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

January 15, 2010

Mr. Charles G. Pardee
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REQUEST TO MODIFY FACILITY OPERATING LICENSE IN SUPPORT OF THE USE OF ISOTOPE TEST ASSEMBLIES (TAC NO. ME1643)

Dear Mr. Pardee:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 190 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1 (CPS). The amendment is in response to your application dated June 26, 2009 (Agencywide Documents Access and Management System (ADAMS) Package No. ML091801065), as supplemented by letters dated November 4, 2009 (ADAMS Accession No. ML093100316), November 17, 2009 (ADAMS Accession No. ML093210561), November 20, 2009 (ADAMS Accession No. ML093280028), December 9, 2009 (ADAMS Package No. ML093440271), December 14, 2009 (ADAMS Accession No. ML093490375), December 16, 2009 (ADAMS Accession No. ML093510232), December 28, 2009 (ADAMS Accession No. ML093630821), and January 11, 2010 (ADAMS Package No. ML100050199).

The amendment would modify CPS License Condition 2.B.(6) and create new License Conditions 1.J, 2.B.(7) and 2.C.(25) as part of a pilot program to irradiate Cobalt (Co)-59 target to produce Co-60. In addition to the proposed license condition changes, the amendment would modify Technical Specification 4.2.1, "Fuel Assemblies," to describe the Isotope Test Assemblies being used.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission s next biweekly *Federal Register* notice.

Sincerely,

C S Goodwin

Cameron S. Goodwin, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosures:
1. Amendment No. 190 to NPF-62
2. Safety Evaluation (Proprietary)
3. Safety Evaluation (Non-Proprietary)

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 190
License No. NPF-62

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated June 26, 2009, as supplemented by letters dated November 4, November 17, November 20, December 9, December 14, December 16, December 28, 2009, and January 11, 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;

   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, changes to paragraph 2.B.(6) of Facility Operating License (FOL) No. NPF-62, as well as adding paragraphs 1.J, 2.B.(7) and 2.C.(25) to FOL No. NPF-62, and paragraph 2.C.(2) of FOL No. NPF-62 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No.190, are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen J. Campbell

Stephen J. Campbell, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: January 15, 2010

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

ATTACHMENT TO LICENSE AMENDMENT NO. 190

FACILITY OPERATING LICENSE NO. NPF-62

DOCKET NO. 50-461

Replace the following pages of the Facility Operating License and 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
License NPF-62	License NPF-62
Page 2	Page 2
Page 3	Page 3
Page 7a	Page 7a
TSs	TSs
4.0-1	4.0-1

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

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-62, 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;

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; and

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 regarding this facility, and pursuant to approval by the Nuclear Regulatory Commission at a meeting on April 10, 1987, Facility Operating License No. NPF-62, which supersedes the license for fuel loading and low power testing, License No. NPF-55, issued on September 29, 1986, is hereby issued to Exelon Generation Company to read as follows:

A. This license applies to the Clinton Power Station, Unit No. 1, a boiling water nuclear reactor and associated equipment (the facility), owned by Exelon Generation Company. The facility is located in Harp Township, DeWitt County, approximately six miles east of the city of Clinton in east-central Illinois and is described in the licensee's Final Safety Analysis Report, as supplemented and amended, and in the licensee's Environmental Report-Operating License Stage, as supplemented and amended.

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

(1) Exelon Generation Company, pursuant to section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the designated location in Harp Township, DeWitt County, Illinois, in accordance with the procedures and limitations set forth in this license;

(2) Deleted

(3) Exelon Generation Company, pursuant to the Act and 10 CFR Part 70, to receive, possess and to 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) Exelon Generation Company, pursuant to the Act and to 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) Exelon Generation Company, 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;

(6) Exelon Generation Company, 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; and

(7) Exelon Generation Company, 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

Exelon Generation Company is authorized to operate the facility at reactor core power levels not in excess of 3473 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 and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 190 are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.15.d, shall be within 24 months, plus the 6 months allowed by SR 3.0.2, as measured from the date of the most recent successful pressure measurement test, or within 6 months if not performed previously.

(24) At the time of the closing of the transfer of CPS and the respective license from AmerGen Energy Company, LLC (AmerGen) to Exelon Generation Company, AmerGen shall transfer to Exelon Generation Company ownership and control of AmerGen Clinton NQF, LLC, and AmerGen Consolidation, LLC shall be merged into Exelon Generation Consolidation, LLC. Also at the time of the closing, decommissioning funding assurance provided by Exelon Generation Company, using an additional method allowed under 10 CFR 50.75 if necessary, must be equal to or greater than the minimum amount calculated on that date pursuant to, and required by 10 CFR 50.75 for CPS. Furthermore, funds dedicated for CPS prior to closing shall remain dedicated to CPS following the closing. The name of AmerGen Clinton NQF, LLC shall be changed to Exelon Generation Clinton NQF, LLC at the time of the closing.

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

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include: (a) an exemption from the requirements of 10 CFR 70.24 for the criticality alarm monitors around the fuel storage area; (b) an exemption from the requirement of 10 CFR Part 50, Appendix J – Option B, paragraph III.B, exempting the measured leakage rates from the main steam isolation valves from inclusion in the combined leak rate for local leak rate tests (Section 6.2.6 of SSER 6); and (c) an exemption from the requirements of paragraph III. B of Option B of 10 CFR Part 50, Appendix J, exempting leakage from the valve packing and the body-to-bonnet seal of valve 1E51-F374 associated with containment penetration 1MC-44 from inclusion in the combined leakage rate for penetrations and valves subject to Type B and C tests (SER supporting Amendment 62 to Facility Operating License No. NPF-62). The special circumstances regarding each exemption, except for item (a) above, are identified in the referenced section of the safety evaluation report and the supplements thereto.

4.0 DESIGN FEATURES

4.1 Site Location

The site for the Clinton Power Station is located in Harp Township, DeWitt County, approximately six miles east of the city of Clinton in east-central Illinois. The exclusion area boundary shall have a radius of 975 meters from the Standby Gas Treatment System vent.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 624 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material, and water rod(s). 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 applicable NRC staff approved codes and methods and 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 nonlimiting core regions.

A maximum of eight GE14i isotope test assemblies will be placed in non-limiting core regions, beginning with the Reload 12 Cycle 13 core reload, with the purpose of obtaining surveillance data to verify that the GE14i assemblies perform satisfactorily in service prior to use of these design features on a production basis. Each GE14i assembly contains a small number of zircaloy-2 clad isotope rods that contain Cobalt-59 targets. These Cobalt-59 targets will transition into Cobalt-60 isotope targets during the cycle irradiation of the assemblies. Details of the GE14i assemblies are contained in NEDC-33505P, "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Clinton Power Station," Revision 0, dated June 2009.

4.2.2 Control Rod Assemblies

The reactor core shall contain 145 cruciform shaped control rod assemblies. The control material shall be boron carbide or hafnium metal, or both.

(continued)

Enclosure 2 contains proprietary information and is not included.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. NPF-62

EXELON GENERATION COMPANY, LLC

CLINTON POWER STATION, UNIT NO. 1

DOCKET NO. 50-461

1.0 INTRODUCTION

By application dated June 26, 2009 (Agencywide Documents Access and Management System (ADAMS) Package No. ML091801065), as supplemented by letters dated November 4 (ADAMS Accession No. ML093100316), November 17 (ADAMS Accession No. ML093210561), November 20 (ADAMS Accession No. ML093280028), December 9 (ADAMS Package No. ML093440271), December 14 (ADAMS Accession No. ML093490375), December 16 (ADAMS Accession No. ML093510232), December 28, 2009 (ADAMS Accession No. ML093630821), and January 11, 2010 (ADAMS Package No. ML100050199), Exelon Generation Company, LLC (EGC) requested changes to the Technical Specifications (TSs) for Clinton Power Station, Unit No. 1 (CPS). The proposed amendment would modify CPS License Condition 2.B.(6) and would create new License Conditions 1.J, 2.B.(7) and 2.C.(25) as part of a pilot program to irradiate cobalt (Co)-59 targets to produce Co-60. EGC is collaborating with Global Nuclear Fuel – America LLC (GNF) and GE – Hitachi Nuclear Energy Americas, LLC (GEH) to develop and implement a pilot program for producing Co-60 in the CPS reactor during power generation. In addition to the proposed license condition changes, EGC also requests an amendment to Appendix A, TS, of the CPS Operating License. This proposed amendment would modify TS 4.2.1 "*Fuel Assemblies*" to describe the Isotope Test Assemblies (ITAs) being used.

Co-60 is intended for use in the medical industry for cancer treatments, blood and instrument sterilization, radiography, in the security industry for imaging, and in the food industry for cold pasteurization or irradiation sterilization. EGC is proposing to load eight ITAs as part of the CPS Reload 12 Cycle 13 core reloads during the January 2010 refueling outage (C13R12).

The Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's regulatory and technical analyses in support of the proposed license amendment, which are described in the licensee's June 26, 2009, application and its supplements. This safety evaluation supports the conclusion 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.

The supplemental information provided by letters dated November 20, December 9, December 14, December 17, December 28, 2009 and January 11, 2010, contained clarifying information and did not expand the scope of the original *Federal Register* notice.

2.0 REGULATORY EVALUATION

The CPS core consists of 624 fuel assemblies of the GE14 type. Each GE14 fuel assembly consists of 92 fuel rods and two large central water rods which are arranged in 10 x 10 lattice array arranged in a channel surrounding the fuel bundle. Fourteen of these fuel rods are part-length rods. The rods are spaced and supported by the upper and lower tieplates as well as fuel rod spacers. The fuel rods consist of high density cylindrical ceramic uranium dioxide (UO_2) pellets, stacked within Zircaloy cladding which is evacuated, backfilled with helium to a specified pressure and sealed with Zircaloy end plugs welded on each end. GE14 fuel design contains two large central water rods which are hollow Zircaloy tubes with several holes around the circumference near each end to allow coolant to flow through.

The fuel channel is open at the bottom. The channel is attached using threaded channel fastener assembly, which also includes the fuel assembly positioning spring. Channel-to-channel spacing is provided by means of spacer buttons located on the upper portion of the channel adjacent to the control rod passage area. The boiling-water reactor (BWR) Zircaloy fuel channel (1) forms the fuel bundle flow path outer periphery for bundle coolant flow, (2) provides surfaces for control rod guidance in the core, (3) provides structural stiffness to the fuel bundle during lateral loadings, (4) minimizes coolant bypass flow at the channel/lower tieplate interface, (5) transmits fuel assembly seismic loadings to the top guide and fuel support of the core internal structures, (6) provides heat sink during loss-of-coolant accident (LOCA), and (7) provides a stagnation envelope for in-core fuel sipping.

EGC is collaborating with GNF and GEH to develop and implement a pilot program for producing Co-60 in the CPS reactor during power generation. EGC plans to load 8 ITAs with Co-59 in to the CPS core as part of the CPS Reload 12 Cycle 13 core reload during the January 2010 refueling outage (C13R12). These ITAs are referred to as GE14i ITAs. The GE14i bundle is identical to the GE14 bundle with the exception of [ ] cobalt isotope rods with cobalt-59. The purpose of the ITA program is to obtain surveillance data to verify that the modified GE14 fuel assemblies with the design features of GE14i ITAs perform satisfactorily in service, prior to use of those features on a production basis on a future date. This proposed test program would result in the introduction of a new type of fuel assembly to the CPS core. Technical design details of GE14i ITAs are in Reference 9.3 and are briefly discussed in Section 3.0 of this safety evaluation.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.90, "*Application for amendment of license, construction permit, or early site permit,*" EGC requests an amendment to the Facility Operating License for CPS to support the pilot program to irradiate Co-59 targets to produce Co-60.

EGC proposes to add a new license Condition 1.J which states:

> *"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."*

This proposed new condition will allow for the production and transfer of Co-60 under the CPS Facility Operating License in accordance with 10 CFR Part 30.

The proposed change modifies CPS license condition 2.B.(6). Condition 2.B.(6) of CPS is revised to state:

> *"Exelon Generation Company, 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."*

This requested change is intended to provide clarification of the term "separation" relative to the removal of enclosed rods containing Co-60 from the CPS core as distinguished from separation of byproduct material from the special nuclear material fuel rods as intended by the original license restriction.

EGC also proposes to add new License Condition 2.B.(7) which states:

> *"Exelon Generation Company, pursuant to the Act and 10 CFR Part 30, to intentionally produce, possess, receive, transfer, and use Cobalt-60."*

This proposed new License Condition is intended to support the pilot bulk isotope generation project at CPS by allowing intentional production of byproduct material during the operation of the CPS facility.

License Condition 2.C.(25) will also be added to the license which states:

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

This new License Condition is intended to ensure that the spent fuel pool walls are not heated from the Co-60 being stored in the spent fuel pool.

Finally, the applicant proposes to add the following paragraph to the end of TS 4.2.1 that will ensure the TS accurately describe all types of fuel assemblies used in the CPS reactor core:

> *"A maximum of eight GE14i isotope test assemblies will be placed in non-limiting core regions, beginning with the Reload 12 Cycle 13 core reload, with the purpose of obtaining surveillance data to verify that the GE14i assemblies perform satisfactorily in service prior to use of these design features on a production basis. Each GE14i assembly contains a small number of zircaloy-2 clad isotope rods that contain Cobalt-59 targets. These Cobalt-59 targets will transition into Cobalt-60 isotope targets during the cycle irradiation of the assemblies. Details of the GE14i assemblies are contained in NEDC-33505P, "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Clinton Power Station," Revision 0, dated June 2009."*

EGC states that it has evaluated the proposed changes to ensure applicable regulations and requirements have been satisfied. EGC states that it has verified that CPS will continue to meet the requirements of 10 CFR Part 30, *Rules of General Applicability to Domestic Licensing of Byproduct Material* and 10 CFR Part 40, *Domestic Licensing of Source Material.*

Section 50.36(c)(4) of 10 CFR, *Technical Specifications* stipulates that "*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 in paragraphs (c) (1), (2), and (3) of this section.*" The requested change that involves ITA design, construction and materials specifications has no adverse impact on safety. The only required change to the TS is the proposed clarification to TS 4.2.1 that briefly describes the implementation of the ITA program in which Co-60 is intentionally produced.

The licensee states that it has assured that it shall continue to meet the requirements of 10 CFR Part 50, "*Domestic licensing production of and utilization facilities.*" EGC states that the proposed amendment does not require any exemptions or relief from regulatory requirements. Further, the proposed amendment do not affect CPS conformance with any General Design Criteria (GDC), specifically Criteria 10, 11, 12, 13 and 29. (Reference 9.20)

This safety evaluation also addresses the impact of the proposed changes on previously analyzed design basis accident (DBA) radiological consequences and the acceptability of the revised analysis results. The regulatory requirements upon which the Accident Assessment Dose Branch (AADB) based its review are the accident dose guidelines in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," as supplemented by accident-specific criteria in Section 15, "Transient and Accident Analyses," of NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plant, LWR Edition," the accident dose criteria in 10 CFR 50.67, "Accident source term," as supplemented in Regulatory Position 4.4 of Regulatory Guide 1.183 (RG1.183), "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and 10 CFR Part 50 Appendix A, General Design Criterion 19 (GDC-19), "Control Room," as supplemented by Section 6.4 of the SRP. Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulations and regulatory guidance provided in the following documents to perform this review:

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors;"
- SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Term;"
- 10 CFR 100, "Reactor Site Criteria;"
- NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants;" and
- 10 CFR Part 50 Appendix A, General Design Criterion 19, "Control Room."

Since the guidance identified above was written for low-enriched uranium (LEU) fuel, the NRC staff had to consider appropriate changes related to the GE 14i isotope test assemblies (ITAS). These adjustments will be addressed in the Technical Evaluation section.

The AADB staff also considered relevant information in the CPS updated final safety analysis report (UFSAR) and technical specifications.

Shipping of fresh and irradiated ITAs will be in accordance with 10 CFR 71, "Packaging and Transportation of radioactive Material," and 49 CFR 173.

3.0 TECHNICAL EVALUATION

3.1 Introduction

EGC is collaborating with GNF and GEH to develop and implement a pilot program for producing Co-60 in the CPS reactor during power generation. EGC plans to load eight ITAs with Co-59 in to the CPS core as part of the CPS Reload 12 Cycle 13 core reload during the January 2010 refueling outage.

Each isotope rod in the GE14i ITA contains Co-59 pellets. 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 fuel rod operating times that 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 CPS core for two to three 2-year cycles or less, as long as they do not exceed the exposure limit and depending on 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 Ci/gram and low specific activity (LSA) as cobalt activity less than 200 Ci/g. The Exelon/GEH program intends to produce HSA cobalt at CPS.

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 mega-electron volt (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 GE off-site facilities for separation and processing.

Attachment 3 of Reference 9.1 (NEDC-33505P, "Safety Analysis report to Support Introduction of GE14i ITAs in CPS (proprietary)," Reference 9.3) contains details regarding the design features of GE14i ITA, CPS 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), and thermal-mechanical evaluations. This safety evaluation report contains an assessment of the applicants' description of the ITAs, as well as nuclear core design and applicability of nuclear, safety analysis methods, licensing evaluations that were performed by GNF and GEH in support of the introduction of these new fuel assemblies to the CPS core.

The NRC staff has also evaluated whether the facility amendment involves a no significant hazards consideration, if operation of the facility in accordance with the proposed amendment as per 10 CFR 50.92, *Issuance of amendment,* would not involve (1) a significant increase in the probability or consequence of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Based on the evaluation, the NRC staff has concluded that the proposed amendment does not involve any significant hazards under the standards set forth in 10 CFR 50.92(c), and, accordingly a no significant hazards consideration determination is justified. (Reference 9.22)

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 10x10 array. The Cobalt isotope rods are positioned in the bundle such that [ ].

GE14i fuel bundle is designed for mechanical, nuclear, and thermal-hydraulic compatibility with the co-resident GE14 design.

The NRC staff has reviewed the design-basis document for the GE14i fuel assembly through auditing process of the document (DB-0011.04, "GE14i Design Basis," Global Nuclear Fuel (Proprietary), April 2009) and agrees the licensee's findings. Highlights of the design aspects are listed below:

- GE14i is a modification of GE14 that replaces a few power rods with segmented isotope rods.

- [
  .]
- Table 1 of DB-0011.04 lists basic information concerning GE14i assembly.
- Figure 2 of DB-0011.04 illustrates the axial spacer and cobalt locations.
- Table 2 of DB-0011.04 shows that cladding on fuel rods is identical to that of GE14. Table 2 also gives the dimensions of GE14i segmented isotope rods and the dimensions of inner and outer tubes of 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 GE14 assembly.
- GE14i limits on burnup, LHGR, power suppression with Gadolinium (Gad) are identical to those of GE14 assembly.

Other new features of GE14i are briefly described below:

<u>Cobalt Target</u>

[

.] 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)

[

].

Inner Tube

[

].

Outer Tube

[

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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 Rod Extensions

[

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[

.]

Defender Lower Tie Plate

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

3.3 Nuclear Design and Methods

3.3.1 Nuclear Core Design

CPS will be inserting eight 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 CPS cycle 13 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 Section 3.3.4).

Selection of non-limiting location 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 statepoint 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 are 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 (Reference 9.2).

[

].
Power suppression options that are used to address these concerns are described in Section 3.3.2.3 of this safety evaluation.

3.3.2 Nuclear Core Design Methods

This section describes the applicability of the current methods and methodologies to the GE14i fuel design. It addresses each of the NRC-approved methodologies (References 9.4, 9.5, and 9.6) 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 isotope bundle, GE14i.

Table 3.2 Summary of GNF Methods Applicability to GE14i

Methodology	Analysis Code and Revision	Supported
Nuclear	TGBLA06	X
	PANAC11	X
Thermal-Hydraulic	ISCOR09	X
Safety Limit MCPR	GESAM02	X
Transient Analysis	ODYNV09	X
	TASC-03	X
Stability	ODYSY05	X
	TRACG04	X
ATWS	TASC-03	X
	ODYNV09	X
Thermal-Mechanical	GSTRM07	X
ECCS-LOCA	LAMB08	X
	TASC-03	X
	SAFER04	X

3.3.2.1 Lattice Physics

Lattice physics calculations are performed using a two-dimensional, fine mesh, few group, transport corrected diffusion theory model, TGBLA06. No modifications to the methodology of TGBLA06 were required to model the GE14i and gadolinium (Gad) rods. The material specifications of the cobalt bearing regions are provided through the standard TGBLA06 input parameters. TGBLA06 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, MCNP5 (Reference 9.8) with ENDFB-VI cross sections.

TGBLA06 generated infinite lattice reactivity, pin fission density distributions, pin power distributions, gamma source distributions and nuclear instrumentation responses are all used in the down stream applications in PANAC11, 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 3-3 of Reference 9.3. These representative uncertainties are consistent with the methodology described in Reference 9.7. Control blade worth from MCNP and TGBLA for various in-channel void fractions are listed in Table 3-2 of Reference 9.3. The current safety limit analysis uncertainties for GE14 are used to model GE14i in this application.

3.3.2.2 Steady-State Core Simulator

PANAC11 is used for design, licensing, and core monitoring of the BWR cores. The combination TRACG04 (thermodynamic stability code) / PANAC11 computer codes are the current GE tool for 3D BWR core physics, Anticipated Operational Occurrences (AOO), and Anticipated Transient without Scram (ATWS) overpressure transients. PANAC11 correctly handles varying axial geometry in nuclear and thermal-hydraulic modeling through use of its lattice-dependent geometry, nodal thermal-hydraulic properties, and axial-meshing routines. This flexibility allows PANAC11 to handle multiple partial-length rods (PLRs), varying rod diameter and other axially varying features. The unique features of GE14i and their impact on PANAC11 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 11 as input quantities. The input parameters which are significant to the processing of thermal limits in PANAC11 are listed in Table 3-4 of Reference 9.3. The NRC staff has determined that no changes to PANAC11 are required to model the thermal performance of GE14i fuel 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 TGBLA06.

The NRC staff agrees that no changes in PANAC11 are required to model the GE 14i fuel design.

Pin Power Reconstruction

The influence of zero-power rods on the PANAC11 pin power reconstruction model was evaluated and no statistically meaningful differences were observed. After reviewing the impact of pin power reconstruction in the GE14i and adjacent fuel assemblies, the licensee determined that the pin power reconstruction was adequate.

The NRC staff has determined that no changes in PANAC 11 are required to model the GE 14i fuel design.

3.3.2.3 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 3-dimensional core simulator program, PANAC11. Instead, based on the 2- and 3-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 2-dimensional models were evaluated with TGBLA06 and the 3-dimensional models were evaluated with PANAC 11 and MCNP5.

[

.] (Reference 9.2)

[

.]

An example of power suppression requirement for the (2, 2) position rod in the dominant zone is illustrated in Figure 2 of Reference 9.2. This analysis is repeated for all rods face or diagonally adjacent to the cobalt rods in all axial zones of the GE14i bundle. The combination result is used to determine the total MLHGR set down required to meet the power suppression requirements. For figure 2 (Reference 9.2) analysis, no MLGHR set down is required prior [
].

Potential reduction in the shutdown margin (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 [ ] 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 GE14i bundle with PANAC11 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.

3.3.3 Analyses Methodology

This section briefly describes the various methodologies used by the applicant for the analysis of GE14i fuel in the CPS core.

Thermal-Hydraulic Methodology

ISCOR09 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 ISCOR09 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 11 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 PANAC11 is capable of modeling the GE14i bundle design for the CPS core.

Safety Limit Methodology

GESAM02 incorporates the implementation of the revised Safety Limit Minimum Critical Power Ratio (LMCPR) methodology using PANAC 11 physics models to calculate CPR distribution. The GE14 uncertainties are used in the evaluation of the safety limit analysis for the CPS. The NRC staff has determined that the capability of GESAM02 to model CPR related uncertainties is adequate for the CPS GE14i bundles and is not impacted by the number of heated rods in GE14i.

Transient Analysis Methodology

GE14i design characteristics will be used in the transient analysis methods of ODYN Version 9 and TASC-03. ODYN 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: 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 ODYN.

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 GE14i in the eight CPS 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 Version 5 (ODYSY05) 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 PANAC11 wrap-up information. The NRC staff has reviewed the provided analyses and agrees that they show that the [ ] zero-power rods in the CPS 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 (1) 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 has determined that the design of the Uranium and Gad rods in the GE14i bundle is identical to the GE14 design and therefore, has no impact on the GSTRM07 methodology.

Emergency Core Cooling System (ECCS) Analysis Methodology

The ECCS analysis methodology applicable to CPS is SAFER/GESTR. 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 in to GESTR is described through SAFER04. 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 percent to 3 percent 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 impact of the [ ] zero-power rods in the CPS GE14i bundles will not impact the adequacy of SAFER/GESTR analysis methodology. The rod-to-rod power distributions and local peaking patterns tested with zero-power rods at Stern Laboratories are presented in Figure 3-5 of Reference 9.3, where cobalt isotope rod(s) or the highest R-factor rod(s) of each pattern are identified. The NRC staff agrees with this analysis.

3.3.4 GEXL+ Correlation

The GEXL, critical quality – boiling length correlation, 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 9.9). 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 applicant 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 and the NRC staff agrees with this determination.

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 9.9.

As a part of the Stern Laboratories 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 3-5 of Reference 9.3.

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 GEH design document [ ] which was audited at the GEH offices by the NRC staff and determined to be acceptable.

The ECPR is determined as:

ECPR = (Predicted Critical Power)/ (Measured Critical Power)

A summary of ECPR statistics provided in Table 3-5 of Reference 9.3 is repeated below.

Number of Data Points	[
Mean ECPR	
Standard Deviation	]

Figure 3-8 of Reference 9.3 compares predicted critical powers to the measured critical powers from GEXL14.

[

.]

Based on the above, 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 9.9) 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 [

].

The NRC staff agrees that the use of a single axial power shape [ ] data only to validate the use of GEXL14 to GE14i is justified primarily based on 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 the lattice design (9x9, 10x10), the part length rod configuration (length, number and location), and spacer designs (type, number, pitch) in the critical quality and boiling length plane independent of the axial power profiles.

3.4 LICENSING EVALUATIONS

Cycle specific limits are established to ensure compliance with licensing limits. Operating limits for the ITAs that assure regulatory compliance are established by performing CPS Cycle 13 reload analysis using NRC-approved methods. Results of the reload analyses are documented in CPS Reload 12 Cycle 13 Supplemental Reload Licensing Report (SRLR) (Reference 9.10). In addition, the licensee is expected to perform core analyses for each cycle of operation subsequent to the initial ITA loading (See Section 4.2).

Licensee has determined and the NRC staff agrees that the events listed in Table 3.3 below are deemed adequate to support the CPS Cycle 13 reload transient analysis.

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

Event		Method
[		
		]

The reactor operating conditions used in the reload licensing analysis for CPS and cycle are listed in Table 3.4 below.

Table 3.4 Reactor Operating Conditions for Cycle 13 Reload Analysis

Parameter	Analysis Value	
	Increased Core Flow Normal Feedwater Temperature	**Increased Core Flow Reduced Feedwater Temperature**
Thermal power, MWt	3473.0	3473.0
Core flow, Mlb/hr	90.4	90.4
Reactor pressure (core mid plane), psia	1056.4	1027.5
Inlet enthalpy, Btu/lb	527.4	508.7
Non-fuel power fraction	0.036	0.036
Steam flow, Mlb/hr	15.17	13.04
Dome pressure, psig	1025.0	997.6
Turbine pressure, psig	936.6	931.2

3.4.1 Evaluation of Abnormal Operational Transients

Cycle specific analyses of the limiting transient events are performed to establish the plant OLMCPR to demonstrate thermal/mechanical compliance and to demonstrate compliance with the ASME overpressure protection criteria.

The CPS Cycle 13 reload licensing analyses include specific modeling of the GE14i ITAs in the determination of the OLMCPR. As discussed in Section 3.3.4 of the safety evaluation, GEXL14 correlation is conservatively applied to the GE14i ITAs. The GE14i ITA uranium and Gad fuel 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 specified in Reference 9.2.

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

3.4.1.1.1 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:

- Maximum Demand Feedwater Controller Failure (FWCF)
- Loss of Feedwater Heating (LFWH)
- Inadvertent High-Pressure Core Spray (HPCS) Pump Start-up
- Pressure Regulator Failure Open (PRFO)
- One relief valve/safety valve (RV/SV) Opening

The CPS Cycle 13 reload licensing analyses includes specific modeling of the GE14i ITAs in determination of the OLMCPR. Plant parameters, which include the [

,] are independent of fuel bundle design and are modeled by methods discussed in Section 3.3.3. [

]

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 eight 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 (Section 3.3.3). Therefore, the GE14 bundles dictate the core average nuclear parameters that affect the transient response. [

] The NRC staff has reviewed the licensee's analyses and determined that they are acceptable.

3.4.1.1.2 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:

- Load Rejection (Turbine Control Valve Fast Closure) with Bypass Failure (LRNBP)
- Load Rejection (Turbine Control Valve Fast Closure) with Bypass (LRWBP)
- Turbine Trip with Bypass Failure (TTNBP)
- Turbine Trip with Bypass (TTWBP)
- Main Steam Isolation Valve Closure with Flux Scram (MSIVF)
- Pressure Regulator Downscale Failure (PRFDS)
- Loss of Auxiliary Power / Loss of Condenser Vacuum
- Loss of Feedwater Flow (LOFW)
- Loss of Instrument Air System

For Cycle 13, as in Cycle 12, the licensee determined that [

] due to reasons specified in Section 3.4.1.1. [

]. 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 Section 3.4.1.1. The MSIVF event is analyzed for overpressure protection and is discussed in Section 3.4.2.2.

The NRC 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, LRNBP and PRFDS events are analyzed as part of the cycle-specific reload licensing analyses. [

] (Reference 9.10)

3.4.1.3 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:

- Trip of One Recirculation Pump
- Trip of Two Recirculation Pumps
- Recirculation Flow Control Failure - Decreasing Flow

The AOOs in this category are bounded by the events in Table 3.3 above. The NRC staff finds that 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. [

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3.4.1.4 Reactivity and Power Distribution Anomalies

AOO events included in this category are those which cause rapid increases in power that are due to increased core flow disturbance events. Increased core flow reduces the void content of the moderator, thereby increasing core reactivity and power level. The events in this category are:

- Rod Withdrawal Error (RWE) at Power
- Mislocated Fuel Assembly Accident
- Misoriented Fuel Assembly Accident
- Startup of an Inactive Recirculation Loop
- Recirculation Manual Controller Failure Increasing Flow

The RWE, Mislocated Fuel Assembly Accident, and the Misoriented Fuel Assembly Accident are potentially limiting events at CPS. [

.]

[

.] 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 agrees that the off-rated limits for CPS are validated as part of reload licensing analyses for application to Cycle 13. (Reference 9.10)

3.4.1.5 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 HPCS Pump Start-up

The NRC staff has determined that in Cycle 13, the Inadvertent HPCS Pump Start-up event will continue to be bounded [

] due to reasons specified in Section 3.4.1.1.

3.4.1.6 Decease in Reactor Coolant Inventory and Other Accidents

Reductions in coolant inventory could result in the coolant becomes 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 DBAs. CPS is a banked position withdrawal sequence (BPWS) plant; therefore, the CRDA analysis is not required per NRC approval documented in Reference 9.21. 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 Section 3.4.5.11 and the radiological consequences of all DBAs are discussed in Section 3.4.3.

3.4.2 Evaluation of Other Transients

3.4.2.1 ATWS

This is an extremely low probability event. This multi-system mal-operation 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 in to 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. All of these features are available at the CPS 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, [

]. 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, 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 CPS is documented in Reference 9.11. The NRC staff has determined that this evaluation demonstrates significant margin to the ATWS acceptance criteria.

3.4.2.2 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 NRC staff agrees that [

.]

3.4.2.3 Single Loop Operation (SLO) Pump Seizure Analysis

This SLO Pump Seizure event was analyzed for GE14 introduction into CPS. The loading of GE14i ITAs will not affect the results of this analysis because a conservative multiplier is applied to the core average void coefficient that results in a void coefficient corresponding to the lower range of values documented in Reference 9.4. The GEXL14 correlation is conservatively applied to the GE14i ITAs; therefore, the NRC staff has determined that [

.]

3.4.2.4 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 General Electric Standard Application for Reactor Fuel (GESTAR II) (Reference 9.4), cycle-independent analyses for a New Fuel Introduction reload application, or as in the case of CPS, plant-specific off-rated limits (Reference 9.10). 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 Section 3.3.4, 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 CPS 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 and have been found acceptable by the NRC staff.

[

] (Reference 9.10).

3.4.2.5 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 12 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 CPS are evaluated as part of the cycle-specific reload licensing analyses. (Reference 9.10)

3.4.3 Design-Basis Accidents and Alternate Source Term (AST)

The CPS DBAs evaluated for Cycle 13 are identified in Chapter 15.0, "Accident Analysis," of the CPS Updated Safety Analysis Report (USAR). The CRDA, MSLB accident outside containment, FHA, and LOCA are licensed under 10 CFR 50.67, "Accident Source Term," utilizing AST methodology per Regulatory Guide (RG) 1.183.

3.4.3.1 Radiological Consequence Analyses

This section provides the NRC staff's evaluation of the DBA analysis results reported in the amendment submittal. The AADB staff reviewed the assumptions, inputs, and methods used by CPS to assess these impacts. When appropriate, the NRC staff performed independent analyses to confirm the conservatism of the CPS analyses. However, the findings of this safety evaluation input are based on the descriptions of CPS's analyses and other supporting information docketed by CPS. Only docketed information was relied upon in making this safety finding.

CPS considered the impact of GE14i ITAS operation on the previously analyzed DBAs. The DBAs considered included:

- CRDA
- LOCA
- FHA
- Cask Drop Accident (USAR 15.7 .5)
- MSLB
- Recirculation Pump Seizure (USAR 15 .3.3)
- Recirculation Pump Shaft Break (USAR 15.3.4)
- Feedwater Line Break Outside Containment (USAR 15.6.6)
- Main Condenser Offgas Treatment System Failure (USAR 15 .7.1 .1)
- Malfunction of Main Turbine Gland Sealing System (USAR 15.7.1 .2)
- Failure of Main Turbine Steam Air Ejector Lines (USAR 15 .7.1 .3)
- Liquid Radwaste Tank Failure (USAR 15.7.3)

<u>CRDA</u>

The CRDA analysis postulates a sequence of mechanical failures that result 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 CPS licensing basis LOCA is analyzed in Section 15.4.9 of the CPS USAR. The impact of eight GE14i ITAs on LOCA radiological consequences was evaluated by CPS. CPS stated:

> The CPS licensing basis CRDA analyzed in Section 15.4.9 of the CPS USAR assumes a failure of 1200 rods (for 10 x 10 fuel). An estimated mass fraction of 0.77 percent of the fuel in the damaged rods is assumed to reach or exceed the initiation temperature of fuel melting. Fuel reaching melt conditions is assumed to release 100% of the noble gas inventory and 50 percent of the iodine inventory. The remaining fuel in the damaged rods is assumed to release 10 percent of both the noble gas and iodine inventories. [
>
> .]. Therefore, the licensing basis CRDA radiological analysis is not impacted by the introduction of eight GE14i assemblies at CPS.

The NRC staff performed independent calculations and confirmed the CPS results as described below. The AADB staff requested additional information regarding the assumption that the release fraction for cobalt in both the cobalt isotope and the GE14i fuel rods is 0.0025 of the core inventory. While the NRC staff agrees with CPS's response that the amount of cobalt remains small compared to amount of fuel and other materials in the core, the NRC staff is not convinced by the justification provided that the assumption is conservative. In a November 20, 2009 response to a request for information, CPS stated that there is no experimental data to support the use of the 0.0025 release fraction for the ITAs. In response to the NRC staff's concern regarding the impact of a higher Co-60 release fraction, CPS performed an assessment assuming a Co-60 release fraction of 0.025 (or ten times the value in RG 1.183). CPS calculated the maximum increase in offsite dose to be 0.4 percent and 0.1 percent for the control room dose.

A study was performed by the NRC staff using the licensee's methodology, as described in Section 3.1.3, "Control Rod Drop Accident," of Reference 9.23, and the higher Co-60 release fraction of 0.025. The NRC staff determined that for the proposed change and using a Co-60 release fraction of 0.025, the proposed change produces a negligible increase in the control room and offsite doses for the CRDA. The use of this release fraction for CPS should not be construed as a precedent that the CPS's Co-60 release fraction value of 0.025 will be found acceptable in another licensing action at CPS or any other reactor site since the 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. In addition, the NRC staff has performed independent calculations and confirmed the licensee's results. The NRC staff finds with reasonable assurance that the licensee's estimates of the total effective dose equivalent due to a CRDA accidents will comply with the requirements of 10 CFR 50.67 and the guidance of RG-1.183.

LOCA

The accident considered is 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 CPS licensing basis LOCA is analyzed in Section 15.6.5 of the CPS USAR. The impact of eight GE14i ITAs on LOCA radiological consequences was evaluated by CPS. CPS stated the following:

[

.]

The NRC staff independently evaluated the impact of the proposed change on the DBA LOCA by performing an assessment of the impact of the proposed change on the postulated primary containment leakage pathway for the LOCA. The assumptions used by the staff are as described in Reference 9.23, Section 3.1.1.4.1, "LOCA – Containment Leakage Pathway."

The AADB staff requested additional information regarding the assumption that the release fraction for cobalt in both the cobalt isotope and the GE14i fuel rods is 0.0025 of the core inventory. While the NRC staff agrees with CPS's response that the amount of cobalt remains small compared to amount of fuel and other materials in the core, the NRC staff was not convinced by the justification provided that the assumption is conservative. In a November 20, 2009 response to a request for information, CPS stated that there is no experimental data to support the use of the 0.0025 release fraction for the ITAs. On December 28, 2009 CPS provided the results of a LOCA assessment which evaluated the impact of increasing the RG 1.183 Co-60 release fraction by a factor of 10 (or 0.025) to determine the sensitivity of the LOCA analysis results to the Co-60 release fraction. The December 28, 2009 response stated that: "The results of the scoping study indicate that there is no observable change in the total dose for the control room within the accuracy of the reported doses. Additionally, this study indicates that there is little impact to the offsite dose even in the unlikely case that Co-60 is released using a potentially increased release fraction."

A sensitivity study was performed by the NRC staff using the methodology in Reference 9.23, Section 3.1.1.4.1 and the Co-60 release fraction of 0.025 to determine whether the LOCA doses were sensitive to the release fraction of Co-60. The NRC staff determined that for the proposed change and using a Co-60 release fraction of 0.025, the proposed change produces a negligible increase in the control room and offsite doses for the LOCA. This finding should not be construed as a precedent that the CPS's melt fraction values will be found acceptable in another licensing action at CPS or any other reactor site since the evaluation is only valid for the current licensing basis as modified by the proposed change.

Based upon this evaluation, the 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 provided Attachment 3 of the June 26, 2009 submittal and a Co-60 release fraction of 0.025.

The NRC staff finds that the licensee's LOCA analysis assumptions and methodology are consistent with the guidance of RG 1.183 with the exception of Co-60 fuel melt assumption which is discussed above. In addition, the NRC staff has performed independent calculations and confirmed the CPS results. The NRC staff finds with reasonable assurance that the licensee's estimates of the total effective dose equivalent due to the LOCA will comply with the requirements of 10 CFR 50.67 and the guidance of RG-1.183.

FHA

The FHA 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 stored fuel bundles. A variety of events which qualify for the class of accidents termed "fuel handling accidents" have been investigated. These included considerations for containment upper pool refueling operations as well as fuel building-pool activities. The accident which produces the largest number of failed spent fuel rods is the drop of a spent fuel bundle onto the reactor core when the reactor vessel head is off.

The CPS licensing basis FHA is analyzed in Section 15.7.4 of the CPS USAR. The impact of eight GE14i ITAs on LOCA radiological consequences was evaluated by CPS.

CPS 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 percent 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 CPS licensing basis FHA is analyzed in Section 15.7.4 of the CPS USAR. The licensing basis FHA postulates that an irradiated GE14 fuel assembly is dropped 34 feet onto the reactor core and fails 172 total rods. Of the failed rods, the entire noble gas fission product inventory and a fraction of the iodine fission product inventory are assumed to be released to the air above the water. All particulates are retained by the water.
>
> [
>
>]. Therefore, the licensing basis FHA radiological analysis is not impacted by the introduction of eight GE14i assemblies at CPS.

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, "Particulate 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)." [

] The NRC staff finds that the assumptions presented and the CPS conclusion that the current FHA analysis is not impacted and is acceptable to the NRC staff. The NRC staff finds with reasonable assurance that the licensee's estimates of the total effective dose equivalent due the FHA will comply with the requirements of 10 CFR 50.67 and the guidance of RG-1.183.

Cask Drop Accident

It is postulated that a spent fuel shipping cask containing irradiated fuel elements is in the process of being removed from the cask storage pool to a shipping car. The cask is suspended from the crane at the maximum crane hook height of approximately 59 feet above the pool floor. This accident is postulated to occur as a consequence of an unspecified failure of the cask lifting mechanism, thereby allowing the cask to fail.

The CPS Cask Drop Accident analysis determined that a dropped cask would not rupture, and no radiological release is associated with this event. Therefore, the radiological consequences are unchanged for operation with ITAs.

Based upon CPS's assessment that the cask will not rupture, the NRC staff agrees that the radiological consequences of this event will remain unchanged and therefore, CPS's use of ITAs will not impact the cask drop accident.

Other Accidents

For the following accident CPS states that the following accidents do not result in fuel damage as a result of the accident, there is no radiological release due to the event, or that the accident is only dependent upon noble gas source terms.

- MSLB
- Recirculation Pump Seizure (USAR 15 .3.3)
- Recirculation Pump Shaft Break (USAR 15.3.4)
- Feedwater Line Break Outside Containment (USAR 15.6.6)
- Main Condenser Offgas Treatment System Failure (USAR 15 .7.1 .1)
- Malfunction of Main Turbine Gland Sealing System (USAR 15.7.1 .2)
- Failure of Main Turbine Steam Air Ejector Lines (USAR 15 .7.1 .3)
- Liquid Radwaste Tank Failure (USAR 15.7.3)

CPS stated in a January 11, 2010 response to a request for additional information that: "During normal operation, transients and design basis accidents other than those involving fuel melt, cobalt isotope rods will not contribute any additional Co-60 to the reactor coolant system and therefore, the cobalt activity in the RCS need not be considered as part of the initial conditions for accident evaluation." Furthermore, CPS states: "Leakage of cobalt (including entire cobalt targets and/or cobalt particulate) from an isotope rod is not a credible event during normal operations, transients or design basis accidents not involving fuel melt accidents (i.e. LOCA and CRDA)."

The NRC staff evaluated these CPS 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 RCS during normal operations. With Co-60 leakage not credible during normal accidents (See Section 3.4.4 of this SE), the proposed change will have no impact on the above accidents. Based upon the licensees statements and the evaluation in Section 3.4.4 (of this SE) the NRC staff agrees that the ITAs will not impact the above eight accidents.

Compliance with licensing limits governing CRDA is assured through adherence to the Banked Position Withdrawal Sequence (BPWS) (Reference 9.21). The associated analyses have generically demonstrated large margin to licensing limits governing acceptable enthalpy insertions. The BPWS analyses demonstrated that the characteristic control rod worth associated with limiting rods in a BPWS sequence are low as compared to that required to challenge the 280 calories per gram (cal/gm) fuel design limit. The reactivity characteristics of GE14i are similar to GE14; therefore, the NRC staff agrees that the introduction of eight GE14i assemblies at CPS will have negligible effects on the existing CRDA margin.

3.4.4 Thermal-Mechanical Evaluation

Thermal-mechanical characteristics of GE14i cobalt isotope rods were evaluated and found acceptable by the NRC staff. The evaluation is contained in the documents that were reviewed during staff's audit of documents related to this license amendment request. Further, the applicant submitted a report to the agency containing the findings of the evaluation (Reference 9.12).

Reference 9.12 evaluates the thermal-mechanical characteristics of GE14i segmented cobalt rod assemblies. The UO_2 and Gad rods in GE14i remain unchanged from those of GE14, therefore, standard uranium dioxide (UO_2) and Gad limits for GE14i bundles are confirmed to be appropriate. 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.

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. The NRC staff agrees with this analysis.

Design Criteria

Although the fuel design criteria in Reference 9.4 are not applicable to the design of GE14i rods, the overall safety criteria defined in Reference 9.4 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 9.4 are sufficient to address cladding integrity. The cladding temperature of GE14i rods is essentially identical to that of GE14 rods [ ]. [ ]

The inputs for the thermal-Mechanical analysis of GE14i rods are consistent with standard fuel rod thermal-mechanical analyses, GSTRM. [:] The NRC staff has reviewed this criteria and finds that it is acceptable.

Evaluation

The following evaluations were performed:

- 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.

[,] thus meeting the internal pressure criteria per Standard Review Plan (Reference 9.13).

The Clad Mechanical Analyses (Fatigue + Creep Rupture & Plastic + Weighted Creep Strain) were performed using NRC approved methodology. The results from these analyses indicate that all calculations meet the design limits with significant margin.

Cladding 1 Percent 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. Reported results indicate that the cladding meets 1 percent strain limit.

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

.]

Analyses were performed to ensure that the cladding does not collapse [

,] The NRC staff finds that GE14i meets the (no) creep collapse requirement. (Reference 9.12)

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

.] In order to ensure that GE14i is [

]. The calculated upper 95 percent 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. (Reference 9.12)
A thermal-mechanical compliance check was performed 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. Since GE14i is less limiting than GE14, the (no) melting criteria will be met for the ITAs for Cycle 13.(Reference 9.10) The NRC staff has concluded that with multiple layers of cladding and design features, there is reasonable assurance that the isotope rod failure will not occur.

3.4.5 Other Evaluations

3.4.5.1 Stability Analysis

An evaluation was performed to assess the impact of GE14i ITAs on thermal-hydraulic instability. Using the methodology in Reference 9.14, 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.

CPS is an Option III plant using the Oscillation Power Range Monitor (OPRM) that uses a microprocessor to monitor groups of Local Power Range Monitor signals as described in Section 15.4 of Reference 9.14. For Option III stability solution, two stability aspects must be considered; the first is the OPRM system setpoint, the second is the Backup Stability Protection System (BSPS). 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, used for stability analysis, computer code acceptance criteria for core and channel decay ratio as specified in Reference 9.15.

A reload Option III stability evaluation was performed in accordance with approved licensing methodology. The stability based OLMCPR as a function of OPRM amplitude setpoint, is determined for two conditions: (1) a postulated oscillation at 45 percent 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 are calculated for Cycle 13 in accordance with the Boiling-Water Reacter (BWR) Owners Group regional mode Delta CPR over Initial MCPR vs. Oscillation Magnitude (DIVOM) guidelines. The Cycle 13 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 15-2, "OPRM Setpoint Versus OLMCPR," of Reference 9.10.

The BSP region boundaries were calculated for CPS Cycle 13 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 15-4, "BSP Region Intercepts for Reduced Feedwater Temperature," and the region boundaries are shown in Figures 8 and 9 of Reference 9.10. The licensee has shown and the NRC staff agrees that Cycle 12 BSP regions are bounding for Cycle 13.

To support the initial introduction of GE14i ITAs into the CPS 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 CPS. 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 CPS core. Table H-1 of Reference 9.10 lists BSP Region Calculated Intercepts for Normal Feedwater Temperature (380°F or higher). Table H-2 of Reference 9.10 lists BSP Region Calculated Intercepts for Reduced Feedwater Temperature (310°F or higher).

3.4.5.2 Appendix R Safe Shutdown Fire

The limiting safety shutdown method for CPS, 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," in Reference 9.11 is Method R which uses Reactor Core Isolation Cooling (RCIC), Safety Relief Valves and Residual Heat Removal (RHR) from a remote shutdown panel. [

] Also 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 fire event evaluation results and acceptance criteria in Reference 9.11 remain applicable for GE14i ITAs.

3.4.5.3 Station Blackout (SBO)

[

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3.4.5.4 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. The licensee has shown that the hex-faced connectors have negligible impact on the pressure drop characteristics of the GE14i fuel (Reference 9.16).

The minimum fuel bundle lift margin is [.] The GE14i bundle weight is [.] 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 in Section 3.4.5.7.

The parameters that are used in the determination of CRGT, such as, [ ] These parameters do not change for GE14i ITAs and thus, the maximum CRGT lift force for GE14 remains applicable for GE14i ITA.

The NRC staff has determined that the introduction of eight GE14i ITAs has no effect on the acoustic and flow-induced loads on jet pump, core shroud and shroud support, which are caused by pressure waves as a result of a recirculation suction line break. [,] which remain unchanged for the GE14i ITA.

3.4.5.5 Reactor Internal Structural Evaluation

An audit was conducted at the GE offices of the documents related to reactor internals structural evaluation.

A qualitative structural assessment of the reactor internal components was performed with respect to the current design-basis evaluation. The evaluation in Section 4.5.7 of Reference 9.3 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 as reported in Section 3.4.5.7 of this safety evaluation (SE) is negligible relative to the structural integrity. The symmetrical location of the GE14i bundles in the CPS core will have no eccentricity on the mass center of the full core.

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 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.

3.4.5.6 Recirculation System Evaluation

An evaluation of the effects of introducing GE14i fuel on Reactor Recirculation System (RRS) performance for CPS 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 since the pump flow rates and recirculation system pressure/temperature is the same value as before GE14i fuel introduction.

The licensee concludes, and the NRC staff agrees that no modifications to RRS equipment or setpoints are required with the introduction of GE14i ITAs at CPS.

3.4.5.7 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.

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 eight GE 14i ITAs. The dominant fuel type, GE14 fuel, dictates the seismic behavior of the core.

The NRC staff agrees that the minor reduction in mass of GE14i fuel may slightly increase its dynamic fuel lift height to [ ] inches, which is still less than the [ ] inches allowable.

3.4.5.8 Neutron Fluence Impact

The introduction of GE14i fuel ITAs into the core will not significantly impact the magnitude of the reactor pressure vessel (RPV) flux 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 624 total number of CPS core loading 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 CPS.

3.4.5.9 Containment Response

The licensee has shown that [

] provided other

key parameters such as [

] do not change

due to the introduction of GE14i ITAs. [

.]

3.4.5.10 ECCS LOCA

GE14i ITAs are loaded in to the CPS 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 fueled rods in Ge14i bundle is less than the number in GE14 by the number of cobalt rods in Ge14i bundle. The MAPLHGR is not averaged over the zero-power rods. Therefore, the NRC staff agrees that the ECCS-LOCA MALPLHGR limits for Cycle 12 for GE14 remain bounding for GE14i ITAs. ECCS-LOCA MAPLHGR limits for all bundles in Cycle 13 for average planar exposure range of 0.0 GWd/MT to 70.0 GWd/MT are listed in Table 16.3-1 of Reference 9.10.

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 9.10. The GE14 10CFR50.46 initial MCPR and SLO multiplier on LHGR and MAPLHGR are applicable to the GE14i ITAs.

[

.] Furthermore, because the PCT and the maximum oxidation values remain within licensing basis, a coolable geometry is assured. The licensing results are applicable to all fuel types in the new cycle (Cycle 13) is listed in Table 3.5 (Reference 9.10, Table 16.1-1).

Table 3.5 Licensing Results for Cycle 13

Fuel Type	Licensing Basis PCT (°F)	Local Oxidation (%)	Core-Wide Metal-Water Reaction (%)
GE14	1570	<1.00	<0.10
GE14i	1570	<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.

In addition, the EOOS such as Automatic Depressurization System Out of Service and the flexibility options such as Maximum Extended Load Line Limit Analysis, SLO, and FFWTR for the current analyses remain applicable for GE14i.

3.4.5.11 Hydrogen Injection

This evaluation is based on Operating Experience (OE) 27774 (OE27774) that describes an incident where a core design change at Fermi 2, a moderate Hydrogen Water Chemistry plant that does not use NobleChem®, resulted in a lower gamma flux in the downcomer region causing a reduction in the hydrogen-oxygen combination reaction. The decreased gamma flux necessitated an increase in the feedwater hydrogen injection rate to maintain stress-corrosion cracking (SCC) mitigation compared to the previous cycle of operation. The key objective of the evaluation of this OE relative to GE14i was to determine whether there is any potential for a decrease in gamma flux in the downcomer region of the CPS core as a result of cobalt bearing rod insertion.

It was determined that, regardless of the location of the cobalt rods, there is no possibility for increased gamma flux in the downcomer region to decrease as a result of GE14i ITAs. Since the CPS plant is a Noble Metal Chemical Addition plant, the recombination of hydrogen and oxygen is expected to be more efficient in the downcomer region due to the decreased gamma flux. Therefore, the NRC staff finds that no negative impact is expected on the hydrogen requirement for SCC mitigation.

3.4.5.12 Post-LOCA Hydrogen Control

A qualitative analysis of the metal-water reaction was performed with respect to the addition of eight GE14i ITA bundles inserted in the CPS Cycle 13 core with the goal of comparing any possible addition of hydrogen produced to the hydrogen produced as a result of the previous extended power uprate analysis with a core of GE14 fuel (Reference 9.11). The RG 1.7, Revision 2 (Reference 9.17) addresses the fact that the decay heat generated in the irradiated fuel rods in post-LOCA core is the primary vehicle for the metal-water reaction. During the two minutes following the LOCA, the cobalt isotope rods will not produce any significant gamma heating greater than the core ambient conditions. Consequently, the cobalt isotope rod temperature will be well below the threshold to cause the metal-water reaction, such that there will be no increase in the quantity of hydrogen produced in a metal-water reaction following LOCA. As such, there are no adverse impacts from this perspective with the utilization of the GE14i ITA.

An evaluation was done to determine any additional hydrogen produced from radiolysis as a result of gamma heating caused by the energy profile of the irradiated cobalt isotope rods within the GE14i bundle. The cobalt isotope rods were found to have a different energy profile than the normal fuel rod within GE14 bundle (Table 4.2 of Reference 9.3). The percentage difference in total core energy throughout the 30-day period is significantly less than 1 percent and therefore, the relative impact on any additional hydrogen production is considered insignificant.

Hydrogen production from corrosion was not considered for this analysis since this is typically the result of changes in coolant chemistry, including pH and metal and coolant temperatures. The introduction of GE14i bundles will not produce changes in these parameters.

The above evaluations indicate that there is no appreciable impact on the ability of the CPS Combustible Gas Control System to mitigate the effects of a combustible gas mixture in a post-LOCA situation with the addition of eight GE14i ITA bundles.

3.4.5.13 Fuel Storage Criticality Safety

This section evaluates the licensee's criticality safety analysis of the fuel storage racks at CPS that are used for the storage of GE14i ITAs. The 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 4.8wt% U-235.

The 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 are replaced by the cobalt target rods in the GE14i ITAs. This replacement introduces neutron absorbers in to 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 4.8 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.

3.4.5.16 Fresh Fuel Shipping

Shipping of GE14i ITA bundle must be done under the regulatory and technical shipping requirements specified in RAJ-II Certificate of Compliance (Reference 9.18) that was issued by the Nuclear Materials Safety and Safeguards office of the NRC. (See Section 4.2 of this SE)

3.4.5.17 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 significantly affect these parameters and therefore the channel performance on GE14i bundles will be the same as on GE14 bundles.

3.4.5.18 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.

3.4.6 Manufacturing Quality Assurance

All aspects of the GE14i ITA program are controlled under the GE Nuclear Energy Quality Assurance Program Description (Reference 9.19). (See Section 4.2)

3.4.7 Post Operational Evaluations

3.4.7.1 Spent Fuel Effects

The licensee has shown and the NRC staff aggress that there are no adverse effects from the introduction of GE14i fuel in the CPS spent fuel pool (SFP), provided guidance for storage of GE14i bundles is followed to minimize the effect of gamma heating on the spent fuel pool concrete walls. Irradiated fuel storage procedures should be modified to specify that the GE14i bundles be stored at least 4 feet from the pool walls. With the 4 foot distance requirement, there is no limit to the duration of time a GE14i bundle may remain in the pool.

The placement of GE14i bundles in the CPS SFP was evaluated for three effects; (1) the effect of additional heat from the gamma decay of Co-60, (2) the effect of increased gamma heating of the SFP walls, and (3) the effect of GE14i bundles on the cleanup portion of the Fuel Pool Cooling and Cleanup system. For the first two cases, the extra radiation from the cobalt isotope rods was conservatively added to the radiation in a normal GE14 bundle. No credit was taken for the removal of [ ] fuel rods in each bundle.

The additional heat from the decay of Co-60 is insignificant compared to the total heat from a normal refueling outage. The small amount extra heat, [ ], added by the cobalt isotope rods poses no additional risk of SFP local boiling over that which was previously analyzed.

The licensee performed an assessment of the gamma radiation effect on the SFP walls where the GE14i bundle is placed 1, 4, and 6 feet from the pool wall. In the GE14i analysis, no credit was taken for shielding provided by the spent fuel and racks in the outer rows, however, water and self-shielding were credited. It was determined that at 4 feet from the walls, the energy deposition from gamma decay was well below that required to cause significant concrete heating.

3.4.7.2 Post Irradiation Handling

Section 4.7.2 of Reference 9.3 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 GE'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 Global Nuclear Fuel/GEH.

The steps in the procedure for segmented rod disassembly are listed in Section 4.7.2.3 of Reference 9.3. The licensee has assured that prior to segmented rod disassembly during the 2012 CPS outage; a similar segmented rod procedure will be prepared and incorporated in to 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. The NRC staff agrees with this analysis.

3.4.7.3 Post-Irradiation Examination

Post-irradiation examination (PIE) of the GE14i ITA bundle and rods will 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 will include:

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

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 will 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 will be inspected for the following:

- Location specific activities along axis
- Cobalt target conditions and status of Nickle 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.

3.4.7.4 Occupational and Public Radiation Doses

The cobalt ITA rods will be double encapsulated to eliminate the potential for release of Co-59 and/or Co-60 into the reactor coolant system or into the plant environs. Therefore, there is no impact of this operation on the amount or type of plant radioactive effluents. The irradiated target rods are handled in the spent fuel pool. The source strength and the radiation levels of the Co-60 rods are bounded by source strength and radiation levels of the spent fuel rods routinely handled in the spent fuel pool. The storage of the irradiated cobalt rods in the spent fuel pool will be performed maintaining a minimum of four feet of clearance between the consolidated/disassembled target rods and the walls of the spent fuel pool. This ensures that the radiation shielding provided by the design of the spent fuel pool remains adequate. The cobalt rods will be shipped and transported to a licensed GE facility using plant procedures for shipping radioactive materials that are adequate to ensure compliance with 10 CFR Part 71 and 49 CFR Part 173. Therefore, the NRC staff finds that the Co-60 production operation as proposed 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 ALARA.

3.5 Conclusion

The NRC staff has reviewed the license amendment request and supporting documents (References 9.1, 9.2, 9.3, 9.10, and 9.22) to modify the facility Operating License of CPS in support of the use of isotope test assemblies. Based on the review of the application and supporting documents, the NRC staff has concluded that the proposed amendment request to modify the CPS facility operating license in support of the use of isotope test assemblies to irradiate Co-59 targets to produce Co-60 has satisfied the regulatory, technical and licensing requirements. The licensee has evaluated the impact of the use of ITAs on previously evaluated transients and design basis accidents for CPS. The NRC staff concludes that the use of these ITAs will not adversely affect accident initiators or precursors, alter the design assumptions, conditions, and configuration, or the manner in which the plant is operated and maintained.

The NRC staff has further determined that the application of the methods described in the licensee's amendment request, as supplemented, demonstrate compliance with 10 CFR Part 20, 10 CFR Part 30, 10 CFR Part 40, 10 CFR 50.46, 10 CFR Part 71, SRP Chapter 4.2, 10 CFR Part 50 Appendix A GDC 10, 11, 12, 13, 29 and 60, and 10 CFR Part 50, Appendix I, "Numerical

Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Reasonably Achievable" for Light-Water-Cooled Nuclear Power Reactor Effluents."

A regulatory commitment submitted by licensee is included in Section 4.1 of this SE. The licensee is expected to implement this commitment in a timely manner as described. Changes to the plans for implementation should be communicated to the NRC staff in accordance with the guidance provided in Nuclear Energy Institute 99-04, "Guidelines for Managing NRC Commitment Changes."

4.0 CONDITIONS AND LIMITATIONS

Licensee must ensure compliance with the following conditions and limitations:

1. Periodic sampling of the Co-60 activity in the reactor coolant should be performed to monitor the integrity of the Co-59/Co-60 target rods. The licensee should incorporate this monitoring and surveillance program in to its procedures.

2. For the subsequent cycles, CPS shall ensure that ITAs are placed in non-limiting locations. The licensee should perform the necessary analyses that assure that these ITAs are not limiting with regard to both thermal margins and reactivity margins, for each cycle of operation.

3. For ITA margin considerations, later in bundle life, the power peaking shifts to the rods adjacent to the water gaps and a reduction in the allowable heat generation rate is used to provide the power suppression. [

   ]. The suppression of MLHGR to accommodate the additional heat flux margin at high exposures shall be calculated.

4. Licensing analyses for each subsequent cycle of operation with ITAs in the core shall be performed.

5. As indicated in Section 4.2.5 of NEDC-33505, the cold water events, fast pressurization and ASME overpressurization events with licensed flexibility options for CPS shall be analyzed as part of the cycle-specific reload licensing analyses.

6. Thermal-mechanical check for each subsequent cycle must be performed to ensure that the Thermal-Mechanical limits for GE14i bundles are not exceeded.

7. Fresh GE14i bundle shipment shall meet the regulatory and technical shipping requirements (10 CFR Part 71) as per RAJ-II Certificate of Compliance issued by the NRC in May 2008.

8. In order to minimize the gamma radiation heating effect on the CPS SFP walls and to prevent long-term concrete degradation, appropriate procedure as described in NEDC-33505 to guide placement of irradiated bundles in the SFP to avoid gamma heating of the walls should be enforced. (See License Conditions)

9. All aspects of the GE14i ITA program, such as zirconium tubing procurement and fabrication, manufacturing and handling of target pellets, welding of the internal and external rod segments, helium leak check of both inner and outer tubes for rod integrity and final isotope bundle assembly, shall be performed under the GE Nuclear Energy Quality assurance program as specified in Section 4.6 of Reference 9.3.

10. Post-Irradiation handling of the cobalt isotope rods shall be performed as prescribed in Section 4.7.2 of Reference 9.3.

11. Post-irradiation examination (PIE) of a GE14i ITA bundle will include poolside visual, poolside gamma scan measurements, poolside combined instrumentation measurement system (COINS) and segmented rod hot cell destructive examination as described in Section 4.7.3 of NEDC-33505. The PIE plan is applied to the end of the first cycle of operation and after subsequent cycles including at the at bundle's EOL. The licensee is required to send a summary report of the results from these inspections to the NRC as soon as they are completed.

5.0 STATE CONSIDERATIONS

In accordance with the commission's regulations, the Illinois State official was notified of the proposed issuance of amendment. The state official had no comments.

6.0 ENVIRONMENTAL CONSIDERATIONS

The amendments change 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 amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (74 FR 66159-66163). Accordingly, the amendments meet 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 amendments.

7.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (NSHCD)

The Commission may issue the license amendment before the expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. This amendment is being issued prior to the expiration of the 60-day period. Therefore, a final finding of no significant hazards consideration follows.

The Commission has made a final determination that the amendment request involves no significant hazards consideration. Under the Commissions regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below.

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes to the license conditions provide clarification and do not impact plant operation in any way. The handling of byproduct material (i.e., Co-60) will continue to be done in accordance with the requirements of 10 CFR 30 and the requirements of the CPS Facility Operating License. The proposed change to TS 4.2.1 also provides clarification and additional description of the proposed ITAs to be used in the CPS core. These changes provide clarification and do not involve an increase in the probability or consequences of an accident previously evaluated.

   The use of the GE14i ITAs, has been evaluated for impact on the previously evaluated transients and design basis accidents for CPS. GE-Hitachi report NEDC-33505P, "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Clinton Power Station," dated June 2009, documents the results of the analyses completed to demonstrate the impact on operation following introduction of the ITAs in the CPS core. The use of these ITAs does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration or the manner in which the plant is operated and maintained. The Cycle 13 (i.e., the first cycle of operation with the GE14i assembly) core, and subsequent cores, will be designed so that the ITAs will be placed in non-limiting locations with respect to thermal limit margins and shutdown margins. The ITAs do not adversely affect the ability of any structures, systems or components (SSCs) to perform their intended safety function to mitigate the consequences of an initiating event within the assumed acceptance limits.

In addition to evaluation of the impact to operation with the introduction of the GE14i assemblies, EGC has also evaluated the effects of these assemblies on post-irradiation conditions. The additional heat from the Co-60 decay is insignificant when compared to the total heat from a normal refueling discharge. The small amount of extra heat added by the cobalt isotope rods poses no additional risk of spent fuel pool (SFP) local boiling over that previously analyzed. The maximum incident radiation due to an irradiated GE14i bundle placed one foot from the spent fuel pool walls is in excess of the radiation that would result in significant gamma heating of the concrete. However, analysis has demonstrated that at four feet, the energy deposition rate is well below that required to cause significant concrete heating. CPS procedures exist to guide placement of irradiated fuel bundles in the SFP to avoid gamma heating of the wall concrete. These procedures will be modified to specify that the irradiated GE14i bundles be stored at least four feet from the pool walls. With the four foot distance requirement in effect, there is no limitation on the amount of time an irradiated GE14i bundle may remain in the pool.

Handling of the licensed transfer casks will be in accordance with the guidance in NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants," using the Fuel Building Crane. These precautions will support safe movement of the casks within the Fuel Building.

The consequences of a previously analyzed event are dependent on the initial conditions assumed in the analysis, the availability and successful functioning of equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The consequences of a previously evaluated accident are not significantly increased by the proposed change. As documented in NEDC-33505P, the proposed change does not affect the performance of any equipment credited to mitigate the radiological consequences of an accident. Evaluation of operation with the GE14i assemblies in the CPS core, demonstrated that the licensing basis radiological analyses are not impacted by the introduction of eight GE14i assemblies at CPS. This includes the analyses done for transients and design basis accident events.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed revision to the CPS license conditions and TS 4.2.1 will not introduce any new or modified equipment since these changes are intended to provide clarification only. These clarifications will not result in operation of the facility in a different way than currently operated.

While the proposed ITA program does result in the introduction of several modified fuel assemblies (i.e., the GE14i assembly), these assemblies are essentially the same as the GE14 assemblies currently in use in the CPS core. The only difference being the use of a number of isotope rods in place of fuel rods. The GE14i assembly was designed for mechanical, nuclear, and thermal-hydraulic compatibility with the GE14 fuel design. The details of the design differences between the GE14 and GE14i are documented in NEDC-33505P. Use of the proposed ITAs does not involve the addition or modification of any plant equipment other than the assemblies modified to include the cobalt target rods. Also, use of the proposed ITAs will not alter the design configuration or method of operation of plant equipment beyond its normal functional capabilities. The ITA program does not create any new credible failure mechanisms, malfunctions or accident initiators.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change to the CPS operating license conditions are intended to provide clarification as to how the generation of byproduct material in the CPS reactor core meets the requirements of 10 CFR Part 30. The proposed change to TS 4.2.1 also provides clarification and additional description of the proposed ITAs to be used in the CPS core. These proposed changes would not affect the design or operation of any equipment important to safety. In addition, since the proposed changes to the license conditions and TS provide clarification only, these changes do not affect the results of any safety calculations.

Cycle specific analyses will be performed for CPS Reload 12 Cycle 13 to establish fuel operating limits for the ITAs that assure compliance with regulatory limits. Results of these analyses will be documented in the CPS Reload 12 Cycle 13 Supplemental Reload Licensing Report. Furthermore, licensing analyses will be performed for the ITAs for each cycle of their operation, wherein the effect of the ITAs is considered for each of the appropriate licensing events and anticipated operational occurrences (AOOs) to establish the appropriate reactor thermal limits for operation.

The proposed introduction of the ITAs has no impact on equipment design or fundamental operation, other than the modifications made to the fuel assembly as part of the program. There are no changes being made to safety limits or safety system allowable values that would adversely affect plant safety as a result of the proposed ITAs. The performance of the systems important to safety is not significantly affected by the use of the proposed ITAs. The margin of safety can be affected by the thermal limits existing at the time of the postulated accident; however, the ITA design has been evaluated and demonstrated to have no significant effect on the calculated thermal limits as described above. The proposed change does not affect safety analysis assumptions or initial conditions and therefore, the margin of safety in the original safety analyses is maintained.

As documented above, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and based on this review, determined that the three standards of 10 CFR 50.92 are satisfied. Therefore, the NRC staff has determined that the amendment involves no significant hazards consideration.

8.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

9.1 Letter RS-09-074 from Exelon Nuclear (J. L. Hansen) to US NRC, "License Amendment Request to Modify Clinton Power Facility Operating License in Support of the Use of Isotope Test Assemblies," Exelon Nuclear, June 26, 2009. (ADAMS Accession No. ML091801064)

9.2 Letter RS-09-150 from Exelon Nuclear (J. L. Hansen) to US NRC, "Additional Information Supporting the Request for a License Amendment to Modify Clinton Power Station Facility Operating License in Support of the Use of Isotope Test Assemblies," Exelon Nuclear, November 4, 2009. (ADAMS Accession Nos. ML093100313, ML093100314)

9.3 NEDC-33505P, Revision 0, "Safety Analysis Report to Support Introduction of GE14i Isotope Test Assemblies (ITAs) in Clinton Power Station (Proprietary)", GE-Hitachi Nuclear Energy, June 2009. (ADAMS Accession No ML091801064)

9.4 NEDE-24011-P-A-16, "General Electric Standard Application for Reactor Fuel (GESTAR II)," GE Hitachi Nuclear Energy, October 2007.

9.5 Letter from US NRC to General electric (G. A. Waterford), "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A-16, GESTAR II – Implementing Improved Steady-State Methods," November 10, 1999.

9.6 Letter MFN 098-96, "Implementation of Improved GE Steady-State Nuclear Methods," July 2, 1996.

9.7 NEDC-32601-P-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," GE Nuclear Energy, August 1999.

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

9.9 NEDC-32851P-A, Revision 4, "GEXL14 Correlation for GE14 Fuel," Global Nuclear Fuel, September 2007
9.10 0000-0099-4244-SRLR, Revision 0, "Supplemental Reload Licensing Report for Clinton Power Station Unit1 Reload 12 Cycle 13," (Attachment to Letter RS-09-171 from Exelon Nuclear (J. L. Hansen) to US NRC), Global Nuclear Fuel, December 14, 2009. (ADAMS Accession No. ML093490375)
9.11 NEDC-32989P, "Safety Analysis Report for Clinton Power Station Extended Power Uprate," GE Nuclear Energy, June 2001.
9.12 GNF-0000-018-6874-R0-P, "GE14i Thermal-Mechanical Evaluation," GNF Proprietary Information, Global Nuclear Fuel, October 2009.
9.13 NUREG-0800, "Standard Review Plan," Revision 3, U. S. Nuclear Regulatory Commissions, March 2007.
9.14 NEDO-31960-A, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," GE Nuclear Energy, November 1995.
9.15 NEDC-32992P-A, "Licensing Topical Report, ODYSY Application for stability Licensing Calculations," GE Nuclear Energy, July 2001.
9.16 GNF-0000-0108-9509-R0-P, "Evaluation of Hydraulic Characteristics of the GE14i Fuel," GNF Proprietary Information, October 2009.
9.17 Regulatory Guide 1.7, Revision 2, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident," US NRC, November 1978.
9.18 NRC Certificate of Compliance No. 9309, Revision 7, for Model RAJ-II Package, US NRC, May 28, 2008
9.19 NEDO-1120904A, Revision 8, "GE Nuclear Energy Quality Assurance Program Description," GE Nuclear Energy, March 1989.
9.20 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," 2009.
9.21 NEDE-24011-P-A-16-US, Revision 16, "General Electric Standard Application for Reactor Fuel, (GESTAR II) (Supplement for United States)," Global Nuclear Fuel, October 2007.
9.22 Letter RS-09-146 from Exelon Nuclear (J. L. Hansen) to US NRC, "Additional Information Supporting the Request for a License Amendment to Modify Clinton Power Station Facility Operating License in Support of the Use of Isotope Test Assemblies," Exelon Nuclear, November 17, 2009. (ADAMS Accession No. ML093210561)
9.23 Letter from the NRC to Exelon Nuclear (C. Crane), "Clinton Power Station, Unit 1: Issuance of Amendment: Re: Application of Alternative Source Term Methodology (TAC No. MB8365), dated September 19, 2005 (ADAMS Accession No. ML052570461)

Principal Contributors: M. Panicker, NRR
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Date: January 15, 2010

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Cameron S. Goodwin, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosures:
1. Amendment No. 190 to NPF-62
2. Safety Evaluation (Proprietary)
3. Safety Evaluation (Non-Proprietary)

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Amendment: ML100050199 NRR-58 *By Memo dated

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