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

PSEG
*Nuclear LLC*

10 CFR 50.90

AUG 12 2010
LR-N10-0306

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

(4) 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 July 28, 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 provided PSEG with a further RAI (RAI3). PSEG provided a partial response to RAI3 in Reference 4 (the

A001
NRR

responses to the RAI3 Questions 1 through 4, 6 through 9, 11, 17 and 18). The remainder of the RAI3 responses are provided in Attachment 1 to this letter (RAI3 Questions 5, 10, and 12 through 16). Attachments 2 through 7 provide additional supporting documentation, as discussed in Attachment 1. Attachment 2 is a calculation that has its own attachment (Attachment 14.2) that contains information which GEH considers to be proprietary. Attachment 14.2 has been extracted from Attachment 2 as a standalone document; it is provided in proprietary form in Attachment 3 to this letter. The associated proprietary RADTRAD files are provided as Attachment 7 to this letter. GEH requests that the proprietary information in Attachment 3 and 7 be withheld from public disclosure, in accordance with the requirements of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4). A signed affidavit supporting this request is included with Attachment 3 and Attachment 7. A non-proprietary version of Attachment 3 is provided in Attachment 4 of this letter. Since all of Attachment 7 is proprietary, no non-proprietary version is required.

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 8/12/10
(Date)

Sincerely,

Robert C. Braun
Senior Vice President Nuclear
PSEG Nuclear LLC

Attachments (7):

Attachment 1 - RAI3 Response
Attachment 2 - Calculation H-1-ZZ-MDC-1880, Revision 4, "Post-LOCA EAB, LPZ, and CR Doses"
Attachment 3 - Attachment 14.2 of Calculation H-1-ZZ-MDC-1880, Revision 4 (GEH Proprietary Information)
Attachment 4 - Attachment 14.2 of Calculation H-1-ZZ-MDC-1880, Revision 4 (GEH Non-Proprietary Information)
Attachment 5 - 10 CFR 50.59 for Calculation H-1-ZZ-MDC-1880, Revision 4 (HC-10-125)
Attachment 6 - RADTRAD Electronic files (CD) for Calculation H-1-ZZ-MDC-1880, Revision 4
Attachment 7 - RADTRAD Electronic files (CD) for Attachment 14.2 of Calculation H-1-ZZ-MDC-1880, Revision 4 (GEH Proprietary Information)

C:

M. Dapas, Regional Administrator (Acting) - 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.

> Note: PSEG provided a partial response to this RAI (RAI3) in a letter dated July 28, 2010 (the responses to the RAI3 Questions 1 through 4, 6 through 9, 11, 17 and 18). The response to the remainder of the RAI3 Questions is provided below (Questions 5, 10, and 12 though 16).

***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 5 Response**

The RADTRAD model presented in AEB-98-03 does credit aerosol deposition and elemental iodine removal for the intact main steam pipe segment that extends from the reactor pressure vessel (RPV) nozzle to the inboard (i.e., first) MSIV. The AEB-98-03 model does not credit aerosol deposition and elemental iodine removal on the main steam pipe segment between the RPV nozzle and the inboard MSIV on the failed main steam line, but it does credit aerosol deposition and elemental iodine removal between the failed inboard MSIV and the outboard (i.e., second) MSIV. The basis for this conclusion is as follows.

Credit for aerosol deposition and element iodine removal in the intact main steam pipe segment is confirmed by the inputs to the RADTRAD input file presented in Appendix C.1 of AEB-98-03 and the information presented in Appendixes A and B of AEB-98-03. Specifically, the two main steam pipe segments that make up the volume nodes for the intact main steam line in AEB-98-03 are described in paragraph 5 on the first page of Appendix A to AEB-98-03. The first pipe segment volume is the length of piping between the reactor vessel and the first MSIV. The second pipe segment volume is the length of piping between the first and second MSIVs. Pathway 14 is shown in Figure 1 of AEB-98-03 as the transfer pathway between compartment volume 8 (MSIV_2-8) and compartment volume 9 (MSIV_3-9) and Pathway 14 is defined on page 4 of Appendix C.1 to AEB-98-03 as the transfer pathway between compartment 8 and compartment 9. The volumes for compartments 8 and 9 are presented on page 3 of Appendix C.1 to AEB-98-03 and are 440 cu ft and 292 cu ft, respectively. Page B-2 of Appendix B to AEB-98-03 indicates that the 440 cu ft volume is the first pipe segment volume of the intact main steam line and the 292 cu ft volume is the second pipe segment volume in the intact main steam line. Pathway 14 thus represents the transfer pathway between the intact main steam line volume between the RPV nozzle and the inboard MSIV and the intact main steam line volume between the inboard and outboard MSIVs. Page 10 of Appendix C.1 of AEB-98-03 presents the filter model for Pathway 14 (i.e., for the transfer of activity from the intact MSIV line segment between the RPV nozzle and the inboard MSIV to the intact MSIV pipe segment between the inboard and outboard MSIVs). A 0 to 30 day flow rate of 4.775 cfm, an aerosol deposition removal efficiency of 71% and an elemental iodine removal efficiency of 50% are indicated for the filter model for Pathway 14 on page 10 of Appendix C.1 of AEB-98-03. The aerosol deposition removal efficiency and elemental iodine removal efficiency apply to the upstream volume to Pathway 14, i.e., to the main steam pipe segment between the RPV nozzle and the inboard MSIV on the intact main steam line. The flow rate of 4.775 cfm is described on page B-2 of AEB-98-03 and is the flow rate from the first intact main steam line pipe segment volume to the second intact main steam line pipe segment volume and is also the flow rate from the second intact main steam line pipe segment volume to the environment.

The absence of credit for aerosol deposition and elemental iodine removal in the failed main steam line pipe segment between the RPV nozzle and the inboard MSIV and the application of credit for aerosol deposition and elemental iodine removal in the failed main steam line between the inboard and outboard MSIVs is confirmed by the inputs to the RADTRAD input file presented in Appendix C.1 of AEB-98-03 and the information presented in Appendixes A and B of AEB-98-03. The failed (i.e., broken) main steam line is modeled as one segment (i.e., volume) of piping between the first and second MSIVs in paragraph 5 on the first page of Appendix A to AEB-98-03. The volume between the first and second MSIVs of the failed main steam line is referred to as compartment volume 7 (MSIV_1-7) on Figure 1 of AEB-98-03. Pathway 12 is shown in Figure 1 of AEB-98-03 as the transfer pathway between the drywell and compartment volume 7. There is no compartment volume identified for the main steam pipe segment between the RPV nozzle and the first MSIV for the failed MSIV line as indicated on Figure 1 and the text in paragraph 5 on the first page of Appendix A to AEB-98-03. Since the pipe segment between the RPV nozzle and the first MSIV is not defined for the RADTRAD model used in AEB-98-03, the RADTRAD model used in AEB-98-03 does not take credit for aerosol deposition or elemental iodine removal for the main steam pipe segment between the RPV nozzle and the first MSIV for the failed main steam line. Pathway 15 is shown in Figure 1 and defined on page 4 of Appendix C.1 to AEB-98-03 as the transfer pathway between compartment volume 7 (MSIV_1-7) and the environment (compartment volume 6) and thus Pathway 15 represents the transfer pathway between the failed main steam line volume between the inboard and outboard MSIVs and the environment. Page 11 of Appendix C.1 of AEB-98-03 presents the filter model for Pathway 15. A 0 to 30 day flow rate of 3.183 cfm, aerosol deposition removal efficiencies of 71%, 85%, 88%, 90%, and 42% over the 30 day period, and an elemental iodine removal efficiency of 50% are indicated for the filter model for Pathway 15. These aerosol deposition removal efficiencies and the elemental iodine removal efficiency apply to the upstream volume of Pathway 15, i.e., to the main steam pipe segment volume between the inboard and outboard MSIV on the failed main steam line. The flow rate of 3.183 cfm is described on page B-2 of AEB-98-03 as the flow rate from the main steam line pipe volume between the MSIVs to the environment.

As discussed in the response to RAI3 Question 15, HCGS Calculation H-1-ZZ-MDC-1880 has been revised (Revision 4). A description of differences between the MSIV leakage pathway model used in Revision 4 of Calculation H-1-ZZ-MDC-1880 to the model used in Revision 3 is provided in RAI3 Question 15 response.

***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 10 Response**

HCGS-UFSAR Section 15.3 states:

> If purging of the containment is chosen, the release will be in accordance with established technical specification limits.

This statement appears to be a holdover from when the Radiological Effluent Technical Specifications (RETS) were included in the Technical Specifications. The previous LCOs that were in the RETS are now included in the Offsite Dose Calculation Manual (ODCM) as Radiological Effluent Controls (REC)[1]. This UFSAR issue has been entered into the PSEG Corrective Action Program.

UFSAR Section 15.3.3.5 discusses the radiological consequences of reactor recirculation pump shaft seizure. The discussion states that the radiological consequences are the same as discussed in Section 15.3.1.5, which concerns reactor recirculation pump trip and discusses the purging of containment. The UFSAR section indicates that under this condition the plant operator can choose to leave the activity in the primary containment or discharge it to the environment under controlled conditions, and includes the statement about purging in accordance with established technical specification limits.

In addition, the following information is provided concerning operation of the purge system.

HCGS Technical Specifications LCO 3.6.1.8 for the drywell and suppression chamber purge system states:

> *"The drywell and suppression chamber purge system, including the 6-inch nitrogen supply line, may be in operation for up to 500 hours each 365 days with the supply and exhaust isolation valves in one supply line and one exhaust line open for containment pre-purge cleanup, inerting, de-inerting, or pressure control.*"*

*The footnote for the LCO states:

> *"Valves open for pressure control are not subject to the 500 hours per 365 days limit, provided the 2-inch bypass lines are being utilized."*

The Technical Specification Bases state that

> *"The 500 hours/365 days limit for operation of the purge valves and the 6" nitrogen supply valve during plant Operational Conditions 1, 2 and 3 is intended to reduce the probability of a LOCA occurrence during the above operational condition when applicable combination of the above valves are open."*
>
> *"Blow-out panels are installed in the CPCS ductwork to provide additional assurance that the FRVs will be capable of performing its safety function subsequent to a LOCA."*
>
> *"The use of the drywell and suppression chamber purge exhaust lines for pressure control during plant Operational Conditions 1, 2 and 3 is unrestricted provided 1) only the inboard purge exhaust isolation valves on these lines and the vent valves on the 2-inch vent paths are used and 2) the outboard purge exhaust isolation valves remain closed. This is because in such a situation, the vent valves will sufficiently choke the flow and additionally the applicable valves will close in a timely manner during a LOCA or steam line break accident and therefore the control room and the site boundary dose guidelines of applicable 10 CFR dose limits will not be exceeded in the event of an accident. The design of the puge supply and exhaust isolation valves and the 6-inch nitrogen supply valve meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations"."*

PSEG Procedure HC.OP-SO.GS-0001, Containment Atmosphere Control System Operation, which outlines the steps necessary for the startup, shutdown and operation of the system,

---

1 HCGS Amendment 121, September 8, 1999

includes prerequisites and limitations for purging the drywell and suppression chamber. The procedure ensures that only one supply and one exhaust line are open at any given time. The procedure calls for performing radioactive gaseous waste sampling in accordance with the ODCM prior to initiation of purge flow.

PSEG Procedure HC.OP-EO.ZZ-0318, Containment Venting, provides the guidance to vent primary containment that would be required during an emergency, or post emergency condition in which drywell pressure could not be maintained below 65 psig or hydrogen and oxygen concentrations could not be maintained below combustible levels.

***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 12 Response**

The implementation of the Extended Power Uprate (EPU) at Hope Creek Generating Station (HCGS) which included consideration of operating within the domain allowed by the Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) did not adversely affect the key parameters described in Reference 11.

As noted in Reference 12-3 below, NEDC-33076P, Revision 2, Safety Analysis Report for Hope Creek Constant Pressure Power Uprate (August 2006), Section 2.2.2, the Linear Heat Generation Rate (LHGR) and the Average Planar Linear Heat Generation Rate (APLHGR) limits did not change with the implementation of EPU with ARTS/MELLLA. In addition, Sections 4.3 and 9.0 of Reference 12-3 describe the effect of EPU with ARTS/MELLLA on design basis events. No new events or new event sequences were identified that could lead to fuel failure, i.e., onset of gap release.

In order to facilitate the NRC review of this response and further validate that the implementation of EPU including consideration of ARTS/MELLLA did not affect the key parameters of Reference 11, GE Hitachi (GEH) performed an evaluation based upon the current HCGS plant and operating configuration. Background information for the evaluation and a description of the evaluation is described below.

The timing of fission product release from perforated fuel rods (i.e., the gap activity release) is based on the BWR-specific value of the timing of the gap activity release phase of a Loss of Coolant Accident (LOCA) as calculated in the BWR Owners Group (BWROG) Report, "Prediction of the Onset of Fission Gas Release From Fuel in Generic BWR," NEDC-32963A, included in Reference 12-1. This report was prepared by the General Electric Company (GE) and was sponsored by the BWROG. GE calculated the minimum time to the onset offission product release from perforated fuel rods to be 121 seconds using a bounding BWR plant configuration and fuel design. The calculation was intended to be generic for all operating BWR plants using licensed BWR fuel.

The evaluation performed by GEH to support this response considered the current HCGS extended operating domain (EPU and ARTS/MELLLA) and fuel types (GE14 and GE14i). Also, consistent with the original BWROG calculation described in Reference 12-1, the GEH evaluation considered conservative values for operating parameters such as Peak Linear Heat Generation Rate (PLHGR) and Initial Minimum Critical Power Ratio (MCPR). These key operating parameters and other related inputs are consistent with parameters provided in Section 4 of Reference 12-2, the HCGS SAFER/GESTR-LOCA report.

The GEH evaluation confirms that the conservative gap activity release value of 121 seconds for a postulated design basis large break LOCA with complete ECCS failure as reported in Reference 12-1 remains applicable to GE14 and GE14i fuel types operating within the currently licensed HCGS expanded operating domain.

<u>Question 12 Response References</u>

12-1 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).

12-2 PSEG Letter to NRC dated 11/07/2005, ADAMS Accession No. ML053250465, SAFER/GESTR-LOCA, Loss of Coolant Accident Analysis for Hope Creek Generating Station at Power Uprate, NEDC-33172P, Revision 0, March 2005.

12-3 Attachment 4 to PSEG letter (LR-N06-0286) to NRC dated September 18, 2006, "Request for License Amendment Extended Power Uprate, Hope Creek Generating Station, Facility Operating License NPF-57, Docket No. 50-354", ADAMS Accession No. ML062680451, LR-N06-0286, NEDC-33076P, Revision 2, Safety Analysis Report for Hope Creek Constant Pressure Power Uprate, (August 2006).

***<u>Question 13</u>***

*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 13 Response**

The impact of EPU on containment peak pressure, temperature and leak flow rate was considered. However, the reduction in primary containment and MSIV leakage at 24 hours is being removed from the calculation. Based on the time constraints for review of the requested amendment, PSEG has decided to remove this element to reduce the number of review issues. Additional discussion on the PSEG model and the revised calculation is provided in the response to RAI3 Question 15.

***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 10$^{th}$ percentile values calculated in AEB 98-03.*

**Question 14 Response**

As discussed in the response to RAI3 Question 15, HCGS Calculation H-1-ZZ-MDC-1880 has been revised (Revision 4). Revision 4 uses a 40$^{th}$ percentile settling velocity from AEB-98-03 (Table A-1).

***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 infomation, 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.*

**Question 15 Response**

a) Calculation H-1-ZZ-MDC-1880 has been revised – see the response to part (d) below. The revised calculation does not rely on the Peach Bottom model or Safety Evaluation for justification (Reference 12). The MSIV model used in the Revision 4 of Calculation H-1-ZZ-MDC-1880 has one MSIV pipe compartment volume for each of the failed and intact main steam lines. A 40$^{th}$ percentile settling velocity was used to determine aerosol deposition in the compartment volumes for both the failed and intact main steam lines. Since only one compartment volume is used instead of two compartment volumes in series, the concerns over what percentile settling velocity to use for the second compartment volume (in order to properly model removal of smaller and lighter aerosol particles for the downstream volume), and the associated parametric study, are no longer applicable.

b) Calculation H-1-ZZ-MDC-1880 has been revised – see the response to part (d) below. The revised calculation does not rely on the Peach Bottom MSIV leakage deposition model for justification.

c) Calculation H-1-ZZ-MDC-1880 has been revised – see the response to part (d) below. The revised calculation does not rely on the Peach Bottom model or Safety Evaluation for justification. (Reference 12).

d) Revision 4 of Calculation H-1-ZZ-MDC-1880 has been issued (Attachment 2 of this submittal[2]) The 10 CFR 50.59 evaluation for Revision 4 is also attached (Attachment 5). The RADTRAD Files for Revision 4 are provided in Attachment 6. The revised calculation does not rely on the Peach Bottom model or Safety Evaluation (Reference 12) and includes elements from Revision 2 of the calculation. Table 1E of the calculation provides a comparison of the differences between Revisions 2, 3 and 4. Table 1E of Revision 4 also provides a cross reference to Section 2 of the calculation that discusses the changes and justifications. A discussion of how the changes relate to Revision 2 is also provided, where appropriate.

The changes in Revision 4 are summarized below:

1. The release of containment airborne activity through the primary containment isolation valves (PCIV) is removed from the calculation. Revision 3 models an increase in the PCIV isolation time to 120 seconds by including a 120 second release of primary coolant

2 Attachment 14.2 of Revision 4 of Calculation H-1-ZZ-MDC-1880 contains GEH proprietary information; it has been extracted from Attachment 2 as a standalone document and is provided in proprietary form in Attachment 3 to this letter. The associated proprietary RADTRAD files are included in Attachment 7 to the letter. A non-proprietary version of Attachment 3 is provided as Attachment 4 of this letter. An affidavit for withholding the Attachments 3 and 7 proprietary information is provided with these attachments.

activity through the PCIV. The removal of this pathway from the calculation is consistent with the current PCIV isolation times and the same approach used in Revision 2.

2. Elemental iodine removal in the main steam lines is calculated using the J. E. Cline methodology in Revision 4. Revision 3 assumes a factor of two reduction in the elemental iodine based on the analysis in AEB-98-03. The Revision 3 approach may not be appropriate to HCGS since AEB-98-03 is specific to Perry. Use of the J. E. Cline methodology is the same as the approach in Revision 2.

3. Main steam isolation valve (MSIV) leakage is modeled using one well mixed volume in each steam line. This is a change from the two compartment model used in Revision 3. It is also different from Revision 2, which used a one compartment model but assumed a plug flow model rather than a well mixed model.

4. Aerosol removal by deposition in the main steam lines is based on the 40$^{th}$ percentile settling velocity. Revision 3 uses a two compartment steam line model for both steam lines. Different settling velocities (50$^{th}$ and 30$^{th}$ percentile) are used in the two compartments to account for changes in the aerosol size distribution between the two compartments. The use of a single settling velocity is consistent with the change to a single compartment model. The 40$^{th}$ percentile settling velocity is the same settling velocity that was used in Revision 2.

5. Containment and MSIV leakage rates are held constant for the loss-of-coolant accident (LOCA) duration, rather than being reduced by 50% after 24 hours. This is a change to the approach used in both Revision 3 and Revision 2, which assumed the 50% reduction after 24 hours.

6. Control room unfiltered inleakage rate is reduced from 350 cfm to 300 cfm. This is a change to the input used in both Revision 3 and Revision 2, and is based on the results of inleakage testing performed in accordance with the Control Room Envelope Habitability Program (TS 6.16).

7. The dose impact of including 12 Isotope Test Assemblies (ITA) in the reactor core is evaluated. The ITAs will contain Co-59 targets that will be irradiated in the reactor core to produce Co-60.

Section 8.3 of Revision 4 provides a discussion of the dose impact associated with the changes in input parameters and methodologies associated with Revisions 2, 3, and 4. The calculation H-1-ZZ-MDC-1880, Revision 4, changes had a net effect of reducing the post-LOCA control room dose relative to the dose calculated in calculation H-1-ZZ-MDC-1880, Revision 3. The offsite EAB and LPZ doses are minimally increased; all the radiological consequences are within regulatory guidelines.

Section 8.3 of Revision 4 includes a table comparing the results from Revisions 2, 3, and 4 for each of the transport pathways analyzed (containment, ESF, and MSIV leakage) for control room, EAB and LPZ doses. The reduction in doses due to containment leakage in Revisions 3 and 4 compared to Revision 2 is primarily due to crediting reduction in airborne elemental iodine from the containment atmosphere resulting from plateout on wetted containment surfaces. The reduction in doses due to MSIV leakage in Revisions 3 and 4 compared to Revision 2 is primarily attributed to using post-LOCA drywell temperature and pressure to determine MSIV volumetric flow rates instead of using the volumetric flow rates associated with standard temperature and pressure conditions. These two changes are the primary factors that offset the increase in ESF leakage.

***Question 16***

*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 16 Response**

As is stated in the response to RAI#19 (Reference 1), the 'resulting loss of dose margin' statement only refers to the increase in dose that is attributed to using a smaller percentile settling velocity (i.e., 30% instead of a larger percentile settling velocity) to account for lighter aerosol particles in the main steam pipe volumes downstream of the outboard MSIVs. It was not explicitly or implicitly indicated that this statement described the overall change in doses. Use of the smaller percentile settling velocity to model lighter aerosol particles downstream of the outboard MSIVs was just one element of Revision 3 to the calculation as discussed in the RAI#19 response (Table 1 of RAI#19). The overall impact of all the elements of Revision 3 of the calculation on the control room, EAB, and LPZ doses is summarized in the response to RAI#19 (Reference 1).

As discussed in the response to RAI3 Question 15, HCGS Calculation H-1-ZZ-MDC-1880 has been revised (Revision 4).

## 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.