April 8, 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 April 8, 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

April 8, 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)

TThe attached draft request for additional information (RAI) was transmitted on April 8, 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

<u>DISTRIBUTION</u>
PUBLIC
LPL1-2 R/F
RidsNrrDorlLpl1-2 Resource
RidsNrrDorlDpr Resource
RidsNrrPMHopeCreek Resource
MPanicker, NRR/DSS/SNPB
MBlumberg, NRR/DRA/AADB
SGarry, NRR/DIRS/IHPB

ACCESSION NO.: ML100990403

OFFICE	LPL1-2/PM
NAME	REnnis
DATE	4/8/10

OFFICIAL RECORD COPY

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 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093640193), 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, the first question below was transmitted to PSEG on January 28, 2010. No response to this question has been received yet. As such, it is included here in order to have a consolidated list of all request for additional information (RAI) questions asked to date.

1) In Table 1, “Equilibrium Cobalt-60 Inventory,” on page 14 of Attachment 7 to the application dated December 21, 2009 (Reference 1), the licensee uses the same values of neutron flux as that used in a similar table[1] for Clinton Power Station (CPS). HCGS and CPS are boiling-water reactors with different rated thermal power levels and number of fuel assemblies. Explain why the fluxes in Table 1 for the two reactors are the same. If the fluxes at the given exposure are different, please repeat the calculations and modify Table 1.

2) HCGS Technical Specification (TS) 5.3.1, “Fuel Assemblies,” currently reads as follows:

> The reactor core shall contain 764 fuel assemblies and shall be limited to those assemblies which have been approved for use in BWRs.

1 Reference page 21 of Attachment 3 to letter dated November 4, 2009, from Exelon to NRC (ADAMS Accession No. ML093100313).

The proposed amendment would revise TS 5.3.1 to add the following:

> A maximum of twelve GE14i Isotope Test Assemblies may be placed in non-limiting core regions, beginning with Reload 16 Cycle 17 core reload, with the purpose of obtaining surveillance data to verify that the GE14i cobalt Isotope Test Assemblies perform satisfactorily in service (prior to evaluating a future license amendment for use of these design features on a production basis). Each GE14i assembly contains a small number of Zircaloy-2 clad isotope rods containing Cobalt-59. Cobalt-59 targets will transition into Cobalt-60 isotope targets during cycle irradiation of the assemblies.

(a) TS 5.3.1 lacks explicit information on the type of clad, type of fuel, type of material of filler rods for potential substitution for fuel rods, approved methodology for fuel design analysis, and information on potential use of a limited number of test assemblies that may be placed in non-limiting locations. Please propose further changes to TS 5.3.1 to address these issues. For example, see TS 4.2.1 of NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4."

(b) In order to adequately describe the specific design of the ITAs which would be allowed to be inserted into the HCGS reactor please add a sentence to the end of proposed TS 5.3.1 similar to the following:

> Specific details regarding the design of the GE14i assemblies are contained in GE-Hitachi report NEDC-33529P, "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station," Revision 0, dated December 2009.

3) Page 4 of Attachment 1 to the application dated December 21, 2009 (Reference 1) indicates that "[t]hese cycle specific analyses will also ensure that the core loading has been designed such that the ITAs will not be the most limiting fuel assemblies at any time during the operating cycles, based on planned control rod patterns." Explain the relationship between the "ITAs not being the most limiting assemblies" and the "planned control rod patterns."

4) Pages 5 and 8 of Attachment 1 to the application dated December 21, 2009 (Reference 1), Attachment 8 to the application dated December 21, 2009, and Section 4.7.1 of NEDC-33529P (Reference 2) indicate that, due to gamma heating effects on concrete, procedures will be modified to specify that irradiated GE14i bundles be stored at least four feet from the spent fuel pool (SFP) walls. 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.

5) Page 5 of Attachment 1 to the application dated December 21, 2009 (Reference 1) states: "In addition to these ITA examinations, cobalt isotope rods will be removed from the ITAs using the fuel prep machine located in the HCGS spent fuel pool. PSEG intends to remove one GE14i assembly after one cycle in the core and a single isotope

rod will be removed from a GE14i assembly for inspection." Please provide the following information regarding the process of removal of the isotope rod from the ITA.

(a) During the removal process, what will be the distance of the ITA from the SFP wall while the ITA is in HCGS fuel prep machine?

(b) How long a time will the assembly normally be in the prep machine during the removal process?

(c) What is the probability that the SFP wall will undergo significant gamma heating during the removal process?

6) In Attachment 8 to the application dated December 21, 2009 (Reference 1), PSEG made a regulatory commitment to "[r]evise applicable Spent Fuel Pool Storage procedures to require storage of irradiated GE14i fuel bundles at least four feet from the wall of the SFP." Consistent with the guidance in SECY-98-224, "Staff and Industry Activities Pertaining to the Management of Commitments Made by Power Reactor Licensees to the NRC," dated September 28, 1998 (ADAMS Accession No. ML992870043), and NRR Office Instruction LIC-100, "Control of Licensing Bases for Operating Reactors" (ADAMS Accession No. ML010660227), escalating a licensee commitment into a legally binding requirement should be reserved for matters that warrant: (1) inclusion in the TSs based on the criteria in 10 CFR 50.36; or (2) inclusion in the license based on determination that the issue is of high safety or regulatory significance. As discussed in 10 CFR 50.36(c)(4), design features to be included in the TSs "are those features of the facility such as materials of construction and geometric arrangements, which if altered or modified, would have a significant effect on safety and are not covered in categories (c) (1), (2), and (3) of this section." Since the proposed commitment to require storage of the ITAs at least four feet from the wall of the SFP relates to a geometric arrangement associated with maintaining integrity of the SFP wall, it appears that this design feature be included in Section 5.0, "Design Features," of the HCGS TSs (specifically, TS 5.6, "Fuel Storage"). Please propose suitable legally binding requirements for storage of the ITAs in the SFP. The response should also address whether additional requirements are necessary with respect to the ITAs while they are located in the fuel prep machine (see RAI 5 above).

7) In response to Clinton Power Station (CPS) RAI Number 10 on page 15 of Attachment 7 to the application dated December 21, 2009 (Reference 1), PSEG stated that "[t]he response to RAI 10(a) is incorporated into Section 2.1, New Design Features, and Section 4.6, Manufacturing Quality Assurance, of NEDC-33529P, Revision 0, "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station"." These afore-mentioned sections do not contain the Table 2, "Cobalt Target Material Content" and Table 3, "Nickel Plating Material Content," included in Exelon's response to RAI 10 for CPS (Reference 3). These two tables list the Cobalt and Nickel coating material compositions of the cobalt pellets that were used for CPS. Please address whether these tables are applicable to the Cobalt and Nickel coating for HCGS. If they are not applicable, provide new tables for the Cobalt and Nickel coating material composition.

8) Provide a detailed engineering sketch of the cross sectional view of a Cobalt isotope rod showing the target placement rod (TPR), inner tube, and outer tube. The drawing should

show diameters of the tubes, thicknesses of the walls of the tubes and sizes of gaps between the TPR, inner and outer tubes. This detailed diagram will enable the NRC staff to verify the licensee's thermal-mechanical evaluation of the GE14i segmented rod and related confirmatory calculations.

9) Table 3-1 on page 33 of NEDC-33529P (Reference 2) lists a summary of methodologies and analysis codes applicable to the GE14i ITAs. Please add a column to this table that lists all references for each of the methodologies and the respective analysis codes with revision numbers. Also include the details of the references in the Reference section of NEDC-33529P.

10) Explain in detail, with assumptions, analysis, and calculations, why the licensee concludes that the GE14i ITAs will not have significant impact on the in-core instrumentation and core monitoring system of the HCGS (as discussed in Section 3.2.3 of NEDC-33529P (Reference 2)). Section 3.2.3 of NEDC-33529P contains insufficient information to complete an effective and efficient review of the material cited. In addition, provide an evaluation of the gamma radiation effects from the GE14i assemblies on other vessel internal components.

11) General Electric (GE) letter MFN 07-040 to the NRC dated January 21, 2007 (ADAMS Accession No. ML072290203), provided an evaluation of potential non-conservatism in the GE Thermal-Mechanical Methodology, GSTRM. Please provide an evaluation of the impact of the information in MFN 07-040 on the adequacy of the use GSTRM model in the thermal-mechanical evaluation of the GE14i fuel bundle. This evaluation should contain justification for the use GSTRM methodology in the following areas of thermal-mechanical design of GE14i:

- Internal pressure design
- Clad mechanical analyses
- Loss-of-coolant accident response
- Cladding strain analysis; and
- GSTRM calculated gap conductance that is used in the stability and transient analyses.

12) Provide a detailed description of the stability methodology mentioned in Section 3.2.6 of NEDC-33529P (Reference 2). The information contained in Section 3.2.6 is not sufficient for a full review of the methodology.

13) Section 4.5.1 of NEDC-33529P (Reference 2) provides a brief qualitative assessment of the impact of GE14i ITAs on thermal-hydraulic instability for HCGS. During an audit performed by the NRC staff supporting the review of the proposed amendment, the staff was informed that a cycle-specific stability analysis will be performed for the up-coming cycle to determine the impact of GE14i ITAs on stability. Please provide details of the stability analysis.

14) Section 4, "Licensing Evaluations" of NEDC-33529P (Reference 2) states that "[c]ycle-specific analyses will be performed for HCGS Reload 16 Cycle 17 to establish fuel operating limits for the ITAs that assure compliance with regulatory limits." Provide the NRC staff with a summary of the HCGS Reload 16 Cycle 17 Supplemental Reload Licensing Report (SRLR) for review and verification of the results of the cycle-specific analyses. This report should be similar to Global Nuclear Fuel report 0000-0099-4244-

SRLR, Revision 0, “Supplemental Reload Licensing Report for Clinton Power Station Unit 1 Reload 12 Cycle 13” attached to Exelon’s letter RS-09-171 dated December 14, 2009, for CPS (ADAMS Accession No. ML093490375).

15) Section 4.2.1 of NEDC-33529P (Reference 2) states that “[t]he GE14i ITAs represent a small fraction of the total bundles in the core. As a result, their impact on the core average nuclear parameters is negligible. Furthermore, the hydraulic characteristics of GE14i ITAs are similar to the GE14 bundles. Therefore, as in HCGS Cycle 16 (Reference 7), a cycle-specific ATWS [anticipated transients without scram] analysis is not required because of the introduction of GE14i ITAs.”

   a) Provide details of the disposition of the ATWS event at HCGS for Cycle 17 and justify that the ATWS acceptance criteria as listed in Section 2.14.2 of Reference 4 has been met.

   b) What would be a minimum threshold number of ITAs in the HCGS core that would require the licensee to perform a reanalysis of the ATWS event?

16) Section 4.3, "Evaluation of Design-Basis Accidents,” of NEDC-33529P (Reference 2) states:

> The HCGS Design-Basis Accidents (DBAs) to be evaluated are identified in Chapter 15.0 of the HCGS Updated Safety Analysis Report (UFSAR). The Control Rod Drop Accident (CRDA), Main Steamline Break (MSLB) accident outside containment, Fuel Handling Accident (FHA), and Loss-of-Coolant Accident (LOCA) are licensed under 10 CFR 50.67, utilizing Alternate Source Term (AST) methodology per Regulatory Guide (RG) 1.183.

Per RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” dated July 2000 (ADAMS Accession No. ML003716792), Regulatory Position C.1.3.2, “Reanalysis Guideline,”

> The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid.

Also, RG 1.183, Section B, “Discussion,” states:

> Although the LOCA is typically the maximum credible accident, NRC staff experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence.

Page 15.0-5 of Standard Review Plan (SRP) 15.0., “Introduction - Transient and Accident Analyses,” Revision 3, dated March 2007 (ADAMS Accession No.

ML070710376) states:

> The reviewer considers the possible case variations of AOOs [anticipated operational occurrences] and postulated accidents presented to verify that the licensee has identified the limiting cases.

The proposed change only provides an evaluation of the impact on the DBAs described above. Please provide an evaluation of the impact of the proposed change on all accidents in the design bases or include a justification why an evaluation of the impact is not needed. If an evaluation of other DBAs is provided, please provide the regulatory bases for the acceptance criteria (i.e., 10 CFR Part 100, 10 CFR Part 50.67) and any regulatory guidance or SRPs used to make this determination.

17) The release fraction for Co-60 used in the design bases analyses assume that the Co-60 is in the fuel cladding and structural materials. For the proposed change, the Co-60 available to be released during a DBA is not mixed with cladding and structural materials, as considered for the RG 1.183 release fractions, but is in high concentrations within the isotope rods. Please justify why the DBA Co-60 release fraction used is applicable or conservative for the proposed isotope test assemblies. Please include any experimental data to justify the proposed release fraction.

18) Section 4.3.4, “Loss-of-Coolant Accident (LOCA),” of NEDC-33529P (Reference 2) states that the HCGS LOCA source term was previously evaluated for an extended power uprate (EPU). The first sentence in the 2nd paragraph of this section makes a statement regarding one of the assumptions for the HCGS EPU LOCA source term. This statement appears inconsistent with a calculation submitted by the licensee in support of the EPU license amendment review. Specifically, the statement in NEDC-33529P appears to be inconsistent with the isotopic core inventory information shown in Section 5.3.1.3 of PSEG Calculation Number H-1-ZZ-MDC-1880, “Post-LOCA EAB, LPZ and CR Doses,” Revision 2IR0 (ADAMS Accession No. ML063110185). Please resolve this apparent inconsistency and provide a revised justification for the impact of the proposed change on the LOCA analysis as necessary.

19) Please provide enough information (i.e., design bases parameters, assumptions or methodologies) to replicate the dose results provided in NEDC-33529P Section 4.3.1, “Control Rod Drop Accident,” and Section 4.3.4, “Loss-of-Coolant Accident (LOCA),” and provide the results of the calculation in rem Total Effective Dose Equivalent. If the <u>only</u> change is to add Co-60 to calculation number H-1-CG-MDC-1795, Revision 4, “Control Rod Drop Accident Radiological Consequences,” and H-1-ZZ-MDC-1880, Revision 2IRO, “Post-LOCA EAB, LPZ and CR Doses,” please state this in your response.

If any design bases parameters, assumptions or methodologies (other than those provided in NEDC-33529P) were changed in the radiological DBA analyses used to support the proposed amendment change, please provide them. If there are many changes it would be helpful to compare and contrast them in a table. Also, please provide a justification for any changes.

The NRC staff has found that the efficiency of the review can be increased by having the calculations available for review. In addition to providing any changes to the current licensing bases and justifications for these changes, the licensee is encouraged to provide above requested information (i.e. design bases parameters, assumptions or methodologies) by providing the modified calculations (LOCA and Contol Rod Drop Accident) including their attachments. As an alternative, the information may be provided in some other format.

20) Section 4.3.2 of NEDC-33529P (Reference 2) states that "[t]he HCGS licensing basis MSLB analyzed in Section 15.6.4 of the HCGS UFSAR [Steam System Piping Break Outside Containment] assumes no fuel damage occurs as a result of the event." Although the analysis assumes that no fuel rods are damaged, there is no explicit statement in NEDC-33529P regarding the isotope rods. Confirm that no damage to the isotope rod occurs because of the event.

21) During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the coolant can plate out in the reactor coolant system (RCS), and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a DBA could send radioactive materials into the environment. A limiting condition of operation (LCO) on the maximum allowable level of radioactivity in the reactor coolant is established, consistent with 10 CFR 50.36(c)(2)(ii), Criterion 2, to ensure, in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 100, "Reactor Site Criteria" and/or 10 CFR 50.67, "Accident Source Term." The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

HCGS TS LCO 3.4.5 "Reactor Coolant System Specific Activity," states that the primary coolant DOSE EQUIVALENT I-131 specific activity of the reactor coolant shall be less than or equal to 0.2 microcuries per gram (µgCi/gm) and less than or equal to $100/\bar{E}$ µgCi/gm. Per the TS Definition 1.11, DOSE EQUIVALENT I-131 is based upon I-131, I-132, I-133, I-134, and I-135. The NRC staff is concerned about whether the LCO adequately addresses a release of Co-60 into the RCS.

While no "fuel damage" is assumed for some DBA events, the current design basis safety analysis conservatively assumes the fuel pins leak. Clarify whether the operational design limit for the isotope rods is no leakage. Since the TSs are derived from the safety analysis, describe how the <u>TSs</u> will ensure that the assumption of no Co-60 leakage from the Co-60 ITA's remains valid. Justify how LCO 3.4.5 remains able to ensure that 10 CFR 50.67 and 10 CFR 100 limits (as applicable), and radiation shielding and plant personnel radiation protection design limits are met, or modify LCO 3.4.5 so that and these limits continue to be met after the proposed change.

<u>References</u>

1. 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).

2 GE-Hitachi proprietary report NEDC-33529P, "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station," Revision 0, dated December 2009 (Attachment 3 to Reference 1). A non-

proprietary version of report (Attachment 4 to Reference 1) is included as part of ADAMS Accession No. ML093640199.

3. Exelon Nuclear letter RS-09-150 to NRC, "Additional Information Supporting the Request for a License Amendment to Modify Clinton Power Station Facility Operating License in Support of the Use of Isotope Test Assemblies," dated November 4, 2009 (ADAMS Accession No. ML093100313).

4. Global Nuclear Fuel proprietary report NEDC-32868P, "GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II)," Revision 3, dated April 2009.