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10 CFR 50.90

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

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

Subject: **Response to Request for Additional Information - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)**

References: (1) Letter from PSEG to NRC, "License Amendment Request Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)," dated December 21, 2009

(2) Letter from PSEG 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

(3) Letter from PSEG 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

In Reference 1, PSEG Nuclear LLC (PSEG) submitted a license amendment request (H09-01) for the Hope Creek Generating Station (HCGS). Specifically, the proposed change would modify License Condition 2.B.(6) and create new License Conditions 1.J and 2.B.(7) as part of a pilot program to irradiate Cobalt (Co)-59 targets to produce Co-60. In addition to the proposed license condition changes, the proposed change would also modify Technical Specification (TS) 5.3.1, "Fuel Assemblies," to describe the specific Isotope Test Assemblies (ITAs) being used.

In References 2 and 3, PSEG Nuclear LLC (PSEG) submitted responses to an NRC Request for Additional Information (RAI) on the license amendment request. Subsequently the NRC has provided PSEG with a further RAI (RAI3). The responses to the RAI3 Questions 1 through 4 and Questions 6 through 9 are provided in Attachment 1 of this letter. The response to RAI3 Question 11 is provided in Attachments 2 (Proprietary) and 3 (Non-proprietary) of this letter. The responses to RAI3 Questions 17 and 18 are provided in Attachment 4 (Proprietary); this document (GEH Report 0000-0120-1959-R1) is GEH Proprietary Information in its entirety, therefore no non-proprietary version is provided. The responses to the remainder of the RAI3 Questions (Questions 5, 10, 12 through 16) will be provided in a subsequent submittal.

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Attachments 2 and 4 to this letter provide information which GEH considers to be proprietary. GEH requests that the proprietary information in Attachments 2 and 4 be withheld from public disclosure, in accordance with the requirements of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4). Signed affidavits supporting this request are included in Attachments 2 and 4 to this letter.

PSEG has reviewed the information supporting a finding of no significant hazards consideration that was provided in Reference 1. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. No new regulatory commitments are established by this submittal.

If you have any questions or require additional information, please do not hesitate to contact Mr. Jeff Keenan at (856) 339-5429.

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

Executed on 7/28/10
(Date)

Sincerely,

Robert C. Braun
Sr, Vice President – Nuclear Operations

Attachments (4)

S. Collins, Regional Administrator - NRC Region I
R. Ennis, Project Manager - USNRC
NRC Senior Resident Inspector - Hope Creek
P. Mulligan, Manager IV, NJBNE
Commitment Coordinator – Hope Creek
PSEG Commitment Coordinator - Corporate

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 <u>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</u>." [emphasis added]

Per Regulatory Position C.5.1.2, non-engineered safety features (ESF) piping outboard of the PCIVs should <u>not</u> 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).

<u>Question 1 Response</u>

All PCIV's isolation times have been maintained at their values prior to issuance of Reference 3. The 120 second change in Reference 3 was never implemented (refer to HCGS TRM Table 3.6.3-1, Revision 0) and will not be implemented. The issue has been entered into the PSEG corrective action program to ensure the 120 second change cannot be implemented in any manner utilizing Reference 3 as the basis. Note that dose consequences of Reference 3 remain bounding for the non-implementation of the 120 second change.

<u>Question 2</u>
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 2 Response

The cited text in 10 CFR 50.59 Evaluation HC 2008-215, Revision 0, was inadvertently stated backwards; the words "less than" should have been written as "greater than". This statement in the 10 CFR 50 59 evaluation is related to the 120 second PCIV closure time change; as noted in the response to Question 1, this change was never implemented and will not be implemented.

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 3 Response

As discussed in the response to Question 1, the 120 second PCIV isolation time change was never implemented and will not be implemented. With no change in PCIV closure times there is no release path to the environment via the containment purge system.

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 4 Response

As stated in response to question 1, the PCIV isolation times have remained at their original design values (the 120 second closure time was never implemented and will not be implemented). Since there is no change in PCIV closure times there is no impact on the water seals.

Question 5

The response to Question 5 will be provided in a subsequent submittal.

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).

<u>Question 6 Response</u>

As stated in response to Question 1, the PCIV isolation times have remained at their original values (the 120 second closure time was never implemented and will not be implemented). Since there is no change to PCIV closure times, no additional analysis is required.

<u>Question 7</u>
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.

<u>Question 7 Response</u>

As stated in response to Question 1, all PCIV's isolation times have remained at their original values (the 120 second closure time was never implemented and will not be implemented). Since there is no change to PCIV closure times, no additional analysis is required.

<u>Question 8</u>
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 <u>other impacts of the delay</u>, 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.

<u>Question 8 Response</u>

As stated in response to Question 1, all PCIV's isolation times have remained at their original values (the 120 second closure time was never implemented and will not be implemented). Since there is no change in PCIV closure times, no additional analysis is required.

<u>Question 9</u>

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.

<u>Question 9 Response</u>

As stated in response to Question 1, all PCIV's isolation times have remained at their original values (the 120 second closure time was never implemented and will not be implemented). Since there is no change in PCIV closure times no evaluation is required.

<u>Question 10</u>

The response to Question 10 will be provided in a subsequent submittal.

<u>Question 11</u>

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 11 Response

The response Question 11 is provided in Attachments 2 (Proprietary) and 3 (Non-proprietary) of this letter.

Questions 12 through 16

The responses to Questions 12 through 16 will be provided in a subsequent submittal.

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 17 Response

The response to Question 17 is provided in Attachment 4 (Proprietary) of this letter.

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.

Question 18 Response

The response to Question 18 is provided in Attachment 4 (Proprietary) of this letter.

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.

RAIs 17 and 18 Response - Consolidated Report 0000-0120-1959-R1 July 2010 (Proprietary)

The header of each page in this document carries the notation "GEH PROPRIETARY INFORMATION{3}." The superscript notation{3} refers to Paragraph (3) of the enclosed affidavit, which provides the basis for the proprietary determination. This document is GEH Proprietary Information in its entirety.

]] The reactivity of GE14i is similar to the current fuel design; therefore, the assumption in the updated version of the report is conservative.

SAR/LAR Impact

None.

NRC RAI 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.

GEH Response

The updated analysis reported in NEDC-33529P adds to the licensing basis CRDA source term a Co-60 inventory [[ ]] This assumption is conservative for three main reasons: the CRDA is a localized event; [[

]]

The CRDA is a localized event, meaning the entire core is not affected but only bundles in the immediate area of the dropped control rod. [[

]] The licensing basis analysis considers a failure of 850 fuel rods (8x8 design), which is equivalent to roughly 14 bundles of a 764-bundle core, and a melt fraction of 0.77% of the failed fuel. [[

]] The assumption [[

]] is therefore conservative.

[[

RAI3 Question 11 Response (Non-Proprietary)

This is a non-proprietary version the RAI3 Question 11 Response from which the proprietary information has been removed. Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here [[ ]].

Note: Each header page also includes a notation to "RW-PSG-KT1-10-079" and "Enclosure 2"; these refer to the GEH letter that provided the material to PSEG