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Nuclear LLC

MAY 11 2010

10 CFR 50.90

LR-N10-0163

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

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.

The NRC provided PSEG a Request for Additional Information (RAI) on the license amendment request. The NRC RAI questions and the PSEG responses are provided in Attachment 1 to this letter, with the exception of RAI Questions 4, 5 and 6; the response to these questions will be provided in a subsequent letter. In addition, an Errata and Addendum (E&A) to NEDC-33529P (Attachment 3 to Reference 1) will be subsequently provided incorporating the changes discussed in the attached responses to RAI Questions 9 and 17.

Attachment 1 to this letter provides information which GEH considers to be proprietary. The proprietary information is identified by bracketed text. GEH requests that the proprietary information in Attachment 1 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 provided in Attachment 2 to this letter. Attachment 3 to this letter provides a nonproprietary version of Attachment1. Attachments 4 and 5 to this letter provide calculations discussed in the response to RAI

Original signed document was never received

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Question 19. Attachment 6 to this letter provides the 10CFR50.59 evaluation related to Attachment 5. Attachment 7 to this letter provides additional proposed changes to the HCGS TSs.

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 May 11, 2010
(Date)

Sincerely,

John F. Perry

John F. Perry
Site Vice President
Hope Creek Generating Station

Attachments (7)

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

Attachment 2
GE-Hitachi Affidavit for Withholding Portions of RAI Responses from Public Disclosure

GE-Hitachi Nuclear Energy Americas LLC
AFFIDAVIT

I, James F. Harrison state as follows:

(1) I am the Vice President, Fuel Licensing, Regulatory Affairs, GE-Hitachi Nuclear Energy Americas LLC ("GEH"), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 of Global Nuclear Fuel-Americas, LLC letter, LRW-PSG-KT1-10-030, Lauren Watts to Don Notigan (Exelon Nuclear), entitled "Responses to Request for Additional Information 3, 7, 8, 9-13, 15, 17, 18, 20, and 21 Related to License Amendment Request to Modify Hope Creek Power Station Facility Operating License in Support of the Use of Isotope Test Assemblies," May 10, 2010. GEH proprietary information in Enclosure 1, which is entitled "Responses to Request for Additional Information 3, 7, 8, 9-13, 15, 17, 18, 20, and 21", is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] A "[[" marking at the beginning of a table, figure, or paragraph closed with a "]]" marking at the end of the table, figure or paragraph is used to indicate that the entire content between the double brackets is proprietary. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GEH is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results including the process and methodology for the design and analysis of the GE14i Isotope Test Assembly. The GE14i Isotope Test Assembly has been developed at a significant cost to GEH.

The development of the GE14i Isotope Test Assembly is derived from the extensive experience database that constitutes a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 10th day of May 2010.

James F. Harrison

James F. Harrison
Vice President, Fuel Licensing,
Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC

Attachment 3
Additional Information Supporting the Request for a License Amendment to Modify HCGS Operating License in Support of the Use of Isotope Test Assemblies
(Non-Proprietary)

ADDITIONAL INFORMATION

SUPPORTING PROPOSED LICENSE AMENDMENT

USE OF ISOTOPE TEST ASEMBLIES FOR COBALT-60 PRODUCTION

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

In reviewing the PSEG letter LR-N09-0290 (LAR H09-01) submittal dated December 21, 2009 (ADAMS No. ML093640193, Reference 1 of this attachment), related to a pilot program to irradiate Cobalt (Co)-59 targets to produce Co-60, for the Hope Creek Generating Station (HCGS), the Nuclear Regulatory Commission (NRC) staff has made a Request For Additional Information (RAI) in order to complete its review:

NRC RAI#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.

RESPONSE TO RAI#1

The flux values provided in Table 1, "Equilibrium Cobalt-60 Inventory," on page 14 of Attachment 7 (non-proprietary attachment) and Attachment 5 (proprietary attachment) of LAR H09-01 are identical to flux values provided for the CPS application because the values and the mathematical expression are generic. As stated in the HCGS response to RAI 9c: "*The response to (c) is generic information that is of general interest to cobalt production.*"

NRC RAI#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.*

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*

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

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.

RESPONSE TO RAI#2

(a) Additional changes to TS 5.3.1, to align with NUREG-1433, will be added. See Attachment 7 of this submittal.

(b) The following sentence: "Details 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," will be added to TS 5.3.1. See Attachment 7 of this submittal.

NRC RAI#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."

RESPONSE TO RAI#3

The GESTAR Lead Test Assembly (LTA) process allows for the introduction of small quantities (less than approximately 2% of the total bundles in a core) of new fuel product designs in non-limiting reactor core locations without the need for full NRC review, evaluation and approval as long as the analysis of the LTAs uses approved methods and meets the approved criteria.

The GE14i design involves only a small change to a fully approved fuel design and utilizes previously licensed materials, bundle designs and analytical methods. Although the pilot project is not being licensed as an LTA program and is undergoing full NRC review, evaluation and approval, the conservative design practice of introducing a quantity of less than 2% of the total bundles in a core into non-limiting core positions is still being employed. This introductory approach is not required but is being utilized for an additional level of conservatism and to be consistent with precedent for introducing new fuel designs.

The placement of the Isotope Test Assemblies (ITA) in the Hope Creek cycle 17 core was addressed with the normal GNF and PSEG core design processes. Specifically for control rod patterns, the procedures are designed to [[

]]

GNF has significant experience with new fuel product line introduction and even has experience introducing segmented fuel rods under the LTA provisions stated above. This LTA process has shown to be valuable in obtaining surveillance data to verify that a fuel bundle design performs satisfactorily in service prior to implementation on a production basis.

NRC RAI#4, 5 and 6

The response to Questions 4, 5 and 6 will be provided in a subsequent letter.

NRC RAI#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.

RESPONSE TO RAI#7

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) listing the Cobalt and Nickel coating material compositions of the cobalt targets that were used for CPS are also applicable to the Cobalt and Nickel plating for HCGS. The information is repeated here for completeness.

Cobalt Target Material Content

Material	% Content
[[	
	]]

Nickel Plating Material Content

Material	% Content
[[	
	]]

NRC RAI#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.

RESPONSE TO RAI#8

Figure 1 below provides the requested sketch.
[[

]]

Figure 2. Isotope Rod Cross Section

NRC RAI#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.

RESPONSE TO RAI#9

Table 3-1 on page 33 of NEDC-33529P (Reference 2) will be modified as follows:

Table 3-2 Summary of GNF Methods Applicability to GE14i

Methodology	Analysis Code	Version	Supported	Reference
Nuclear	TGBLA	06	X	3, 20
	PANAC	11	X	
Thermal Hydraulic	ISCOR	09	X	21
Safety Limit MCPR	GESAM	02	X	22, 23, 24
Transient Analyses	ODYNM	10	X	25, 26, 27
	TASC	03	X	28
Stability	ISCOR	09	X	21
	PANAC	11	X	3, 20
	ODYSY	05	X	29
	TRACG	04	X	30
ATWS	TASC	03	X	28
	ODYNM	10	X	31
Thermal Mechanical	GSTRM	07	X	32, 33
ECCS-LOCA	LAMB	08	X	34
	TASC	03	X	28
	SAFER	04	X	35

NEDC-33529P Section 6 Reference additions:

20. NEDE-30130-P-A, “Steady State Nuclear Methods,” April 1985.
21. The ISCOR code is not approved by name. However, the SER supporting approval of NEDE-24011-P Rev. 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R.Gridley (GE) finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in reactor internal pressure differences, Transient, ATWS, Stability, and LOCA applications is consistent with the approved models and methods.”
22. NEDC-32601P-A, Methodology and Uncertainties for Safety Limit MCPR Evaluations, August 1999
23. NEDC-32694P, Power Distribution Uncertainties for Safety Limit MCPR Evaluation
24. NEDE-24011-P-A on Cycle Specific Safety Limit MCPR (TAC Nos. M97490, M99069 and M97491), March 11, 1999, Amendment 25.
25. NEDO-24154-A, Qualification of the One-Dimensional Core Transient Model (ODYN) for BWRs, Vol. 1, August 1986.
26. NEDO-24154-A, Qualification of the One-Dimensional Core Transient Model (ODYN) for BWRs, Vol. 2, August 1986.
27. NEDE-24154-P-A, Qualification of the One-Dimensional Core Transient Model (ODYN) for BWRs, Vol. 3, August 1986.

28. NEDC–32084P–A Rev. 2, TASC–03A – A computer program for Transient Analysis of a Single Channel, July 2002.
29. NEDC–32992P–A, ODYSY Application for Stability Licensing Calculations, July 2001.
30. NEDO–32465A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Application, August 1996.
31. NEDC-24154-P-A, Supplement 1 - Volume 4, Revision 1, Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, February 2000.
32. MFN-036-85, Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A Amendment 7 to Revision 6, "GE Standard Application for Reactor Fuel Letter", C.O. Thomas (NRC) to J. S. Charnley (GE), March 1, 1985.
33. MFN-082-85, Letter, C. O. Thomas (NRC) to J. S. Charnley (GE), Acceptance For Referencing of LTR NEDE-24011-P-A-6, Amendment 10, "GE Standard Application for Reactor Fuel," May 28, 1985.
34. NEDE-20566-P-A, "General Electric Company Analytical Model for Loss-of-Coolant analysis in Accordance with 10CFR50 Appendix", Volumes 1-3, September 1986.
35. NEDE-23785-1-PA, Revision 1, "GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident," Volumes II and III, October 1984.

<u>NRC RAI#10</u>

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.

RESPONSE TO RAI#10

The major sources of gammas or photons in the operating reactor are from the fission events and the neutron capture events. To form the TIP instrumentation correlation constants (J-factors), gamma sources from each material region and signal contribution (attenuation) factors are used to determine the amount of gamma energy deposited in the Gamma TIP detector.

The released energy from a neutron capture reaction in Co-59 is approximately 7.5 MeV per capture (mass defect) and approximately 2.5 MeV per decay from the subsequent decay of Co-60 to Ni-60. The total energy (approximately 10.0 MeV/event) from the neutron capture and the subsequent decay of Co-60 is assumed to occur at the time of neutron capture. This assumption will over-estimate the gamma energy from the cobalt material early in life but the error will reduce as the contribution from the decay of Co-60 increases. The assumption that the decay energy is released at time zero is consistent with the TGBLA assumption for all explicitly modeled fission product isotopes with similar half-lifes.

The methodology for determining the gamma energy heat deposition in the gamma detector incorporates the energy released from each nuclear reaction event (capture, fission, and decay), spatial location of the event, attenuation due to material between the event and the gamma detector, and the energy deposition in the gamma detector configuration. This methodology has been used in BWRs since the mid 1980s.

To determine the contribution from the cobalt material in total Gamma TIP signal, an evaluation was performed with the energy released from the cobalt capture and decay defined as zero. This demonstrates the error that would result if the gamma production in the cobalt rods were ignored. Due to the location and source strength of the cobalt isotope rods, the total gamma energy deposition in the gamma TIP detector from the cobalt material in four surrounding GE14i lattices is approximately [[ ]] or less. By including the cobalt gamma energy release model in the Gamma TIP detector signal correlation, the impact of the cobalt material on the accuracy of the Gamma TIP signal is reduced to a level significantly below [[ ]].

For the neutron in-core instrumentation (LPRMs), the in-core instrumentation signal-to-lattice power relationship is formed using the thermal detector J-factor. The thermal detector J-factor provides the relationship between the lattice power and the signal generated by the LPRM detector. The cobalt rods are explicitly modeled in the GE14i design and the impact of the cobalt neutron absorption is incorporated in the thermal J-factors and neutron flux predictions at the LPRM instrumentation location.

With the inclusion of the cobalt material effects in the lattice physics model, the perturbations on the instrumentation (gamma or neutron) signal from cobalt material are captured, and the adequacy of the in-core instrumentation is assured.

Additionally, the replacement of a fission material bearing fuel rod with a cobalt isotope rod will result in approximately a factor of 10 reduction in the gamma energy emitted from that rod location, as supported by NEDC-33529P, Section 4.4. The gamma energy from the fission material bearing rod is generated from prompt fission gammas, delayed fission gammas, and neutron capture gammas in actinides and fission products. Only after reactor shutdown and subsequent decay of short half life fission products and actinides in the ITA will the gamma from Co-60 decay become a significant contributor to the total gamma energy production. Therefore, the effects on the material characteristics of instruments or local vessel internals will be bounded by what is seen from a UO_2 fuel rod at that location.

NRC RAI#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.*

RESPONSE TO RAI#11

In MFN 07-040 GNF evaluated a potential non-conservatism in the GSTRM thermal-mechanical calculations. Specifically, [[ ]] model on the GSTRM fuel temperature, fuel design analyses and downstream safety analyses have been evaluated. As reported in the MFN 07-040 the evaluated condition does not constitute a reportable condition per 10 CFR 21. The NRC staff's evaluations of MFN 07-040 and associated supplements recommend an additional [[ ]] for the GSTRM fuel rod internal pressure analyses to address [[ ]] model in GSTRM (Reference R-1).

As requested in this RAI, the applicability of the GSTRM methodology to the GE14i design analyses, including the MFN 07-040 evaluation/conclusions and the NRC staff recommendations, has been evaluated and the following conclusions have been made.

- LHGR limits for the full length UO_2 rod, partial length UO_2 rod and Gadolinia containing rods have been updated to include the additional [[ ]] for the GSTRM rod internal pressure analyses (Reference R-2). GE14i bundles for Hope Creek Generating Station are designed with these revised LHGR limits and will be monitored in the core based on these revised LHGR limits.

- NRC staff recommended [[ ]] is not applicable for the GE14i cobalt isotope rods as fuel failure due to excessive internal pressure is not a likely failure mechanism for these isotope rods. [[ ]] and also there is no fission gas release from the cobalt targets to increase the rod internal pressure during the irradiation. As a net result, the rod internal pressure at the end of life is significantly below the reactor system pressure and thus the fuel failure due to high rod internal pressure is not a likely failure mechanism for these rods and no additional pressure design margin is required.

- MFN 07-040 also demonstrated the applicability of GSTRM for the cladding mechanical analyses, loss-of-coolant accident response, cladding strain analyses and the gap conductances generated by GSTRM for the transient and stability analyses. The cladding mechanical/strain analyses and the downstream safety analyses are [[ ]] and thus the application of GSTRM with its conservative uncertainties treatment is adequate for those analyses. NRC staff's evaluation of the MFN 07-040 also did not recommend any additional design margins for these calculations.

References:

[R-1] Appendix F of the NRC SER for NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," July 21, 2009.

[R-2] Appendix C of the NEDC-32868P Revision 3, "GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)" April 2009.

NRC RAI#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.

RESPONSE TO RAI#12

Detailed descriptions of the stability analysis and methodology mentioned in Section 3.2.6 of NEDC-33529P (Attachment 3 of Reference 1) are described in Section S.4 of GESTAR (Reference 2 in Attachment 3 of Reference 1).

Hope Creek Generating Station implements the Option III stability Long-Term Solution (LTS). The plant and cycle-specific calculations required for the Option III stability LTS are described in Section S.4.1 of Reference R-2. The cycle-specific stability analyses are also described in the HCGS response to Clinton Power Station RAI Number 6 in Attachment 5 of Reference 1.

The approval status of the codes mentioned in Section 3.2.6 of Reference R-1 are summarized in the table below.

Computer Code	Version or Revision	NRC Approved	Comments
ISCOR	09	Y(1)	NEDE-24011-P Rev. 0 SER
PANACEA	11	Y	NEDE-30130P-A (2)
ODYSY	05	Y	NEDC-32992P-A
TRACG	04	N(3)	NEDO-32465-A

(1) The ISCOR code is not approved by name. However, the SER supporting approval of NEDE-24011P, Rev. 0, by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GE) finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in stability applications is consistent with the approved models and methods.

(2) The physics code PANACEA provides inputs to the transient code ODYN or to TRACG. The improvements to PANACEA that were documented in NEDE-30130-P-A were incorporated into ODYN by way of Amendment 11 of GESTAR II (NEDE-24011-P-A). The use of TGBLA Version 06 and PANACEA Version 11 in this application was initiated following approval of Amendment 26 of GESTAR II by letter from A. A. Richards (NRC) to G. A. Watford (GE) Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady-State Methods," (TAC NO. MA6481), November 10, 1999.

(3) TRACG02 has been approved in NEDO-32465-A by the US NRC for the stability DIVOM analysis. The current licensed thermal power (extended power uprate conditions) stability analysis is based on TRACG04, which has been shown to provide essentially the same or more conservative results in DIVOM applications as the previous version, TRACG02. The use of TRACG04 at HCGS was introduced with acceptance of License Amendment 163, implementing Average Power Range Monitor, Rod Block Monitor Technical Specifications concurrent with Maximum Extended Load Line Limit Analysis at HCGS.

NRC RAI#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.

RESPONSE TO RAI#13

The cycle-specific stability analysis will be performed for the upcoming cycle and provided to the NRC by July 8, 2010.

NRC RAI#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).

RESPONSE TO RAI#14

The HCGS Reload 16 Cycle 17 Supplemental Reload Licensing Report (SRLR) will be provided to the NRC by August 4, 2010.

NRC RAI#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?*

RESPONSE TO RAI#15

(a) HCGS has margin to the ATWS acceptance criteria as shown in the cycle-independent calculation results documented in Reference R-1. This document is also referenced in NEDC-33529P. The GE14i ITA geometry and enrichment are similar to GE14. Therefore, the differences in nuclear characteristics of the GE14i bundle design will not be any greater than what is expected when transitioning to a different BWR fuel design. The impact of a core-wide [[ ]] increase in ODYN void coefficient on the ATWS analysis has been assessed for reactor cores consisting of [[ ]] fuel designs. The results documented in Tables 1 & 2 provide evidence that a [[ ]] increase in void coefficient has a [[ ]] on suppression pool temperature and [[ ]] change in peak vessel overpressure.

It is known that the plant response during an ATWS event is primarily affected by plant characteristics (SRV capacity, SLCS operating parameters, ATWS recirculation pump trip, etc). Minute changes in fuel design being loaded in small quantities (<2% batch fraction) does not impact the conclusions of Reference R-1. As such, ATWS is treated on a plant specific, cycle independent manner.

Table 1 Peak Vessel Pressure Void Coefficient Study

Event and Description	Exposure	Peak Vessel Pressure (MPa)
PRFO Base Case	BOC	[[
PRFO with [[ ]] void coefficient increase	BOC	
PRFO Base Case	EOC	
PRFO with [[ ]] void coefficient increase	EOC	]]

Table 2 Peak Suppression Pool Temperature Void Coefficient Study

Event and Description	Exposure	Peak Pool Temperature (°C)
PRFO Base Case	BOC	[[
PRFO with [[ ]] void coefficient increase	BOC	
PRFO Base Case	EOC	
PRFO with [[ ]] void coefficient increase	EOC	]]

References

R-1. GE Nuclear Energy, "Safety Analysis Report for Hope Creek Constant Pressure Power Uprate," NEDC-33076P, Revision 2, August 2006.

(b) The proposed changes to TS 5.3.1 in LAR H09-01 specifically state: "A maximum of twelve GE14i Isotope Test Assemblies may be placed in non-limiting core regions." As part of the ITA pilot program 12 is the maximum number of ITAs that will be placed in the HCGS core. The analysis provided in LAR H09-01 provides adequate technical justification for operation with 12 ITAs. It is known that the plant response during an ATWS event is primarily affected by plant characteristics (SRV capacity, SLCS operating parameters, ATWS recirculation pump trip, etc). Minute changes in fuel design being loaded in small quantities (<2% batch fraction) does not impact the ATWS analysis conclusions.

NRC RAI#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.*

RESPONSE TO RAI#16

A detailed explanation of all probable isotope rod failure modes is provided in Section 2.2 of NEDC-33529P (Attachment 3 to LR-N09-0290, LAR H09-01). Section 2.2 also describes key protective design features; the isotope rods will operate at a significantly lower heat generation rate compared to fuel rods, the isotope rods have a double layer of Zircaloy encapsulation before exposure of the nickel-plated cobalt targets, and the isotope rods have Zircaloy connections at all spacer locations. Section 2.2 provides a technical basis to conclude that isotope rods are not more vulnerable to common failure modes than normal fuel rods during operation. Section 2.3, Online Failure Detection, also provides the HCGS ability to measure changes in cobalt-60 activity and take appropriate response. The response to RAI#21 of this attachment provides further discussion on failure modes and cobalt detection.

Section 4.3 of NEDC-33529P, Evaluation of Design-Basis Accidents, identifies that the HCGS Design Basis Accidents (DBAs) to be evaluated are identified in Chapter 15 of the HCGS Updated Final Safety Analysis Report (UFSAR). The section states that Control Rod Drop Accident (CRDA), Main Steam Line 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. In addition to these four events, Chapter 15 of the HCGS UFSAR identifies five other events classified as limiting faults. These events are:

6. Reactor Recirculation Pump Shaft Seizure (UFSAR 15.3.3)
7. Reactor Recirculation Pump Shaft Break (UFSAR 15.3.4)
8. Instrument Line Break (UFSAR 15.6.2)
9. Feedwater Line Break – Outside Primary Containment (UFSAR 15.6.6)
10. Gaseous Radwaste Subsystem Leak or Failure (UFSAR 15.7.1)

As discussed in the response to RAI#21, leakage of cobalt (including entire cobalt targets and/or cobalt particulate) from an isotope rod in an ITA is not a credible event during normal operations, transients or design basis accidents not involving fuel melt accidents (i.e., Loss of Coolant Accident and Control Rod Drop Accident). None of the additional five postulated events involve fuel failures or fuel melt; therefore, isotope rod failure or leakage is not credible during any of these events. Therefore, the radiological consequences for these five events are unchanged for a core operating with isotope test assemblies. The five events are described below.

1. <u>Reactor Recirculation Pump Shaft Seizure (UFSAR 15.3.3)</u>

The seizure of a reactor recirculation pump is a design basis accident that does not result in the failure of fuel. Since no fuel rod failures occur due to the recirculation pump shaft seizure, no GE14i isotope rod failures are postulated to occur, and the consequences of this event will be unchanged in operation with GE14i.

While the consequence of this transient does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression chamber via SRV operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Because this transient does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity in the primary containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with established technical specification limits.

2. Reactor Recirculation Pump Shaft Break (UFSAR 15.3.4)

This event is less severe than the Reactor Recirculation Pump Shaft Seizure event, and consequences of this event are considered to be bounded by the shaft seizure event in HCGS UFSAR 15.3.3. Since no fuel rod failures occur due to the recirculation pump shaft break event, no GE14i isotope rod failures are postulated to occur, and the consequences of this event will be unchanged in operation with GE14i.

While the consequence of this transient does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression chamber via SRV operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Because this transient does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity in the primary containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with established technical specification limits.

3. Instrument Line Break (UFSAR 15.6.2)

The Instrument Line Break involves the postulation of a small break in a steam or liquid line inside or outside containment but within a controlled release structure.

No fuel damage is associated with this accident. Since no fuel rod failures occur due to the instrument line break, no GE14i isotope rod failures are postulated to occur, and the consequence of this event will be unchanged in operation with GE14i. As a result of depressurizing the Reactor Coolant System, normal operating concentrations of iodine and noble gases can be released including consideration of iodine spiking. The analysis results indicate that offsite and control room doses are small fractions of 10 CFR 50.67 guidelines.

4. Feedwater Line Break – Outside Primary Containment (UFSAR 15.6.6)

To evaluate the pipe breaks in a large liquid process line outside primary containment, the failure of a feedwater line is assumed. The feedwater line break outside primary containment results in no fuel failures. Since no fuel rod failures occur due to the feedwater line break outside primary containment, no GE14i isotope rod failures are postulated to occur, and the radiological consequences of this event will be unchanged in operation with GE14i.

Though there is no fuel damage as a consequence of this accident, the activity in the main condenser hotwell prior to occurrence of the break is released. The radiological release consideration is primarily one of iodine release. Noble gas activity in the condensate is considered negligible. The analysis results indicate that offsite and control room doses are small fractions of 10 CFR 50.67 guidelines.

5. Gaseous Radwaste Subsystem Leak or Failure (UFSAR 15.7.1)

The Gaseous Radwaste Subsystem Leak or Failure does not affect the nuclear fuel as there is no reactor core transient associated with this event. Since no fuel rod failures occur due to the gaseous radwaste subsystem failure, no GE14i isotope rod failures are postulated to occur, and the consequences of this event will be unchanged in operation with GE14i. Branch Technical Position 11-5 identifies that only radioactive noble gases (xenon and krypton) are to be considered to be released to the environment since the assumed transit time is long enough to permit major radioactive decay of oxygen and nitrogen isotopes. The branch technical position also identifies that particulates and radioiodines are assumed to be removed by pretreatment, gas separation, and intermediate radwaste treatment equipment. The analysis results indicate that offsite dose is a small fraction of 10 CRF 100 guidelines.

NRC RAI#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.

RESPONSE TO RAI#17

The design of the Isotope Test Assemblies (ITAs) is such that the nickel-plated cobalt (Co) targets in the ITAs are isolated from the reactor environment by a double layer of Zircaloy encapsulation. Because there is no uranium fuel present in the cobalt isotope rods, the isotope rods have much lower heat generation than fuel rods. It is expected that the lower heat generation rate and double Zircaloy barrier features of cobalt isotope rods would justify the assumption that the fraction of cobalt released from the passive isotope rods during a design basis LOCA or CRDA would be equal to or less than the fraction of cobalt released from other passive materials present in the reactor core. However, no experimental data can be provided as further justification for this expectation. Therefore, the methodologies in sections 4.3.1 and 4.3.4 of NEDC-33529P have been updated (as shown below) to include analysis of potentially higher cobalt release fractions for CRDA and LOCA dose evaluations, respectively. The previously assumed release fraction of 0.0025, which is consistent with the recommended post-LOCA cobalt release fraction in RG 1.183, was [[ ]] and analyzed for CRDA and LOCA. For both accidents, assuming the [[ ]] the dose impact of introducing 12 ITAs at HCGS remains negligible.

Updated Sections of NEDC-33529P:

4.3.1 Control Rod Drop Accident

The HCGS licensing basis CRDA analyzed in Reference A1 assumes a failure of 850 rods (8x8 fuel). The mass fraction of fuel in the damaged rods that reaches or exceeds the initiation temperature of fuel melting is estimated to be 0.77%. Fuel reaching melt conditions is assumed to release 100% of the noble gas inventory and 50% of the iodine inventory. [[

]] Therefore, the licensing basis CRDA radiological analysis is not impacted by the introduction of 12 GE14i assemblies at HCGS.

As described in Reference 9, compliance with licensing limits governing CRDA is assured through adherence to the Banked Position Withdrawal Sequence (BPWS). The associated analyses have generically demonstrated large margin to licensing limits governing acceptable enthalpy insertions. The BPWS analyses demonstrated that the characteristic control rod worth associated with limiting rods in a BPWS sequence are low as compared to that required to challenge the 280 cal/gm fuel design limit. The reactivity characteristics of GE14i are similar to GE14; therefore, the introduction of 12 GE14i assemblies at HCGS will have negligible effects on the existing CRDA margin. In addition to similar fuel reactivity characteristics, the impact on the rod worths is constrained by other design factors such as shutdown margin and in-sequence rod worths.

4.3.4 Loss-of-Coolant Accident (LOCA)

The HCGS LOCA source term was previously evaluated in Reference A2. The impact of 12 GE14i assemblies on the HCGS licensing basis LOCA source term and radiological consequences was evaluated.

[[

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The introduction of 12 GE14i bundles at HCGS presents no significant impact on the AST LOCA source term.

6. References

9. GE Hitachi Nuclear Energy, "General Electric Standard Application for Reactor Fuel (Supplement for United States)," NEDE-24011-P-A-16-US, Revision 16, October 2007.
A1. Calculation H-1-CG-MDC-1795, Revision 5, "Control Rod Drop Accident Radiological Consequences", June 2007.
A2. Calculation H-1-ZZ-MDC-1880, Revision 3, "Post-LOCA EAB, LPZ and CR Doses", September 2009.

NRC RAI#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.

RESPONSE TO RAI#18

The analysis documented in NEDC-33529P Section 4.3.4 has been modified (see RAI#17 response) to correctly consider the Co-60 present in the HCGS licensing basis post-LOCA radiological consequences evaluation source term. The revised analysis is consistent with the HCGS licensing basis methodology as documented in Calculation H-1-ZZ-MDC-1880 Revision 3. The conclusion that the introduction of 12 GE14i bundles at HCGS presents no significant impact on the licensing basis LOCA source term is still supported.

NRC RAI#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 only 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.

RESPONSE TO RAI#19

The radiological analyses examining the effect of introducing GE14i ITAs on licensing basis CRDA and post-LOCA doses have been updated (see RAI#17 response) to be consistent with HCGS calculations H-1-CG-MDC-1795 Revision 5 and H-1-ZZ-MDC-1880 Revision 3 and present results in rem Total Effective Dose Equivalent (TEDE). The alternative RADTRAD analyses performed and their differences from these HCGS calculations are described in the revised sections 4.3.1 and 4.3.4 of NEDC-33529P.

The current revisions of Calculations H-1-CG-MDC-1795 (Revision 5)[2], "Control Rod Drop Accident Radiological Consequences," and H-1-ZZ-MDC-1880 (Revision 3), "Post-LOCA EAB, LPZ and CR Doses," are provided as Attachments 4 and 5 of this submittal. During an April 6 audit supporting the review of this LAR, the NRC asked for clarification on two issues related to H-1-ZZ-MDC-1880, and subsequently asked for the 10CFR50.59 evaluation that was performed for H-1-ZZ-MDC-1880, Revision 3. The two issues identified during the audit are discussed below, including discussion and comparison tables on parameters, assumptions and methodologies in the current licensing basis; the 10CFR50.59 evaluation is provided as Attachment 6 to this submittal.

<u>Issue 1: For the HCGS LOCA dose calculation H-1-ZZ-MDC-1880, Revision 3, specific</u> to the <u>MSIV leakage, why did the doses go down from Revision 2 to Revision 3?</u>

The HCGS full scope Alternative Source Term (AST) license amendment request, and subsequent Amendment 134, dated October 3, 2001, included an aerosol deposition model for the Main Steam Isolation Valve (MSIV) leakage path based on the guidance in NRC document AEB 98-03 (Reference 5). The aerosol deposition model that was subsequently included in H-1-ZZ-MDC-1880, Revision 2, was developed using the following very conservative assumptions:

2 Revision 5 only corrected typographical errors; Revision 4 was previously docketed to support the HCGS EPU amendment request (ADAMS ML063110190)

6. The Technical Specification MSIV leakage rates of 150 scfh (2.5 cfm) and 50 scfh (0.833 cfm) were modeled without reducing these leak rates to address post-LOCA primary containment pressure and temperature conditions. A 50% reduction in the MSIV leak rate was credited after 24 hours.

7. One volume node for each release path – MSIV failed line and intact line – was modeled, with one aerosol removal efficiency per path. Although HCGS has seismically supported main steam lines beyond the outboard MSIVs, the piping upstream and downstream of the outboard valve was modeled as a single volume.

8. One aerosol settling velocity of 40^{th} percentile was used for both MSIV failed and intact lines, upstream and downstream of the outboard MSIVs.

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.

In 2009, PSEG NUCLEAR initiated a revision to the HCGS LOCA analysis in H-1-ZZ-MDC-1880, Revision 2, to (1) allow for keeping the primary containment isolation valves (PCIVs) open for 120 seconds post LOCA, and (2) increase allowable Engineered Safety Feature (ESF) leakage from 1.0 to 2.85 gpm. H-1-ZZ-MDC-1880, Revision 3, was updated as follows:

8. Each piping segment upstream and downstream of the outboard MSIVs in the MSIV failed and intact lines were modeled as well mixed volumes. Two well mixed volumes for each MSIV release path is consistent with AEB 98-03 (See Table 1, Item 1).

9. The MSIV leakage in the release path was reduced based on the post-LOCA drywell and wetwell pressure and temperature, which significantly reduced the MSIV leakage and consequently reduced the resulting doses from the MSIV leakage paths (Table 1, Items 2 and 3).

10. The aerosol removal in each MSIV release path was divided between two well mixed volumes, which created two aerosol removal filters in a series configuration that reduced the aerosol released to the environment by factors of about 4 and 10 for the MSIV failed and intact lines, respectively (as shown in the computations provided in Table 1, Items 4 and 5). The MSIV leakage dose from these release paths were reduced proportionately. The aerosol removal filter efficiencies were calculated using the horizontal projected area (diameter x length) of the main steam piping.

11. Hold-up times of 9.32 hrs and 29.52 hrs were credited for the MSIV failed and intact lines, respectively, in Revision 2. The hold-up time credit is not appropriate for the well mixed volumes; therefore, hold-up times are not credited in H-1-ZZ-MDC-1880, Revision 3, which is conservative with respect to radiological consequences (Table 1, Item 6).

12. The maximum primary containment isolation valve (PCIV) isolation time was increased to 120 seconds (Table 1, Item 7). The open PCIVs present a release path to the environment for airborne containment activity due to the radionuclide inventory in the reactor coolant system liquid which is not considered in H-1-ZZ-MDC-1880, Revision 2.

13. The allowable ESF leak rate was increased from 1.0 gpm to 2.85 gpm to facilitate acceptable results for future plant maintenance surveillances (Table 1, Item 8).

14. The removal of the elemental iodine by wall deposition on wetted surfaces inside containment is modeled in accordance with NUREG-0800, Standard Review Plan 6.5.2 (Table 1, Item 9).

Table 1 presents a comparison of all differences between H-1-ZZ-MDC-1880, Revisions 2 and 3.

Table 1
Comparison of MSIV Leakage Modeling in H-1-ZZ-MDC-1880, Revision 2 versus Revision 3

Item No.	Design Input Information	H-1-ZZ-MDC-1880 Revision 2	H-1-ZZ-MDC-1880 Revision 3
1	Well mixed volume	Both release paths - MSIV failed and intact steam line releases are model as a single node volumes.	Both release paths are model as two well mixed volume nodes based on AEB 98-03.
2	MSIV Leakage Rate In MSIV Failed Line	Assumed 150 scfh for 0-24 hrs and 75 scfh for > 24 hrs (2.5/1.25 cfm). No reduction in MSIV leakage credited for drywell post-LOCA condition.	Reduced based on the post-LOCA containment pressure and temperature (0.808, 0.446, and 0.223 cfm for 0-2, 2-24 and 24-720 hr, respectively), which significantly reduced MSIV leakage.
3	MSIV Leakage Rate In MSIV Intact Line	MSIV leakage of 100 scfh was divided between two intact MS line - 50 scfh/line (0.8334 and 0.417 cfm for 0-24 and 24-720 hr, respectively). No reduction in MSIV leakage credited for drywell post-LOCA condition.	MSIV leakage of 100 scfh was allocated to one intact MS line - 100 scfh/line with the leakage reduction based on the post-LOCA containment pressure and temperature (0.539, 0.297, and 0.149 cfm for 0-2, 2-24 and 24-720 hr, respectively), which significantly reduced MSIV leakage.
4	Aerosol deposition efficiency - MSIV failed line	One aerosol removal efficiency was calculated for both the MSIV failed and intact lines. The use of one aerosol removal efficiency of 98.32% for MSIV failed line resulted in 1.68% of aerosols released to the environment.	Two aerosol removal efficiencies (85.92% for the piping segment between the inboard and outboard MSIVs and 96.96% for the segment beyond the outboard MSIV) were calculated for MSIV failed piping segments, which resulted in 0.43% of aerosol released to the environment. This reduced the aerosol release by about a factor of 4 (1.68/0.43 = 3.91).

Table 1
Comparison of MSIV Leakage Modeling in H-1-ZZ-MDC-1880, Revision 2 versus Revision 3

Item No.	Design Input Information	H-1-ZZ-MDC-1880 Revision 2	H-1-ZZ-MDC-1880 Revision 3
5	Aerosol deposition efficiency - MSIV intact line	One aerosol removal efficiency of 99.46% was calculated for one intact line well mixed volume node resulting a 0.54% aerosol released the environment.	Two aerosol removal efficiencies (97.32% for the piping segment between the RPV nozzle and outboard MSIV and 97.95% for the segment beyond the outboard MSIV) were calculated for MSIV intact piping segments, which resulted in 0.055% of aerosol released to the environment. This reduced the aerosol release by about a factor of 10 (0.54%/0.055% = 9.82)
6	MSIV Leakage Holdup Time	Hold up times of 9.32 hrs and 29.52 hrs were credited for the MSIV failed and intact lines respectively	Holdup times are not credited for any MSIV leakage path, which is conservative with respect to radiological consequences..
7	Primary containment isolation valves (PCIVs)	PCIVs not modeled as a release path (i.e., the PCIV release path is isolated prior to the onset of the AST gap release).	Primary containment isolation valves (PCIVs) remain open for 120 seconds.
8	Allowable ESF Leakage Rate	Allowable ESF leak rate of 1.0 gpm (modeled as 2.0 gpm)	Allowable ESF leak rate increased to 2.85 gpm (modeled as 5.7 gpm). This increased the ESF leakage dose by a factor of 2.85 (= 2.85 gpm /1.0 gpm).
9	Elemental iodine removal by wetted surface deposition	Not Credited	Credited

The combined effects of the above changes are such that the doses resulting from the MSIV leakage path are reduced substantially. This also demonstrates that the aerosol deposition model in the original AST license amendment based on Revision 2 to H-1-ZZ-MDC-1880 was extremely conservative.

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.

For comparison, Table 2 summarizes the differences between the EXELON Peach Bottom AST MSIV leakage model (as implemented in LOCA Analysis PM-1077, Revision 1) and the HCGS MSIV leakage model (as implemented in H-1-ZZ-MDC-1880, Revision 3).

Table 2
Comparison of Peach Bottom and Hope Creek MSIV Leakage Aerosol Deposition Model

Variable Parameter	Peach Bottom AST Analysis - PM-1077, Rev 1					Hope Creek AST Analysis - H-1-ZZ-MDC-1880, Rev 3				
	MSIV Failed Line			Intact Line		MSIV Failed Line			Intact Line	
	RPV To Inboard MSIV	Between Inboard and Outboard MSIVs	Between Outboard MSIV and TSV	Between RPV and Outboard MSIV	Between Outboard MSIV and TSV	RPV To Inboard MSIV	Between Inboard and Outboard MSIVs	Between Outboard MSIV and TSV	Between RPV and Outboard MSIV	Between Outboard MSIV and TSV
Piping Integrity Assumed	Remains Intact		Remains Intact			Ruptured - not credited	Remains Intact			
Aerosol Deposition	Not Credited		Credited			Not Credited	Credited			
Piping Volume Dilution	Credited		Credited			Not Credited	Credited			
Drywell P/T Related MSIV Leak Rate Reduction	Not Applicable		Credited			N/A	Credited			
Holdup Time	Not Credited					Not Credited				
Deposition Velocity Distribution	Not Credited		40 Percentile*			Not Credited	50 Percentile	30 Percentile	50 Percentile	30 Percentile
Elemental Iodine Removal	Not Credited		Credited (50%)			Not Credited	Credited (50%)			

* Peach Bottom LOCA analysis in PM-1077, Rev 1 uses the 40 percentile aerosol deposition velocity in both MSIV failed and intact lines. PM-1077, Rev 1, Appendix A documents the parametric study, which demonstrates that the results in the calculation using the 40 percentile aerosol deposition velocity is bounding for a lower deposition of the lighter aerosol particles in the piping downstream of the outboard MSIVs due to conservatism in the calculation by neglecting the aerosol and elemental iodine removal in the piping segment between the inboard and outboard MSIVs in the MSIV failed line.

Table 3 lists the input parameters associated with the AST methodology differences between the PB and HCGS AST LOCA calculations. In addition to the Tables 2 and 3 AST methodology differences, the most significant modeling differences between the PB AST LOCA analysis and the HCGS AST LOCA analysis are:

3) Containment Leakage – The PB model has an initial higher containment leakage rate (0.7 weight %/day vs. 0.5 weight %/day)

4) ESF Leakage – The PB model has a higher ESF leak rate (5 gpm vs. 2.85 gpm)

Table 3
Comparison of Design Input Related to Methodology Differences
AST LOCA Calculations PM-1077, Revision 1 versus H-1-ZZ-MDC-1880, Revision 3

Item No.	Design Input Information	H-1-ZZ-MDC-1880 Revision 3	PM-1077 Revision 1
1	Containment Elemental Iodine Removal by Wetted surface Area Model	Standard Review Plan 6.5.2	Standard Review Plan 6.5.2
2	Particulate (Aerosol) Deposition/Plateout Model	Powers' 10 percentile model	Powers' 10 percentile model
3	Total MSIV Leak Rate Through All Four Lines	250 scfh for < 24 hrs @ 50.6 psig 125 scfh for > 24 hr @ 50.6 psig	360 scfh for < 38 hrs @ 49.1 psig 180 scfh for > 38 hrs @ 49.1 psig
4	MSIV Leak Rate Through Line With MSIV Failed	150 scfh for <24 hrs @ 50.6 psig 75 scfh for > 24 hrs @ 50.6 psig	205 scfh for < 38 hrs @ 49.1 psig 102.5 scfh for >38 hrs @ 49.1 psig
5	MSIV Leak Rate Through First Intact Line	100 scfh for <24 hrs @ 50.6 psig 50 scfh for >24 hrs @ 50.6 psig	155 scfh for < 38 hrs @ 49.1 psig 77.5 scfh for > 38 hrs @ 49.1 psig
6	Maximum PCIV Closure (Isolation) Time	120.0 sec	Instantaneously
7	Iodine Specific Activity	0.2 μCi/g DE I-131	N/A
8	Noble Gas Specific Activity	100/Ē μCi/g	N/A
9	Maximum RCS Noble Gas Release Rates μCi/sec		
	KR-83M	3.40E+03	N/A
	KR-85M	6.10E+03	
	KR-85	2.00E+01	
	KR-87	2.00E+04	
	KR-88	2.00E+04	
	XE-131M	1.50E+01	
	XE-133M	2.90E+02	
	XE-133	8.20E+03	
	XE-135M	2.60E+04	
	XE-135	2.20E+04	
	XE-138	8.90E+04	
10	Maximum RCS Iodine Activity μCi/g		
	I-131	1.30E-02	N/A
	I-132	1.20E-01	
	I-133	8.90E-02	
	I-134	2.40E-01	
	I-135	1.30E-01	

Issue 2: For volumes after the outboard MSIV, what is the justification for using a 30th percentile deposition velocity?

Subsequent to HCGS 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 many successful AST license amendments for the EXELON fleet. 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. This resulted in some loss of dose margin.

The NRC staff concluded in AEB 98-03, page 11, that:

> "Given the conservatism associated with using a well-mixed model for the entire length of pipe and a number of additional conservatisms inherent in the piping deposition analysis, the use of a 10th percentile settling velocity with a well-mixed model is not appropriate. Additional conservatisms include additional deposition by thermophoresis, diffusiophoresis, and flow irregularities; additional deposition as a result of hygroscopicity; and a possible plugging of the leaking MSIV by aerosols. Given the conservatism of the well-mixed assumption, we believe it is acceptable to use median values (as compared to more conservative values) for deposition."

Therefore, a 50th percentile aerosol settling velocity is used in main steam piping upstream of the outboard MSIV, where the majority of heavier aerosol particle are expected to be deposited. The remaining lighter aerosol particles experience lesser gravitational deposition in the piping beyond the outboard MSIV. This mechanism is modeled using the 30th percentile aerosol settling, which is a median value between the 10th and 50th percentile settling velocities. The use of a lower 30th percentile settling velocity reduces the removal of the remaining lighter aerosols, and is conservative. The use of a lower settling velocity further increases the resulting doses. 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.

Table 4

Hope Creek MSIV Leakage Aerosol Deposition Model - Compliance With AEB 98-03 Methodology

Variable Parameter	AEB 98-03				Hope Creek AST Analysis - H-1-ZZ-MDC-1880, Revision 3					
	MSIV Failed Line		Intact Line		MSIV Failed Line			Intact Line		
	RPV To Inboard MSIV	Between Inboard and Outboard MSIVs	RPV To Inboard MSIV	Between Inboard and Outboard MSIVs	RPV To Inboard MSIV	Between Inboard and Outboard MSIVs	Between Outboard MSIV and TSV	RPV To Inboard MSIV	Between Inboard and Outboard MSIVs	Between Outboard MSIV and TSV
Piping Integrity Assumed	Ruptured - not credited in analysis	Remains Intact	Remains Intact	Remains Intact	Ruptured - not credited in analysis	Remains Intact	Remains Intact	Remains Intact	Remains Intact	Remains Intact
Aerosol Deposition	Not Credited	Credited	Credited	Credited	Not Credited	Credited	Credited	Credited	Credited	Credited
Piping Volume Dilution	Not Credited	Credited	Credited	Credited	Not Credited	Credited	Credited	Credited	Credited	Credited
Holdup Time	Not Credited	Not Credited	Not Credited	Not Credited	Not Credited	Not Credited	Not Credited	Not Credited	Not Credited	Not Credited
Deposition Velocity Distribution	Used 40 and 50 Percentile velocity for sensitivity study				Not Credited	50 Percentile	30 Percentile	50 Percentile		30 Percentile
Elemental Iodine Removal	Not Credited	Credited (50%)	Credited (50%)	Credited (50%)	Not Credited	Credited (50%)	Credited (50%)	Credited (50%)	Credited (50%)	Credited (50%)

Perry (AEB 98-03 pilot plant) does not have the seismically supported main steam line beyond the outboard MSIV; therefore, unlike the Hope Creek Plant, the main steam line between the outboard MSIV and Turbine Stop Valve (TSV) is not modeled.

The 10CFR50.59 evaluation that was done supporting Revision 3 of Calculation H-1-ZZ-MDC-1880 is provided as Attachment 6 to this submittal. While the 50.59 process/evaluation format is not designed to document the detail provided in the above discussion on the Revision 3 methodology changes, the evaluation does sufficiently describe and evaluate the Revision 3 methodology changes, and appropriately concludes that the calculation revision does not result in a departure from a method of evaluation that would require prior NRC approval.

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.

NRC RAI#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.

RESPONSE TO RAI#20

No damage to cobalt isotope rods occurs due to a MSLB event at HCGS, and no cobalt is released from the cobalt isotope rods. Cobalt isotope rods are significantly less likely to fail than fuel rods during operation, transients and design basis accidents not involving fuel melt (see discussion in RAI#21 response). Any event where no fuel damage is assumed can safely use the assumption that no isotope rod damage occurs.

NRC RAI#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/Ē 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.

RESPONSE TO RAI#21

The operational design limit for the isotope rods is no leakage. Furthermore, leakage of cobalt (including entire cobalt targets and/or cobalt particulate) from an isotope rod in an ITA is not a credible event during normal operations, transients or design basis accidents not involving fuel melt accidents (i.e., Loss of Coolant Accident and Control Rod Drop Accident). Based on regulatory guidance provided for fuel melt design basis accidents, it is conservatively assumed that cobalt (Co) isotope rods melt along with the fuel rods during a fuel melt design basis accident. The negligible impact of ITAs on CRDA and LOCA radiological consequences is addressed in the revised NEDC-33529P Sections 4.3.1 and 4.3.4, (See response to RAI#17).

The isotope rod design, discussed in Section 2.1 of NEDC-33529P provides multiple features to prevent cobalt isotope rod failures. The main features that provide multiple levels of safety for the cobalt isotope rods are:

- The nickel-plated cobalt targets are encapsulated with two layers of Zircaloy-2 cladding
- The solid Zircaloy-2 connections between cobalt rod segments are located at each spacer location (debris fretting failures normally occur at spacer locations)
- The heat generation rate of a cobalt isotope rod is significantly less than a typical fuel rod

GNF has experience with segmented rods in previous Lead Test Assembly programs. Introduction of a small number of isotope rods into non-limiting locations in the core add to the argument that leakage of cobalt is not a credible event during normal or transient events.

The GE14i materials and bundle configuration were purposely selected to be the same as GE14; the design that GNF has now deployed in approximately 26,000 bundles with over 10 years of successful operating experience. Of the over 70,000 rods in the HCGS core, only a small quantity will be cobalt bearing rods. The selection of the well-established bundle design for HCGS further reduces risk and performance uncertainty.

An explanation of isotope rod failures is provided Section 2.2 of NEDC-33529P. The failure mechanisms addressed include:

- Fuel handling accidents
- Manufacturing defects and assembly error
- Pellet cladding interaction
- Corrosion
- Primary hydriding
- Cladding creep collapse
- Rod bow
- Unthreading of segments
- Stress
- Seismic and flow induced vibration
- Internal fret from inner capsule
- Spacer location fretting
- Mid-span fretting
- Failures during disassembly

To further expand upon the failure modes discussed in NEDC-33529P, additional multiple levels of failure considerations are discussed below:

<u>Targets being mechanically pulverized, worn-out by fluid flow, corroded or otherwise damaged while still inside the inner tube or capsule to compromise nickel coating and release cobalt.</u>

In addition to the failure modes required to compromise the inner and outer cladding not being credible, this failure scenario itself is not credible for multiple reasons. The nickel plating of the targets is harder than all the Zircaloy-2 components that surround it. The nickel would therefore not be the material to grind or wear. It is more likely that the Zircaloy-2 tubing or canister grind or wear than the nickel. The coolant flow into an opening in the outer cladding and into an opening in the inner cladding would not have the necessary flow rate to cause any significant wear of any internal isotope rod components.

Additionally, there are no forces to excite the targets and sustain vibration or wear. Even considering the unlikely case that targets were to become excited, the magnitude of the displacement of the isotope rod and, in turn, [[ ]] would be so small that damage to inner tubing is highly implausible.

Finally, nickel is chosen as plating or alloying material in many applications, including BWR alloys partly, because of its ability to withstand severe operating conditions involving corrosive environments.

<u>Targets escaping segment assembly through a cladding hole and being mechanically pulverized to release cobalt.</u>

In addition to the failure modes required to compromise the inner and outer cladding not being credible, this failure scenario itself is also not credible for multiple reasons. If an inner tube were to be compromised, the [[ ]]. Two layers of cladding would have to be breached at the exact same axial and radial position and the breach would have to be greater than the size of a target for any targets to escape. After escape, the target would have to find a mechanical pulverizing mechanism against a material harder than nickel. This scenario is considered highly implausible.

<u>Targets escaping segment assembly resulting from canister [[ ]] and release of cobalt</u>.

In addition to the failure modes required to compromise the inner and outer cladding not being credible, this failure scenario itself is also not credible for multiple reasons. In the remote chance that full circumferential failure of the inner and outer cladding occurred at the same location, on the same end of the same segment, the rod-to-rod and rod-to-channel spacing of the surrounding rods and/or fuel channel is too small to allow a [[ ]] and release targets. [[ ]]

Regarding coolant flow into the opening after two full circumferential failures, as described above, the nickel plating of the targets is harder than all the Zircaloy-2 components that surround it. The nickel would therefore not be the material to experience significant flow induced wear. Additionally, the coolant flow [[ ]] would not have the necessary flow rate to cause any significant wear of any internal isotope rod components. The nickel plating of the targets would remain intact to prevent cobalt release into the coolant.

Even assuming [[ ]] and targets escaping from the segment, the targets would have to find a mechanical pulverizing mechanism against a material harder than nickel. These scenarios are considered highly implausible.

In summary, there are no plausible mechanisms for both the outer and inner cladding of an isotope rod to be compromised such that cobalt targets come in contact with the reactor coolant. If it is assumed that some unknown event were to occur such that the outer and inner cladding of the same rod segment (there are 9 independent rod segments in each cobalt isotope rod) were compromised, there is no plausible mechanism for cobalt targets to lose their nickel coating and release cobalt. The nickel-plating on the targets is harder than the Zircaloy-2 cladding materials surrounding them, so any wear associated with component interaction would be to the softer Zircaloy parts.

Additionally, combining any of these non-credible events such that the outer and inner cladding of the same segment were compromised there is no plausible mechanism to align the breach points to allow a cobalt target to escape or allow [[ ]] to release targets to the coolant. Coolant flow is also not sufficient to negatively affect the plating on the targets.

Even adding these multiple levels of non-credible events, the segmented rod structure, with 9 individual double encapsulated containers, also ensures that the number of cobalt targets that can escape is limited to a small volume fraction of the targets in a single rod. This additional characteristic ensures that, in the event of multiple levels of failure that result in a single isotope rod segment failure, cobalt activity release is limited.

Traditional design basis analysis assumes some leakage of fuel rods, which is incorporated into technical specifications (TS) and is consistent with the design basis analyses. As described above, isotope rods have multiple layers of cladding and design features beyond a fuel rod's single layer of cladding <u>and</u> the isotope rods essentially act as a passive component in the operation of the bundle. Leakage of cobalt from an isotope rod is not a credible event during normal operations, transients or design basis accidents not involving fuel melt.

Fuel leakage is characterized by release of highly volatile gaseous fission products after failure of a single layer of cladding. Isotope rod leakage is characterized by the release of a low volatility metal (i.e., cobalt in target and/or particulate form) after the failure of an outer layer of cladding, an inner layer of cladding and compromising nickel plating. [[ ]]

In summary, by design and definition, isotope rod failure is not credible and isotope rod leakage does not need to be incorporated into TS to remain consistent with traditional design basis analyses. If there is no fuel melt due to an accident, the source term available for release to the environment is based upon the activity in the RCS during normal operations. With Co-60 leakage not credible during design basis accidents not involving fuel melt, the proposed change will have no impact on the source term.

Although isotope rod leakage is not a credible event during normal operations, transient and design basis accidents not involving fuel melt, it should be further noted that the existing TS surveillance requirements and periodic reactor coolant sampling detect Co-60 activity. HCGS

TS 3.4.5, "RCS Specific Activity," has, in addition to an equivalent I-131 specific activity limit, a limit for RCS gross specific activity (E-bar). E-bar is defined in the TS as:

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

> $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

HCGS Surveillance Requirement (SR) 4.4.5 requires that, in Mode 1, radiochemical analysis for E-bar determination shall be performed at least once per 6 months (there is also a further requirement for E-bar that a sample to be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was last subcritical for 48 hours or longer.). SR 4.4.5 also requires, in Modes 1, 2 and 3, a Gross Beta and Gamma Activity Determination every 72 hours. Consequently, the existing LCO 3.4.5 remains adequate for ensuring dose limits and radiation shielding and plant personnel radiation protection design limits are met with GE14i ITAs installed.

References for Attachment 1 (unless uniquely identified in individual responses)

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.

5. 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