October 7, 2010

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear LLC
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: USE OF ISOTOPE TEST ASSEMBLIES FOR COBALT-60 PRODUCTION (TAC NO. ME2949)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 184 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station (HCGS). This amendment consists of changes to the Technical Specifications and Facility Operating License in response to your application dated December 21, 2009, as supplemented by letters dated May 11, June 10, June 24, June 29, July 28, August 3, August 12, September 10, and September 17, 2010.

The amendment allows the production of Cobalt-60 by irradiating Cobalt-59 targets located in modified fuel assemblies called Isotope Test Assemblies (ITAs). The amendment allows up to 12 ITAs to be loaded into the HCGS reactor core beginning with the fall 2010 refueling outage. The modified fuel assemblies are planned to be in operation as part of a pilot program. The purpose of the pilot program is to obtain data to verify that the modified fuel assemblies perform satisfactorily in service prior to use on a production basis. The Cobalt-60 is ultimately intended for use in the medical industry for use in cancer treatments, and blood and instrument sterilization; in the radiography and security industries for imaging; and in the food industry for cold pasteurization and irradiation sterilization.

Enclosure 2 transmitted herewith contains sensitive unclassified information. When separated from Enclosure 2, this document is decontrolled.

The Nuclear Regulatory Commission (NRC) staff has determined that its safety evaluation (SE) for the subject amendment contains proprietary information pursuant Title 10 of the *Code of Federal Regulations*, Section 2.390. Accordingly, the NRC staff has prepared a redacted, publicly available, non-proprietary version of the SE. Both versions of the SE are enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

RB Ennis

Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 184 to
   License No. NPF-57
2. Proprietary SE
3. Non-Proprietary SE

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 184
License No. NPF-57

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by PSEG Nuclear LLC dated December 21, 2009, as supplemented by letters dated May 11, June 10, June 24, June 29, July 28, August 3, August 12, September 10, and September 17, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-57 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 184, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: October 7, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 184

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Facility Operating License with the revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
2	2
3	3
12	12

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
xxiii	xxiii
5-4	5-4
5-5	5-5

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. NPF-57, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

J. The receipt, production, possession, transfer, and use of Cobalt-60 as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30.

2. Based on the foregoing findings and approval by the Nuclear Regulatory Commission at a meeting on July 21, 1986, the License for Fuel Loading and Low Power Testing, License No. NPF-50, issued on April 11, 1986, is superseded by Facility Operating License NPF-57 hereby issued to PSEG Nuclear LLC (the licensee), to read as follows:

A. This license applies to the Hope Creek Generating Station, a boiling water nuclear reactor, and associated equipment (the facility) owned by PSEG Nuclear LLC. The facility is located on the licensee's site on the east bank of the Delaware River in Lower Alloways Creek Township, Salem County, New Jersey. The facility is located approximately eight miles southwest of Salem, New Jersey and is described in the PSEG Nuclear LLC Final Safety Analysis Report, as supplemented and amended, and in the Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) PSEG Nuclear LLC, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the above designated location in Salem County, New Jersey, in accordance with the procedures and limitations set forth in this license;

(2) Deleted

(3) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Mechanical disassembly of the GE14i isotope test assemblies containing Cobalt-60 is not considered separation.

(7) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 30, to intentionally produce, possess, receive, transfer, and use Cobalt-60.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of 3840 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 184, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Inservice Testing of Pumps and Valves (Section 3.9.6, SSER No. 4)*

This License Condition was satisfied as documented in the letter from W. R. Butler (NRC) to C. A. McNeill, Jr. (PSE&G) dated December 7, 1987. Accordingly, this condition has been deleted.

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* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

h. actions to be taken if acceptance criteria are not satisfied; and

i. verification of the completion of commitments and planned actions specified in its application and all supplements to the application in support of the EPU license amendment request pertaining to the steam dryer prior to power increase above 3339 MWt.

PSEG Nuclear LLC shall provide the related EPU startup test procedure sections to the NRC staff prior to increasing power above 3339 MWt.

4. The following key attributes of the program for verifying the continued structural integrity of the steam dryer shall not be made less restrictive without prior NRC approval:

   a. During initial power ascension testing above CLTP, each test plateau increment shall be approximately 5 percent of 3339 MWt;

   b. Level 1 performance criteria; and

   c. The methodology for establishing the stress spectra used for the Level 1 and Level 2 performance criteria.

   Changes to other aspects of the program for verifying the continued structural integrity of the steam dryer may be made in accordance with the guidance of NEI 99-04.

5. During the first scheduled refueling outage after Cycle 15 and during the first two scheduled refueling outages after reaching full EPU conditions, a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer in accordance with BWRVIP-139 inspection guidelines.

6. The results of the visual inspections of the steam dryer shall be reported to the NRC staff within 90 days following startup from the respective refueling outage. The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report within 60 days following the completion of all Cycle 15 power ascension testing. A supplement shall be submitted within 60 days following the completion of all EPU power ascension testing.

(23) Irradiated GE14i fuel bundles shall be stored at least four feet from the wall of the Spent Fuel Pool.

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DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material and water rods. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

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

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. The absorber material has a nominal absorber length of 143 inches.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

   1. 1250 psig on the suction side of the recirculation pump.
   2. 1500 psig from the recirculation pump discharge to the jet pumps.

c. For a temperature of 575°F.

5.4 REACTOR COOLANT SYSTEM (continued)

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 21,970 cubic feet at a nominal steam dome saturation temperature of 547°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.

b. A nominal 6.308 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 199' 4".

CAPACITY

5.6.3 The spent fuel storage pool shall be limited to a storage capacity of no more than 4006 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 184 TO FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

Proprietary information pursuant to
Title 10 of the *Code of Federal Regulations* (10 CFR), Section 2.390
has been redacted from this document.
Redacted information is identified by blank space enclosed within double brackets
as shown here [[ ]].

Enclosure 3

1.0 INTRODUCTION

By letter dated December 21, 2009, as supplemented by letters dated May 11, June 10, June 24, June 29, July 28, August 3, August 12, September 10, and September 17, 2010 (References 1 through 10), PSEG Nuclear LLC (PSEG or the licensee) requested changes to the Technical Specifications (TSs) and Facility Operating License (FOL) for the Hope Creek Generating Station (HCGS). The proposed amendment would allow the production of Cobalt-60 by irradiating Cobalt-59 targets located in modified fuel assemblies called Isotope Test Assemblies (ITAs). The amendment would allow the licensee to load up to 12 ITAs into the HCGS reactor core beginning with the fall 2010 refueling outage (HCGS Reload 16 Cycle 17). The modified fuel assemblies, also referred to as GE14i ITAs, are planned to be in operation as part of a joint pilot program with Global Nuclear Fuel - Americas, LLC (GNF) and GE - Hitachi Nuclear Energy Americas, LLC (GEH). The purpose of the pilot program is to obtain data to verify that the modified fuel assemblies perform satisfactorily in service prior to use on a production basis. The Cobalt-60 is ultimately intended for use in the medical industry for use in cancer treatments, and blood and instrument sterilization; in the radiography and security industries for imaging; and in the food industry for cold pasteurization and irradiation sterilization.

The supplements dated May 11, June 10, June 24, June 29, July 28, August 3, August 12, September 10, and September 17, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 2, 2010 (75 FR 9445).

2.0 REGULATORY EVALUATION

2.1 Background

As described in TS 5.3.1, the HCGS reactor core consists of 764 fuel assemblies. Currently, all of the fuel assemblies are of the GE14 design. As discussed in Section 2 and Table 2-1 of Attachment 4 to PSEG's application dated December 21, 2009 (GEH Report NEDO-33529), each GE14 fuel assembly consists of 92 fuel rods and two large central water rods contained in a 10 x 10 array. The two water rods encompass eight fuel rod positions.

The proposed amendment would allow the licensee to load up to 12 GE14i ITAs into the HCGS reactor core instead of GE14 fuel assemblies. The GE14i ITAs are identical in design to the GE14 fuel assemblies except that each ITA contains a small number [[ ]] of cobalt isotope rods instead of fuel rods (i.e., less than 92 fuel rods in each ITA). Each cobalt isotope rod is segmented. Initially, the cobalt isotope segments (which contain the cobalt targets) consist of Cobalt-59 (Co-59), a naturally occurring stable isotope. After the ITAs are loaded in the reactor core, the Cobalt-59 targets will absorb neutrons during plant operation and will transition into Cobalt-60 (Co-60) targets.

Post-irradiation handling of the ITAs and cobalt isotope rods is discussed in detail in Section 4.7.3 of Attachment 4 to PSEG's application dated December 21, 2009. In general terms, the handling process includes the following:

- After receiving the desired specific activity, the GE 14i ITAs will be removed from the reactor along with other used GE14 fuel assemblies during a refueling outage and will be placed in the spent fuel pool (SFP).
- Following the refueling outage, the cobalt isotope rods are removed intact from the ITAs using the fuel preparation machine in the SFP. The cobalt isotope rods are replaced with mechanically equivalent stainless steel rods to maintain integrity of the stored assembly.
- The cobalt isotope rods are disassembled into rod segments.
- The Cobalt-60 segments are loaded into an NRC-approved shipping cask.
- The cask will be shipped from HCGS to the GEH Vallecitos Nuclear Center (VNC) facility in Sunol, California for examination and subsequent processing for commercial use of the Cobalt-60.

As discussed in Attachment 1 to PSEG's application dated December 21, 2009, once the 12 ITAs are loaded into the HCGS reactor core, they are planned to remain in the core for three to four 18-month operating cycles, depending on achieving the desired specific activity. However, PSEG intends to remove one ITA after one cycle in the core and a single isotope rod will be removed and shipped to VNC for inspection. This rod will be replaced with a new cobalt isotope rod and the ITA will be returned to the reactor.

2.2 <u>Proposed License and TS Changes</u>

The proposed amendment would modify the HCGS FOL and TSs as discussed below.

<u>New License Condition 1.J</u>

The licensee proposed to add new License Condition 1.J to allow for the production and transfer of Cobalt-60 in accordance with the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 30, "Rules of general applicability to domestic licensing of byproduct material." As shown in Attachment 2 to the application dated December 21, 2009, the new License Condition would read as follows:

> The receipt, production, possession, transfer, and use of Cobalt-60 as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30.

<u>Change to License Condition 2.B.(6)</u>

Currently, License Condition 2.B.(6) reads as follows:

> PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

PSEG proposed a change to the above license condition to provide clarification of the term "separation" relative to the removal and disassembly of the cobalt isotope rods (discussed above in Safety Evaluation (SE) Section 2.1). As shown in Attachment 2 to the application dated December 21, 2009, the following sentence would be added to License Condition 2.B.(6):

> Mechanical disassembly of the GE14i isotope test assemblies containing Cobalt-60 is not considered separation.

New License Condition 2.B.(7)

The licensee proposed to add new License Condition 2.B.(7) to allow intentional production of Cobalt-60 during operation of HCGS. As shown in Attachment 2 to the application dated December 21, 2009, the new License Condition would read as follows:

> PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 30, to intentionally produce, possess, receive, transfer, and use Cobalt-60.

New License Condition 2.C.(23)

The licensee proposed a new license condition to address concerns regarding the potential effect on SFP wall integrity due to gamma heating effects from the GE14i ITAs. As shown in Attachment 4 to the supplement dated June 10, 2010, new License Condition 2.C.(23) would read as follows:

> Irradiated GE14i fuel bundles shall be stored at least four feet from the wall of the Spent Fuel Pool.

Changes to TS 5.3.1, "Fuel Assemblies"

Currently, TS 5.3.1 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 [boiling water reactors].

The licensee proposed to revise TS 5.3.1 to more specifically define the types of fuel assemblies that would be authorized for use in the HCGS reactor core. As shown in Attachment 4 to the supplement dated September 10, 2010, TS 5.3.1 would read as follows:

> The reactor core shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material and water rods. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

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

2.3 Regulatory Requirements and Guidance Documents

The regulatory requirements and guidance documents the NRC staff considered in its review of the proposed amendment included the following:

- 10 CFR 20.1101, "Radiation protection programs," which requires, in part, that the licensee use, to the extent practicable, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are "as low as is reasonably achievable" (ALARA).

- 10 CFR 20.1201, "Occupational dose limits for adults," which, in part, limits the annual occupational dose to 5 roentgen equivalent man (rem) total effective dose equivalent (TEDE).

- 10 CFR 20.1301, "Dose limits for individual members of the public," which, in part, limits the annual dose to a member of the public to 0.1 rem TEDE.

- 10 CFR 50.36, "Technical specifications," which requires that the TSs include items in five specific categories. As required by 10 CFR 50.36(c)(4), "Design features," are one of the categories to be included. Specifically, design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have significant effect on safety and are not covered by the categories described in paragraphs (c)(1), (c)(2), and (c)(3) of 10 CFR 50.36.

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," which establishes standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance.

- 10 CFR 50.67, "Accident source term," which sets limits for the radiological consequences of a postulated design-basis accident.

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 10, "Reactor design," which requires, in part, that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- 10 CFR Part 50, Appendix A, Criterion 11, "Reactor inherent protection," which requires, in part, that the reactor core be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

- 10 CFR Part 50, Appendix A, Criterion 12, "Suppression of reactor power oscillations," which requires, in part, that the reactor core be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

- 10 CFR Part 50, Appendix A, Criterion 19, "Control room," which requires, in part, that adequate radiation protection be provided to permit access and occupancy of the control room (CR) under accident conditions.

- 10 CFR Part 50, Appendix A, Criterion 60, "Control of releases of radioactive materials to the environment," which requires, in part, means to suitably control the release of radioactive materials in gaseous and liquid effluents.

- 10 CFR Part 50, Appendix I, "Numerical Guides For Design Objectives and Limiting Conditions for Operation To Meet The Criterion "As Low As Is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," which establishes numerical guides to meet the ALARA criterion for radioactive effluents.

- 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," which provides requirements for siting a power reactor and includes offsite dose limits associated with the exclusion area boundary (EAB) and low population zone (LPZ).

- Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," which provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports" (hereinafter referred to as SRP), Section 4.2, "Fuel System Design," which, in part, provides guidance to NRC staff in performing reviews associated with the requirements in General Design Criterion (GDC) 10 and 10 CFR 50.46.

- SRP Section 6.4, "Control Room Habitability System," which, in part, provides guidance to NRC staff in performing reviews associated with the requirements in 10 CFR 50.67 and GDC 19.

- SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," which, in part, provides guidance to NRC staff in performing reviews associated with the requirements in 10 CFR 50.67.

- NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," which provides estimates of the source term release into containment, in terms of timing, nuclide types, quantities, and chemical form, given a severe core-melt accident.

3.0 TECHNICAL EVALUATION

3.1 Introduction

PSEG is collaborating with GNF and GEH to develop and implement a pilot program for producing Co-60 in the HCGS reactor during power generation. PSEG plans to load 12 ITAs with Co-59 into the HCGS core as part of the HCGS Reload 16 Cycle 17 core reload during the fall 2010 refueling outage.

Each isotope rod in the GE14i ITA contains Co-59 targets. The GE14i ITAs will be placed in the reactor core where they will stay for varying amounts of time, depending upon the thermal neutron flux and the desired specific activity. The term "varying amounts of time" refers to the operating time of the fuel rods. The ITAs are limited by the accumulated time period in the reactor core spent at operating temperature and by the peak pellet exposure. The peak pellet exposure limit of GE14i ITAs is identical to the GE14 limit. This will enable the GE14 and GE14i bundles to remain in the core for up to five 18-month cycles, as long as they do not exceed the exposure limit and depending on subsequent core designs. For the HCGS core, the bundles are planned to remain in the core for three to four 18-month cycles depending on the subsequent core designs.

The term "desired specific activity" refers to specific activities that are sought in the radioactive cobalt industry which considers high specific activity (HSA) as cobalt greater than 200 Curies (Ci)/gram and low specific activity (LSA) as cobalt activity up to 200 Ci/gram. The PSEG/GEH program intends to produce HSA Cobalt-60 at HCGS.

While in the reactor core, a Co-59 nucleus absorbs a neutron and is converted into a Co-60 nucleus. The resulting irradiated isotope rods will contain Co-60 which emits two gammas with energy of 1.33 million electron-volts (MeV) and 1.17 MeV that are ideal for medical applications. After the duration of irradiation in the reactor, these rods will be transported to GEH off-site facilities for separation and processing.

Reference 22 contains details regarding the design features of the GE14i ITAs, HCGS nuclear core design details that include thermal-hydraulic, safety limit, transient and stability methodologies, and licensing evaluations that include abnormal operational transients, anticipated transient without scram (ATWS), design-basis accidents (DBAs), thermal-mechanical evaluations, as well as other evaluations that support loading of the ITAs into the HCGS core.

The NRC staff reviewed the information provided by the licensee to ensure that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

3.2 GE14i Fuel Product Description

The GE14i bundle is identical to GE14 bundle except the GE14i consists of [[ ] fuel rods, [[ ]] cobalt isotope rods, and two large central water rods, that encompass eight fuel rod positions, arranged in a 10x10 array. The cobalt isotope rods are positioned in the bundle such that [[ ]]

The GE14i bundle is designed for mechanical, nuclear, and thermal-hydraulic compatibility with the co-resident GE14 design. Highlights of the GE14i design aspects are listed below:

- GE14i is a modification of GE14 that replaces a few power rods with segmented isotope rods.
- [[ ]]
- Fuel pellets are identical in the GE14 and GE14i bundles with the exception that GE14i contain segmented isotope rods instead of fuel pellets.
- GE14i water rods are identical to those of GE14 bundles.
- GE14i spacers are identical to GE14 spacers.
- GE14i tie plates are identical to those of GE14 assemblies and they use Defender plates (See below)
- GE14i channel is identical to that of a GE14 assembly.
- GE14i limits on burnup and power suppression with Gadolinium (Gd) are identical to those of GE14 assembly.

Other new features of the GE14i bundle are briefly described below.

Cobalt Target

[[ ]] Table 3.1 below lists the contents of the cobalt targets and the nickel plating on the cobalt targets from the material Certificate of Compliance.

Table 3.1 - Cobalt Target and Nickel Plating Material Content

Material	Cobalt Target Material Content (%)	Nickel Plating Material Content (%)
[[		
		]]

Target Placement Rod (TPR)

[[

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Inner Tube

[[

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Outer Tube

[[

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[[

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Inner Tube Caps

[[

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Male and Female Threaded Connectors

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[[

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Lower and Upper End Plug Extensions

[[

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[[

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Defender Lower Tie Plate

The GE14i bundle incorporates the Defender lower tie plate. The Defender lower tie plate maintains the same resistance to foreign material debris as the GE14 fuel assemblies.

3.3 Cobalt Isotope Rod Failure Mechanism Controls

The GE14i bundle and isotope rod design described in Section 3.2 provides multiple features to prevent cobalt isotope rod failures. The main features that provide multiple levels of safety are:

- Two layers of encapsulation before exposure of nickel-coated cobalt targets;
- Solid Zircaloy connections at all spacer locations; and
- Significantly lower heat generation rate in isotope rods compared to fuel rods.

SE Sections 3.3.1 through 3.3.3 describe the probable cobalt isotope rod failure mechanisms and controls in place to mitigate the rod failure, and/or the consequences of the failure during loading, operation, offloading and disassembly of the ITAs.

3.3.1 Fuel Handling Accident

The radiological consequence of a fuel handling accident (FHA) is evaluated in SE Section 3.5.3. The double containment design of the cobalt isotope rod provides protection against any release compared to normal fuel rods. The absence of gaseous fission products in the isotope rods ensures radiological consequences from any release are bounded by those of a standard fuel assembly. Also, the licensee's administrative controls put in place to protect against fuel handling errors of normal fuel assemblies are similarly applied to GE14i ITAs.

3.3.2 Manufacturing Defects and Other Fuel Failure Mechanisms

Section 4.6 of Reference 22 describes the fabrication processes and materials for the GE14i ITAs. The manufacturing process is handled and controlled under the same quality controls as fuel rods and other fuel bundle components throughout the fabrication process to protect against manufacturing defects or assembly damage. Details of the manufacturing quality assurance for the GE14i manufacturing process is discussed in SE Section 3.5.6.

3.3.3 Other Rod Failure Controls

Pellet-Cladding Interaction

Since there are no fuel pellets in the cobalt isotope rods, there is no pellet-cladding interaction for the cobalt rods.

Corrosion

The lack of fuel in the isotope rod results in lower clad temperature and therefore significantly lower corrosion rates. The double containment design of the isotope rods prevents content release in comparison to normal fuel rods. Both GNF manufacturing controls and PSEG reactor water chemistry controls which protect against fuel rod cladding corrosion are similarly applied to the ITAs.

Primary Hydriding

Since the isotope production rods are subjected to the same control standards as fuel rods, there are no additional sources of hydrogen in the cobalt isotope rods during reactor operation.

Cladding Creep Collapse

A creep collapse analysis was performed as part of the thermal-mechanical evaluation of the ITAs (See SE Section 3.5.4) and it was determined that the [[

]]

Rod Bow

Rod bow results in clad deformation that is due to creep of the Zircaloy material when subjected to operational temperatures, fluences, and axial loading from the rod weight. The isotope rods use standard fuel rod tubing material for the outer structure. The creep of the Zircaloy cladding in the segmented rods is bound by the creep in the standard fuel rods due to the following reasons:

- [[ ]]
- The fluence at the outer tubing is no higher than that for a standard fuel rod.
- [[ ]]

Unthreading of Segmented Rods

[[ ]]

There are no counteracting torsional loads on the rod segments to encourage unthreading.

Stress

The stress in the cladding during operation of the cobalt isotope rods was evaluated. The layers of protection during the operation of the cobalt isotope rods are listed below:

- [[ ]]
- Since the spacers interface with the solid Zircaloy connectors, and not with the cladding, [[ ]]
- [[ ]]
- Nominal cladding temperature is assumed to be the same as that of coolant. [[ ]] The only power generated in the cobalt isotope rod is from gamma heating.

Thus GE14i cobalt isotope rods are adequately protected with respect to cladding stress.

Seismic and Flow-Induced Vibration (FIV)

[[

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GNF has successfully used segmented fuel rods and lead use assembly programs with no evidence of cracking caused by vibration.

Internal Fretting from Inner Capsule

[[

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Spacer Location Fretting

[[

]]

Mid-Span Fretting

The Defender debris filter mitigates this potential failure mechanism of the cobalt isotope rods. If the outer cladding tube is perforated, the inner cladding provides a barrier preventing release of cobalt targets to the reactor coolant. If both the inner and outer cladding tubes are breached, the two breach points would need to be aligned and of sufficient size to allow the nickel-plated cobalt targets to escape. If cobalt targets reach the coolant, the nickel coating will prevent release of cobalt from the targets to the reactor coolant.

Failures during Disassembly

The licensee assures that numerous administrative and procedural controls are in place to prevent failures during disassembly of the cobalt isotope rods. If isotope rod segments cannot be readily separated by unthreading the male-female connections as intended, they may be separated by a torque-induced failure of a necked point on the threaded extension of the male threaded connector. GEH's testing has shown that this failure consistently occurs at the intended location, preventing inner capsule and cobalt targets from escaping and locking the male extension into the female connector's threading.

GEH's testing and the GE14i double encapsulation design of cobalt targets demonstrate that it is less likely for the cobalt targets to reach the fuel pool cleanup system than pieces of fuel pellets. If, in the event that a cobalt target was released in the spent fuel pool, the cobalt target would fall to the bottom of the pool because of its higher density than water. The HCGS spent fuel pool (SFP) cooling system takes suction well above the top of the spent fuel bundles. Therefore, it is highly unlikely that cobalt targets will be swept from the bottom of the SFP into the cooling system.

Online Failure Detection

Online monitoring methods to indicate whether cobalt isotope rod integrity has been compromised are listed below:

- The HCGS plant chemistry sampling program provides detection capability to measure significant increases in Co-60 activity and takes appropriate actions; which may include plant shutdown.
- The reactor water sampling procedures for HCGS include periodic sampling for Co-60 activity.
- If an entire cobalt target becomes exposed, where plant radiation monitors provide detection capabilities, necessary warning and appropriate action can be taken which may include plant shutdown.
- If the target were to become lodged to a location remote to the plant radiation monitors, significant increase in radioactivity would be detected while performing radiological surveys during operation or shutdown.

The NRC staff has concluded that the double encapsulation of the cobalt targets, nickel plating of the cobalt targets, the lower heat generation in the isotope rods than in the fuel rods, and the radiation detection capabilities at the HCGS, have provided multiple levels of safety for the cobalt isotope rods.

3.4 Nuclear Design and Methods

3.4.1 Nuclear Core Design

HCGS is proposing to insert 12 GE14i bundles into its core for the ITA program. The objective of the program is to review and confirm the ITA performance and provide confidence in overall design prior to inserting large numbers of GE14i fuel assemblies for Co-60 production. The HCGS cycle 17 core is designed such that the ITAs are placed in non-limiting locations with respect to thermal limit and shutdown margins.

The definition of a non-limiting location for thermal margins is a bundle location that does not result in the highest core Maximum Fraction of Limiting Critical Power Ratio (MFLCPR), Maximum Fraction of Linear Power Density, and Maximum Average (nodal) Power Ratio. MFLCPR is the ratio of operating limit minimum critical power ratio (OLMCPR) to limiting assembly maximum critical power ratio (MCPR). The operating limit MCPR for the ITA bundles has an added additional margin of [[ ]] compared to the GE14 bundles such that the bundles are monitored to the same MFLCPR margin value (see SE Section 3.4.4 for further discussion).

Selection of non-limiting locations based on reactivity margins is completed by considering any four-bundle cell containing a single ITA and ensuring that the cell does not result in the minimum core shutdown margin (SDM) value at any exposure state point throughout the cycle. The ITA cell includes an additional [[ ]] SDM with respect to other limiting SDM cells at the same cycle exposure state point.

During the cycle of introduction, and for subsequent cycles, three-dimensional analyses will be performed. These analyses are used to determine the non-limiting locations in the core. The licensee has stated that as a result of detailed core design analyses, the ITAs shall remain in non-limiting locations during the subsequent cycles as was done for the cycle of introduction.

[[

]] Power suppression options that are used to address these concerns are described in SE Section 3.4.2.3.

The licensee stated in Attachment 7 of Reference 1 that three-dimensional analyses will be performed for subsequent cycles to ensure that the ITAs will remain in non-limiting locations as was done for the cycle of introduction.

3.4.2 Nuclear Core Design Methods

This section describes the applicability of the current methods and methodologies to the GE14i design. It addresses each of the NRC-approved methodologies (References 23, 24, and 25) that are used in the analyses, and provides the qualification of methods in support of GE14i geometry and characteristics. The codes used for the methodologies have been approved for use by the NRC. Many of the methods are unaffected by either of the unique characteristics, namely, the impact of the non-power producing cobalt isotope rods and the impacts of the connector sections of the isotope rods. The remaining methods require explanation as to how they are qualified for this application.

Table 3.2 lists the summary of the status of the applicability of codes and methodologies to the GE14i bundle. Many of the listed methods are unaffected by the impacts of the non-power producing cobalt isotope rods and the impacts of the connector regions of the cobalt isotope rods. Those methods which are affected by these characteristics require explanations that are included in the relevant sections of this SE.

Table 3.2 - Summary of GNF Methods Applicability to GE14i Bundle

	Analysis Code	Version	Supported	Reference (see SE Section 8.0)
Nuclear	TGBLA	06	X	24, 37
	PANAC	11	X	
Thermal-Hydraulic	ISCOR[1]	09	X	See [1]
Safety Limit MCPR	GESAM	02	X	26, 38, 39
Transient Analysis	ODYNM	10	X	40, 41, 42
	TASC	03	X	45
Stability	PANAC	11	X	24, 37
	ODYSY	05	X	43
	TRACG	04	X	44
ATWS	TASC	03	X	45
	ODYNM	10	X	46
Thermal-Mechanical	GSTRM	07	X	47
Emergency Core Cooling System (ECCS)/Loss-of Coolant Accident (LOCA)	LAMB	08	X	48
	TASC	03	X	45
	SAFER	04	X	49

[1] The ISCOR code is not approved by name. However, the SE supporting approval of NEDE-24011-P Revision 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.

3.4.2.1 Lattice Physics

Lattice physics calculations are performed using a two-dimensional, fine mesh, few group, transport corrected diffusion theory model, TGBLA. No modifications to the methodology of TGBLA were required to model the GE14i UO_2 and gadolinium (Gd) rods. The material specifications of the cobalt bearing regions are provided through the standard TGBLA input parameters. TGBLA qualification was performed by comparison of Co-60 inventory as a function of lattice exposure and in-channel void history with results from Monte Carlo code, MCNP (Reference 27) with ENDFB-VII cross sections.

TGBLA generated infinite lattice reactivity, pin fission density distributions, pin power distributions, gamma source distributions and nuclear instrumentation responses are all used in the downstream applications in PANAC, a three-dimensional core simulator code. The fission density uncertainty comparison for various in-channel void fractions for GE14i and GE14 is listed in Table 3-2 of Reference 22. These representative uncertainties are consistent with the methodology described in Reference 26. Control blade worth from MCNP and TGBLA for various in-channel void fractions are listed in Table 3-3 of Reference 22. The current safety limit analysis uncertainties for GE14 are used to model GE14i in this application.

3.4.2.2 Steady-State Core Simulator

PANAC is the three-dimensional core physics code used for design, licensing, and core monitoring of the boiling-water reactor (BWR) cores. PANAC correctly handles varying axial geometry in nuclear and thermal-hydraulic modeling through the use of its lattice-dependent geometry, nodal thermal-hydraulic properties, and axial-meshing routines. This flexibility allows PANAC to handle multiple partial-length rods (PLRs), varying rod diameter and other axially varying features. The unique features of GE14i and their impact on PANAC are listed below.

Zero-Power Rods

[[

]] Therefore, cobalt isotope rods can be treated as non-power generating (zero-power) rods. These zero-power rods impact the calculation of the heated perimeter, average fuel rod temperature, average planar linear heat generation rate (APLHGR) and the fuel pin LHGR for the isotope bearing bundles. For purposes of critical power, average fuel rod temperature, average planar power, and peak UO_2 rod power, all gamma energy is assumed to be deposited in the fuel rods. The exact number of heated rods, zero-power rods, and total rods is provided to PANAC as input quantities. The input parameters, which are significant to the processing of thermal limits in PANAC, are listed in Table 3-4 of Reference 22. The NRC staff has determined that no changes to PANAC are required to model the thermal performance of the GE14i design.

Nodal Quantities

The impacts on nodal reactivity, nodal pin power distributions, and nodal instrument response functions are explicitly provided by lattice physics evaluations with TGBLA.

Pin Power Reconstruction

The influence of zero-power rods on the PANAC pin power reconstruction model was evaluated by GEH and no statistically meaningful differences were observed. After reviewing the impact of pin power reconstruction in the GE14i and adjacent fuel assemblies, GEH determined that the pin power reconstruction was adequate.

The NRC staff has determined that for the nuclear design of the GE14i fuel assembly no changes to PANAC are required.

3.4.2.3 ITA Margin Considerations

Neutron absorption cross-section of the connector section is lower than that of the cobalt bearing section of the ITAs. Therefore, additional margins need to be applied to the LHGR limit and the cell SDM limit. The connector/spacer zones are not modeled directly in the three-dimensional core simulator program, PANAC. Instead, based on the two- and three-dimensional modeling of the connector/spacer zones performed as part of the design studies, appropriate assumptions to accommodate cobalt isotope rod geometric modeling were determined. The two-dimensional models were evaluated with TGBLA and the three-dimensional models were evaluated with PANAC and MCNP.

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Potential reduction in the SDM in the control rod cell that contains the GE14i bundle is expected due to the increase in reactivity of the lower neutron cross-section connector zone relative to the cobalt isotope bearing zone. An additional margin of [[ ]] Δk SDM in the control rod cell containing the GE14i bundle will provide necessary margin to accommodate this geometric modeling assumption. This additional margin is determined by explicitly modeling all axial zones in the GE14i bundle with PANAC and evaluating the change in control rod worth of control blades adjacent to the GE14i bundles. The NRC staff has determined that this methodology is acceptable.

3.4.2.4 Applicability of GE Methods to Expanded Operating Domains Licensing Topical Report

To support previous applications for extended power uprates (EPUs), GEH evaluated the impact of BWR operation at higher void conditions characteristic of an EPU and maximum extended load line limit analysis plus (MELLLA+) operation on most of its licensing analytical methods as described in GE licensing topical report (LTR) NEDC-33173P. The NRC staff has previously reviewed the LTR to evaluate the applicability of GE's analytical methods to operation at EPU and MELLLA+. The NRC staff concluded that the LTR is acceptable with several limitations and conditions as discussed in Reference 50. Since the LTR was applicable to GE14 bundles, GEH has evaluated the limitations and conditions for applicability to the GE14i bundles to support the proposed HCGS amendment. Appendix A to Reference 22 summarizes GEH's evaluation of each of the limitations and conditions for use of the GE14i bundles at HCGS. The NRC staff has reviewed the disposition of each of the limitations in conditions as described in Appendix A to Reference 22 and agrees with GEH's conclusions regarding whether each of the limitations and conditions applies to the proposed amendment. For those limitations and conditions that are applicable, the NRC staff also agrees with GEH's evaluation of how the limitations and conditions are met. As such, the NRC staff concludes that the licensee has adequately addressed the limitations and conditions in the LTR for the GE14i ITAs.

3.4.3 Methodologies

This section briefly describes the various methodologies used by the licensee for the analysis of GE14i bundles in the HCGS core.

Thermal-Hydraulic Methodology

ISCOR is a thermal-hydraulic core analysis program where different fuel types can be designated to represent various types of bundles in a core. The introduction of various PLR rod heights or zero-power rods such as GE14i can be readily handled by ISCOR since parameters can be varied axially to account for changes in the number of rods, water rod diameters, etc., in the lattice at different axial locations.

PANAC uses the "New Dix" void-quality correlation in its thermal-hydraulic treatment and accounts for bundle leakage and water rod flow by parameterized input from ISCOR simulations. The NRC staff has determined that PANAC is capable of modeling the GE14i bundle design for the HCGS core.

In-Core Instrumentation

The licensee has analyzed the effect of gamma radiation from GE14i assemblies on vessel internal components and determined that the GE14i ITAs will not have significant impact on the in-core instrumentation and core monitoring system.

The major source of gamma energy in the operating reactor is from the fission events and the neutron capture events. The local power range monitoring system (LPRM) that detects localized neutron flux in the core consists of an in-core assembly with metal instrument tube LPRMs, their associated cable and a dry traversing in-core probe (TIP). The TIP is used in the calibration of the LPRM detectors and the update of the core monitoring system parameters that incorporate LPRM and TIP data into the thermal limit calculations. TIP instrumentation correlation constants (J-factors), gamma sources from each material, and signal contribution (attenuation) factors are used to determine the gamma energy deposited in the gamma TIP detector.

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. As discussed in the licensee's submittal dated May 11, 2010 (Reference 2), 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. The evaluation showed that the total gamma energy deposition in the gamma TIP detector from cobalt material in four surrounding GE14i lattices is approximately [[ ]] or less. Therefore, by including the cobalt gamma energy release model in the gamma TIP detector signal correlation, the impact of cobalt material on the accuracy of the TIP signal is reduced to a level significantly below [[ ]]

For the LPRMs, the in-core instrumentation signal-to-lattice power relationship is constituted using the thermal detector J-factor that 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. By including the cobalt effects in the lattice physics model, both the gamma and neutron perturbation on instrumentation are captured and the adequacy of the in-core instrumentation is assured.

The replacement of fuel rods with cobalt isotope rods results in approximately a factor of 10 reduction in the gamma energy emitted from that rod location. Gamma energy from the fission material bearing rod is generated from prompt gammas, delayed fission gammas, and neutron capture gammas from actinides and fission products. Cobalt-60 decay will become significant contributor of gamma energy only after reactor shutdown. Therefore, the effects on the instruments or vessel internals will be bounded by the radiation effects from a fuel rod at that location.

The NRC staff has determined that the GE14i ITAs will not have any significant impact on the in-core instrumentation and core monitoring system in the HCGS core.

Safety Limit Methodology

GESAM incorporates the implementation of the revised Safety Limit Minimum Critical Power Ratio (SLMCPR) methodology using PANAC physics models to calculate CPR distribution. The GE14 uncertainties are used in the evaluation of the safety limit analysis for the HCGS. The NRC staff has determined that the capability of GESAM to model CPR related uncertainties is adequate for the HCGS GE14i bundles and is not impacted by the number of heated rods in the GE14i bundles.

Transient Analysis Methodology

The GE14i design characteristics will be used in the transient analysis methods of ODYNM and TASC. The ODYNM code consists of a one-dimensional representation of the reactor core, and the recirculation and control system model. These two models are coupled to each other. A steady-state initialization is made initially, and then the parameters for the transient are calculated. The recirculation and control systems are solved for the steady-state conditions. The steady-state initialization in the recirculation and control model provides the loop pressure drop, core exit pressure, core inlet flow, and enthalpy to the one-dimensional reactor core model. These values are used in the reactor core model to calculate the neutron kinetics, thermal-hydraulics, and fuel parameters for the steady-state conditions. The steady-state thermal-hydraulic solution permits the calculation of the steady-state fuel temperature distribution. During the transient, the recirculation and control system model calculates the time derivatives. The reactor core model calculates the new neutron flux, thermal-hydraulic parameters such as reactor core exit quality, flow and pressure as input to the recirculation and control system model, and fuel temperatures. The recirculation and control system model calculates the loop pressure drop and the reactor core model calculates the core pressure drop. These pressure drops are compared and updated iteratively.

The TASC code analyzes one-dimensional single-channel hydraulic and heat transfer transients and calculates the local hydraulic conditions for a single channel and the rod temperatures for each rod group in that channel. It is used to calculate the change in CPR during transient conditions. TASC is capable of analyzing a single fuel bundle and is applicable for all fuel types. TASC predicts the thermal-hydraulic response to the BWR transient (i.e., the transient CPR, the bundle fuel rod temperatures, the void fractions, the bundle pressure drop, and the initial onset of loss of nucleate boiling). TASC receives transient information generated by the system codes such as ODYNM.

The NRC staff has reviewed the licensee's analysis and determined that the analysis provided shows that the impact of the [[ ]] zero-power rods in each of the 12 HCGS GE14i bundles will not affect the capability of the transient analysis methods.

Stability Methodology

The applicant employs ODYSY to predict hydrodynamic stability for both a single channel and a full reactor core. ODYSY is a best-estimate, Engineering Computer Program (ECP) which incorporates a linearized, small perturbation, frequency domain model of the reactor core and associated coolant circulation system. It will predict both core-wide mode coupled thermal-hydraulic and reactor kinetic instabilities, and single channel thermal-hydraulic instabilities.

ODYSY is capable of modeling axially-varying bundle designs. ODYSY obtains the GE14i geometry from the ISCOR system and provides adequate results for the GE14i bundle design. ODYSY receives neutronic information through the PANAC wrap-up information.

The NRC staff has reviewed the provided analyses and determined that they show that the [[ ]] zero-power rods in the HCGS GE14i bundles have no impact on the adequacy of the stability methods.

Fuel Rod Thermal-Mechanical Methodology

The PRIME model and computer program has been developed to provide best-estimate predictions of the thermal and mechanical performance of (U, GdO_2) light-water reactor (LWR) nuclear fuel rods experiencing variable power histories. The PRIME code was developed from the GESTR-Mechanical (GSTRM) code by incorporating new models to address specific high exposure mechanisms identified and quantified since the original development of GSTRM and approval of GSTRM and its associated application methodology by the NRC in 1985.
The NRC staff identified a potential non-conservatism in the GSTRM thermal-mechanical calculations supporting the GE14 fuel design. In Reference 51, GE evaluated the 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 Reference 51, GE confirmed that the evaluated-condition does not constitute a reportable condition per 10 CFR 21. The NRC staff recommended an additional [[ ]] for the GSTRM fuel rod internal pressure analyses to address the [[ ]] model in GSTRM.

In response to an NRC staff request for additional information regarding the applicability of the GSTRM methodology to the GE14i design analyses, the licensee has made the following conclusions as discussed in Reference 2:

- LHGR limits for the full length UO_2 rods (FLRs), partial length UO_2 rods, and Gadolinia bearing rods have been updated to include the additional [[ ]] for the GSTRM rod internal pressure analyses. The GE14i bundles for HCGS are designed with these revised LHGR limits and will be monitored during HCGS operation.
- The NRC staff recommendation of additional rod internal pressure for UO_2 rods is not applicable for the GE14i cobalt isotope rods since the fuel failure for these rods are not due to excessive internal pressure. Due to [[ ]] and no fission gas release from the cobalt isotope rods due to radiation, the rod internal pressure at the end of life (EOL) is significantly below the system pressure and thus the fuel failure due to high rod internal pressure is not a likely failure mechanism for these rods.

The NRC staff has determined that the design of the Uranium and Gadolinia rods in the GE14i bundle is identical to the GE14 design and therefore, has no impact on the GSTRM methodology.

<u>Emergency Core Cooling System (ECCS) Analysis Methodology</u>

The ECCS analysis methodology applicable to HCGS is SAFER/GESTR (Reference 49). The GESTR code predicts fuel rod thermal and mechanical performance for variable operating power histories. This code also considers a set of nested, iterative calculations or loops in which the fuel and cladding temperatures, hot gap size and rod internal pressure are determined in sequence.

The input for GE14i zero-power rods to GESTR is described through SAFER. The GESTR fuel rod characteristic data is based on GE14 fuel rod evaluations. No changes to the GESTR fuel characteristics are required as a result of the use of GE14 UO_2 fuel rod design characteristics.

The gamma energy generated in the cobalt isotope rods is assumed to be deposited in the uranium fuel rods. The total gamma energy generated in the cobalt isotope rods varies from 2% to 3% of the total gamma energy released in the lattice as a function of lattice exposure and void history. The analysis shows that this assumption will provide a small conservatism in the SAFER/GESTR analysis.

The rod-to-rod power distributions and local peaking patterns tested with zero-power rods at Stern Laboratories are presented in Figure 3-6 of Reference 22, where cobalt isotope rod(s) or the highest R-factor rod(s) of each pattern are identified.

The NRC staff concludes that the impact of the [[ ]] zero-power rods in the HCGS GE14i bundles will not affect the adequacy of SAFER/GESTR analysis methodology.

3.4.4 GEXL+ Correlation

The critical quality – boiling length correlation, GEXL+ (GEXL14) was developed to accurately predict the onset of boiling transition (BT) in BWR fuel assemblies during both steady-state and reactor transient conditions. The GEXL14 correlation used in the core design and safety analysis is intended to accurately predict the critical power performance of the fuel assembly and the thermal margin for the operating cycle (Reference 28). In the GEXL correlation, critical quality is expressed as a function of boiling length, thermal diameter, system pressure, lattice geometry, local rod peaking pattern, mass flux, R-factor, and annular flow length. The R-factor is an input to the GEXL+ correlations and it accounts for the effects of the pin power distributions and the geometry of the assembly/channel/spacer on the assembly critical power.

The GE14i ITAs are identical to the GE14 fuel bundles except for the cobalt isotope rods in GE14i. Due to the similarity between GE14i and GE14, the GEXL14 correlation can be applied to the GE14i ITAs. The licensee has demonstrated the applicability of the GEXL14 correlation to the GE14i ITAs by comparing the GEXL14 prediction to the critical power data with zero-power rods in the GE14i bundle.

GEXL critical power correlation (GEXL14) for the GE14 fuel was developed using the ATLAS critical power facility and the Stern Laboratory. The database used in the GEXL14 correlation covers wide ranges of fluid conditions and a number of rod-to-rod power distributions with a [[ ]] and is validated against [[ ]] data generated in the Stern Laboratories as described in Reference 28.

Full scale critical power and pressure drop testing for a simulated GE14 fuel bundle was performed in the Stern Laboratories test facility. As a part of the Stern testing for the GE14 fuel, critical power data was collected with zero-power rods and [[ ]] Four different rod-to-rod power distributions were tested for a wide range of inlet flow and inlet subcooling conditions at a pressure of 1000 psia. The rod-to-rod power distributions or local peaking patterns tested with zero-power rods at Stern Laboratories are presented in Figure 3-6 of Reference 22, where target rod(s), the highest R-factor rod(s), of each pattern were identified with a green background color. The four peaking patterns were designated as patterns J1, J2, J3, and D0xx as shown in Figure 3-6 of Reference 22. Peaking patterns J1/J2/J3 have [[ ]] zero-power rods and pattern D0xx has [[ ]] zero-power rods. Mass flux and inlet subcooling conditions are plotted in Figure 3-7 of Reference 22. Typical bundle axial power shape is illustrated in Figure 3-8 of Reference 22. [[

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The NRC staff has determined that the use of a single axial power shape [[ ]] data to validate the use of GEXL14 to GE14i is justified primarily due to the prior experience in GEXL correlations, which had shown that the critical power data correlated well in the critical quality and boiling length plane independent of the axial power profiles. The axial power shape sensitivity has been well predicted by the GEXL correlation for a wide range of different designs such as lattice design (9x9, 10x10), the PLR configuration, and spacer designs.

As discussed in Reference 22, a statistical analysis was performed for the GE14 database with zero-power rods consisting of [[ ]] data points obtained from the Stern Laboratories test assembly. To facilitate the statistical evaluation of the predictive capability of the GEXL14 correlation, the concept of an experimental critical power ratio (ECPR) is defined and used. The details of the analysis are given in a GEH design document which was audited by the NRC staff.

The ECPR is determined as: ECPR = (Predicted Critical Power)/ (Measured Critical Power)

A summary of ECPR statistics provided in Table 3-5 of Reference 22 is repeated below.

Number of Data Points	[[
Mean ECPR	
Standard Deviation	]]

Figure 3-9 of Reference 22 compares predicted critical powers to the measured critical powers from GEXL14.

[[

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Based on the above, the NRC staff finds that the licensee has demonstrated the applicability of the GEXL14 correlation to the GE14i ITAs by comparing GEXL14 prediction to the critical power data with zero-power rods in the GE14 bundle. The R-factor methodology (Reference 28) was applied in generating the R-factors for the test assembly with zero-power rods as part of the overall evaluation of the applicability of GEXL14 to GE14i. Therefore, the R-factor methodology is confirmed applicable to GE14i. The GEXL correlation, on average, conservatively predicted the critical powers for the zero-power rod test data obtained at the Stern Laboratories for the GE14 bundle with an [[

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The NRC staff has evaluated the applicability of the above analyses methodologies to the HCGS core with GE14i ITAs and determined that it is acceptable.

3.5 Licensing Evaluations

As discussed in Reference 22, cycle-specific limits are established to ensure compliance with licensing limits. Operating limits for the ITAs were established for HCGS Reload 16 Cycle 17 by performing reload analysis using NRC-approved methods. Results of the reload analyses are documented in HCGS Reload 16 Cycle 17 Supplemental Reload Licensing Report (SRLR) (Reference 7). In addition, the licensee is expected to perform core analyses for each cycle of operation subsequent to the initial ITA loading.

The list of events analyzed for HCGS Cycle 17 is shown in Table 3.3 below.

Table 3.3 - List of Analyzed Events for the Reload License with GE14i ITAs in the Core

	Event	Method
A.	**Fuel Thermal Margin Events**	
1.	Generator Load Rejection with Bypass Failure	ODYN
2.	Turbine Trip with Bypass Failure	ODYN
3.	Feedwater Controller Failure-Maximum Demand	ODYN
4.	Loss of Feedwater Heating	3D-simulator
5.	Rod Withdrawal Error at Rated Power	3D-simulator
6.	Mislocated Fuel Assembly	3D-Simulator
7.	Misoriented Fuel Assembly	TGBLA
B.	**Limiting Transient Overpressure Events**	
1.	Main Steam Isolation Valve Closure with Scram on High Flux (Failure of Direct Scram)	ODYN

The reactor operating conditions used in the reload licensing analysis for HCGS Cycle 17 are listed in Table 3.4 below. Table 3.5 lists the pressure relief and safety valve configuration for the plant.

Table 3.4 - Reactor Operating Conditions for Cycle 17 Reload Analysis

Parameter	Analysis Value			
	ICF NFWT	LCF NFWT	ICF RFWT	LCF RFWT
Thermal power, MWth	3840.0	3840.0	3840.0	3840.0
Core flow, Mlb/hr (Million lb/hr)	105.0	94.8	105.0	94.8
Reactor pressure (core mid plane), psia	1036.0	1034.0	1013.4	1011.6
Inlet enthalpy, Btu/lb	526.3	523.8	511.0	507.3
Non-fuel power fraction	0.036	0.036	0.036	0.036
Steam flow, Mlb/hr	16.80	16.78	14.75	14.73
Dome pressure, psig	1005.0	1005.0	983.6	983.6
Turbine pressure, psig	945.8	945.9	937.3	937.4

ICF = Increased Core Flow, LCF = Low Core Flow, NFWT = Normal Feedwater Temperature, RFWT = Reduced Feedwater Temperature

Table 3.5 - Pressure Relief and Safety Valve Configuration

Valve Type	Number of Valves	Lowest Setpoint (psig)
Safety/Relief Valve	14	1141.2

3.5.1 Evaluation of Abnormal Operational Transients

Cycle-specific analyses of the limiting transient events are performed to establish the plant Operating Limit Minimum Critical Power Ratio (OLMCPR) to demonstrate thermal/mechanical compliance and to demonstrate compliance with the ASME overpressure protection criteria.

The HCGS Cycle 17 reload licensing analyses include specific modeling of the GE14i ITAs in the determination of the OLMCPR. As discussed in SE Section 3.4.4, the GEXL14 correlation is conservatively applied to the GE14i ITAs. The GE14i ITA UO_2 and Gadolinia (Gd) fuel rod mechanical designs are identical to the GE14 fuel rods and, therefore the normal GE14 thermal and mechanical overpower LHGR limits ensure compliance with the thermal-mechanical licensing requirements.

The HCGS abnormal operational transients evaluated to support the introduction of GE14i ITAs into the HCGS core are identified in the following subsections. The evaluations have been reviewed by the NRC staff and found acceptable.

3.5.1.1 Decrease in Reactor Coolant Temperature

Reactor vessel water (moderator) temperature reduction results in an increase in core reactivity. This could lead to fuel-cladding damage. The events in this category are:

- Loss of Feedwater Heating (LFWH)
- Feedwater Controller Failure - Maximum Demand (FWCF)
- Pressure Regulator Failure Open (PRFO)

- Inadvertent Main Steam Relief Valve - Open
- Inadvertent RHR Shutdown Cooling Operation

The HCGS Cycle 17 reload licensing analyses includes specific modeling of the GE14i ITAs in determination of the OLMCPR. Plant parameters, which include the steam line volume, steam line pressure losses, Turbine Control Valve (TCV)/Turbine Stop Valve (TSV) closure times, scram time, and the associated trip and delay times, impact the core average response of the limiting events in this category. Such plant parameters are independent of fuel bundle design and are modeled by methods discussed in SE Section 3.4.3. In Cycle 17, as in Cycle 16, the inadvertent RHR Shutdown Cooling Operation, PRFO, and Inadvertent Main Steam Relief Valve Opening events are found to be bounded by the LFWH and FWCF events.

The transient response is affected by the core average reactivity characteristics. However, the licensee determined that the introduction of GE14i ITAs has a negligible impact on the core average nuclear parameters affecting the transient response because the 12 GE14i ITAs represent a small fraction of the total bundles in the core and the hydraulic characteristics of the GE14i ITAs are similar to the GE14 bundles (see SE Section 3.4.3). Therefore, the GE14 bundles dictate the core average nuclear parameters that affect the transient response. Consequently, for Cycle 17, as in Cycle 16, only FWCF and LFWH events are analyzed as part of the cycle-specific licensing analyses. The results of the Cycle 17 analysis of LFWH and FWCF events are listed in Reference 7. The results show that these events meet the regulatory and thermal-hydraulic design requirements for these anticipated operational occurrences (AOOs). Therefore, it can be concluded that the reload licensing scope listed in Table 3.3 bounds all other AOOs in this category. The NRC staff has reviewed the licensee's analyses and determined that they are acceptable.

3.5.1.2 Increase in Reactor Pressure

Increase in nuclear system pressure increases the possibility of rupturing the reactor coolant pressure boundary (RCPB). Increasing the pressure also collapses the voids in the core-moderator, thereby increasing core reactivity. This could lead to fuel cladding damage. The events in this category are:

- Pressure Regulator Failure - Closed (PRFDS)
- Generator Load Rejection with Bypass (LRWBP)
- Generator Load Rejection with Bypass Failure (LRNBP)
- Turbine Trip with Bypass Failure (TTNBP)
- Turbine Trip with Bypass (TTWBP)
- Main Steam Isolation Valve Closures
- Loss of Alternating Current (AC) Power
- Loss of Condenser Vacuum
- Loss of Feedwater Flow (LOFW)
- Failure of RHR Shutdown Cooling

For Cycle 17, as in Cycle 16, the licensee determined that the TTNBP and LRNBP events will continue to bound the PRFDS, LRWBP, TTWBP, Loss of AC Power, Loss of Condenser Vacuum, LOFW and Failure of RHR Shutdown Cooling events due to reasons specified in SE Section 3.5.1.1. The FWCF event includes a system pressure increase due to the turbine trip from reactor high water level. However, the FWCF event is categorized as a reactor coolant temperature decrease event and is discussed in SE Section 3.5.1.1. The Main Steam Isolation Valve Closure with Flux Scram (MSIVF) event is analyzed for overpressure protection and is discussed in Section 3.5.2.2.

The NRC staff has reviewed the submitted information and finds that the GE14i ITAs do not impact the core average response of the limiting events in this category because the core average nuclear characteristics are dictated by GE14 bundles. The TTNBP and LRNBP events are analyzed as part of the cycle-specific reload licensing analyses. The results of the Cycle 17 analysis of these events are listed in Reference 7. The NRC staff finds that the results show that these events meet the licensing and thermal-hydraulic design requirements for these AOOs.

3.5.1.3 Decrease in Reactor Coolant Flow Rate

A reduction in the core coolant flow rate causes the cladding to overheat as the coolant becomes unable to adequately remove the heat generated by the fuel. The events in this category are:

- Reactor Recirculation Pump Trip
- Recirculation Flow Control Failure - Decreasing Flow

The AOOs in this category are bounded by the events in Table 3.3 above. The decrease in core flow causes a decrease in reactor power, and consequently, the thermal limits are not challenged. The core-wide decrease in reactor power instigated by decreasing core flow is a BWR characteristic that remains unchanged with the introduction of the GE14i ITAs. Therefore, none of the above events will be analyzed for Cycle 17 specific reload licensing analyses due to the introduction of GE14i ITAs in HCGS. The NRC staff finds this acceptable.

3.5.1.4 Reactivity and Power Distribution Anomalies

AOO events included in this category are those which cause rapid increases in power. The events in this category are:

- Rod Withdrawal Error (RWE)
- Control Rod Maloperation (System Malfunction or Operator Error)
- Mislocated Fuel Assembly Accident
- Misoriented Fuel Assembly Accident
- Abnormal Startup of Idle Recirculation Loop
- Recirculation Flow Control Failure with Increasing Flow

The RWE and the Misoriented Fuel Assembly Accident are potentially limiting events at HCGS. The RWE bounds the Control Rod Maloperation event. These RWE and Misoriented Fuel

Assembly accident events have the potential to set an OLMCPR very close between the beginning of cycle (BOC) and middle of cycle (MOC) exposure range. The RWE and Misoriented Fuel Assembly Accident are analyzed as part of the cycle-specific reload licensing analyses (Reference 7). The Control Rod Maloperation event is not analyzed for Cycle 17. The HCGS Cycle 17 reload licensing analyses includes specific modeling of GE14i ITAs.

In Cycle 17, the off-rated power and flow limits continue to bound the Abnormal Startup of Idle Recirculation Loop event and the Recirculation Flow Control Failure with Increasing Flow event, which result in a fast recirculation flow run out, due to reasons discussed in SE Section 3.5.1.1. The slow recirculation flow run out event has been previously analyzed to develop the flow-dependent MCPR and LHGR limits. Based on the analysis above, the NRC staff concludes that the off-rated limits for HCGS are validated as part of reload licensing analyses for application to Cycle 17.

3.5.1.5 Increase in Reactor Coolant Inventory and Other Accidents

Increasing coolant inventory could result in excessive moisture carryover to the main turbine, feedwater turbines, etc. The event in this category is:

- Inadvertent High Pressure Coolant Injection (HPCI) Start-up

The severity of the HPCI event is affected by plant parameters such as steam line volume, steam line pressure losses, TCV/TSV closure times, scram time and HPCI system flow capacity. These plant parameters are independent of fuel bundle designs and are modeled by appropriate transient analysis methodology, ODYNM. The ODYNM model is updated on a cycle- specific basis to incorporate the most recent HCGS plant characteristics. Therefore, as in Cycle 16, the Inadvertent HPCI Start-up event will continue to be bounded by the cycle-specific reload licensing events listed in Table 3.3.

3.5.1.6 Decease in Reactor Coolant Inventory and Other Accidents

Reductions in coolant inventory could result in the coolant becoming less able to remove the heat generated in the core. The events in this category are:

- Control Rod Drop Accident (CRDA)
- Main Steam Line Break Accident (MSLB)
- Fuel-Handling Accident (FHA)
- Loss-of-Coolant Accident (LOCA)

All events in this category are limiting events or design-basis accidents (DBAs). HCGS is a banked position withdrawal sequence (BPWS) plant; therefore, the CRDA analysis is not required to be analyzed for each reload, as documented in Reference 35. The MSLB and LOCA events result in decrease in reactor coolant inventory. The LOCA analysis, as a result of the introduction of GE14i ITAs, is discussed in SE Section 3.5.5.11 and the radiological consequences are discussed in SE Section 3.5.3.

3.5.2 Evaluation of Other Transients

3.5.2.1 Anticipated Transients Without Scram (ATWS)

An ATWS is an extremely low probability event. This multi-system maloperation event is postulated in order to determine the capability of plant design. The evaluation of the ATWS events is not a design basis requirement. However, specific requirements for ATWS are provided in 10 CFR 50.62, "Requirements for reduction of risk from ATWS events for light-water-cooled nuclear power plants." Specifically, BWRs are required to have an alternate rod injection system (10 CFR 50.62(c)(3)), a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel 86 gpm equivalent borated water (10 CFR 50.62(c)(4)) and equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. All of these features are available at the HCGS plant. The current licensing basis ATWS analyses demonstrate reactor integrity, containment integrity and fuel integrity. Reactor integrity is demonstrated by ensuring that peak reactor vessel pressure is less than the American Society of Mechanical Engineers (ASME) Service Level C limit. Containment integrity is demonstrated by ensuring that the peak suppression pool temperature is below the maximum allowed bulk suppression pool temperature and containment pressure is less than the containment design pressure limit. Fuel integrity is demonstrated by ensuring that the Peak Cladding Temperature (PCT) and fuel cladding oxidation is below the 10 CFR 50.46, "Acceptance criteria for ECCS for light-water nuclear power plants," limits.

The ATWS response is primarily affected by the key plant characteristics, which include the ATWS - Recirculation Pump Trip (ATWS-RPT) and Safety/Relief Valve and Standby Liquid Control System (SLCS) operating parameters. Since the GE14i ITAs represent a small fraction of the total bundles in the core and since their hydraulic characteristics are similar to the GE14 bundles (see SE Section 3.5.4.2), their impact on the core average nuclear parameters is negligible. Therefore, a cycle-specific ATWS analysis is not required because of the introduction of GE14i ITAs.

The fuel and cycle independent ATWS evaluation for HCGS is documented in Reference 29. The NRC staff has determined that this evaluation demonstrates significant margin to the ATWS acceptance criteria.

3.5.2.2 ASME Overpressure Protection

The acceptance limit for pressurization events is the ASME Code allowable peak pressure of 1375 psig, which is 110% of the design pressure. ASME overpressure protection is demonstrated by the analysis of an assumed closure of all Main Steam Isolation Valves (MSIVs) with no credit for the direct scram signal on MSIV closure (MSIVF). A scram is assumed to occur on high neutron flux in the reactor core. The presence of 12 GE14i ITAs does not impact plant characteristics such as scram delay time or the core average nuclear characteristics.

As in Cycle 16, the MSIVF event continues to bound all other pressurization events due to the reasons specified in SE Section 3.5.1.1 and is analyzed as part of the cycle-specific reload licensing analysis that includes specific modeling of GE14i ITAs.

3.5.2.3 Single Loop Operation (SLO) Pump Seizure Analysis

This SLO Pump Seizure event was analyzed for GE14 introduction into HCGS. [[

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3.5.2.4 Applicability of Off-Rated Limits to GE14i ITAs

The off-rated power/flow limits are constructed to assure that thermal limits are not violated when a transient event (AOO) is initiated while the reactor is operating at an off-rated power/flow condition. The off-rated limits (or multipliers) are confirmed applicable for new fuel designs as outlined in the General Electric Standard Application for Reactor Fuel (GESTAR II) (Reference 23), cycle-independent analyses for a new fuel introduction reload application, or as in the case of HCGS, plant-specific off-rated limits (Reference 7). The main bundle characteristic that influences the transient response and operating thermal limits is the critical power performance of the new fuel. As discussed in SE Section 3.4.4, the GEXL14 correlation is conservatively applied to the GE14i ITAs. In addition, since the GE14i bundles represent a small fraction of the total bundles in the HCGS core and the hydraulic characteristics and fuel mechanical design of the GE14i ITAs are similar to the GE14 bundles, the impact on the core average nuclear parameters that affect the transient response is negligible. Therefore, the power and flow dependent limits are applicable to the GE14i ITAs.

The off-rated power and flow limits are confirmed and scaled to adjust for the cycle-specific SLMCPR as part of the cycle-specific reload licensing analyses for HCGS Cycle 17. The off-rated power dependent limits and off-rated flow dependent limits are listed in the Cycle 17 HCGS reload licensing report (Appendix D of Reference 7).

The NRC staff concludes that the cycle-specific reload analyses using the GESTAR II methods provide reasonable assurance that the off-rated power/flow limits will be properly constructed with respect to consideration of the impact of the GE14i ITAs.

3.5.2.5 Flexibility and Equipment Out-of-Service (EOOS) Options

The thermal-hydraulic characteristics, fuel mechanical design, and critical power performance of the GE14i ITAs are similar to those of GE14 fuel. The impact on the core average parameters that affect the transient response is negligible because the GE14i ITAs represent a small fraction of the total fuel bundles in the core. Therefore, the NRC has determined that the flexibility and EOOS options supported in the Cycle 16 reload analyses remain unchanged and continue to be supported with the introduction of the GE14i ITAs. The cold-water events, fast pressurization and ASME overpressurization events in combination with the licensed flexibility options for HCGS are evaluated as part of the cycle-specific reload licensing analyses (Reference 7).

Section 8 of Reference 7 presents the operational domains and flexibility options and EOOS that are supported by the reload licensing analysis, and are listed below:

- Extended operating domain (EOD): Maximum Extended Load Line Limit (MELLLA) (94.8% core flow at rated power).
- Increased core flow (ICF, 105% at rated power).
- One recirculation pump out-of-service or single loop operation (SLO).
- Recirculation pump trip out-of-service.
- Safety relief valve out-of-service.

3.5.3 Radiological Consequence Analyses

This section provides the NRC staff's evaluation of the DBA analysis results reported in the amendment submittal. The staff reviewed the assumptions, inputs, and methods used by PSEG to assess these impacts. When appropriate, the staff performed independent analyses to confirm the conservatism of the licensee's analyses. However, the conclusions in this SE are based on the descriptions of the licensee's analyses and other supporting information docketed by PSEG.

The licensee considered the impact of GE 14i ITAs operation on the previously analyzed DBAs. The DBAs considered included:

- Control Rod Drop Accident (CRDA) (UFSAR Section 15.4.9)
- Loss-of-Coolant Accident (LOCA) (UFSAR Section 15.6.5)
- Fuel Handling Accident (FHA) (UFSAR Section 15.7.4)
- Main Steam Line Break (MSLB) (UFSAR Section 15.6.4)
- Reactor Recirculation Pump Seizure (UFSAR Section 15.3.3)
- Reactor Recirculation Pump Shaft Break (UFSAR Section 15.3.4)
- Instrument Line Break (UFSAR Section 15.6.2)
- Feedwater Line Break Outside Containment (UFSAR Section 15.6.6)

- Gaseous Radwaste Subsystem Leak or Failure (UFSAR Section 15 .7.1)

3.5.3.1 Control Rod Drop Accident

The CRDA analysis postulates a sequence of mechanical failures that results in the rapid removal (i.e., drop) of a control rod, upon which a reactor trip will occur. Localized damage to fuel cladding is expected to occur, resulting in a breach of the fuel cladding. The temperature of a small fraction of the fuel in the breached rods will be sufficient to cause localized melting.

The HCGS licensing basis CRDA is analyzed in Section 15.4.9 of the HCGS UFSAR. The licensee provided an assessment of the impact of the GE14i ITAs on the CRDA in the December 21, 2009, submittal in NEDC-33529P, Section 4.3.1, "Control Rod Drop Accident." In Attachment 1 of the May 11, 2010, response to the staff's request for additional information (RAI) regarding the assumption that the release fraction for cobalt is 0.0025 of the core inventory, the licensee provided the following update to NEDC-33529P, Section 4.3.1:

> The HCGS licensing basis CRDA analyzed in Reference A1 assumes a failure of 850 rods (8 x 8 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. [[
>
>]]

The licensee's May 11, 2010, response states that no experimental data could be provided to justify the assumed Co-60 release fraction. In response to NRC staff's concern regarding the impact of a higher Co-60 release fraction, the May 11, 2010, response also provided an assessment assuming a Co-60 release fraction of [[ ]] The licensee determined that the calculated maximum increase in offsite dose to be [[

]]

In the July 28, 2010, response to an NRC staff RA), the licensee provided additional justification for the assumption that the equivalent of [[

]] The licensee stated that this assumption is conservative because the CRDA is a localized event, and [[

]] Based upon these arguments and the assessment performed in SE Section 3.5.4.1, the NRC staff agrees that the assumption that [[ ]] is conservative for the CRDA analysis.

In the September 10, 2010, supplement (Reference 9) the licensee stated:

> The changes to the LOCA analysis model have no impact on the CRDA evaluation provided in NEDC-33529P. The evaluation results and conclusions in Section 4.3.1, [are] based on the Calculation H-1-CG-MDC-1795 (Revision 5). "Control Rod Drop Accident Radiological Consequences," remain unchanged. The changes to the LOCA analysis do not affect the CRDA analysis. This includes assumed control room inleakage; the CRDA analysis did not credit CREFS initiation. Revision 5 of the CRDA calculation only corrected typographical errors; Revision 4 was previously docketed to support the HCGS EPU Amendment 174. Therefore the parameters, inputs and assumptions used in Amendment 174 remain valid.

The NRC staff assessed the CRDA using the licensee's previously-accepted methodology and those described above including a higher Co-60 release fraction of [[ ]]. In the NRC staff SE for HCGS Amendment No. 174, "Extended Power Uprate," (Reference 18), the staff stated that the assumptions and methods used for the CRDA are in accordance with RG 1.183. Therefore, the staff only assessed the changes of the release fraction and the additional Cobalt-60 due to the GE14i ITAs. The staff performed a confirmatory calculation using the licensee's assumptions and confirmed the licensee's results.

The NRC staff notes that its acceptance of the assumed release fraction is based on a Co-60 release fraction of [[ ]] for the proposed license amendment, and should not be solely relied upon as the licensing basis to support any other proposed licensing action. This evaluation is only valid for the current licensing basis as modified by the proposed change.

The NRC staff finds that the licensee's CRDA analysis assumptions and methodology, with the exception of the Co-60 release fraction, are consistent with the guidance of RG 1.183 as previously approved in HCGS Amendment No. 174. The licensee's calculations determined a negligible maximum increase in control room and offsite doses [[ ]] which is within the uncertainty of the calculations. The staff compared the doses estimated by licensee to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations. The staff finds, with reasonable assurance, that the licensee's estimates of the total effective dose equivalent due to design-basis accidents will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183.

3.5.3.2 Loss-of-Coolant Accident (LOCA)

The LOCA event considered is a double-ended rupture of the largest pipe in the reactor coolant system (RCS). The objective of this postulated DBA is to evaluate the ability of the plant design to mitigate the release of radionuclides to the environment in the unlikely event that the emergency core cooling system (ECCS) is not effective in preventing core damage. A LOCA is a failure of the RCS that results in the loss of reactor coolant that, if not mitigated, could result in fuel damage including a core melt. Thermodynamic analyses, done using a spectrum of RCS break sizes, show that the ECCS and other plant safety features are effective in preventing significant fuel damage. Nonetheless, the radiological consequence portion of the LOCA analysis conservatively assumes that ECCS is not effective and that substantial fuel damage occurs.

The impact of 12 GE14i ITAs on LOCA radiological consequences was evaluated by HCGS. The accident analyses provided to support the GE 14i ITAs are documented in Calculation H-1-ZZ-MDC-1880, Revision 4, "Post-LOCA EAB, LPZ, and CR Doses," (Reference 8, Attachment 2) and supporting Technical Evaluation 80102291-0040, "Effect on LOCA Radiological Consequences of Increased MSIV Flow Rate Based on Main Steam Line Temperature and a Single Main Steam Piping Compartment Volume" (Reference 9, Attachment 3). Some parameters, inputs, assumptions and results of these current analyses are different from those evaluated in HCGS Amendment No. 174. In Attachment 1, "Evaluation of Changes to Accident Analyses," to Reference 9, the licensee provided the differences between Amendment No. 174 and the proposed analyses. The Attachment identifies changes (parameters, inputs and assumptions) to the analysis model, evaluates the effect of each change, and justifies the use of the resulting accident model.

The LOCA analysis is used to demonstrate the adequacy of the HCGS engineered safety features (ESF) systems to mitigate the radiological consequences of a LOCA. The analysis includes the evaluation of three potential release pathways following a LOCA:

1. Containment leakage,
2. Post-LOCA leakage from ESF systems outside containment, and
3. Main steam isolation valve (MSIV) leakage.

The changes that are made to each of these pathways are discussed in detail below. In addition, the effect of the changes to parameters, inputs and assumptions on the following items is also addressed below:

4. Control Room Model
5. Reactor Core Inventory
6. Effect of ITAs
7. LOCA Summary

Table L1 of this SE summarizes the results of the licensee's radiological consequence calculations. Table L2 of this SE provides parameters and assumptions used in radiological consequences calculations for License Amendment 174 (labeled Amendment 174) and those proposed to support the change to use GE14i ITAs (labeled GE 14i ITAs). The differences in major parameters and assumptions between these calculations are shown by highlighting the values in bold. The licensee used the parameters in the column labeled GE 14i ITAs in its radiological consequence calculations. The staff used the same values in its confirmatory dose calculations. Tables L1 and L2 are located after Section 8.0 of this SE.

1. Containment Leakage Pathway

The radioactive material released from the core enters the drywell atmosphere and is mixed in the drywell and wetwell (suppression chamber) volumes. It subsequently leaks from primary containment to the secondary containment and then leaks to or is exhausted to the environment. All of the assumptions and parameters used to evaluate the amount of activity released through this pathway remain the same as the Amendment No. 174 assumptions and parameters, except as indicated below.

Containment Leak Rate

In Amendment No. 174, the containment leak rate was established as 0.5% per day in accordance with the HCGS TSs. At 24 hours following the accident, the leak rate was reduced by a factor of two to 0.25% per day based on the reduction in pressure in the containment.

The licensee responded to the NRC staff's RAI Question 13, in Attachment 1 to the August 12, 2010, supplement. As stated in this response, the licensee eliminated the assumption that the containment leak rate is reduced at 24 hours. The licensee used a containment leak rate of 0.5% per day at the start of the accident and held the value constant over the course of the accident (30 days). This change is consistent with Appendix A, Regulatory Position 3.7 of RG 1.183.

Mixing in Primary Containment

In Amendment No. 174, it is assumed that the activity released to the drywell atmosphere is instantaneously mixed in the total primary containment volume (the sum of the drywell and wetwell volumes).

In the analysis to justify use of GE14i ITAs, the licensee assumes that initially the activity released to the drywell is uniformly mixed in the volume of the drywell only. After 2 hours, it is assumed the activity is mixed in the total primary containment volume. This change was made based on the recognition that processes that cause mixing between the drywell and wetwell (e.g., blowdown to the suppression pool, operation of vacuum breakers, etc.) do not operate continuously, although eventually there should be fairly good mixing between the two volumes. Assuming that there is no mixing for the first 2 hours conservatively accounts for the time dependent nature of the mixing. This change is more conservative than originally accepted in Amendment No. 174 and is consistent with Appendix A, Regulatory Position 3.1 of RG 1.183.

Fission Product Source Term Behavior for First 24-Hours of LOCA

In Amendment No. 174, the evaluation does not take credit for removal of any activity from the containment atmosphere, except for removal of aerosol activity by natural deposition. No credit is taken for the activity that may be removed by the operation of the containment sprays, even though it is likely they will be operating following a LOCA. The GE 14i ITAs justification also does not credit operation of the containment sprays for activity removal.

The NRC staff review of the radiological consequences for the HCGS EPU review (i.e., Amendment No. 174) was based, in part, on the previous staff review for HCGS Amendment No. 134 (Reference 21) which revised the TSs to permit an increase in the allowable leak rate for the main steam isolation valves (MSIVs) and deleted the MSIV sealing system based on the use of an alternate source term. As discussed in the NRC staff SE for HCGS Amendment No. 134:

> Based on engineering judgment, the staff believes that, for the first 24 hours into the postulated LOCA, the fission product source term behavior, its transport, and release to the environment, will be entirely dominated by thermal hydraulic conditions in the drywell and in the containment (drywell leakage, steam production and condensation, and mixing), and by aerosol removal mechanisms (containment spray and aerosol deposition) independent of suppression pool water pH and iodine reevolution from the suppression pool to the containment atmosphere. Consequently, any postulated radiological consequences at any point on the boundary of the exclusion area for a 24-hour period will not be affected by iodine reevolution and pH control.

The NRC staff finds that the proposed amendment does not impact the staff's previous conclusions (stated above) that the fission product source term behavior, for the first 24 hours into the postulated LOCA, is independent of suppression pool water pH and iodine re-evolution from the suppression pool to the containment atmosphere. Following this 24-hour period, the suppression pool pH will be maintained greater than 7 due to the buffering action created by sodium pentaborate injection from the standby liquid control system. Maintaining pH greater than 7 will ensure that iodine is not re-evolved from the suppression pool.

2. <u>Post-LOCA ESF Systems Leakage Pathway</u>

This leakage pathway involves the circulation of suppression pool water in ESF systems outside the primary containment. The components of the ESF systems are expected to leak, and the amount of leakage is monitored and controlled by a TS-required program. All of the assumptions and parameters used in the analysis for the GE14i ITAs are the same as the assumptions and parameters used in the analysis for Amendment No. 174, except as described below.

ESF Leak Rate

In the September 10, 2010, supplement (Reference 9), the licensee stated:

> Amendment 174 assumes an ESF leak rate of 1 gpm, which is doubled in the ESF systems leakage pathway model. The Co-60 LAR [license amendment request] assumes an ESF leak rate of 2.85 gpm, which is also doubled in the ESF systems leakage pathway model. The change in leak rate was made to provide operational margin. The assumed leak rate of 2.85 gpm has been

> incorporated as acceptance criterion in Hope Creek's leakage reduction program, which is maintained in accordance with Technical Specification 6.8.4.a.

In the September 17, 2010, supplement (Reference 10) the licensee stated:

> The HCGS Leakage Reduction Program, ER-HC-1051, does adjust operational ESF leakage to account for accident conditions. If a leak is identified at normal operating or other conditions, adjustments are made for accident conditions.

The NRC staff finds that the doubling of the ESF leakage pathway model and use of 2.85 gpm for the leakage rate acceptance criterion is consistent with Appendix A, Regulatory Position 5.2 of RG 1.183. The staff further finds the adjustment of operational ESF leakage to account for accident conditions consistent with, "NRC Regulatory Issues Summary (RIS) 2006-04, Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (Reference 19).

3. MSIV Leakage Pathway

The main steam lines penetrate both the primary and secondary containment boundaries and therefore represent a release pathway that bypasses secondary containment. The main steam lines at HCGS have an inboard MSIV, outboard MSIV and a turbine stop valve (TSV). The post-LOCA flow rate through these valves is based on the TS leak test limit of 250 standard cubic feet per hour (scfh), with a maximum leak rate through any one valve of 150 scfh. The licensee conservatively assumes that one steam line ruptures between the reactor pressure vessel (RPV) and the inboard MSIV. This is referred to as the failed line. Then it is assumed that the inboard MSIV also fails on the failed line, and the maximum amount of leakage (150 scfh) is through the failed line. The remaining lines are referred to as the intact lines, and the remaining leakage (100 scfh) is through these lines. All of the assumptions and parameters used in the analysis to justify use of GE14i ITAs are the same as the assumptions and parameters used in the analysis for Amendment No. 174, except as described below.

Steam Line Volumes

The MSIV leakage pathway model contains steam line volumes that are used to model the flow of radioactivity from the drywell to the environment. In Amendment No. 174, the licensee assumed the shortest steam line is the failed line. The licensee assumes the volume of the failed line is based on the volume of the steam line between the RPV and the TSV minus the volume between the inboard and outboard MSIV. For the intact lines, two lines were assumed. The volumes were based on the next two shortest steam lines and the steam line volumes between the RPV and the TSV.

In the analysis to justify use of GE14i ITAs, the failed steam line volume is based on the volume between the outboard MSIV and the TSV. This change was made to conservatively ignore deposition in the portion of the steam line that is open to the drywell (i.e., the volume between the inboard and outboard MSIVs). The drywell atmosphere in this pipe segment may be reduced by deposition, but the activity could be replaced by higher activity drywell air. The most conservative approach is to ignore deposition in the volume between the inboard and outboard MSIVs. Similarly for the intact steam line, the volume modeled is the volume between the outboard MSIV and the TSV. Use of this volume ignores the steam line between the RPV and the inboard MSIV, which is part of the drywell volume. This model also ignores the volume between the inboard and outboard MSIVs. The GE 14i ITAs justification only considers one steam line with a flow rate of 100 scfh rather than two steam lines with a flow rate of 50 scfh each.

The change in deposition volume has two effects on the MSIV leakage model. First, the smaller volumes result in a more rapid turnover of the activity in the deposition volume, decreasing the holdup time and the corresponding decay of radioactivity. This is also the effect of using a single intact steam line since the total volume of the intact steam lines is smaller. Second, the effective removal efficiency for the steam line is decreased because the deposition area of the smaller volumes is smaller. Therefore, these changes to the steam line volumes are conservative.

The NRC staff finds the method of ignoring deposition and holdup in the portion of the steam line that is open to the drywell (i.e., the volume between the inboard and outboard MSIVs), and the volume in the steamline between the RPV and MSIV to be conservative. Crediting deposition in these volumes is subject to many uncertainties including the potential impact of: (1) flow which could inhibit gravitational settling; (2) higher activity drywell air entering and replacing the air in these volumes; and (3) vaporization of any deposited fission products. Since the MSIVs provide a containment barrier, the volume up to MSIV barrier is part of the containment source volume and should not be credited separately for holdup. In addition, ignoring these volumes is consistent with the methods of modeling this pathway as shown in Figure 1 and Appendix C.1 of AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," (AEB 98-03) dated December 9, 1998 (Reference 20). In HCGS Amendment No. 134 (Reference 21) and Amendment No. 174, the use of AEB 98-03 was found acceptable.

Steam Line Flow Rates

The steam line flow rates are based on leak testing limits for the MSIVs, which are given in standard cubic feet per hour (scfh). The LOCA analysis for Amendment No. 174 uses the TS limits directly as inputs to the model. These values are based on standard conditions of atmospheric pressure and temperature (68 degrees Fahrenheit (°F), 14.7 pounds force per square inch absolute (psia)).

An alternative to using the flow rates at standard conditions is to use the flow rates that correspond to the test conditions assumed in the surveillance procedure that is used to demonstrate compliance with the TSs. In the analysis for the GE14*i* ITAs, the leak rates into the steam lines are adjusted for the peak pressure and temperature in containment. The resulting change in flow rates is shown in Table L2 of this SE. The total flow rate into the steam lines changes from 4.167 cubic feet per minute (cfm) (250 scfh at standard conditions) to 1.347 cfm (250 scfh at containment conditions).

A similar change is made to the flow rates out of the steam lines to the environment. In Amendment No. 174, the total flow rate is 250 scfh. In the analysis for the GE 14i ITAs, the flow rate out of the steam lines is based on a steam line temperature of 550 °F and atmospheric pressure. The steam line temperature of 550 °F is bounding for the limiting steam line temperature of 546 °F identified in the HCGS line index. As indicated in Table L2 of this SE, this results in an increase in the flow rate out of the steam lines from 4.167 cfm (250 scfh at standard conditions) to 7.966 cfm (250 scfh at steam line conditions). This is a conservative change to the MSIV pathway model since it decreases the holdup time in the steam line, and results in less decay.

The calculated flow rates are consistent with those calculated and used in AEB 98-03. In Amendment Nos. 134 and 174, the use of AEB 98-03 was found acceptable.

In the analysis that supported Amendment No. 174, the steam line flow rate was set at maximum values initially and then reduced by a factor of two at 24 hours. In response to the NRC staff's request for additional information Question 13 (Reference 8, Attachment 1) regarding the GE 14i ITA analysis, the assumption of a reduction in leak rate at 24 hours was eliminated. The staff finds this change conservative because it is consistent with Appendix A, Regulatory Position 6.2 of RG 1.183.

Steam Line Flow Model

There are two flow models that are commonly used to model the transport of the containment atmosphere through the steam lines. The first is the plug flow model, which assumes that all activity that enters the steam line moves at the same flow rate down the length of the steam line (no mixing). Thus, the residence time in the steam line (the holdup time) is determined by the flow rate and the steam line volume. The second is the well mixed model described in AEB 98-03. This model assumes that when activity enters a steam line volume, it is uniformly mixed throughout the volume, so that some of the activity is available for release immediately from the steam line.

The analysis for Amendment No. 174 assumes plug flow through the steam lines, whereas the analysis used to support the GE 14i ITA's assumes a well mixed model. During a design-basis LOCA, the flow pattern in the main steam line could be plug flow, well-mixed, or some combination of the two. With temperature gradients along the length of the pipe, some degree of mixing is expected to occur. For the same leak rate into the main steam line, plug flow is expected to result in less offsite release than well-mixed flow, because the concentration of the material released to the environment is at the concentration of the material in the plug at the end of the pipe. Plug flow effectively results in a longer fission product transport time in the pipe and more deposition in the pipe. Therefore, the NRC staff finds this change conservative and is consistent with Appendix A, Regulatory Position 6.3 of RG 1.183.

Steam Line Deposition Area

In the analysis for Amendment No. 174 the licensee assumed the entire internal surface of the steam line is available for aerosol deposition. Since the primary method for deposition of aerosols is gravitational settling, it is logical that the deposition would occur only on the bottom of the steam line. In the analysis used to support the GE14i ITAs, the deposition area was changed to the projected horizontal area (length of credited horizontal piping multiplied by the pipe diameter) of the steam line. This change is consistent with the information provided in RIS 2006-04.

Steam Line Deposition Rates and Removal Efficiencies

The analysis in Amendment No. 174 credits aerosol deposition in the steam lines for the duration of the accident. In the analysis used to support the GE 14i ITAs, credit for deposition of both elemental and aerosol activity is terminated at 96 hours, even though the release continues out to 30 days. The NRC staff finds this change will yield a small, conservative increase in aerosol activity released and is consistent with Regulatory Position 6.3 of RG 1.183.

The aerosol effective removal efficiencies are recalculated for the change in steam line volume indicated above. The aerosol effective removal efficiencies are also a function of the flow rate through the main steam line. The recalculated aerosol effective removal efficiencies are based on the flow rate from the steam line to the environment, which is maximized by using a steam line temperature of 550 °F for the duration of the accident.

For the elemental iodine removal efficiencies, the steam line volumes used to calculate the removal efficiencies are slightly larger than the volumes used for the aerosol removal efficiencies. The licensee stated that this may be slightly non-conservative, but since the elemental iodine removal efficiency is based on the ratio of surface area to volume, this effect will be small. The NRC staff performed a sensitivity study to determine the impact of changing the elemental iodine removal coefficients. Based upon the current modeling assumptions, the staff agrees that this effect would be small. In addition, there is reasonable assurance that the effect would be offset by conservatisms discussed below.

The elemental iodine removal efficiencies are also based on the time-dependent temperature in containment. The same time-dependent temperature distribution was used in the analyses for both Amendment No. 174 and the GE 14i ITAs. It is recognized that initially the steam line will be at a higher temperature than the containment, and that it will take some time for the steam line temperature to come into equilibrium with the containment atmosphere. The higher temperature decreases the deposition velocity for elemental iodine, so not considering the steam line temperature in the evaluation of the effective removal efficiencies for elemental iodine is non-conservative. In the September 10, 2010, supplement, the licensee stated that the effect is very small. The licensee stated and the staff confirmed that assuming a steam line of 550 °F for the first 96 hours, increases the control room dose by less than 2%. To offset the non-conservative modeling assumption, the licensee stated the following in the September 17, 2010, supplement:

> However, the assumptions used to address the steam line temperature are conservative. In particular, the flow rate from the drywell into the steam line is based on a constant drywell temperature of 298 °F and pressure of 50.6 psig for the duration of the accident. The drywell temperature and pressure will drop below these values within a day or two, causing the flow rate into the steam line to decrease. Basing the flow rate on a constant drywell temperature and pressure conservatively overestimates the amount of activity entering the steam line. The flow rate out of the main steam line is also based on constant temperature and pressure, which are the maximum steam line temperature and atmospheric pressure, even though this temperature will decrease substantially over the course of the accident and the initial pressure will be higher than atmospheric. This produces a conservative estimate of the aerosol effective

removal efficiency. It also produces a shorter holdup time in the steam line, which is conservative.

The use of the constant maximum steam line temperature to estimate the elemental iodine removal efficiencies is conservative since the steam line temperature will drop to drywell conditions within the first 96 hours of the accident, although the effect is small (less than a 2% increase in dose). Using a more realistic steam line temperature profile will result in an increase in the elemental iodine removal efficiency. This will result in doses that are less than the doses calculated for a constant steam line temperature, although these doses are likely to be greater than the doses calculated using the drywell temperature profile. Given the small effect of constant temperature elemental iodine removal on the doses, the use of the drywell temperature profile for the iodine removal efficiency, as described in Revision 4 of calculation H-1-ZZ-MDC-1880, when combined with the conservative assumptions described above results in a conservative estimate of the doses due to releases through the MSIV leakage pathway.

The NRC staff reviewed the above justification. Based upon: (1) a licensee study that shows a small impact of less than a 2% increase in doses when a steamline temperature of 550 °F is used to model elemental plateout in the steamline; (2) licensee statements that "when combined with the conservatism described above" (constant pressures and temperatures for 96 hours to calculate flows in and out of the steam line, etc.) that the analysis "results in a conservative estimate of the doses;" (3) the current MSIV modeling assumptions (i.e., drywell elemental iodine plateout credit, etc.); and (4) the use of a elemental iodine deposition model (Cline model) and drywell temperatures previously found acceptable in Amendment Nos. 134 and 174, the staff finds that using the drywell temperatures to determine elemental removal in the steamlines is appropriate for HCGS.

The NRC staff acknowledges that aerosol settling is expected to occur in the main steamline piping. Resolution of staff concerns regarding the amount of deposition that should be credited have been addressed in a proposed revision to RG 1.183 issued as draft guide (DG) 1199. However, the guidance has not been finalized. The licensee proposed a model based on the methodology of AEB 98-03, but included some additional conservatism using a 40th percentile settling velocity in an attempt to address the NRC staff's concerns regarding AEB 98-03. The 40th percentile aerosol settling velocity is a smaller value than 50th percentile settling velocity, and estimates less aerosol settling than the 50th percentile settling velocity used in AEB 98-03. The licensee model also uses one deposition volume in each line which is more conservative than the two volume model used in one of the AEB 98-03 steam line models. The NRC staff finds the 40th percentile settling velocity proposed by the licensee acceptable because it is consistent with what was found previously acceptable in Amendment Nos. 134 and 174. The staff notes that its acceptance of the assumed AEB 98-03 model is based on what was previously accepted in conjunction with the present conservatism in the proposed model.

4. <u>Control Room Model</u>

The calculation of the post-LOCA doses to the control room operator includes credit for the operation of the control room emergency filter system (CREFS). The assumptions and parameters used in the control room model for the GE 14i ITAs are the same as the assumptions and parameters used in the analysis to support Amendment No. 174, except as indicated below.

Control Room Unfiltered Inleakage

The control room unfiltered inleakage use in the analysis to support Amendment No. 174 is 350 cfm. In the analysis to support the GE 14i ITAs, the unfiltered inleakage was decreased to 250 cfm. The amount of unfiltered inleakage is a plant-specific parameter that is confirmed by testing performed in accordance with the Control Room Envelope Habitability Program, which is implemented in accordance with TS 6.16. Control room inleakage tests indicate that there is less than 200 cfm of inleakage into the HCGS control room and that essentially all of the inleakage is filtered. As part of the implementation of the GE14i ITAs amendment, the acceptance criterion in the procedure for control room inleakage testing will be changed to 250 cfm. The NRC staff finds this change is consistent with Regulatory Position 4.2.3 of RG 1.183.

5. Reactor Core Inventory

Reactor Power Level

The core inventory used in the LOCA analysis that is the basis for Amendment No. 174 is based on a reactor power level of 4031 MWt and a maximum discharge bundle exposure. The core inventory used in the analysis to support the GE 14i ITAs is based on a reactor power level of 3917 MWt and the average core inventory. This reactor power level is the current licensed power level of 3840 MWt plus 2% for instrument uncertainty consistent with RG 1.183. The lower power level reduces the amount of activity in the core that is available for release and therefore reduces the doses, although the change in burnup associated with the change from maximum discharge bundle to core average will also affect the amount of activity available for release. The licensee stated that the net effect of the two changes is expected to be small (less than 3%). The NRC staff finds this change for the LOCA analysis is consistent with Regulatory Position 3.1 provided in RG 1.183.

6. Effect of ITAs

The licensee evaluated the impact of the GE14i ITAs and provided the results of this analysis in the December 21, 2009, submittal. In the submittal, the licensee concluded that the licensing basis post-LOCA onsite and offsite doses at HCGS remain within regulatory limits established in 10 CFR 50.67.

The NRC staff requested additional information regarding the assumed release fraction for cobalt. In the May 11, 2010, response to a request for additional information the licensee stated:

> 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 [S]ections 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.

In the September 10, 2010, supplement, the licensee stated:

> The additional changes incorporated into the Technical Evaluation 80102291-0040 will not alter the conclusions of Attachment 14.2 [to calculation H-1-ZZ-MDC-1880, Revision 4] that the resulting dose consequences remain unchanged. There is no change to the Co-60 ITA [GE14i ITAs] inventory; the increase in the doses due to the additional Co-60 activity is insignificant.

An analysis was performed by the NRC staff using the licensee's methodology, as described in Calculation H-1-ZZ-MDC-1880, Revision 4, "Post-LOCA EAB, LPZ, and CR Doses," (Reference 8, Attachment 2) and supporting Technical Evaluation 80102291-0040 (Reference 9, Attachment 3). A summary of the licensee's inputs used by the staff is provided in Table L2. The staff also used a higher Co-60 release fraction of [[ ]]. The staff determined that the proposed change produces a negligible increase in the control room and offsite doses for the LOCA. The staff notes that the assumed release fraction is based on a sensitivity analysis of the impact of the release fraction to the overall dose for the proposed license amendment, and should not be solely relied upon as the licensing basis to support any other proposed licensing action. This evaluation is only valid for the current licensing basis as modified by the proposed change.

Based upon this evaluation, the NRC staff confirmed the licensee's assessment that the impact of the proposed change on the LOCA would be negligible using the licensee's LOCA assumptions and a Co-60 release fraction of [[ ]]

7. <u>LOCA Summary</u>

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes as described above. Assumptions and parameters used in the staff's confirmatory analysis are in Table L2 of this SE. Based upon the information provided by the licensee, the staff concludes that the licensee used analysis methods and assumptions that are conservative or consistent with the guidance of RG 1.183. The NRC staff compared the radiation doses estimated by the licensee to the 10 CFR 50.67(b)(2) acceptance criteria and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the exclusion area boundary (EAB), low-population zone (LPZ), and CR doses for the LOCA will continue to comply with the criteria.

3.5.3.3 Fuel Handling Accident

The fuel handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism resulting in the dropping of a raised fuel assembly onto other fuel assemblies. A variety of events that qualify for the class of accidents termed "fuel handling accidents" has been considered. The accident that produces the largest number of ruptured spent fuel rods is the drop of a spent fuel assembly into the reactor core when the reactor vessel head is off.

The HCGS licensing basis FHA is analyzed in Section 15.7.4 of the HCGS UFSAR. The impact of 12 GE14i ITAs on the FHA radiological consequences was evaluated by the licensee.

In the December 21, 2009, submittal the licensee stated the following:

> The existing GE14 fuel handling accident analysis takes the available potential energy from a dropped fuel assembly and calculates the number of failed fuel rods, assuming the rods fail by 1% strain in compression using a number of conservative assumptions. Given the reduced weight of the GE14i fuel assembly, the potential energy from a dropped fuel assembly is reduced and the resulting number of failed rods is also reduced.
>
> The HCGS licensing basis FHA is analyzed in Section 15.7.4 of the HCGS UFSAR. The licensing basis FHA postulates that an irradiated 8x8 fuel bundle is dropped 32.95 feet onto the reactor core and fails 124 rods. Of the failed rods, 8% of the I-131, 10% of the Kr-85, 5% of the other noble gases and halogen inventories, and 12% of the alkali metal inventory of the damaged rods are released from the rods. All other particulates are retained by the water.
>
> Reference 1 [GNF, "GE14i Thermal-Mechanical Evaluation," GNF-0000-0108-6874-RO, October 2009] documents that radiological consequences from a FHA involving the GE14 design are bounded by consequences from a FHA involving the 8x8 fuel design. [[
>
>]].
> Therefore, the licensing basis FHA radiological analysis is not impacted by the introduction of 12 GE14i assemblies at HCGS.

The NRC staff finds that the licensee's FHA analysis assumptions and methodology are consistent with the guidance of RG 1.183. Per Regulatory Position 3, Appendix B of RG 1.183, "[p]articulate radionuclides, with exception of cesium iodide (CsI) are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor)." Since Co-60 is expected to be released as a particulate during FHA conditions, the Co-60 will be retained by the water in the fuel pool or reactor cavity. Also, since the GE14i assemblies are not as heavy as the other fuel assemblies previously analyzed, the potential energy from a dropped fuel assembly is reduced and the resulting number of failed rods is also reduced. Based upon the discussion above, the proposed FHA doses will be bounded by the previous HCGS analysis. Therefore, the staff determined with reasonable assurance that the licensee's estimates of the total effective dose equivalent due to the FHA will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183 and, therefore, is acceptable.

3.5.3.4 Other Accidents

In the May 11, 2010, response to a request for additional information (Question 16), the licensee states that 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., LOCA and CRDA). None of the six postulated events listed below 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 six events are unchanged for a core operating with isotope test assemblies.

- Main Steam Line Break (MSLB) (UFSAR Section 15.6.4)
- Reactor Recirculation Pump Seizure (UFSAR Section 15.3.3)
- Reactor Recirculation Pump Shaft Break (UFSAR Section 15.3.4)
- Instrument Line Break (UFSAR Section 15.6.2)
- Feedwater Line Break Outside Containment (UFSAR Section 15.6.6)
- Gaseous Radwaste Subsystem Leak or Failure (UFSAR Section 15 .7.1)

The NRC staff evaluated these HCGS statements for the above accidents. If there is no core melt due to the accident, the source term available for release to the environment is based upon the activity in the reactor coolant system during normal operations. With Co-60 leakage not credible during normal operations (see SE Section 3.5.4.1), the proposed change will have no impact on the above accidents. Based upon the licensee's statements and the evaluation in SE Section 3.5.4.1, the staff agrees that the ITAs will not impact the above 6 accidents.

3.5.4 Thermal-Mechanical and Hydraulic Evaluation

3.5.4.1 Thermal-Mechanical Evaluation

Thermal-mechanical characteristics of the GE14i cobalt isotope rods were evaluated as discussed in Reference 22. As part of its review, the NRC staff performed an audit to gain greater understanding and to verify the information provided by the licensee.

The UO_2 and Gd rods in GE14i remain unchanged from those of GE14, therefore, standard UO_2 and Gd limits for GE14i bundles are applicable. GE14i cobalt rods consist of [[

]] The assemblies are then assembled into a full-length rod with outer characteristics essentially identical to GE14 full length rod.

The LHGR envelope assumed for these analyses is [[

]]

Heat Deposition in GE14i Rods

Minor heating is expected for these GE14i rods, due to the gamma and neutron absorption in the cobalt pellets and other rod components. While the decay of Co-60 produces high energy gammas and is dependent only on Co-60 concentration in the cobalt pellets, the produced gammas represent a small percentage compared to fission gammas in the bundle and their high mean free path make it unlikely that they will interact with the cobalt rod which produces them. Since the gamma flux is proportional to bundle power, the GE 14i LHGR will follow bundle power throughout rod life, including power transients.

The heat deposition in the GE14i rods is estimated based on the energy deposition compared to the peak rod in the bundle, assuming that the peak rod is operating on the GE 14 UO_2 LHGR limit of 13 .4 kW/ft. This calculation determines a maximum heat deposition rate of [[ ]] including heat deposited in the cladding. This analysis conservatively assumes a LHGR of [[ ]] for the cobalt rods and evaluated margins for failures for this rod.

Design Criteria

Although the fuel design criteria in Reference 23 are not applicable to the design of GE14i rods, the overall safety criteria defined in Reference 23 are still applicable for the GE14i rod design. Since the outer cladding for the GE14i rods is essentially unchanged from the GE14 rods, the design evaluations from Reference 23 are sufficient to address cladding integrity. The cladding temperature of GE14i rods is essentially identical to that of GE14 rods [[

]]

Inputs

The inputs for the thermal-mechanical analysis of GE14i rods are consistent with standard fuel rod thermal-mechanical analyses, GSTRM. [[

]]

Evaluation

The following evaluations were performed by GEH:

- Statistical internal pressure design ratio calculation;
- Statistical fatigue, creep rupture, fatigue + creep rupture, and plastic + weighted creep design ratio calculations;
- Permanent Strain during a pressurization type event, initiated from full power with Mechanical Overpower (MOP) applied;
- Worst-tolerance cladding creep collapse analysis; and
- Statistical steady-state temperature evaluation during an AOO with thermal overpower of (TOP) [[ ]] applied.

[[ ]] As such, overpressure failures are not plausible.

The clad mechanical analyses (Fatigue + Creep Rupture and Plastic + Weighted Creep Strain) were performed using an NRC-approved methodology. The results from these analyses indicate that all calculations meet the design limits with significant margin.

The cladding 1% permanent strain for core-wide AOO's initiated from full power with MOP is a [[ ]] In order to ensure that GE14i is less limiting than GE14, GE14i was analyzed to higher value of MOP. The reported results indicate that the cladding meets the 1% strain limit.

The worst tolerance permanent strain analysis uses the worst-tolerance values, including cladding and fuel dimensions. [[ ]]

Analyses were performed by GEH to ensure that the cladding does not collapse [[ ]] This analysis is similar to the analysis performed to ensure no collapse between fuel pellets due to densification. Inputs to this analysis are: initial cladding ovality, conservative value for rod pressure, the heat generated in GE14i and [[ ]] For this analysis, no credit is taken for the [[ ]] The analysis concluded that cladding creep collapse will not occur [[ ]]

The possibility of melting of the internal components of GE14i, [[

]] In order to ensure that GE14i is [[
]] The calculated upper 95% peak temperature for the nickel coating was significantly less than the nickel melting temperature. Thus, the NRC staff finds that the (no) melting design criterion was met.

As discussed in Reference 7, a thermal-mechanical compliance check was performed by GEH for all analyzed transients to assure that the fuel will operate without violating the thermal-mechanical design limits. These limits are designed such that reactor operation within these limits provides assurance that the fuel will not exceed any thermal-mechanical design or licensing limits during all modes of operation. The NRC staff has determined that with multiple layers of cladding and design features, there is reasonable assurance that the isotope rod failure will not occur.

3.5.4.2 Hydraulic Evaluation

This section summarizes the evaluation of the pressure drop characteristics of the GE14i fuel compared to the GE14 fuel. GE14i fuel is different from GE14 fuel in two aspects:
(1) replacement of [[ ]] fuel rods with cobalt isotope rods; and (2) hex-faced connectors are used for the assembly and disassembly of the isotope rods.

Since isotope rods are considered zero-power rods due to negligible amount of heat produced, these cold rods change the void generation and void/flow distribution patterns which may have an impact on the pressure drop characteristics of the GE14i fuel. [[

]] This hydraulic evaluation compares the pressure drop data with zero-power rods to those without zero-power rods, and the impact of hex-faced connectors (Hex) on coolant flow area changes.

<u>Impact of Cold Rods</u>

Testing for the GE14 fuel, critical power and pressure drop data were collected with zero-power rods and [[ ]] (see SE Section 3.4.4). Four different rod-to-rod power distributions with a wide range of inlet flow and inlet subcooling conditions were tested with zero-power rods. Peaking patterns J1/J2/J3 had [[ ]] zero-power rods and pattern D0xx had [[ ]] zero-power rods. The pressure drop difference for the cold rod data are listed in Table 3.6.

Table 3.6 - Pressure Drop Difference

	[[		
Mean, kPa			
St. Dev, kPa			
No. of Data			]]

Based on the results of the testing, the NRC staff concludes that the cold rod impact on pressure drop characteristics of GE14i fuel is negligible and the impact is within the uncertainty for the GE14 fuel.

Impact of the Hex

There is a small area increase at the short length of the Hex region [[ ]] over the standard circular rod region of the active fuel length. Table 3.7 compares the flow areas for GE14i fuel circular region and the Hex region.

Table 3.7 - Flow Area Comparison

Elevation	**GE14i Flow Area, in^2**		
	Circular Region (A)	**Hex Region (B)**	**Ratio (B/A)**
Fully Rodded	[[		
Partially Rodded			]]

The small area change at the short length of the Hex will not have any significant impact on flow/void distribution.

The NRC staff, upon review of the available information, has determined that the zero-power rods and the hex-faced connectors have negligible impact on the pressure drop characteristics of the GE 14i fuel.

3.5.5 Other Evaluations

3.5.5.1 Stability Analysis

An evaluation was performed by GEH to assess the impact of GE14i ITAs on thermal-hydraulic instability. Using the methodology in References 31 and 35, a review was performed on the GE14i ITAs to demonstrate that an ITA is very unlikely to result in single-channel instability. An assessment was performed based on the GE14i bundles to evaluate the impact on decay ratio. Decay ratio is a measure of the stability of an oscillating system and is defined as the value of one peak in the oscillation to the amplitude of the peak immediately preceding it. The amplitude is measured relative to the average amplitude of the signal. A stable system is characterized by a decay ratio of less than 1.0; an unstable system has a decay ratio greater than 1.0. Decay ratios greater than 1.0 are referred to as growth rates.

HCGS is an Option III plant, and continues to use the Option III system for Cycle 17. The Option III design provides automatic detection and suppression of reactor instability events; as such, reliance on operator actions to suppress instability events is minimized. The Option III design provides a high degree of defense-in-depth. Each of four independent trip channels monitors signals from a large number of LPRM detectors. A local group of LPRMs known as Oscillation Power Range Monitor (OPRM) cells are distributed throughout the core so that each trip channel provides monitoring of the entire core. Thus, the system is fully capable of detecting both core-wide and regional modes of oscillation.

For the Option III stability solution, two stability aspects must be considered; the first is the OPRM system setpoint, the second is the Backup Stability Protection. The Option III stability Backup Stability Protection (BSP) regions provide protection in the case that the OPRM system is inoperable. The BSP regions are calculated on both plant- and cycle-specific bases. The BSP region is expanded or contracted each cycle in accordance with the specific ODYSY code acceptance criteria for core and channel decay ratio as specified in Reference 31.

A reload Option III stability evaluation was performed in accordance with an approved licensing methodology as discussed in Section 15 of Reference 7. The stability-based OLMCPR as a function of OPRM amplitude setpoint, is determined for two conditions: (1) a postulated oscillation at 45% rated core flow quasi steady-state operation (SS); and (2) a postulated oscillation following a two recirculation pump trip (2PT) from the limiting rated power operation state point.

The OPRM-setpoint-dependent OLMCPR (SS) and OLMCPR (2PT) values were calculated for Cycle 17 in accordance with the Boiling-Water Reactor Owners Group (BWROG) regional mode Delta CPR over Initial MCPR vs. Oscillation Magnitude (DIVOM) guidelines. The Cycle 17 Option III evaluation provides adequate protection against violation of the SLMCPR for the two postulated reactor instability events as long as the plant OLMCPR is equal to or greater than OLMCPR (SS) and OLMCPR (2PT) for the selected OPRM setpoint. The OPRM setpoints for two-loop operation are conservative relative to single-loop operation (SLO) and are, therefore, bounding. The results are listed in Tables 15-1, "Relationship between OPRM Successive Confirmation Count Setpoint and OPRM Amplitude Setpoint," and Table 15-2, "OPRM Setpoint Versus OLMCPR," of Reference 7.

The BSP region boundaries were calculated for HCGS Cycle 17 for normal and reduced feedwater temperature operation. The end points of the regions are defined in Tables 15-3, "BSP Region Intercepts for Normal Feedwater Temperature," and the region boundaries are shown in Figure 52 of Reference 7. The NRC staff finds that the licensee has shown that the Cycle 17 BSP region is conservative and bounds the calculated BSP region endpoints.

Appendix H of Reference 7 contains additional results for BSP region boundaries for reduced feedwater temperature (FFWTR) operation. Tables H-1 and H-2 define the endpoints of the BSP regions for reduced feedwater temperature operation. FFWTR BSP region boundaries are based on 343°F which is an 88°F reduction from the rated normal feedwater temperature of 431.6°F. Figures H-1 and H-2 of Reference 7 illustrates the BSP region boundaries for feedwater heaters out-of-service (FWHOOS) and for FFWTR operation, respectively.

To support the initial introduction of GE14i ITAs into the HCGS core, the licensee performed additional calculations for BSP region end points and decay ratios. These calculated power/flow points and decay ratios represent typical calculations for HCGS. The NRC staff finds that the plant- and cycle-specific calculations provide reasonable assurance that the thermal-hydraulic stability, as prescribed by Option III with respect to the size of the BSP regions, is maintained with ITAs in the HCGS core. Table I-1 of Reference 7 lists BSP Region Calculated Intercepts for Normal Feedwater Temperature.

3.5.5.2 Decay Heat Assessment

A comparative core decay heat assessment between GE14 and GE14i was performed using the [[

]]

The NRC staff audited the licensee's documents that provided the detailed calculations performed by GEH for HCGS (DRF 0000-0100-5497 Revision 2). [[

]]

The NRC staff concludes that replacing 12 GE14 bundles with 12 GE14i bundles will not cause a significant increase in core decay heat at HCGS.

3.5.5.3 Appendix R Safe Shutdown Fire

The limiting safe shutdown event for HCGS, with respect to 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," is mitigated with Reactor Core Isolation Cooling (RCIC), Safety Relief Valves and Residual Heat Removal (RHR) from a remote shutdown panel. [[

]] RCIC is used from a remote shutdown panel to maintain the water level above the top of active fuel, and the peak cladding temperature (PCT) for GE14i ITA is the initial steady state fuel temperature, which is well below the Appendix R PCT limit of 1500°F. The NRC staff finds that the change due to introduction of the GE14i ITAs has a negligible effect on the fire-related safe shutdown analysis previously evaluated by the staff as part of the HCGS EPU review (Reference 18).

3.5.5.4 Station Blackout (SBO)

[[

]] The NRC staff finds that this change is considered to be negligible for the licensing-basis SBO analysis.

3.5.5.5 Containment Response

[[

]] The NRC staff has determined that replacement of 12 GE14 assemblies with 12 GE14i ITAs at HCGS will not have any significant impact on containment response analysis at HCGS.

3.5.5.6 Reactor Internal Pressure Difference

This section describes the evaluation of maximum pressure drop for reactor internals, the minimum fuel bundle lift margin, and maximum control rod guide tube (CRGT) lift force, as well as acoustic and flow-induced loads on a jet pump, core shroud and shroud support.

The thermal-hydraulic design for the GE14i bundle closely matches the overall pressure drop of previous designs. The main differences are that GE14i bundle has [[ ]] cobalt isotope rods replacing the fuel rods and the GE14i ITA's hex-faced connectors used for the assembly and disassembly of the isotope rods. Also, the isotope rods are practically considered as zero-power rods or cold rods, since heat generated in them due to gamma ray deposition is very small compared to the heat generated in the fuel rods. The existence of the cold rods changes the void generation and void/flow distribution patterns, which has an impact on the pressure drop characteristics in the GE14i fuel. The licensee has determined that the cold rod impact on the pressure drop characteristics of the GE14i fuel is negligible and the impact is within the uncertainty for the GE14 fuel (see SE Section 3.5.4.2). The licensee has shown that the hex-faced connectors have negligible impact on the pressure drop characteristics of the GE14i fuel.

The minimum fuel bundle lift margin is [[

]] The GE14i bundle weight is [[ ]] than that of GE14 bundle. Other key parameters are unchanged due to the similar thermal-hydraulic design. Therefore, the GE14i bundle results in [[ ]] than the minimum fuel bundle lift margins for GE14. The limiting faulted condition fuel lift margin for GE14i is [[ ]]. The impact on the fuel lift load and other reactor internal loads due to decreased fuel lift margin is assessed by structural analysis as discussed in SE Section 3.5.5.7.

The parameters that are used in the determination of the maximum CRGT lift force are [[

]] These parameters do not change for GE14i ITAs and thus, the maximum CRGT lift force for GE14 remains applicable for GE14i ITA.

The introduction of 12 GE14i ITAs in HCGS core has no effect on the acoustic and flow-induced loads on the jet pump, core shroud and shroud support, which are caused by pressure waves as a result of a recirculation suction line break. [[

]]

The NRC staff concludes that the replacement of 12 GE14 bundles with 12 GE14i ITAs in HCGS core will not cause any significant change in the reactor internal pressure differences (RIPD).

3.5.5.7 Reactor Internals Structural Evaluation

An audit was conducted at the GE offices of the documents related to reactor internals structural evaluation (DRF 0000-0106-6565) to confirm information provided by the licensee in Reference 22.

A qualitative structural assessment of the reactor internal components was performed with respect to the current design-basis evaluation. The evaluation in Section 4.5.8 of Reference 22 demonstrates that operation with GE14i ITAs will have no adverse effect on the structural integrity of the reactor internals relative to seismic loading. The weight variation of the full core (i.e., with 12 GE14i bundles versus with 12 GE14 bundles) is negligible relative to the structural integrity.

All applicable Normal, Upset, Emergency, and Faulted condition loads for GE14i ITAs such as seismic loads, acoustic and flow induced loads, fuel lift loads, RIPDs, system flow loads, core flow loads, and thermal loads, as appropriate, were considered in the licensee's assessment. These loads are either bounded by, remain unaffected, or have an insignificant effect on the structural integrity of the reactor internals with respect to the current design basis evaluation. Therefore, the NRC staff has concluded that the introduction of GE14i ITAs has an insignificant effect on the structural integrity of the reactor internal components.

3.5.5.8 Recirculation System Evaluation

An evaluation of the effects of introducing GE14i fuel on Reactor Recirculation System (RRS) performance for HCGS was performed by the licensee. The evaluation is based on clean equipment conditions and does not consider the potential effects of crud deposition on jet pumps, which lowers their efficiency.

For the recirculation system evaluation, the primary impact of introducing a different fuel assembly would be a core pressure drop change. The evaluation results show that the core pressure drop change is negligibly small with the introduction of the GE14i fuel bundles. As a result, there is no change in the recirculation system pressures, temperatures, pump flow rate and reactor recirculation pump motor brake horsepower. Also, there is no change to the recirculation pump required or available Net Positive Suction Head (NPSH) since the pump flow rates and recirculation system pressure/temperature is the same value as before GE14i fuel introduction.

The NRC staff finds that no modifications to RRS equipment or setpoints are required with the introduction of GE14i ITAs at HCGS.

3.5.5.9 Seismic and Dynamic Response

The audit conducted at the GE offices by the NRC staff reviewed the documents related to seismic and dynamic response from testing of the fuel assemblies to confirm information provided by the licensee in Reference 22. Due to the negligible full core weight variation [[ ]] impact, the seismic/dynamic behavior of the core, the reactor internals, and the balance of plant will not be affected by the introduction of 12 GE 14i ITAs. The dominant fuel type, GE14 fuel, dictates the seismic behavior of the core. The minor overall fuel bundle mass difference will not impact the seismic adequacy. The maximum fuel applicable acceleration increase to [[ ]] G for horizontal and [[ ]] G for vertical are both within the [[ ]] G horizontal and [[ ]] G vertical allowable peak seismic accelerations.

Dynamic fuel lift load analysis is not required for HCGS in accordance with the Mark 1 containment licensing basis. The NRC staff concludes that the seismic/dynamic behavior of the core and the internals, the balance of plant and the primary structure will not be affected by the introduction of 12 GE14i ITAs into the HCGS core.

3.5.5.10 Neutron Fluence Impact

The introduction of GE14i fuel ITAs into the core will not significantly impact the magnitude of the reactor pressure vessel (RPV) fluence since the reactor power is unchanged and the core-wide void and relative power distribution remains approximately the same.

RPV fluence is highly dependent on the core peripheral bundle power distribution, which is dependent on the cycle operating plan and the core loading pattern. The loading pattern constraints and limitations are applicable to each reload fuel cycle, regardless of the fuel type. The substitution of neutron absorber material for fuel in the few rods of the GE14i ITAs will have insignificant impact on the power density of the fuel bundle. Further, the number of ITAs is only a small fraction of the total number of bundles in the HCGS core (i.e., 12 out of 764 bundles). This is not expected to significantly impact the core-wide power distribution and peripheral bundle power.

Changing from one fuel type to another with different part length rod (PLR) designs may cause slight variation in the axial flux distribution. However, GE14i fuel uses the same PLR design as GE14; thus, no variation in the axial flux distribution is expected.

Therefore, the NRC staff finds that the introduction of GE14i fuel will not have any significant impact on the current overall fluence values for HCGS.

3.5.5.11 ECCS LOCA

GE14i ITAs are loaded into the HCGS core at non-limiting locations and not in the hot channel with respect to ECCS LOCA Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits. The number of fuel rods in GE14i bundle is less than the number in GE14 by the number of cobalt rods in the GE14i bundle. The MAPLHGR is not averaged over the zero-power rods. The ECCS-LOCA MALPLHGR limits for Cycle 16 for GE14 remain bounding for the GE14i ITAs. ECCS-LOCA MAPLHGR limits for all bundles in Cycle 17 for average planar exposure range of 0.0 GWd/MT to 70.0 GWd/MT are listed in Table 16.3-1 of Reference 7 and are reproduced in Table 3.8.

Table 3.8 - MAPLHGR Limits

Average Planar Exposure		MAPLHGR Limit
GWd/MT	**GWd/ST**	**kW/ft**
0.00	0.00	12.82
16.00	14.51	12.82
21.09	19.13	12.82
63.50	57.61	8.00
70.00	63.50	5.00

The SLO multiplier on LHGR and MAPLHGR, and the ECCS analytical initial MCPR values applicable to each fuel type in the new cycle core are shown in the Table 16.3-2 of Reference 7 and reproduced below in Table 3.9. The GE14 10 CFR 50.46 initial MCPR and SLO multiplier on LHGR and MAPLHGR are applicable to the GE14i ITAs.

Table 3.9 - Initial MCPR and Single Loop Operation Multiplier on LHGR and MAPLHGR

Fuel Type	Initial MCPR	SLO Loop Operation Multiplier on LHGR and MAPLHGR
GE14	1.250	0.80
GE14i	1.250	0.80

[[

]] Furthermore, because the PCT and the maximum oxidation values remain within licensing basis, a coolable geometry is assured. The licensing results, applicable to all fuel types in the new cycle (Cycle 17), are listed in Table 16.1-1 of Reference 7 and are reproduced below in Table 3.10.

Table 3.10 - Licensing Results for Cycle 17

Fuel Type	Licensing Basis PCT (°F)	Local Oxidation (%)	Core-Wide Metal-Water Reaction (%)
GE14	1380	<1.00	<0.10
GE14i	1380	<1.00	<0.10

The introduction of the GE14i ITAs does not affect the reflooding capability of the ECCS or the operation of the core spray systems, thus assuring long term core cooling. Therefore, the NRC staff has determined that the five acceptance criteria established by 10 CFR 50.46 remain satisfied with the introduction of the GE14i ITAs.

3.5.5.12 Hydrogen Injection

As discussed in Reference 22, GEH performed an evaluation regarding the potential impact on hydrogen water chemistry (HWC) requirements due to the proposed introduction of the GE14i ITAs into the HCGS core.

As discussed in HCGS Updated Final Safety Analysis Report Section 10.4.7.2.1, the HWC system is provided to inject gaseous hydrogen into the suction side of the secondary condensate pumps at an injection rate necessary to provide intergranular stress corrosion cracking (IGSCC) protection of the recirculation piping. The addition of hydrogen reduces the oxygen content in the reactor water and reduces the corrosion potential of the water. As discussed in the NRC staff's SE for HCGS Amendment No. 176 (Reference 30), in 2007, HCGS implemented the GE NobleChem™ process (noble metal chemical addition (NMCA)) which allows the HWC system hydrogen injection rate to be reduced significantly.

As discussed in Reference 22, an Institute of Nuclear Power Operations operating experience (OE) report describes an incident where a core design change, at a HWC plant that does not use NMCA, resulted in a lower gamma flux in the down-comer region of the reactor causing a reduction in the hydrogen-oxygen combination reaction. The decreased gamma flux necessitated an increase in the hydrogen injection rate to maintain IGSCC mitigation compared to the previous cycle of operation. The key objective of the GEH evaluation of this OE relative to GE14i was to determine whether there is any potential for a decrease in gamma flux in the down-comer region of the HCGS core as a result of cobalt bearing rod insertion.

GEH concluded that BWRS that use NMCA with the HWC system do not rely on the down-comer dose rate to catalyze the hydrogen-oxygen recombination. Instead, the noble metals catalyze the reaction at the metal surface. Therefore, GEH further concluded that there is no negative impact on the hydrogen requirements for IGSCC protection with the introduction of GE14i ITAs into the HCGS core. The NRC staff agrees with the GEH conclusion.

3.5.5.13 Post-LOCA Hydrogen Control

HCGS Amendment No. 160 (Reference 52) eliminated the requirements associated with hydrogen recombiners, and hydrogen and oxygen monitors to support the implementation of a 2003 revision to 10 CFR 50.44, "Combustible gas control for nuclear power reactors" for HCGS. The revised 10 CFR 50.44 no longer defines a design-basis LOCA hydrogen release and eliminates the requirements for hydrogen control systems to mitigate such releases. As such, the NRC staff finds that the proposed introduction of the GE14i ITAs into the HCGS core will have no impact on post-LOCA hydrogen control.

3.5.5.14 Fuel Storage Criticality Safety

This section evaluates the licensee's criticality safety analysis of the fuel storage racks at HCGS that are used for the storage of GE14i ITAs. The licensee's original analyses evaluated the peak reactive GE14 lattice that meets the fuel storage rack reactivity safety limits at a maximum bounding uniform enrichment of no less than 4.9 wt% U-235.

The licensee's reanalysis assumes that mechanically-equivalent stainless steel rods will be used to replace any isotope target rods that are removed from the bundle in order to maintain mechanical integrity of the stored bundle. Use of the mechanically-equivalent stainless steel rods lends greater stability to the system and displaces the interstitial water in order to conserve the relative moderator effects of the previous analyses.

For criticality safety, the only difference between GE14i and a standard GE14 bundle is the [[ ]] fuel rods that are replaced by the cobalt target rods in the GE14i ITAs. This replacement introduces neutron absorbers into the core. The displaced enrichment may be either removed from the assembly entirely, or it may be placed within other locations within the same bundle or bundles not utilizing isotope rods as allowed by fuel and core design constraints.

The maximum bounding uniform enrichments of no less than 4.9 wt% U-235 assumed in the original GE14 models ensure that the models are insensitive to the spatial distribution of fissile material. Therefore, the potential enrichment displacement proposed by the GE14i ITA is already conservatively factored into the original GE14 models. For these reasons, the NRC staff finds that the GE14 fuel storage rack reactivity safety limits, including infinite multiplication factor (k-infinity) design limits, are appropriate for use with GE14i ITAs.

3.5.5.15 Fresh Fuel Shipping

Reference 22 states that shipping of GE14i ITA bundle will be done under the requirements specified in the RAJ-II Certificate of Compliance (CoC) that was issued by the NRC on May 28, 2008 (Reference 33). Since the technical requirements specified in Section 5(b)(1) of the CoC pertain specifically to "enriched commercial grade uranium or enriched reprocessed uranium, uranium oxide or uranium carbide fuel rods enriched to no more than 5.0 weight percent in U-235," these technical requirements do not apply to the cobalt isotope rods since these rods do not contain uranium.

There is no licensing impact on the fresh fuel shipping container criticality analysis since these [[ ]] cobalt isotope rod locations are analyzed as containing 5% enriched UO_2 rods in Chapter 6 of the RAJ-II Safety Analysis Report (SAR) which bounds the cobalt isotope rods from a criticality safety standpoint. Since the GE14i bundle with [[ ]] total UO_2 rods which is outside the range of 91-100 as specified in Table 3 of CoC, this condition is bounded by the RAJ-II analysis under both normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

The GE14i bundle gadolinium (Gd) requirements, minimum number of Gd rods as well as minimum Gd enrichment, are identical to those specified for a GE14 bundle. Since the locations of cobalt isotope rods are analyzed as those containing enriched UO_2 rods in the RAJ-II safety analysis report, this bounds the cobalt isotope rods from a criticality standpoint under NCT and

HAC. The NRC staff concludes that shipment of fresh GE14i bundles in the RAJ-II container is acceptable.

3.5.5.16 Fuel Channel Distortion

Channel distortion that can cause channel interference is a function of the fluence gradient (fluence bow), early life control (shadow bow) and the pressure gradient across the channel (channel bulge). The NRC staff finds that the presence of non-fueled rods does not significantly affect these parameters and therefore the channel performance on GE14i bundles will be the same as on GE14 bundles.

3.5.5.17 Fuel Conditioning Guidelines

The fuel conditioning guidelines are based on the peak nodal powers in the bundle and the thresholds are exposure dependent. The presence of a small number of cobalt isotope rods does not modify these guidelines.

3.5.5.18 Emergency Operating Procedure Data

Table 4.2 of Reference 22 lists the GE14i data for revising the applicable emergency operating procedures. This list contains values for cold and hot shutdown boron concentrations, decay heat fraction 10 minutes after shutdown, decay heat as a function of time after shutdown, values for minimum active fuel length fraction that must be covered to maintain a peak clad temperature (PCT) less than 1500°F with injection and to maintain PCT less than 1800°F without injection, minimum bundle steam flow required to maintain PCT less than 1500°F, maximum core uncover time before PCT exceeds 1500°F and physical properties of fuel and cladding. The NRC staff reviewed this list and found it acceptable.

3.5.6 Manufacturing Quality Assurance

All aspects of the GE14i ITA program will be controlled under the GE Nuclear Energy Quality Assurance Program Description (Reference 34).

3.5.7 Post Operational Evaluations

3.5.7.1 Spent Fuel Pool Effects

In response to an NRC staff request for additional information, PSEG and GEH provided details of analysis and calculations related to the effects on the spent fuel structure from the introduction of GE14i irradiated fuel in the HCGS spent fuel pool (Reference 53). The analysis and calculations were performed in two parts: (1) detailed incident gamma energy analysis and calculations; and (2) detailed gamma energy deposition analysis and calculations.

3.5.7.1.1 Incident gamma energy analysis and calculations

The gamma energy incident at the concrete spent fuel pool wall from two types of irradiated fuel bundles was analyzed for HCGS at various post-shutdown time intervals. The two fuel types evaluated were GE14 and GE14i ITA. The computer codes that were used for this analysis are:

(1) MCNP for the photon transport calculation; (2) TGBLA for lattice pin-by-pin material specifications and geometry; and (3) ORIGN01P to determine radionuclide composition for the MCNP calculations.

The following assumptions were made for the incident gamma analysis.

- [[[
  ]]
- Peak pellet exposure of [[ ]] for both GE14 and GE14i fuel assemblies.
- A conservative value of [[ ]] bundle power for EOL conditions.
- [[ ]]
- A conservative value of [[ ]] of Cobalt-60 activity for a GE14i bundle.
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]

[[

]] Table 3.11 lists the single bundle gamma incident energies at the HCGS SFP wall based on the information provided in References 3 and 53.

Table 3.11 - Gamma Incident Energies at the HCGS SPF Wall

Fuel Type	Post-shutdown decay time	Distance from closest pins to concrete wall	MCNP Result: Incident energy at concrete wall, MeV/ $(cm^2\text{-}sec)$	[[ ]]	[[ ]]
1 foot cases					
GE14i ITA	[[ ]]	1ft	[[	]]	1.00E+11
GE14i ITA	[[ ]]	1ft	[[		]]
GE14i ITA	[[ ]]	1ft	[[		]]
GE14	[[ ]]	1ft	[[		]]
GE14	[[ ]]	1ft	[[		]]
4 feet cases					
GE14i ITA	[[ ]]	4 ft	[[	]]	2.4E+08
GE14	[[ ]]	4 ft	[[		]]

Per ANSI/ANS-6.4-2006, "Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants," (Reference 54) incident fluxes less than 10^{10} MeV/$(cm^2\text{-}sec)$ result in negligible heating of the concrete. Therefore, a GE14i bundle placed 4 feet from the SFP wall will cause negligible heating of the concrete. In addition, because the 4-foot incident energy is so far below the threshold, any array of GE14i bundles will also cause negligible heating of the concrete when stored at least 4 feet away from the SFP wall.

3.5.7.1.2 Gamma Energy Deposition Analysis and Calculations

The gamma energy deposition in the concrete SFP wall from GE14i and GE14 fuel types was analyzed for HCGS at a post-shutdown interval of 24 hours. The photon transport and interaction calculations were performed using MCNP-05P, which is a GEH/GNF Level 2 version of the Monte Carlo code MCNP5 developed by the Los Alamos National Laboratory.

The following assumptions were made for the gamma energy deposition analysis.

- [[

]]

- Peak pellet exposure of [[ ]] for both GE14 and GE14i fuel assemblies.
- A conservative value of [[ ]] bundle power for EOL conditions.
- [[ ]]
- A conservative value of [[ ]] of Cobalt-60 activity for a GE14i bundle.
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]

[[ ]]

Per NUREG/CR-6927, "Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures - A Review of Pertinent Factors" (Reference 55), there is a possibility of some concrete degradation due to a total integrated gamma dose above 10^{10} Rad. Therefore, the MCNP calculation was extended to determine the energy deposition rate (dose rate) to the concrete walls of the HCGS SFP. This calculation assumed a conservative value [[ ]]

The dose rates in the first cubic centimeter of concrete, where the dose is highest, were integrated to determine the time to reach 10^{10} Rad and the results are shown in Table 3.12. No credit was taken for decay of the fission products or cobalt from the 24-hour dose rates. Since the time to reach an integrated dose to the wall greater than 10^{10} Rad at 4 feet is significantly greater than the life of the plant, a single or an array of GE14i bundles will not cause a long term concrete degradation. Therefore, there is no limitation on the amount of time an irradiated GE14i bundle may remain in the SFP at 4 feet from the walls.

Table 3.12 - Irradiated Fuel Dose Rate and Integrated Dose at the HCGS SPF Wall

Fuel Type	Distance from Closest Pins to Concrete Wall, feet	Dose Rate Rad/hour	Years to Integrated Dose of 10^{10} Rad
GE14i	1	[[	]]
GE14i	4	[[	]]

The licensee has shown that there are no adverse effects from the introduction of GE14i fuel in the HCGS spent fuel pool, provided guidance for storage of GE14i bundles is followed to minimize the effect of gamma heating on the SFP concrete walls. As discussed in SE Section 2.2, a new license condition will be added that will require that irradiated GE14i bundles be stored at least 4 feet from the walls of the SFP. With the 4-foot distance requirement (i.e., from outside edge of the bundle to the SFP wall), there is no limit to the duration of time a GE14i bundle may remain in the pool.

Based on the licensee's analyses, the NRC staff has determined that at 4-foot distance from the SFP walls, the irradiated GE14i ITAs will not cause any substantial heating or degradation of the SFP concrete walls.

3.5.7.1.3 Gamma Heating of SFP Walls during Isotope Rod Removal

Attachment 1 of PSEG's application dated December 21, 2009, states that the cobalt isotope rods will be removed intact from the ITAs using the fuel preparation machine, located in the HCGS SFP, after achieving the desired specific activity of the Co-60 targets for shipping to the GEH facility. The NRC staff requested the licensee to provide additional information regarding the possibility that the SFP wall will undergo significant heating during the removal process.

In response to the staff's request, GEH/PSEG performed a gamma energy deposition analysis and calculation, as well as thermal analysis and calculations for the gamma heating of the SFP wall during the removal process (References 3 and 53). The gamma energy deposition analysis and calculations performed were similar to the calculations presented in SE Sections 3.5.7.1.1 and 3.5.7.1.2.

The licensee has indicated that during the removal process, the closest the ITAs will be to the SFP wall is approximately 14.75 inches, and it is expected that an ITA will normally be in the fuel preparation machine for [[ ]] Due to the close proximity of the GE14i bundle to the SFP wall during the removal process, the incident flux calculated in SE Section 3.5.7.1.1 yielded 1.0×10^{11} MeV/(cm^2-sec) 1 foot from the SFP wall. Therefore, the licensee performed detailed concrete heatup calculations using the energy deposition rate for a GE14i bundle in the fuel preparation machine.

The heatup calculation was performed at a conservative distance of 12 inches from the fuel pool wall. Fission product and cobalt activity were calculated as described in SE Sections 3.5.7.1.1 and 3.5.7.1.2. The temperature rise in the concrete was calculated for this energy deposition using an ANSYS Release 11.0 SP-1, ANSYS Incorporated, finite element analysis. The calculation assumed conservative parameters for energy deposition and temperature rise in the concrete. The concrete wall temperature profile resulting from the conservative material property energy deposition is illustrated in Figure 3 of Reference 53. The maximum temperature reached is [[ ]] which is [[ ]] higher than the peak temperature of the SFP water under normal operating conditions. Since the maximum calculated temperature is less than 65°C (149°F), no special consideration needs to be given to temperature effects as specified in ANSI/ANS-6.4-2006 standard.

Therefore, the NRC staff concluded that the temperature rise due to gamma heating has no detrimental effect on the concrete and there is no probability of significant gamma heating in the SFP wall for any period of time while the GE14i ITA is in the fuel preparation machine.

Note, as discussed in SE Sections 2.2 and 3.5.7.1.2, a new license condition will be added that will require that irradiated GE14i bundles be stored at least 4 feet from the walls of the SFP. This license condition relates to the long-term storage of the irradiated GE14i fuel bundles in the SFP and is not intended to preclude the activities performed for the short period of time the GE14i bundles are in the fuel preparation machine.

3.5.7.2 Post-Irradiation Handling

Section 4.7.3 of Reference 22 describes various processes during the post-irradiation handling of the cobalt isotope rods. This includes the removal of cobalt isotope rods from the discharged GE14i fuel bundle and their replacement with equivalent steel rods to maintain the integrity of the stored bundle. Following the first cycle of operation, the licensee will perform an ITA fuel inspection during the outage. As part of this, a single rod will be sent to GEH's Vallecitos Nuclear Center in California for inspection.

Fuel rod removal and replacement is performed by GEH's fuel examination services (FES) team using the procedures in place at GNF/GEH.

The steps in the procedure for segmented rod disassembly are listed in Section 4.7.3.3 of Reference 22. The licensee has assured that prior to segmented rod disassembly during the HCGS outage following Cycle 17, a similar segmented rod procedure will be prepared and incorporated into the fuel inspection process.

The licensee has included design features in the manufacturing of the target rods to protect cobalt encapsulation integrity if segment disassembly problems occur. A male/female connection has a thread size that will allow for disassembly after years of irradiation. However, if disassembly under normal conditions is not possible, this will not be a problem since the male end plug of a cobalt isotope rod has been designed with a strategic break zone so that large amounts of torque will force a fracture at this known breaking point, not the location of cobalt targets. Furthermore, the broken male component is locked into the female receptor preventing any debris inside the fuel pool. Prototype tests have shown the failure torque to be high enough to prevent failure during normal operation, but low enough for contingency plans with existing tooling.

3.5.7.3 Post-Irradiation Examination

Post-irradiation examination (PIE) of the GE14i ITA bundle and rods may include all or part of four inspections: poolside visual, poolside gamma scan measurements, poolside combined instrumentation measurement system (COINS), and segmented rod hot cell destructive exam. This PIE plan applies to the end of the first cycle of operation and subsequent fuel cycles and at the bundle's end of life (EOL).

Poolside visual examination may include:

- A full bundle periphery visual examination of all mechanical elements.
- Assessment of rod-to-rod spacing of the cobalt isotope rods relative to nearby fuel rods.
- Assessment of rod growth of the cobalt rods relative to nearby fuel rods.
- Assessment of spacer cells with the cobalt rods removed to verify no abnormal growth.

A GE14i rod visual examination may include: (1) visual examination of one or more cobalt isotope rods after brushing to remove the crud; and (2) visual examination of one or more brushed fuel rods adjacent to cobalt isotope rods.

Gamma scanning is a non-destructive method that is used to measure the relative fission product inventory in irradiated nuclear fuel rod or the gamma activity of a cobalt isotope rod. A multi-channel analyzer is used to determine the gamma discrete energy levels in order to determine the activity of all isotopes of interest for a decay chain. The gamma scan of the cobalt isotope rods can give the specific isotopic of activity over the rod's length.

The COINS system is used to measure the corrosion and lift-off for a single fuel rod that has been removed from a bundle. The poolside COINS will be performed on cobalt isotope rods in order to non-destructively obtain information about outer surface corrosion and diameter. The segmented rod hot cell destructive examination will be performed on a cobalt rod at the GEH Vallecitos Nuclear Center. The hot cell examination will include the following:

- Vibration and corrosion.
- Inner and outer oxide layer thickness.

The cobalt targets may be inspected for the following:

- Location of specific activities along axis.
- Cobalt target conditions and status of Nickel plating.
- Vibration and corrosion.
- Ease with which cobalt targets are released.

The NRC staff has determined that the results of these examinations can confirm the successful performance of the GE14i bundle design.

3.5.7.4 Occupational and Public Radiation Doses

<u>Design Considerations</u>

[[

]] The potential for cobalt rod failure resulting in Co-59 and Co-60 releases into the reactor coolant system is significantly reduced as compared to standard fuel rods. This occurs since:

- The Co-59 targets are nickel-plated and loaded into a segmented cobalt isotope rod which separates the targets using zircaloy connections at all spacer locations.

- The cobalt target placement rods are double encapsulated with Zircaloy-2 cladding to minimize the potential for release of Co-59 and Co-60 into the reactor coolant system and subsequently into the plant environs. The double encapsulation provides a high degree of cladding integrity due to the second layer of sealed cladding. The GE14i zirconium tubing and components are procured, fabricated, and handled under the same quality controls as standard production fuel rods, and target pellets are handled with similar quality controls as UO_2 pellets.

- The Co-59 activation to Co-60 results in significantly lower heat generation compared to fission that occurs in fuel rods, significantly reducing corrosion rates.

- There are no fission gases generated within the cobalt isotope rods, resulting in lower internal gas pressure than standard fuel rods.

In addition, radioactivity monitoring of the reactor coolant system is routinely provided for Co-60 activity, allowing for early detection of potential leakage, and if necessary, for plant shutdown and removal of the ITAs such as to meet "as low as reasonably achievable" (ALARA) criteria and public dose limits.

<u>ITA handling evaluation to ensure occupational doses are ALARA</u>

Upon completion of the activation cycle, the ITAs are removed from the reactor using existing refueling equipment and are temporarily stored in the spent fuel pool. The process of ITA handling during disassembly will be conducted at a minimum of approximately 9 - 10.5 feet underwater to provide shielding for occupational worker protection. The cobalt rods will be separated in the spent fuel pool [[ ]] using a previously-tested segment disassembly design involving threaded connections. Underwater tooling that is provided to perform the separation of the cobalt isotope rods into segments includes a Fuel Rod Collet Grapple, Fuel Rod Side Grapple, and a Six Rod Universal Storage Rack.

The receiving basket is hung from a depth of approximately 10.5 feet underwater. The limiting dose consideration has been evaluated at the pool surface to be [[ ]] for a completely filled cobalt rod segment receiving basket. These procedures provide for an adequate radiation shielding to keep occupational radiation exposures ALARA consistent with regulatory requirements.

<u>Co-60 rods - packaging and transport</u>

The Co-60 rods will be shipped in a GE Model 2000 cask suitable for shipping cobalt isotope rods. GEH will provide HCGS with a site-specific procedure for cask usage, including removing the cask overpak, transferring the cask to the fuel floor, handling the cask lid, transferring the cask to and from the spent fuel pool cask pit, transferring the cask to the reactor building, and reinstalling the overpak. The existing HCGS reactor building crane is suitable for handling a fully loaded Model 2000 cask which has a gross assembly weight of 33,550 pounds. This is approximately 17 tons, or 13% of the crane capacity of 130 tons.

Comparison of change to regulatory criteria

For public radiation protection, the proposed changes utilize procedures and engineering controls that are sufficient to meet the general design objectives established in GDC-60, "Control of radioactive materials to the environment." The procedures and controls to be used include the use of double-encapsulated cobalt rods using Zircaloy-2 cladding, quality control measures in the manufacturing of the Zircaloy-2 cladding, lower heat generation, lower internal gas pressure, and lack of fission products generated in the cobalt rods. These controls minimize the potential for release of Co-59 and Co-60 into the reactor coolant system and subsequently into the plant environs, thereby providing reasonable assurance that there will be no impact on the licensee's ability to meet the 10 CFR Part 50 Appendix I ALARA criteria for radioactive effluents and 10 CFR 20.1301 public dose limits.

For occupational radiation protection, the proposed change utilizes procedures and engineering controls that effectively incorporate the ALARA principles of time, distance, and shielding by keeping the cobalt rod segments underwater or in the GE Model 2000 cask. These procedures are sufficient to meet the occupational dose ALARA requirements in 10 CFR 20.1101(b) and occupational dose limits in 10 CFR 20.1201.

Based on the above considerations, the NRC staff has concluded that the Co-60 production process: (1) will not adversely impact the licensee's ability to maintain occupational and public radiation doses to within the applicable limits in 10 CFR Part 20, the design objectives of 10 CFR Part 50, Appendix I; and (2) is consistent with the ALARA criteria.

3.6 Evaluation of License and TS Changes

The proposed amendment would revise the HCGS facility operating license (FOL) and TSs as discussed in SE Section 2.2. The NRC staff has reviewed the proposed changes as discussed below.

FOL Changes

The NRC staff finds that new license condition 1.J, revised license condition 2.B.(6), and new license condition 2.B.(7) provide appropriate definition of the licensee's authority concerning the receipt, production, possession, transfer, and use of Cobalt-60.

The NRC staff finds that new license condition 2.C.(23) provides appropriate requirements to provide reasonable assurance that SFP wall integrity will not be effected by gamma heating effects from the GE14i ITAs. This issue is discussed in more detail above in SE Sections 3.5.7.1.1, 3.5.7.1.2, and 3.5.7.1.3.

TS Changes

In a request for additional information (RAI), the NRC staff stated that the proposed changes to TS 5.3.1, as shown in PSEG's application dated December 21, 2009, lacked specific 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. The NRC staff requested the licensee to propose further changes to TS 5.3.1 to address these issues. The staff also referenced TS 4.2.1 of NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," as an example of the information that should be included.

In response to the RAI, the licensee, in its supplement dated May 11, 2010, proposed additional changes to TS 5.3.1 to align it with the information shown in NUREG-1433. Specifically, the licensee proposed that the first paragraph in TS 5.3.1 would read as follows:

> The reactor core shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material and water rods. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

The staff's RAI also requested that TS 5.3.1 be revised to adequately describe the specific design of the ITAs which would be allowed to be inserted into the HCGS reactor. The licensee proposed to revise the second paragraph so that it would read as follows:

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

The words in the "Design Features" portion of the Standard TSs (e.g., NUREG-1433) pertaining to fuel assemblies were developed, in part, based on the considerations in Supplement 1 to Generic Letter (GL) 90-02, "Alternative Requirements for Fuel Assemblies in the Design Features Section of the Technical Specifications," dated July 31, 1992 (ADAMS Accession No. ML031140545). The supplement to the GL was issued, in large part, to clarify the limitations on the application of currently approved analytical methods used in the analysis of reconstituted fuel. Reconstituted fuel refers to modification of fuel assemblies by substitution of a fuel rod with a replacement rod or filler rod (e.g., if the fuel rod was damaged or leaking). The supplement to the GL also clarified limitations on the use of lead test assemblies (LTAs) in the reactor core. LTAs are used to generate in-reactor operating experience for new fuel assembly design features or materials in order to validate performance and model predictions. The supplement to the GL also provided guidance on application of the provisions of 10 CFR 50.59 when making changes to the reactor core design.

The NRC staff finds that the first paragraph in the proposed HCGS TS 5.3.1 provides sufficient information to address the type of clad, type of fuel, and the type of rods for potential substitution for fuel rods. In addition, the staff finds that the proposed TS provides adequate controls on substitution of fuel rods to provide reasonable assurance that the reactor core will be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded.

The last sentence in the first paragraph of the proposed TS 5.3.1 states that:

> A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

The licensing basis on use of LTAs (also called "lead use assemblies") is defined by the NRC-approved methods in GESTAR II (Reference 23). Specifically, Section 1.2.1.B of GESTAR II cites References 32 and 36 as the basis on the acceptable method for use of LTAs. As discussed in Reference 32, this method allows that "the number of LTAs inserted at any one time are numerically small, less than about 2 percent of the core." As discussed in Reference 36, as long as the analysis of the LTAs uses approved methods and meets the approved criteria, the LTAs may be used without prior NRC review and approval (i.e., uses the provisions of 10 CFR 50.59). As discussed in the licensee's supplement dated May 11, 2010:

> Although the [GE14i ITA] pilot project is not being licensed as an [GESTAR II] 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 the 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 NRC staff reviewed the GE14i ITAs by conducting a technical review which is more detailed than the use of NRC-approved methods in GESTAR II for LTA use. Further, the NRC staff has concluded that the GE14i ITAs are to be considered as part of the total number of fuel bundles that would be allowed to be inserted in the core as part of an LTA program. The NRC will allow up to 15 LTAs/ITAs (2% of the total of 764 fuel assemblies) to be placed in the reactor core in non-limiting core regions. With a maximum number of 12 GE14i ITAs inserted, this would leave an additional 3 LTAs that could be used by the licensee after following the NRC-approved methods of GESTAR II for use of LTAs

The NRC staff finds that the proposed change to TS 5.3.1, with respect to LTAs, clarifies the licensee's current authority on use of LTAs under the provisions of 10 CFR 50.59, and therefore is acceptable.

The NRC staff finds that the second paragraph in the proposed HCGS TS 5.3.1 provides sufficient information to describe the specific design of the ITAs which would be allowed to be inserted into the HCGS reactor.

Based on the above considerations, the NRC staff finds that the proposed changes to TS 5.3.1 are acceptable.

3.7 Technical Evaluation Conclusion

Based on the discussion in SE Sections 3.1 through 3.6, the NRC staff concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey (NJ) State Official (NJ Department of Environmental Protection (DEP)) was notified of the proposed issuance of the amendment. In a letter dated April 19, 2010 (ADAMS Accession No. ML101170113), the State Official provided the following comments on the proposed amendment:

> The NJ DEP met with PSEG personnel and management responsible for this program to better understand the changes involved. We will review the results of the cycle specific analyses that have not been completed and are committed to be provided by July 8, 2010. We also recommend that the following sentence be added to the proposed revision to Technical Specification Section 5.3.1 for completeness: Details of the GE14i assemblies are contained in NEDC-33529P, "Safety Analysis Report to Support Introduction of GE 14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station," Revision 0, dated December 2009.

In a letter dated August 18, 2010 (ADAMS Accession No. ML102440026), the State Official provided the following comments:

> In our letter to you dated April 19, 2010, we committed to review the cycle specific reload analyses when completed. We have reviewed the Supplemental Reload Licensing Report for Hope Creek Cycle 17 that was submitted by PSEG Nuclear to the NRC in a letter dated August 3, 2010. As a result of this review our only comment continues to be recommending that the following sentence be added to the proposed revision to Technical Specification Section 5.3.1 for completeness: Details of the GE14i assemblies are contained in NEDC-33529P, "Safety Analysis Report to Support Introduction of GE 14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station," Revision 0, dated December 2009.

The licensee, in Attachment 7 to its supplement dated May 11, 2010, proposed further changes to TS 5.3.1 consistent with the change proposed by the State Official.

5.0 PUBLIC COMMENTS

On March 2, 2010, the NRC staff published a "Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing," in the *Federal Register* associated with the proposed amendment request (75 FR 9445). In accordance with the requirements in 10 CFR 50.91, the notice provided a 30-day period for public comment on the proposed no significant hazards consideration (NSHC) determination. Public comments were received regarding the proposed amendment (References 11 through 15). Some of the issues discussed in the public comments do not specifically pertain to the proposed NSHC determination. However, the NRC staff has addressed both the issues within the scope of the proposed NSHC and those that are not within the scope. A summary of the comments and the NRC staff responses, grouped by issue, are addressed below.

5.1 Proposed Changes to TS 5.3.1

Public Comment

In the licensee's application dated December 21, 2009, PSEG proposed to modify TS 5.3.1, in part, to state that:

> Each GE14i assembly contains a small number of Zircaloy-2 clad isotope rods containing Cobalt-59.

In Reference 14, a public comment was made that the proposed changes to TS 5.3.1 in the licensee's application dated December 21, 2009, did not explicitly limit the number of isotope rods within the GE14i assemblies since the words "small number" are vague. The comment suggested that TS 5.3.1 be revised to replace the above proposed change with words such as:

> Each GE14i assembly contains the number of Zircaloy-2 clad isotope rods specified in NEDC-33529P, dated December 2009.

NRC Response

The NRC staff requested the licensee to address the above public comment in an RAI dated April 8, 2010 (ADAMS Accession No. ML100990403). In response to the RAI, the licensee proposed additional changes to TS 5.3.1 in its supplement dated May 11, 2010. The proposed changes would leave the above-quoted sentence containing the words "small number," as-is. However, the licensee proposed to add the following new 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.

The NRC staff finds that the proposed changes are sufficient to define the number of isotope rods within the GE14i assemblies such that the licensee would need prior NRC approval to increase the number of isotope rods, in accordance with 10 CFR 50.90, rather than make the change under the provisions of 10 CFR 50.59.

5.2 Limiting Fuel Assemblies

Public Comment

In the licensee's application dated December 21, 2009, PSEG stated that cycle-specific analyses will 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.

In Reference 14, a public comment was made that PSEG should make a commitment to not operate the core with the ITAs being the most limiting fuel assemblies.

NRC Response

As discussed above in SE Section 2.2, the proposed amendment will revise TS 5.3.1, "Fuel Assemblies," to state, in part, that the ITAs "may be placed in non-limiting core regions." As such, the NRC staff concludes that a regulatory commitment is not necessary.

5.3 Gamma Radiation Effects on Spent Fuel Pool Walls - Fuel Preparation Machine

Public Comment

In Reference 14, a public comment was made stating that there was an inconsistency in the licensee's application dated December 21, 2009. Specifically, the application states that due to gamma radiation heating effects on concrete, GE14i assemblies are restricted to a location 4 feet from the spent fuel pool walls. The application also states that the cobalt isotope rods will be removed intact from the ITAs using the fuel preparation machine in the spent fuel pool. The public comment referenced Section 9.1.4.2.3.1 of the HCGS UFSAR which states that the fuel preparation machine is mounted on the wall of the fuel storage pool (i.e., closer than 4 foot restriction stated in the application).

NRC Response

The NRC's evaluation of the issue is discussed above in SE Section 3.5.7.1.3. The NRC staff concluded that the temperature rise due to gamma heating has no detrimental effect on the SFP concrete and that there is no probability of significant gamma heating in the SFP wall for any period of time while the ITA is in the fuel preparation machine.

5.4 Gamma Radiation Effects on Reactor Vessel Internal Components

Public Comment

In Reference 14, a public comment raised concerns about the gamma radiation effects from the GE14i assemblies on reactor vessel internal components.

NRC Response

The NRC's evaluation of the issue is discussed above in SE Section 3.4.3. The NRC staff concluded that the gamma radiation effects on incore instrumentation or vessel internals from the cobalt isotope rods will be bounded by the effects from a fuel rod at that location.

5.5 Potential Fuel Loading Errors in Core

Public Comment

In Reference 14, a public comment raised concerns about potential fuel loading errors and their effect due to introduction of the GE14i assemblies into the HCGS core.

NRC Response

As discussed above in SE Section 3.5, the list of analyzed events, for the planned introduction of the GE14i assemblies into the HCGS core, included a mislocated fuel assembly and a misoriented fuel assembly. Results of the reload analyses are documented in Reference 7. The analyses were performed using NRC-approved methods.

5.6 Substitution of Administrative Controls for Design Features

Public Comment

In Reference 14, a public comment raised concerns that PSEG was substituting administrative controls for design features. Specifically, the comment raised a concern regarding potential mislocation of one or more of the GE14i assemblies into storage locations within 4 feet of the spent fuel pool walls thus causing a potential for damage to the structural integrity of the spent fuel pool (i.e., due to gamma radiation heating effects).

NRC Response

In Attachment 8 to the licensee's application dated December 21, 2009, PSEG made a regulatory commitment to "[r]evise applicable Spent Fuel Pool Storage procedures to require storage of irradiated GE14i fuel bundles at least four feet from the wall of the SFP."

In an RAI dated April 8, 2010 (ADAMS Accession No. ML100990403), the NRC staff stated, in part, that:

> Consistent with the guidance in SECY-98-224, "Staff and Industry Activities Pertaining to the Management of Commitments Made by Power Reactor Licensees to the NRC," dated September 28, 1998 (ADAMS Accession No. ML992870043), and NRR Office Instruction LIC-100, "Control of Licensing Bases for Operating Reactors" (ADAMS Accession No. ML010660227), escalating a licensee commitment into a legally binding requirement should be reserved for matters that warrant: (1) inclusion in the TSs based on the criteria in 10 CFR 50.36; or (2) inclusion in the license based on determination that the issue is of high safety or regulatory significance.

> Please propose suitable legally binding requirements for storage of the ITAs in the SFP.

In response to the RAI, the licensee, in its supplement dated June 10, 2010, proposed to add a new license condition which would state:

> Irradiated GE14i fuel bundles shall be stored at least four feet from the wall of the Spent Fuel Pool.

The NRC staff finds that the proposed license condition provides suitable legally binding requirements for storage of the ITAs in the SFP consistent with the guidance provided in SECY-98-224.

The NRC's evaluation of the gamma radiation heating effects from the GE14i assemblies on the spent fuel pool walls is discussed above in SE Section 3.5.7.1. The NRC staff concluded that the GE14i assemblies will have negligible heating effect on the concrete when stored at least 4 feet from the spent fuel pool wall.

5.7 <u>Safety Culture at HCGS</u>

Public Comment

Comments were made raising concerns about the safety culture at HCGS (References 13 and 15). The comments cited a number of items reported in the news media and in NRC letters to the licensee in the 2003 to 2006 timeframe such as:

- Substantive cross-cutting issue related to ability of the licensee to identify problems and resolve them effectively, particularly regarding instances of ineffective problem evaluations and untimely, ineffective corrective actions.
- Willingness of the licensee to defer needed maintenance.
- Licensee prioritizing production such that it had a negative impact on safety.
- Plant being in danger of creating an unacceptable chilled environment for raising issues and making appropriate operational decisions.

Based on the above issues, the commenter's concluded that it was too soon to add the additional burden of a pilot project that could prove distracting from the necessary improvements the plant must make to improve its safety performance. The commenter's also stated that the plant should focus on improving its safety program rather than adding new procedures or products.

NRC Response

The issues raised in the public comments relate to safety conscious work environment (SCWE) issues identified by the NRC staff as part of the reactor oversight process at HCGS. As a result of these issues, the NRC staff increased inspection oversight in the area of SCWE at HCGS in August 2004. The increased oversight included a number of inspections and other actions, such as increased senior NRC management site visits to monitor the licensee's progress in addressing the SCWE issues. As discussed in a letter from the NRC to the licensee dated August 31, 2006 (ADAMS Accession No. ML062430643), the NRC staff determined that substantial, sustainable improvements in the SCWE had been achieved by the licensee. Based on these results, the substantive cross-cutting issue in the SCWE area was closed. The NRC staff is currently performing baseline inspections at HCGS and there are no current SCWE issues. Accordingly, the SCWE issues were deemed to not be applicable to the NRC staff decision regarding whether or not the license amendment request should be granted.

5.8 Transportation Risks

Public Comment

Comments were made raising the concern about the risks in transporting Cobalt-60 from HCGS to other facilities, thus opening up a new pathway for accidents or radiation exposure to communities surrounding HCGS and along the route to the end user of the Cobalt-60 (References 13 and 15).

NRC Response

As discussed in Section 4.7.3 of Attachment 4 to PSEG's application dated December 21, 2009 (GEH Report NEDO-33529), following irradiation in the HCGS reactor, the fuel assemblies with cobalt isotope rods will be removed from the reactor along with other used fuel assemblies during a refueling outage. The cobalt isotope rods will be removed from the fuel assemblies and disassembled into segments. The Cobalt-60 segments will be placed into an NRC-approved shipping cask. The cask will be shipped from HCGS to the GEH Vallecitos Nuclear Center (VNC) facility in Sunol, California for examination and subsequent processing for commercial use of the Cobalt-60.

Regulating the safety of the transportation of nuclear materials is the joint responsibility of the NRC and the U.S. Department of Transportation (DOT). The NRC oversees the safety of the transportation of nuclear materials through a combination of regulatory requirements, transportation package certification, quality assurance, inspections, and a system of monitoring to ensure that safety requirements are being met. Organizations are authorized to ship radioactive material in a package approved for use under the general licensing provisions of 10 CFR Part 71. For a transportation package to be certified by the NRC, it must be shown by actual test or computer analysis to withstand a series of hypothetical accident conditions.

The transportation of Cobalt-60 from HCGS to VNC and from VNC to commercial users is not within the scope of the proposed amendment since that aspect is covered under the general licensing provisions of 10 CFR Part 71 and the applicable DOT regulations (e.g., Title 49 of the CFR). Further information regarding specific requirements for transportation of nuclear materials is provided on the NRC's website at: http://www.nrc.gov/materials/transportation.html

5.9 Food Irradiation

Public Comment

Comments were made raising concerns regarding use of Cobalt-60 for irradiation of food (References 13 and 15). The comments indicated that irradiated food has been rejected by consumers and, as such, there is no demand or need for irradiated food.

NRC Response

As discussed in the licensee's application dated December 21, 2009, the Cobalt-60 is ultimately intended for use in the medical industry for use in cancer treatments, and blood and instrument sterilization; in the radiography and security industries for imaging; and in the food industry for cold pasteurization and irradiation sterilization. The entities involved with potential end uses of the Cobalt-60 produced at HCGS are subject to meeting applicable NRC regulations for byproduct materials (e.g., 10 CFR Part 30). However, the end uses of the Cobalt-60 are not specifically within the scope of PSEG's license amendment request.

5.10 Miscellaneous Issues

Public Comment

Comments were made pertaining to the following miscellaneous issues:

- Support for production of Cobalt-60 at HCGS (References 11 and 12).
- Support for construction of more nuclear power plants (Reference 12).

NRC Response

Consistent with the NRC's regulations associated with issuance of a license amendment in 10 CFR Part 50, the scope of the NRC staff review focused on whether there is reasonable assurance that the activities authorized by the amendment can be conducted without endangering the health and safety of the public and will be conducted in compliance with the NRC's regulations. As such, the above issues were deemed to not be applicable to the NRC staff decision regarding whether or not the license amendment request should be granted.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration (75 FR 9445). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to

10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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8.0 REFERENCES

1. PSEG letter (LR-N09-0290) to NRC dated December 21, 2009, "License Amendment Request Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)" (ADAMS Package Accession No. ML093640193).

2. PSEG letter (LR-N10-0163) to NRC dated May 11, 2010, "Response to Request for Additional Information - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)" (ADAMS Package Accession No. ML101390320).

3. PSEG letter (LR-N10-0210) to NRC dated June 10, 2010, "Response to Request for Additional Information - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)" (ADAMS Accession No. ML101830004).

4. PSEG letter (LR-N10-0217) to NRC dated June 24, 2010, "Supplemental Safety Analysis Report Information - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)" (ADAMS Accession No. ML101880037).

5. PSEG letter (LR-N10-0239) to NRC dated June 29, 2010, "Cycle-Specific Stability Analysis - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)" (ADAMS Package Accession No. ML101820083).

6. PSEG letter (LR-N10-0289) to NRC dated July 28, 2010, "Response to Request for Additional Information - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)" (ADAMS Accession No. ML102250203).

7. PSEG letter (LR-N10-0290) to NRC dated August 3, 2010, "Reload 16 Cycle 17 Supplemental Reload Licensing Report (SRLR) - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)" (ADAMS Package Accession No. ML102240335).

8. PSEG letter (LR-N10-0306) to NRC dated August 12, 2010, "Response to Request for Additional Information - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)" (ADAMS Package Accession No. ML102371025).

9. PSEG letter (LR-N10-0341) to NRC dated September 10, 2010, "Supplement - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)" (ADAMS Accession No. ML102640105).

10. PSEG letter (LR-N10-0357) to NRC dated September 17, 2010, "Supplement - License Amendment Request (H09-01) Supporting the Use of Co-60 Isotope Test Assemblies (Isotope Generation Pilot Project)" (ADAMS Accession No. ML102630066).

11. Public Comments on *Federal Register* Notice 75 FR 9445, Letter from William E. Morris to NRC dated March 16, 2010 (ADAMS Accession No. ML100920547).

12. Public Comments on *Federal Register* Notice 75 FR 9445, Letter from William H. Day to NRC dated March 25, 2010 (ADAMS Accession No. ML100920548).

13. Public Comments on *Federal Register* Notice 75 FR 9445, Comments from 243 Individuals to NRC dated March 31, 2010 (ADAMS Accession No. ML100970126).

14. Public Comments on *Federal Register* Notice 75 FR 9445, Letter from David A. Lochbaum, Union of Concerned Scientists, to NRC dated March 26, 2010 (ADAMS Accession No. ML100920549).

15. Public Comments on *Federal Register* Notice 75 FR 9445, Letter from Leigh Davis (Edible Garden Project), Dena Mottola Jaborska (Environment New Jersey), Wenonah Hauter (Food & Water Watch), Amy Goldsmith (New Jersey Environmental Federation), Jeff Tittel (New Jersey Sierra Club), William S. Kibler (Soutch Branch Watershed Association), Jennifer M. Coffey (Stony Brook-Millstone Watershed Association) and Norm Cohen (UNPLUG Salem) to NRC dated April 1, 2010 (ADAMS Accession No. ML100950037).

16. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792).

17. HCGS Calculation H-1-CG-MDC-1795, Revision 5, "Control Rod Drop Accident Radiological Consequences," dated June 7, 2007 (ADAMS Accession No. ML101390315).

18. HCGS Amendment No. 174, "Extended Power Uprate," dated May 14, 2008 (ADAMS Package Accession No. ML081230540).

19. NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (ADAMS Accession No. ML053460347).

20. NRC Report AEB 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," dated December 9, 1998 (ADAMS Accession No. ML011230531).

21. HCGS Amendment No. 134, "Increase in Allowable Main Steam Isolation Valve (MSIV) Leakage Rate and Elimination of MSIV Sealing System," dated October 3, 2001 (ADAMS Accession No. ML012600176).

22. GEH Report NEDC-33529P, Revision 0, "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station," dated December 2009 (ADAMS Accession No. ML093640200). Note, this report which was submitted as Attachment 3 to PSEG's application dated December 21, 2009, is non-

publicly available since it contains GEH proprietary information. A non-proprietary version of the report, NEDO-33539 (Attachment 4 to the application), is publicly available (ADAMS Accession No. ML093640199).

23. GNF Licensing Topical Report NEDE-24011-P-A-16, "General Electric Standard Application for Reactor Fuel (GESTAR II)," dated October 2007 (ADAMS Accession No. ML091340081, non-publicly available).

24. NRC letter to General Electric (GE) dated November 10, 1999, "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, "GESTAR II" - Implementing Improved GE Steady-State Methods," (ADAMS Package Accession No. ML993230387).

25. GE letter (MFN 098-96) to NRC dated July 2, 1996, "Implementation of Improved GE Steady-State Nuclear Methods," (ADAMS Accession No. ML070400507).

26. GE Nuclear Energy Report NEDC-32601-P-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," dated August 1999 (ADAMS Package Accession No. ML003740166).

27. "MCNP-A General Monte Carlo N-Particle Transport Code," Version 5, Los Alamos National Laboratory, LA-UR-03-1987, dated April 2003.

28. GNF Report NEDC-32851P-A, Revision 4, "GEXL14 Correlation for GE14 Fuel," dated September 2007 (ADAMS Package Accession No. ML072620192).

29. GE Nuclear Energy Report NEDC-33076P, Revision 2, "Safety Analysis Report for Hope Creek Constant Pressure Power Uprate," dated August 2006 (ADAMS Accession No. ML062690073, non-publicly available). Publicly available version of report (ADAMS Accession No. ML062690086).

30. HCGS Amendment No. 176, "Hydrogen Water Chemistry Low Power Restriction," dated March 4, 2009 (ADAMS Accession No. ML090280496).

31. GE Report NEDO-31960-A, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," dated November 1995 (ADAMS Legacy Library Accession No. 9603130121).

32. GE letter (MFN-156-81) to NRC dated August 24, 1981, "Lead Test Assembly Licensing," (ADAMS Legacy Library Accession No. 8108280228).

33. NRC Certificate of Compliance No. 9309, Revision 7, dated May 28, 2008, for Model RAJ-II Package (ADAMS Package Accession No. ML081540596).

34. GE Nuclear Energy Report NEDO-11209-04A, Revision 8, "GE Nuclear Energy Quality Assurance Program Description," dated March 31, 1989 (ADAMS Legacy Library Accession No. 8909200061).

35. GNF Licensing Topical Report NEDE-24011-P-A-16-US, Revision 16, "General Electric Standard Application for Reactor Fuel, (GESTAR II) (Supplement for United States)," dated October 2007 (ADAMS Accession No. ML091340082, non-publicly available).

36. NRC letter to GE dated September 23, 1981, "Lead Test Assembly Licensing," (ADAMS Legacy Library Accession No. 8110090006).

37. GE Report NEDE-30130-P-A, "Steady State Nuclear Methods," dated April 1985 (ADAMS Legacy Library Accession No. 8505090321).

38. GE Nuclear Energy Report NEDC-32694P-A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluation," dated August 1999 (ADAMS Package Accession No. ML003740166).

39. NRC letter accepting Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) on Cycle Specific Safety Limit MCPR, dated March 11, 1999 (ADAMS Accession No. ML993140059).

40. GE Licensing Topical Report NEDO-24154-A, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volume 1, dated August 1986 (ADAMS Accession No. ML062690107).

41. GE Licensing Topical Report NEDO-24154-A, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volume 2, dated August 1986 (ADAMS Accession No. ML071070420).

42. GE Licensing Topical Report NEDO-24154-A, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volume 3, dated August 1986 (ADAMS Legacy Library Accession No. 8610240198).

43. GE Nuclear Energy Licensing Topical Report NEDC-32992P-A, "ODYSY Application for Stability Licensing Calculations," dated July 2001 (ADAMS Package Accession No. ML012610606).

44. GE Nuclear Energy Licensing Topical Report NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," dated August 1996 (ADAMS Accession No. ML072260045, non-publicly available).

45. GE Nuclear Energy Report NEDC-32084P-A Revision 2, "TASC-03A - A Computer Program for Transient Analysis of a Single Channel," dated July 2002 (ADAMS Package Accession No. ML100220495).

46. GE Report NEDC-24154-P-A, Supplement 1 - Volume 4, Revision 1, Qualification of the One- Dimensional Core Transient Model for Boiling Water Reactors, dated February 2000.

47. NRC letter to GE dated March 1, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, Amendment 7 to Revision 6, "GE Standard Application for Reactor Fuel Letter" (ADAMS Legacy Library Accession No. 8503110227, non-publicly available).

48. GE Report NEDE-20566-P-A, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K," Volumes 1-3, dated September 1986 (ADAMS Legacy Library Accession No. 8705070021).

49. GE Licensing Topical Report NEDE-23785-1-PA, Revision 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident," Volumes II and III, dated October, 1984 (ADAMS Accession Nos. ML102230241 and ML102230240, non-publicly available).

50. NRC letter to GEH, "Final Safety Evaluation for GE Hitachi Nuclear Americas, LLC Licensing Topical Report NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains"" dated July 21, 2009 (ADAMS Accession No. ML083530224).

51. GE Energy letter (MFN 07-040) to NRC dated January 21, 2007, "Part 21 Notification: Adequacy of GE Thermal-Mechanical Methodology, GSTRM," (ADAMS Package Accession No. ML072290203).

52. HCGS Amendment No. 160, "Elimination of Requirements for Hydrogen Recombiners and Hydrogen/Oxygen Monitors Using the Consolidated Line Item Improvement Process," dated August 9, 2005 (ADAMS Package Accession No. ML052220436).

53. Attachment 4 to PSEG letter (LR-N10-0289) to NRC dated July 28, 2010 (see Reference 6 above), GEH Document DRF 0000-0103-7804, Revision 0, "Response to NRC RAIs for Hope Creek Generating Station, Related to "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Hope Creek Generating Station," Draft RAI 17 and 18," (ADAMS Accession No. ML102250205, non-publicly available).

54. American National Standards Institute (ANSI) Standard ANSI/ANS-6.4-2006, "Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants," dated September 2006.

55. NRC Report NUREG/CR-6927, "Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures - A Review of Pertinent Factors," dated February 2007 (ADAMS Accession No. ML070850183).

Table L1
Post-LOCA Dose (rem-TEDE) With GE 14i Fuel Rods

Post-LOCA Activity Release Path	Control Room	EAB	LPZ
Containment Leakage	5.28E-01	3.87E-01	1.47E-01
ESF Leakage	2.42E+00	5.22E-01	2.64E-01
MSIV Leakage	9.99E-01	2.20E+00	4.79E-01
CR Filter Shine	1.29E-02	0.00E+00	0.00E+00
Total	**3.96E+00**	**3.11E+00**	**8.90E-01**
10 CFR 50.67 Limit	**5.00E+00**	**2.50E+01**	**2.50E+01**

Table L2
Parameters and Assumptions Used in Radiological Consequence Calculations for a LOCA

Parameter	Amendment No. 174	GE 14i ITAs
Reactor power	**4,031 MWt (max. discharge bundle exposure)**	**3917 MWt (Core Average)**
Drywell air volume	1.69E+5 ft^3	1.69E+5 ft^3
Containment air volume	3.06E+5 ft^3	3.06E+5 ft^3
Primary Containment Dilution Volume		
0 – 2 hours	**3.06E+5 ft^3**	**1.69E+5 ft^3**
2 hours – 30 days	**3.06E+5 ft^3**	**3.06E+5 ft^3**
Reactor building air volume	4.0E+6 ft^3	4.0E+6 ft^3
Containment leak rate to environment		
0 -24 hours	0.5% per day	0.5% per day
1 -30 days	**0.25% per day**	**0.5% per day**
Reactor building pressure drawdown time	375 seconds	375 seconds
Aerosol deposition rate in drywell	10 percentile in RADTRAD	10 percentile in RADTRAD
Elemental iodine deposition rate in containment	**Not credited**	**SRP 6.5.2 Methodology**
Elemental iodine decontamination factor	**Not credited**	**200**
Reactor building mixing efficiency	50%	50%
FRVS vent exhaust filter efficiencies		
Elemental iodine	90%	90%
Organic iodine	90%	90%
Aerosol (particulate)	99%	99%
FRVS recirculation filter efficiencies		
Elemental iodine	Not credited	Not credited
Organic iodine	Not credited	Not credited
Aerosol (particulate)	99%	99%
FRVS recirculation flow rate	1.08E+5 cfm	1.08E+5 cfm
ECCS leak rate	**1 gpm**	**2.85 gpm**
ECCS iodine partition factor	10%	10%
ECCS leak initiation time	0 minutes	0 minutes
Sump volume	1.18E+5 ft^3	1.18E+5 ft^3

Table L2
Parameters and Assumptions Used in Radiological Consequence Calculations for a LOCA

Parameter	Amendment No. 174	GE 14i ITAs
MSIV leak rate **Into Steam Line**		
All four lines	250 scfh **(4.167 cfm)**	250 scfh **(1.347 cfm)**
Line with MSIV failed	150 scfh **(2.50 cfm)**	150 scfh **(0.808 cfm)**
First intact line	50 scfh **(0.8333 cfm)**	**100 scfh (0.539 cfm)**
Second intact line	50 scfh **(0.8333 cfm)**	
MSIV leak rate **To Environment**		
All four lines	250 scfh **(4.167 cfm)**	250 scfh **(7.966 cfm)**
Line with MSIV failed	150 scfh **(2.50 cfm)**	150 scfh **(4.783 cfm)**
First intact line	50 scfh **(0.8333 cfm)**	**100 scfh (3.183 cfm)**
Second intact line	50 scfh **(0.8333 cfm)**	
Aerosol settling velocity on main steamlines	8.1E-4 meters/second	8.1E-4 meters/second
Aerosol settling area (well-mixed region volumes)		
MSIV faulted line	**1398 ft^3**	**1065 ft^3**
MSIV intact lines	**1476 ft^3**	**1062 ft^3**
Steam line transport model	**Plug flow**	**Well mixed**
Steam line deposition period	**0 – 30 days**	**0 – 96 hours**
Control room volume	8.5E+4 ft^3	8.5E+4 ft^3
CREFS outside air intake flow	1000 cfm	1000 cfm
CREFS recirculation flow	2600 cfm	2600 cfm
Control room isolation time	30 minutes	30 minutes
Unfiltered air in leakage rate into control room		
0 to 30 minutes	500 cfm	500 cfm
30 minutes to 30 days	**350 cfm**	**250 cfm**
CREFS filter efficiencies		
Elemental iodine	99%	99%
Organic iodine	99%	99%
Aerosol (particulate)	99%	99%

The Nuclear Regulatory Commission (NRC) staff has determined that its safety evaluation (SE) for the subject amendment contains proprietary information pursuant Title 10 of the *Code of Federal Regulations*, Section 2.390. Accordingly, the NRC staff has prepared a redacted, publicly available, non-proprietary version of the SE. Both versions of the SE are enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/ra/
Richard B. Ennis, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 184 to License No. NPF-57
2. Proprietary SE
3. Non-Proprietary SE

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ADAMS Accession Nos.:
Package: ML102700263
Cover letter and Amendment: ML102700271
Proprietary SE: ML102700301
Non-Proprietary SE: ML102710156 * Concur via e-mail

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