Exelon Nuclear

Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

RS-10-010

10 CFR 50.90

January 11, 2010

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: 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

References:

1. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "License Amendment Request to Modify Clinton Power Station Facility Operating License in Support of the Use of Isotope Test Assemblies," dated June 26, 2009

2. Letter from U. S. NRC to Mr. Charles G. Pardee (Exelon Generation Company, LLC), "Clinton Power Station, Unit No. 1 – Request for Additional Information Related to License Amendment Request to Modify Clinton Power Station Facility Operating License in Support of the Use of Isotope Test Assemblies (TAC No. ME1643)," dated November 2, 2009 (ADAMS Accession No. ML093030218)

3. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. 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 20, 2009

4. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. 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 December 16, 2009

5. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. 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 December 28, 2009

6. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. 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 17, 2009

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to the facility operating license for Clinton Power Station (CPS), Unit 1. Specifically, the proposed change would modify CPS 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, EGC also requests an amendment to Appendix A, Technical Specifications (TS), of the CPS Facility Operating License. This proposed change would modify TS 4.2.1, "Fuel Assemblies," to describe the Isotope Test Assemblies (ITAs) being used. In Reference 2, the NRC requested that EGC provide additional information in support of their review of Reference 1. The NRC request for additional information and the specific EGC responses were provided in Reference 3. At the request of the NRC, EGC also provided additional clarification of the Reference 3 RAI 2 response in Reference 4.

On November 30, 2009, EGC and the NRC conducted a conference call to discuss the responses provided in Reference 3. During this conference call additional information was requested concerning the EGC response to RAI 4. A number of subsequent conference calls were also conducted to address the request for clarification on the EGC response to RAI 4. In a call on December 17, 2009 with the CPS NRC Project Manager, EGC was asked to provide an evaluation of the need for a new TS to address the addition of the ITAs in the CPS core. Specifically, the NRC requested that EGC perform an evaluation against the requirements of 10 CFR 50.36, "Technical specifications," to determine if a new TS addressing a limit on cobalt in the reactor coolant system is required. This requested evaluation was provided in the attachment to Reference 5.

In addition to the 10 CFR 50.36 evaluation, EGC was asked to perform an evaluation to determine the impact from the use of a potentially higher release fraction for Co-60. EGC has completed this evaluation and the results of the evaluation are provided in Attachment 1 to this letter.

A follow-up conference call was conducted between the NRC and EGC on January 5, 2010. Additional questions were asked by the NRC reviewer concerning the previous EGC responses for additional information. In an email from the CPS NRC Project Manager to Timothy A. Byam on January 6, 2010, an additional set of questions was provided in support of the NRC review of References 3, 4, and 5. The responses to these questions are provided in Attachment 2 to this letter.

Attachment 2 contains information which GE – Hitachi Nuclear Energy Americas, LLC (GEH) considers to be proprietary. The proprietary information is identified by bracketed text. GEH requests that the proprietary information in Attachment 2 be withheld from public disclosure, in accordance with the requirements of 10 CFR 2.390 paragraph (a)(4). A signed affidavit supporting this request is provided in Attachment 3.

Attachment 4 to this letter provides a non-proprietary version of the responses in Attachment 2.

EGC has reviewed the information supporting a finding of no significant hazards consideration that was provided to the NRC in Reference 5. 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.

In order to support refueling activities, approval of this amendment is required by January 15, 2010.

If you have any questions concerning this letter, please contact Mr. Timothy A. Byam at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 11th day of January 2010.

Respectfully,

Jeffrey L. Hansen
Manager – Licensing
Exelon Generation Company, LLC

Attachments:

1. Release Fraction Sensitivity Study in Support of Resolution to NRC Concern Identified During Review of CPS ITA LAR

2. 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 (Proprietary)

3. GE Affidavit for Withholding Portions of RAI Responses from Public Disclosure

4. 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 (Non-Proprietary)

ATTACHMENT 1

Clinton Power Station Technical Evaluation

Release Fraction Sensitivity Study in Support of Resolution to NRC Concern Identified During Review of CPS ITA LAR

Technical Evaluation

Passport # 927236-17

Release Fraction Sensitivity Study in Support of Resolution to NRC Concern Identified During Review of CPS ITS LAR

REASON FOR EVALUATION / SCOPE

This Technical Evaluation is to document a sensitivity study to show the effects of increased Co-60 during a loss of coolant accident (LOCA). This study is being performed in response to a discussion with the NRC in response to a request for additional information (RAI). This is in support of RAI response RS-09-181 (Ref. 4).

DETAILED EVALUATION

From CPS Calculation C-020, Rev. 3A (Ref. 1), the current AST analysis assumes 648,369 Curies of Co-60 in the core. Note that Co-60 is not a fuel fission product. Rather, it is assumed to be present as corrosion and activation products on fuel and reactor system components. This value is the NRC conservative default value used in the RADTRAD v3.03 computer program (Ref. 2).

In accordance with NEDC-33505P, Rev. 0 (Ref. 3), 1,000,000 Ci Co-60 is added into the RADTRAD nuclide core inventory. However, in order to assess the issue with a potentially higher release fraction, the total inventory (including the original value for conservatism) is multiplied by a factor of 10 in accordance with the phone discussion we had with Mark Blumberg on December 17, 2009 (Ref. 4). This was then converted into a new total activity in Ci/MWth for input into the RADTRAD .nif file. This resulted in a value of 0.4657E+04 Ci/MWth based on 3543 MWth, which includes 102% of licensed reactor power (Ref. 1) Therefore, 650,000 + 1,000,000 = 1,650,000 *10 = 16,500,000 Ci. 16,500,000 Ci / 3543 MWth = 0.4657E+04 Ci/MWth.

All 22 RADTRAD plant scenario files were verified to produce the same results as in Calculation C-20, Rev. 3A (Ref. 1).

Each RADTRAD case was re-run using the updated Nuclide Inventory Files (.nif) as described above, obtaining a new set of results. Since the Control Room dose was the limiting dose (closest to the 10 CFR 50.67 limit), only the CR TEDE dose was evaluated for this study. Results for offsite doses can be inferred based on the ratio of fractional changes. Results of this study can be seen in Table 1 (attached).

CONCLUSIONS/FINDINGS

RADTRAD Run #	Calc C-20 Rev. 3A (Rem TEDE)		Using B.I.G. Source Term (Rem TEDE)		Fractional Change
1	7.630E-02		7.637E-02		0.0009
2	3.989E-03		3.989E-03		0.0000
3	2.351E-01		2.352E-01		0.0005
4	4.092E-02		4.092E-02		0.0001
5	2.545E-04		2.545E-04		0.0001
6	1.208E-06		1.208E-06		0.0003
7	4.855E-01		4.861E-01		0.0012
8	5.836E-02		5.837E-02		0.0001
9	2.545E-04		2.545E-04		0.0001
10	1.208E-06		1.208E-06		0.0003
11	2.896E-02		2.903E-02		0.0025
12	4.345E-03		4.345E-03		0.0000
13	6.085E-05		6.085E-05		0.0000
14	2.886E-07		2.886E-07		0.0000
15	1.104E-01		1.104E-01		0.0000
16	2.620E-02		2.620E-02		0.0000
17	7.300E-01		7.301E-01		0.0002
18	2.376E-01		2.376E-01		0.0000
19	1.588E+00		1.588E+00		0.0000
20	1.717E-02		1.717E-02		0.0000
21	4.748E-01		4.760E-01		0.0024
22	2.239E-02		2.239E-02		0.0001

			Average Fractional Change:	0.0004
RADTRAD	4.1406	4.1428		
Shine Dose	0.590	0.590		
Total Dose	4.7306	4.7328		

Using conservative assumptions, there is no observable change in the total dose for the Control Room within the accuracy of the reported doses.

REFERENCES

1. CPS Calculation C-020, Rev. 3A; "Reanalysis of Loss of Coolant Accident (LOCA) Using Alternative Source Terms" March 21, 2006

2. S.L. Humphreys et al., "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," Version 3.03, NUREG/CR-6604, USNRC, April 1998.

3. NEDC-33505P, Rev. 0: "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITA) in Clinton Power Station" June 2009 (Proprietary)

4. RS-09-181, Subject: "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"

ATTACHMENTS

NOTE: RADTRAD Outputs have been truncated to reduce the size of the outputs.

1. RADTRAD Nuclide Inventory File as Used in Calculation C-020, Rev. 3A
2. RADTRAD Nuclide Inventory File as Used in This Evaluation
3. RADTRAD Output Case 1
4. RADTRAD Output Case 2
5. RADTRAD Output Case 3
6. RADTRAD Output Case 4
7. RADTRAD Output Case 5
8. RADTRAD Output Case 6
9. RADTRAD Output Case 7
10. RADTRAD Output Case 8
11. RADTRAD Output Case 9
12. RADTRAD Output Case 10
13. RADTRAD Output Case 11
14. RADTRAD Output Case 12
15. RADTRAD Output Case 13
16. RADTRAD Output Case 14
17. RADTRAD Output Case 15
18. RADTRAD Output Case 16
19. RADTRAD Output Case 17
20. RADTRAD Output Case 18
21. RADTRAD Output Case 19
22. RADTRAD Output Case 20
23. RADTRAD Output Case 21
24. RADTRAD Output Case 22

Preparer: **See Milestone** **Date:** **01/05/2009**
T.J. Mscisz

Independent Reviewer: **See Milestone** **Date:** **01/05/2010**
J. DeLaRosa

Approved: **See Milestone** **Date:** **01/05/2010**
M. Heger

ATTACHMENT 3

GE Affidavit for Withholding Portions of RAI Responses from Public Disclosure

GE-Hitachi Nuclear Energy Americas LLC
AFFIDAVIT

I, Edward D. Schrull, state as follows:

(1) I am Vice President, Regulatory Affairs, Services Licensing, GE-Hitachi Nuclear Energy Americas LLC ("GEH"), 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, JMD-EXN-LH1-10-004, J. Michael Downs, GNF, to Timothy Byam, Exelon Nuclear, "Responses to Request for Additional Information Related to License Amendment Request to Modify Clinton Power Station Facility Operating License in Support of the Use of Isotope Test Assemblies," dated January 11, 2010. The proprietary information contained in Enclosure 1 of that letter, which is entitled "Responses to Request for Additional Information Related to License Amendment Request to Modify Clinton Power Station Facility Operating License in Support of the Use of Isotope Test Assemblies," is identified by a dark red font and 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 11th day of January 2010.

Edward D. Schrull
Vice President, Regulatory Affairs
Services Licensing
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Rd.
Wilmington, NC 28401
edward.schrull@ge.com

ATTACHMENT 4

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
(Non-Proprietary)

ATTACHMENT 4

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 (Non-Proprietary)

NRC RAI 1:
It is unclear as to whether leakage from the ITAs is a credible event during normal operations or a design-basis accident or transient. Please indicate whether your analysis considers this a credible scenario and if not provide justification. If this is a credible scenario, please address how operational restrictions will be established to ensure the initial conditions of your design-basis will be met (including operations related to NUREG-0737).

RESPONSE 1:
Leakage of cobalt (including entire cobalt targets and/or cobalt particulate) from an isotope rod in an Isotope Test Assembly (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. Sensitivity studies on the effects of increased levels of Co-60 in the core have been performed for Loss of Coolant Accident (LOCA) and Control Rod Drop Accident (CRDA) design bases accidents. The studies have shown negligible effects of increased cobalt on the accident analyses.

The isotope rod design discussed in Section 2.1 of NEDC-33505 (Reference 1 Attachment 3) 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 approximately [[ ]] less than a typical fuel rod

Additionally, GNF experience with segmented rods in previous Lead Use Assembly programs and introduction of only [[ ]] 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 ITA (i.e., 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. Only [[ ]] out of over 57,000 rods in the Clinton Power Station (CPS) core will be cobalt bearing rods. The selection of this well-established bundle design for CPS, which has never experienced a fuel failure, further reduces risk and performance uncertainty.

An explanation of isotope rod failures was provided in the response to RAI 9(a) on November 4, 2009 (Reference 2). The failure mechanisms addressed included:

- 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

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 very small volume. 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 and dose to plant personnel and the public is protected.

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 and the isotope rods essentially act as a passive component in the operation of the bundle. As stated above, 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. [[ ]] Therefore, 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.

To further expand upon the failure modes listed in the response to RAI 9(a) in Reference 2, additional multiple levels of failure considerations are included below:

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.

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.

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

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.

Targets escaping segment assembly resulting from canister [[ ]] and release of cobalt.

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.

Detection in the Remote Chance of Cobalt Release

As discussed earlier, in the extremely remote chance of cobalt exposure to the reactor coolant, any wear to the cobalt targets would be slow, and an increasing trend would be detected by the plant water chemistry monitoring as a gradual rise in the Co-60 activity values. It is highly unlikely for a breach of the isotope rods to occur such that a sudden large spike in Co-60 values would be seen. In the remote event of increased cobalt levels, administrative action can then be taken to mitigate further release of cobalt. Since CPS is historically a low cobalt plant, changes to Co-60 activity levels in the reactor coolant would be readily apparent.

The reactor water sampling program for CPS describes the frequencies of analysis, chemistry control specifications, and corrective actions for reactor water chemistry control. This program also defines the requirements for the Reactor Water Chemistry control based on Reference 3.

NRC RAI 2:

The 50.36 evaluation provided on December 28, 2009 contained a scoping assessment which evaluated the impact of the proposed change on control room dose during a design basis Loss of Coolant Accident (LOCA). Per Regulatory Guide 1.183, "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 to evaluate the response of a facility's engineered safety features." Likewise, Standard Review Plan 15.0 states: "The *reviewer considers the possible case variations of AOOs and postulated accidents presented to verify that the licensee has identified the limiting cases." Please provide an assessment of the impact of a potentially higher release fraction for Co-60 on other accidents, which consider core melt (i.e. control rod drop accident) or provide a justification why the LOCA continues to be the limiting accident with the proposed change.*

RESPONSE 2:

Using a similar approach as was used in the LOCA scoping study (see Attachment 1 of this letter), a release fraction increase of 10 was applied to Co-60 for the melted cobalt isotope rods during a CRDA. Conservatively assuming all cobalt isotope rods in one ITA in the proximity of the dropped blade melt during a CRDA (which is conservative since only 12 rods melt in the CPS CRDA licensing basis), the maximum increase in the offsite dose is 0.4%. The resulting offsite dose is more than two orders of magnitude below the 6.3 rem total effective dose equivalent (TEDE) regulatory dose criterion. The maximum increase to the control room dose is 0.1%; the resulting control room dose remains more than one order of magnitude below the 5 rem TEDE regulatory dose criterion.

NRC RAI 3:

The 50.36 evaluation, provided on December 28, 2009, stated that gross gamma radioactivity rate gases are measured at the condenser evacuation system pretreatment monitor station and is limited by LCO 3.7.5, "Main Condenser Offgas." It also states that the reasoning for not having a reactor coolant gross activity technical specification is provided in BWR-12 which states:

> *The offgas pretreatment sample provides a more representative sample of the noble gases that would be released in the event of a main steam line failure outside containment than did the reactor coolant sample taken from the reactor recirculation system as part of the former gross specific activity requirement. The offgas pretreatment monitor includes a setpoint which responds to release rates above a specified level which is established to ensure that untreated releases would not result in a whole body dose that exceeds a small fraction of the limits of 10 CFR 100.*

The measurement of condenser offgas is likely insensitive to Co-60 in the reactor coolant system. While, radioactive gas is more likely to be release to the environment for typical fuel rods, design basis calculations do include the effects of particulate radioactivity (i.e. See Regulatory Guide 1.183, Regulatory Position 5.5.4). The staff believes that an evaluation of the effects of particulates is appropriate for the proposed change because the dose contribution due to particulates could become important at high concentrations of dose significant particulates. The NRC staff is concerned that the 50.36 evaluation does not consider the impact of particulates on the design basis MSLB (MSLB). Therefore, the NRC staff requests the following information:

a) *Provide an analysis of the MSLB which shows that if all the available Co-60 and particulates due to moisture carryover are considered that the regulatory dose limits are met (or)*

b) *Provide an analysis of the MSLB which shows that for a given Co-60 reactor coolant concentration, defined by suitable controls (i.e. limiting condition of operation, license condition, etc.) and particulates due to moisture carryover are considered that the regulatory dose limits are met.*

RESPONSE 3:
The 10 CFR 50.36 evaluation as documented in Reference 4, was intended to document why the existing TS were acceptable for ensuring CPS compliance with the 10 CFR 50.67 and 10 CFR 100 limits as applicable and why a new TS to monitor cobalt in the reactor coolant system was not necessary. BWR-12 was referenced since it provided the basis for why CPS TS Limiting Condition for Operation (LCO) 3.4.8, "RCS Specific Activity," did not contain a limit for reactor coolant system gross specific activity. It was not the intent of this discussion to provide a basis for justifying the assumptions used in the accident analyses for CPS.

CPS has been licensed for Alternative Source Term (AST) as documented in License Amendment 167 (Reference 5). In support of this amendment, Exelon Generation Company, LLC (EGC) performed the necessary analyses in accordance with the guidance provided in Regulatory Guide (RG) 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000. EGC performed the Main Steam Line Break AST analysis utilizing the assumptions defined in RG 1.183 Appendix D, "Assumptions for Evaluating the Radiological Consequences of a BWR Main Steam Line Break Accident." The above RAI 3 question specifically references RG 1.183 Appendix E, "Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident," Regulatory Position 5.5.4. This regulatory position is an assumption for the PWR analysis and there is no equivalent assumption for BWRs. The CPS BWR offgas system contains a series of adsorbers and filters designed to remove the particulates from the offgas mixture prior to release. Therefore, based on the BWR offgas system design, the assumptions for transport of the particulates to the environment do not exist for the BWR. There is no moisture carryover issue in the BWR similar to that addressed in the Appendix E for the PWR assumptions described in Regulatory Position 5.5.4.

Based on the above, EGC believes that the Main Steam Line Break (MSLB) analysis as evaluated in Attachment 3 Reference 1 and supported in the RAI response provided in Reference 6 does not need to be revised to address the concerns associated with moisture carryover and additional cobalt in the reactor coolant system. The assumptions identified in RG 1.183 Appendix D continue to be applicable and conservative for the proposed introduction of a limited number of ITAs at CPS. As described in Regulatory Position 4.2 in RG 1.183 Appendix D the total mass of coolant released is assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure. The Main Steam Isolation Valves are assumed to close in no more than 5 seconds after receiving a signal to close in accordance with CPS TS Surveillance Requirement (SR) 3.6.1.3.6. This significantly limits the amount of any particulates in the reactor coolant system from being released during a Main Steam Line Break.

NRC RAI 4:
The 50.36 evaluation, provided on December 28, 2009, does not reflect an analysis of the proposed change on design basis accidents other than the MSLB or LOCA. The December 16, 2009, responses to a request for additional information provide an assessment of the impact of the proposed change on several other accidents. The impact of the proposed change for many of these accidents states that radiological analysis concludes that no fuel failures or isotope rod failures result due to the event and, therefore, the radiological consequences are unchanged. These arguments appear to assume that the ITS rods do not contribute to any additional Co-60 in the reactor coolant system during normal operations since the effect of the normal reactor coolant system activity is not considered in the evaluation.

This appears to be different from the analysis performed for the Main Condenser Off Gas Treatment System Failure and MSLB design basis analyses that justifies the change using a different method. These analyses assume that no gross specific activity requirements are necessary because LCO 3.7.5 provides reasonable assurance that the reactor coolant gross specific activity is maintained at a sufficiently low level to preclude offsite doses from exceeding a small fraction of the limits of 10 CFR Part 100. Clinton appears to assume that any Co-60 introduced into the reactor coolant system will not contribute to dose because it is in particulate form.

While the ITS rods are designed to not leak, traditional design basis analysis assumes some leakage (which is incorporated into technical specifications and is consistent with the design basis analyses).
Based upon the above discussion, the NRC staff requests the following information for accidents other than fuel melt accidents or the MSLB accident.

c) *Provide an analysis for each of these accidents which shows that if all the Co-60 in the ITS rods were introduced into that reactor coolant system, that regulatory dose limits are met (or)*
d) *Provide an analysis for each of these accidents that shows that for a given Co-60 reactor coolant concentration, defined by suitable controls (i.e. limiting condition of operation, license condition, etc.) that the applicable regulatory dose limits are met.*

RESPONSE 4:
As stated in response to Question 1, leakage of cobalt (including entire cobalt targets and cobalt particulate) from an isotope rod is not a credible event during normal operations, transients or design basis accidents not involving fuel melt accidents (i.e. LOCA and CRDA). Based on regulatory guidance provided for fuel melt design basis accidents, it is conservatively assumed that cobalt isotope rods melt along with the fuel rods during a fuel melt design basis accident. Sensitivity studies on the effects of increased levels of Co-60 in the core have been performed for LOCA and CRDA design bases accidents. The studies have shown negligible effects of increased cobalt on the accident analyses. For design basis accidents other than fuel melt accidents, no fuel failures or isotope rod failures result due to the events and, therefore, the radiological consequences remain unchanged.

During normal operation, transients and design basis accidents other than those involving fuel melt, cobalt isotope rods will not contribute any additional Co-60 to the reactor coolant system and therefore, the cobalt activity in the Reactor Coolant System (RCS) need not be considered as part of the initial conditions for accident evaluation.

Additionally, the analyses for Main Condenser Offgas Treatment System Failure and MSLB have no gross specific activity requirements because applicable LCOs provide reasonable assurance that the reactor coolant specific activity is maintained at a sufficiently low level to preclude offsite doses from exceeding a small fraction of the regulatory dose limits. Since leakage of cobalt from an isotope rod is not a credible event during normal operations, transients and all design basis accidents except LOCA and CRDA, there will be no Co-60 introduced into the reactor coolant system and therefore, no contribution to dose from Co-60 particulate. This ensures that regulatory dose limits remain unchallenged.

Traditional design basis analyses do assume some leakage of fuel rods, which is incorporated into technical specifications and is consistent with the design basis analyses. As described in Question 1, isotope rods have multiple layers of cladding and design features beyond a fuel rod's single layer of cladding and the isotope rods essentially act as a passive component in the operation of the bundle. As stated in response to Question 1, leakage of cobalt from an isotope rod is not a credible event during normal operations, transients or design basis accidents not involving fuel melt. Because isotope rod leakage is not considered a credible event during normal operation or transients, unlike fuel rod failure, isotope rod leakage does not need to be incorporated into TS to remain consistent with traditional design basis analyses.

Additionally, 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. [[
]] Therefore, by design and definition, isotope rod failure is not a credible event and isotope rod leakage does not need to be incorporated into technical specifications to remain consistent with traditional design basis analyses.

Regardless of the improbability of all failure mechanisms, in the remote chance of cobalt exposure to the reactor coolant, any wear to the cobalt targets would be very slow, and an increasing trend would be detected by the plant water chemistry monitoring as a gradual rise in the Co-60 activity values. It is even more highly unlikely for a breach of the isotope rods to occur such that a sudden large spike in Co-60 values would be seen. In the remote event of increased cobalt levels, administrative action can then be taken to mitigate further release of cobalt and to ensure that regulatory dose limits are not challenged. Since CPS is historically a low cobalt plant, changes to Co-60 activity levels in the reactor coolant would be readily apparent.

The current reactor water sampling program for CPS describes the frequencies of analysis, chemistry control specifications, and corrective actions for reactor water chemistry control. This program also defines the requirements for the Reactor Water

Chemistry control based on Reference 3. These controls ensure that applicable regulatory dose limits are met for the safety of the public and plant personnel.

As stated in Question 1, even adding multiple levels of non-credible events, the segmented rod structure, with nine individual double encapsulated containers, also ensures that the number of cobalt targets that can escape is limited to a very small volume. 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. This release is detectable by the current reactor water chemistry monitoring program and administrative action can be taken to limit dose to plant personnel and the public well below regulatory dose limits.

<u>NRC RAI 5:</u>

Per the December 16, 2009, responses to additional information, the CPS Main Condenser Off Gas Treatment System Failure design basis radiological analysis is based on a 100,000 μCi/sec after 30 minutes delay noble gas source term. This appears to be consistent with page 4 of calculation CC-AA-309-1001, Revision 2 for the MSLB. The technical specifications are derived from the safety analysis. Technical specification 3.7.5, "Main Condenser Offgas," states: "The radioactivity rate of the noble gases measured at the offgas recombiner effluent shall be less than 289 mCi/second after decay of 30 minutes." This is less limiting than the assumption of 100 mCi/sec after 30 minutes in the safety analysis. Please justify why the analysis assumptions are less conservative than those monitored by the technical specifications.

<u>RESPONSE 5:</u>

In Reference 4, EGC provided an explanation as to why the current CPS TS LCO 3.4.8 no longer has a limit for RCS gross specific activity. When the improved standard TS were developed for the BWR/6 design (i.e., NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6") the limit for the gross specific activity was removed under the approved Standard Technical Specification Change Traveler BWR-12. The basis for this change was that since the following is true the requirements associated with the gross specific activity in TS LCO 3.4.8 are unnecessary.

(1) The reactor coolant limit on DOSE EQUIVALENT I-131 adequately assures that offsite doses will not exceed small fractions of the limits of 10 CFR 100 (10 CFR 50.67 for the current CPS licensing basis) in the event of a MSLB outside containment, and

(2) The gross gamma radioactivity rate of the noble gases measured at the condenser evacuation system pretreatment monitor station is limited by TS LCO 3.11.2.7 (see CPS LCO 3.7.5, "Main Condenser Offgas") to a value that provides reasonable assurance the reactor coolant gross specific activity is maintained at a sufficiently low level to preclude offsite doses from exceeding a small fraction of the limits of 10 CFR 100.

The only reason this discussion was provided in the attachment to Reference 4 was to explain why the CPS TS LCO 3.4.8 no longer had a gross specific activity limit. It was not provided to support the use of the ITAs under the proposed amendment request.

The current CPS TS LCO 3.7.5 wording is based on NUREG-1434 and calculation of the radioactivity rate is consistent with the standard. There is no justification as to why the analysis assumption is less conservative than the TS LCO. As documented in the Bases for TS 3.7.5, the LCO is conservatively established to ensure compliance with the assumptions of the Main Condenser Offgas Treatment System failure event. This does appear to be a discrepancy since the analysis basis as described in CPS USAR Section 15.7.1.1 is less conservative. EGC will initiate an issue report within our corrective action program and will work with the industry to address the issue. However, as described above, this issue does not impact the proposed introduction of the ITAs in the CPS core.

The Main Condenser Offgas Treatment System failure analysis basis is not impacted by the proposed introduction of the ITAs to the CPS core. The basis for this analysis, as documented in the question above and referred to in TS LCO 3.7.5, is noble gas radioactivity rate in the offgas system. The introduction of the ITAs will not affect the amount of noble gas present in the reactor coolant system or in the offgas system. The difference between the LCO 3.7.5 noble gas rate and the accident analysis assumption as documented in CPS USAR 15.7.1.1 has no bearing on the proposed amendment request.

References:

1. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "License Amendment Request to Modify Clinton Power Station Facility Operating License in Support of the Use of Isotope Test Assemblies," dated June 26, 2009

2. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. 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

3. BWR Vessel Internals Program (BWRVIP) Report BWRVIP-190, "BWR Water Chemistry Guidelines – 2008 Revision", TR-1016579

4. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. 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 December 28, 2009

5. Letter from U. S. NRC to Mr. Christopher M. Crane (AmerGen Energy Company, LLC), "Clinton Power Station, Unit 1 – Issuance of an Amendment – Re: Application of Alternative Source Term Methodology (TAC No. MB8365)," dated September 19, 2005

6. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting the Request for a License Amendment to Modify

ATTACHMENT 4

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 (Non-Proprietary)

Clinton Power Station Facility Operating License in Support of the Use of Isotope Test Assemblies," dated November 20, 2009