isotope rods reaching melting conditions. Please justify why the assumption in the updated version of the report is conservative.

Question 12
Table 4 of Regulatory Guide 1.183 (Reference 5) provides the "LOCA Release Phases," and the time for onset of a gap activity release for boiling water reactors. This gap release onset is 120 seconds and is based upon Reference 11. The gap release onset is dependent upon the fuel assembly linear heat generation rate and the method of operation of the facility. HCGS has undergone a power uprate and operates at conditions that appear to be outside the operating parameters assumed in Reference 5. These include increased linear power density and other changes such as the ability to operate under an expanded operating domain resulting from the implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLA). Given these changes, justify the assumption that the gap activity is released at 120 seconds or provide a revised analysis that incorporates the current operational parameters for your facility.

Question 13
Per the Revision History on page 5 of Calculation H-1-ZZ-MDC-1880, "Post-LOCA EAB, LPZ and CR Doses" (Reference 3), Revision 2 of the calculation was issued to assess the radiological impacts at a power level of 3917 MWt (i.e., in support of the power uprate).

Per Section 4.6.3 or Reference 3:

"The primary containment and the MSIVs are assumed to leak at the allowable Technical Specification peak pressure leak rate for the first 24 hours (Ref. 10.1, RGP A.3.7). This leakage is reduced to 50% of its TS value after the first 24 hours through day 30 (per Ref. 10.1, Sections 3.7 and 6.2). The flow is cut in half during these 29 days because the driving pressure after the first 24 hours averages less than 12.0 psig (that is, less than one-fourth of the Technical Specification peak pressure of 50.6 psig) and because flow is proportional to the square root of the pressure. The implied driving pressure for the 29-day period is 50.6 psig / 4 or 12.65 psig (27.35 psia). The post-LOCA pressure versus time curve for Case C in Reference 10.15 indicates that the pressure reaches a second peak of 15.8 psig at 7 hours and then decreases to the end of the curve at 8 hours. UFSAR Figure 6.2-40 indicates that the pressure is 12.7 psig at 24 hours, is less than 12 psig at 33 hours, and then drops off rapidly at 42 hours. UFSAR Figure 6.2-39 indicates that the average for the 29 days is only about 6 psig. (Note: If the KP system (MSIV Sealing System) deletion is credited in UFSAR Figure 6.2-39, the average for the 29 days is estimated to be only about 4 psig.) Thus the calculation is sufficiently conservative with respect to the 50% leakage rates based on the containment pressure behavior after 24 hours."

Other BWR Mark I designs of lower power levels do not assume this reduction in leakage until 36 hours into the accident. Was the impact of increased power level on the containment peak pressure and leakage considered? If not, justify the reduction of MSIV leakage at 24 hours given the revised power level.

Question 14
Figure 10, of Reference 3 provides a linear fit to AEB 98-03 (Reference 8) data. The fit at the 30% uncertainty is nonconservative and unjustified. Please justify why the fit is appropriate instead of interpolating between the 40% and 10th percentile values calculated in AEB 98-03.

Question 15
Reference 6, states that the proposed MSIV leakage release model is also consistent with the guidance of RG 1.183, Sections A6.1 through A6.5 and is consistent with the most recent NRC guidance as promulgated through NRC reviews and acceptance of the MSIV leakage models for the Peach Bottom Atomic Power Station. Contained in the list of references for Reference 6, is Reference 15 which is the "Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments RE: Application of Alternative Source Term Methodology," (Reference 12). Page 18 of the safety evaluation in Reference 12 stated the following:

"The NRC staff acknowledges that aerosol settling is expected to occur in the MSL piping, however, because of NRC staff concerns regarding the AEB-98-03 report and the lack of additional confirmatory information, it is not clear how much deposition (i.e., which settling velocity value) is appropriate for BWR MSL MSIV leakage analysis. The licensee has used a model based on the methodology of AEB-98-03, but included some additional conservatism to address NRC staff questions on the applicability of the AEB-98-03 methodology to Peach Bottom. The licensee assumed in its analysis that the TS allowed leakage at accident peak pressure is only from the two shortest MSLs. The licensee assumed a break of the shortest MSL with inboard MSIV failure leaking at the maximum TS leakage. Additionally, the licensee assumed that the second shortest intact MSL leaks at the maximum remaining TS allowed flow leakage at peak accident pressure. The other two MSLs were assumed to have no leakage for the proposed AST LOCA analysis. In addition, the licensee used the AEB-98-03 model with a 40th percentile settling velocity which is more conservative than use of the median settling velocity noted as reasonable in AEB-98-03. The licensee also showed in its parametric study that use of the 40th percentile settling velocity used for the credited MSL piping in its design analysis was bounding for the range of settling velocities described in AEB-98-03 for the piping between the outboard MSIVs to the TSVs. Given this information, the NRC staff finds the Peach Bottom MSL settling model for aerosol deposition used in its AST LOCA analysis to be reasonably conservative."

The NRC staff utilized the sum total of several considerations to form the approval for Peach Bottom. These considerations included the conclusions of a parametric study which are given below:

"The aerosol gravitational deposition and elemental iodine removal in the post-LOCA MSIV leakage release paths are conservatively modeled for Peach Bottom and maintain the conservative characteristics of the method described in AEB 98-03.

The MSIV leakage model is structured to provide a conservative bound for a large range of settling velocities including very fine aerosol particles represented by a lower settling velocity,

lesser deposition, and higher dose as well as coarser aerosol particles represented by a higher settling velocity, larger deposition, and lower dose.

The conservative model used in the licensee design analysis provides an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensates for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion."

Given the many considerations and the number of documents involved, the NRC is uncertain whether the MSIV leakage methodology used by the licensee is consistent with the Peach Bottom model. To aid the NRC staff in making these determinations please supply the following information:

a) The parametric study for the Hope Creek LOCA using the same methods and parameters as was performed by Peach Bottom. Justify that the conclusions of this study are consistent with the Peach Bottom conclusions and demonstrate that the 10 CFR 50.67 criteria are met.

b) A justification why the Hope Creek licensing basis MSIV leakage deposition model credits aerosol and elemental deposition between the inboard and outboard MSIVs in the failed line, when the model it is stated to be consistent with (the Peach Bottom model) does not credit aerosol or elemental deposition in this volume.

c) Provide a table of each item considered in the Peach Bottom Safety Evaluation (Reference 12) Section 3.2.2.8, "Assumptions on Main Steam Isolation Valve Leakage," and a justification how the Hope Creek model complied with these considerations or

d) Justify why the above information is not needed.

<u>Question 16</u>
Page 20 of Attachment 3 to Reference 1 states:

"Subsequent HCGS plant receiving its AST license amendment, the industry and NRC gained experience with, and an understanding of, aerosol deposition in the main steam lines following a LOCA. The NRC informed some AST license amendment applicants of a concern related to the modeling of lighter aerosol particles, which experience lesser gravitational deposition in the seismically supported lines beyond the outboard MSIVs. This concern was addressed in AST license amendments for the Peach Bottom (PB) plant. While this issue was not identified as an industry concern, PSEG NUCLEAR made the prudent decision to address the concern in H-1-ZZ-MDC-1880, Revision 3, by updating the aerosol deposition model with respect to the latest regulatory developments (see discussion below). **This resulted in some loss of dose margin.**" [emphasis added]

Page 22 of Attachment 3 to Reference 1 states:

"The net impact of the MSIV, PCIV and ESF leak rate changes was an increase in the control room dose from 4.16 to 4.17 Rem TEDE, a decrease in the Exclusion Area Boundary (EAB) dose from 3.10 to 1.43 Rem TEDE, and a decrease in the Low Population Zone (LPZ) dose from 0.696 to 0.548 Rem TEDE."

This overall change which included a methodology change and relaxation of the amounts of leakage (increasing PCIV closure times to 120 seconds, and ESF leak rates from 1 gallon per minute to 2.85 gallons per minute) appears to have not resulted in a loss of dose margin. The EAB and LPZ dose margin has increased (a decrease in doses increases the margin to the dose limit therefore increasing dose margin). The increase in control room doses is negligible. Given the overall change does not appear to have resulted in a loss of margin, it unclear how the methodology change described above could have resulted in a loss of dose margin. Given this conflict justify how updating of the aerosol deposition model results in a loss of dose margin.

Question 17
The licensee's letter dated June 10, 2010 (Reference 13), provided a response to NRC RAI#4 concerning the gamma heating effect on the spent fuel pool (SFP) walls. The NRC's RAI stated, in part, that:

"Please provide the **detailed analysis, assumptions and calculations** that led to the conclusion that the effect of gamma heating on the HCGS SFP walls will be minimized if the GE14i bundles are stored four feet from the SFP walls and that there is no limitation on the amount of time a GE14i bundle may remain in the SFP at this location." [emphasis added]

The licensee's response provided inputs, assumptions and results of the calculations. However the detailed analysis and calculations were not provided. Please submit this information for NRC staff review.

Question 18
The licensee's letter dated June 10, 2010 (Reference 13), provided a response to NRC RAI#5 concerning the process for removal of the isotope rods from the Isotope Test Assemblies (ITAs). Part "c" of the RAI requested information regarding the probability that the SFP wall will undergo significant gamma heating during the removal process. The license's response indicated that a calculation was performed to address this issue. Please submit the calculation that was performed for NRC staff review.

References

1. PSEG letter LR-N10-0163 to NRC, “Response to Request for Additional Information - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project),” dated May 11, 2010 (ADAMS Package Accession No. ML101390320 containing 5 documents, Attachment 1 is ML101390319, Attachment 2 and 3 are ML101390314, Attachment 4 is ML101390315, Attachment 5 is ML101390316 (Reference 3) and Attachment 6 is ML101390318 (Reference 6).) The letter is in ML101390314).

2. PSEG letter LR-N09-0290 to NRC, “License Amendment Request Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project),” dated December 21, 2009 (ADAMS Package Accession No. ML093640193, letter is contained in ADAMS Accession No. ML093640198).

3. Calculation H-1-ZZ-MDC-1880, Revision 3, “Post-LOCA EAB, LPZ and CR Doses”, (Attachment 5 to Reference 1, ADAMS Accession No. ML101390316).

4. Hope Creek Technical Evaluation DCR 80096650-0210, Revision 0, “Technical Evaluation to Determine post-LOCA Design Functional Impact on Systems & Components Located Downstream of Outboard Containment Isolation Valves which are Expected to Remain Open for 120 seconds at the Hope Creek Generating Station (HCGS),” dated November 15, 2009.

5. Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” dated July 2000 (ADAMS Accession No. ML003716792).

6. HCGS 50.59 Evaluation No. HC 2008-215, “Leakage Reduction Program Calculation, Revision 0,” Attachment 6 to Reference 1 (ADAMS Accession No. ML101390318)

7. HCGS Updated Final Safety Analysis, Revision 17, dated June 23, 2009.

8. J. Schaperow et al., “Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term,” U.S. Nuclear Regulatory Commission, AEB 98-03, December 9, 1998.

9. NUREG-0800, Standard Review Plan 6.2.3, “Secondary Containment Functional Design,” Revisoin 3.

10. NUREG-0800, Standard Review Plan, 3.6.1, “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,” Revision 3.

11. U.S. Nuclear Regulatory Commission safety evaluation entitled, “Grand Gulf Nuclear Station, Unit 1 - Acceptance of Boiling Water Reactors Owners Group (BWROG) Report, ’Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR,’ July 1996, TAC M98744” (ADAMS Legacy Library Accession No. 9909150040).

12. “Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments RE: Application of Alternative Source Term Methodology (TAC Nos: MD2295 and MC2296),” August 23, 2006 ADAMS Accession No. ML082320406).

13. PSEG letter LR-N10-0210 to NRC, “Response to Request for Additional Information - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project),” dated June 10, 2010.