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

December 31, 2014

Mr. Michael J. Pacilio
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS REGARDING NETCO INSERTS (TAC. NOS. MF2489 AND MF2490)(RS-13-148)

Dear Mr. Pacilio:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 253 to Renewed Facility Operating License No. DPR-29 and Amendment No. 248 to Renewed Facility Operating License No. DPR-30 for Quad Cities Nuclear Power Station, Units 1 and 2 (QCNPS). The amendments are in response to your application dated July 16, 2013, as supplemented by letters dated September 18, 2013, January 22, April 7, August 12, and November 11, 2014. The amendments allow the use of specific neutron absorbing rack insert (NETCO-SNAP-IN®) that can be installed into the QCNPS spent fuel pool (SFP) storage racks and credited as a replacement for the neutron absorbing properties of the Boraflex panels.

As discussed in the Attachment to the supplement dated November 11, 2014, Exelon Generation Company, LLC (EGC, the licensee) made a commitment to several performance objectives related to a Rack Insert Surveillance Program (RISP). The NRC has determined that proper implementation of the commitment is essential to ensuring the timely identification and mitigation of any degradation if the aluminum boron carbide rack inserts. Therefore, the QCPNS licenses have been amended to reflect the following requirements for the RISP.

While fuel assemblies are in the SFPs for QCPNS, Units 1 and 2, the licensee shall implement and maintain a RISP to ensure the timely identification and mitigation of degradation of the aluminum boron carbide rack inserts in either units SFP. The RISP must:

1. Ensure that coupon evaluations of Boron-10 areal densities are performed by a qualified laboratory;

2. Ensure that insert evaluations are performed to verify that any service wear is within expected parameters;

Enclosure 4 transmitted herewith contains sensitive unclassified information. When separated from Enclosure 4, this document is decontrolled.

3. Ensure that the evaluations are performed at intervals not to exceed four years for coupon Boron-10 areal density, and 10 years for insert service wear;

4. Ensure that if any inserts are identified as potentially failing the minimum certified Boron-10 areal density criterion, based on correlation of the coupon evaluation or insert service wear evaluation results to inserts, or other abnormal indications, EGC will take affected inserts out of service until it can be positively demonstrated that the minimum certified Boron-10 areal density criterion (0.0116 g/cm^2) is met for each insert; and

5. Submit a report to the NRC, within 90 days following completion of evaluations associated with Item 4 above, that describes the testing results, assessments performed, and interim and long-term corrective actions for abnormal indications.

As discussed in the Attachment to the supplement dated November 11, 2014, EGC made a commitment to revise the Updated Final Safety Analysis Report to add a description of the Composite Surveillance Program. The addition of this commitment as an implementation condition of the amendment was discussed with Mr. Ken Nicely of your staff on December 10, 2014.

A copy of the related SE is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Brenda Mozafari

Brenda Mozafari, Senior Project Manager
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-254 and 50-265

Enclosures:
1. Amendment No. 253 to DPR-29
2. Amendment No. 248 to DPR-30
3. Safety Evaluation (Non-Proprietary)
4. Safety Evaluation (Proprietary)

cc w/encls 1, 2, and 3

Distribution via Listserv

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

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 253
Renewed License No. DPR-29

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

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated July 16, 2013, as supplemented by letters dated September 18, 2013, January 22, April 7, August 12, and November 11, 2014, 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 and paragraph 3.B.of Renewed Facility Operating License No. DPR-29 is hereby amended to read as follows:

Enclosure 1

B Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 253, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the Operating License is amended as indicated in the attachment to this license amendment and subject to the following License Condition:

While fuel assemblies are in the spent fuel pools (SFPs) for Quad Cities Nuclear Power Station Units 1 and 2, the licensee shall implement and maintain a Rack Insert Surveillance Program (RISP) to ensure the timely identification and mitigation of degradation of the aluminum boron carbide rack inserts in either units SFP. The RISP must:

1. Ensure that coupon evaluations of Boron-10 areal densities are performed by a qualified laboratory;

2. Ensure that insert evaluations are performed to verify that any service wear is within expected parameters;

3. Ensure that the evaluations are performed at intervals not to exceed four years for coupon Boron-10 areal density, and 10 years for insert service wear;

4. Ensure that if any inserts are identified as potentially failing the minimum certified Boron-10 areal density criterion, based on correlation of the coupon evaluation or insert service wear evaluation results to inserts, or other abnormal indications, EGC will take affected inserts out of service until it can be positively demonstrated that the minimum certified Boron-1 O areal density criterion (0.0116 g/cm^2) is met for each insert; and,

5. Submit a report to the NRC, within 90 days following completion of evaluations associated with Item 4 above, that describes the testing results, assessments performed, and interim and long-term corrective actions for abnormal indications.

4. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of the date of issuance. Implementation of the amendment shall include revision of the Updated Final Safety Analysis Report as described in the Attachment to the licensee's letter dated November 11, 2014.

FOR THE NUCLEAR REGULATORY COMMISSION

Travis L. Tate

Travis L. Tate, Chief
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Renewed Facility Operating License

Date of Issuance: December 31, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 253

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Replace the following page of Renewed Facility Operating License DPR-29 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
4	4
9	9
10	10

Replace the following pages of 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
4.0-2	4.0-2

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 253 are hereby incorporated into this renewed operating License. The licensee shall operate the facility in accordance with the Technical Specifications.

C. The licensee shall maintain the commitments made in response to the March 14, 1983, NUREG-0737 Order, subject to the following provision:

The licensee may make changes to commitments made in response to the March 14, 1983, NUREG-0737 Order without prior approval of the Commission as long as the change would be permitted without NRC approval, pursuant to the requirements of 10 CFR 50.59. Consistent with this regulation, if the change results in an Unreviewed Safety Question, a license amendment shall be submitted to the NRC staff for review and approval prior to implementation of the change.

D. Equalizer Valve Restriction

Three of the four valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation with one bypass valve open to allow for thermal expansion of water.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined sets of plans[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Quad Cities Nuclear Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 2," submitted by letter dated May 17, 2006.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 249.

F. The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Reports dated July 27, 1979, with supplements dated November 5, 1980, and

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan

AA. Upon implementation of Amendment No. 238 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.4.4, in accordance with TS 5.5.13.c(i), the assessment of CRE habitability as required by Specification 5.5.13.c(ii), and the measurement of CRE pressure as required by Specification 5.5.13.d, shall be considered met. Following implementation:

(1) The first performance of SR 3.7.4.4, in accordance with Specification 5.5.13.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from September 21, 2006, the date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(2) The first performance of the periodic assessment of CRE habitability, Specification 5.5.13.c(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from September 21, 2006, the date of the most recent successful tracer gas test, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(3) The first performance of the periodic measurement of CRE pressure, Specification 5.5.13.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.

BB. While fuel assemblies are in the spent fuel pools (SFPs) for Quad Cities Nuclear Power Station Units 1 and 2, the licensee shall implement and maintain a Rack Insert Surveillance Program (RISP) to ensure the timely identification and mitigation of degradation of the aluminum boron carbide rack inserts in either unit's SFP. The RISP must:

(1) Ensure that coupon evaluations of Boron-10 areal densities are performed by a qualified laboratory;

(2) Ensure that insert evaluations are performed to verify that any service wear is within expected parameters;

(3) Ensure that the evaluations are performed at intervals not to exceed four years for coupon Boron-10 areal density, and 10 years for insert service wear;

(4) Ensure that if any inserts are identified as potentially failing the minimum certified Boron-10 areal density criterion, based on correlation of the coupon evaluation or insert service wear evaluation results to inserts, or other abnormal indications, EGC will take affected inserts out of service until it can be positively demonstrated that the minimum certified Boron-10 areal density criterion (0.0116 g/cm^2) is met for each insert; and,

(5) Submit a report to the NRC, within 90 days following completion of evaluations associated with Item 4 above, that describes the testing results, assessments performed, and interim and long-term corrective actions for abnormal indications.

4. This renewed operating license is effective as of the date of issuance and shall expire at midnight on December 14, 2032.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A – Technical Specifications
2. Appendix B – Environmental Protection Plan

Date of Issuance: October 28, 2004

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

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

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the UFSAR;

b. A nominal 6.22 inch center to center distance between fuel assemblies placed in the storage racks;

c. The combination of U-235 enrichment and gadolinia loading shall be limited to ensure fuel assemblies have a maximum k-infinity of 0.8991 as determined at 4°C (39.2°F) in the normal spent fuel pool in-rack configuration; and

d. The installed neutron absorbing rack inserts having a Boron-10 areal density ≥ 0.0116 g/cm^2.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 666 ft 8.5 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3657 fuel assemblies for Unit 1 and 3897 fuel assemblies for Unit 2.

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

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 248
Renewed License No. DPR-30

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

   A. The application for amendment by the Exelon Generation Company, LLC (the licensee) dated July 16, 2013, as supplemented by letters dated September 18, 2013, January 22, April 7, August 12, and November 11, 2014, 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 and paragraph 3.B. of Renewed Facility Operating License No. DPR-30 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 248, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the Operating License is amended as indicated in the attachment to this license amendment and subject to the following License Condition:

While fuel assemblies are in the spent fuel pools (SFPs) for Quad Cities Nuclear Power Station Units 1 and 2, the licensee shall implement and maintain a Rack Insert Surveillance Program (RISP) to ensure the timely identification and mitigation of degradation of the aluminum boron carbide rack inserts in either units SFP. The RISP must:

1. Ensure that coupon evaluations of Boron-10 areal densities are performed by a qualified laboratory;

2. Ensure that insert evaluations are performed to verify that any service wear is within expected parameters;

3. Ensure that the evaluations are performed at intervals not to exceed four years for coupon Boron-10 areal density, and 10 years for insert service wear;

4. Ensure that if any inserts are identified as potentially failing the minimum certified Boron-10 areal density criterion, based on correlation of the coupon evaluation or insert service wear evaluation results to inserts, or other abnormal indications, EGC will take affected inserts out of service until it can be positively demonstrated that the minimum certified Boron-1 O areal density criterion (0.0116 g/cm^2) is met for each insert; and,

5. Submit a report to the NRC, within 90 days following completion of evaluations associated with Item 4 above, that describes the testing results, assessments performed, and interim and long-term corrective actions for abnormal indications.

4. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days of the date of issuance. Implementation of the amendment shall include revision of the Updated Final Safety Analysis Report as described in the Attachment to the licensee's letter dated November 11, 2014.

FOR THE NUCLEAR REGULATORY COMMISSION

Travis L. Tate

Travis L. Tate, Chief
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Renewed Facility Operating License

Date of Issuance: December 31, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 248

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Replace the following page of Renewed Facility Operating License DPR-30 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
4	4
9	9
10	10

Replace the following pages of 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
4.0-2	4.0-2

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 248, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. The licensee shall maintain the commitments made in response to the March 14, 1983, NUREG-0737 Order, subject to the following provision:

> The licensee may make changes to commitments made in response to the March 14, 1983, NUREG-0737 Order without prior approval of the Commission as long as the change would be permitted without NRC approval, pursuant to the requirements of 10 CFR 50.59. Consistent with this regulation, if the change results in an Unreviewed Safety Question, a license amendment shall be submitted to the NRC staff for review and approval prior to implementation of the change.

D. Equalizer Valve Restriction

Three of the four valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation with one bypass valve open to allow for thermal expansion of water.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Quad Cities Nuclear Power Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 2," submitted by letter dated May 17, 2006.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 244.

F. The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Reports dated July 27, 1979 with supplements dated

---

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

Y. The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

Z. Upon implementation of Amendment No. 233 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.4.4, in accordance with TS 5.5.13.c.(i), the assessment of CRE habitability as required by Specification 5.5.13.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.13.d, shall be considered met. Following implementation:

(1) The first performance of SR 3.7.4.4, in accordance with Specification 5.5.13.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from September 21, 2006, the date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(2) The first performance of the periodic assessment of CRE habitability, Specification 5.5.13.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from September 21, 2006, the date of the most recent successful tracer gas test, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(3) The first performance of the periodic measurement of CRE pressure, Specification 5.5.13.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

AA. While fuel assemblies are in the spent fuel pools (SFPs) for Quad Cities Nuclear Power Station Units 1 and 2, the licensee shall implement and maintain a Rack Insert Surveillance Program (RISP) to ensure the timely identification and mitigation of degradation of the aluminum boron carbide rack inserts in either unit's SFP. The RISP must:

(1) Ensure that coupon evaluations of Boron-10 areal densities are performed by a qualified laboratory;

(2) Ensure that insert evaluations are performed to verify that any service wear is within expected parameters;

(3) Ensure that the evaluations are performed at intervals not to exceed four years for coupon Boron-10 areal density, and 10 years for insert service wear;

(4) Ensure that if any inserts are identified as potentially failing the minimum certified Boron-10 areal density criterion, based on correlation of the coupon evaluation or insert service wear evaluation results to inserts, or other abnormal indications, EGC will take affected inserts out of service until it can be positively demonstrated that the minimum certified Boron-10 areal density criterion (0.0116 g/cm^2) is met for each insert; and,

(5) Submit a report to the NRC, within 90 days following completion of evaluations associated with Item 4 above, that describes the testing results, assessments performed, and interim and long-term corrective actions for abnormal indications.

4. This renewed operating license is effective as of the date of issuance and shall expire at midnight on December 14, 2032.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A – Technical Specifications
2. Appendix B – Environmental Protection Plan

Date of Issuance: October 28, 2004

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

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

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the UFSAR;

b. A nominal 6.22 inch center to center distance between fuel assemblies placed in the storage racks;

c. The combination of U-235 enrichment and gadolinia loading shall be limited to ensure fuel assemblies have a maximum k-infinity of 0.8991 as determined at 4°C (39.2°F) in the normal spent fuel pool in-rack configuration; and

d. The installed neutron absorbing rack inserts having a Boron-10 areal density ≥ 0.0116 g/cm^2.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 666 ft 8.5 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3657 fuel assemblies for Unit 1 and 3897 fuel assemblies for Unit 2.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 253 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-29

AND

AMENDMENT NO. 248 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-30

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

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

Enclosure 3

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 253 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-29

AND

AMENDMENT NO. 248 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-30

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

1.0 INTRODUCTION

By application to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated July 16, 2013, as supplemented by letters dated September 18, 2013, January 22, April 7, August 12, and November 11, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML13199A032, ML13261A518, ML14024A496, ML14101A213, ML14224A445, and ML14317A753, respectively), Exelon Generation Company (EGC), LLC, the licensee for Quad Cities Nuclear Power Station, Units 1 and 2 (QCNPS), requested an amendment to the technical specifications (TSs) to replace credit for Boraflex in the nuclear criticality safety analysis with NETCO-SNAP-IN® rack inserts.

The supplements dated September 18, 2013, January 22, April 7, August 12, and November 11, 2014, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration dated July 8, 2014 (79 FR 38577).

2.0 REGULATORY EVALUATION

2.1 Regulatory Requirements

Paragraph 50.36(c)(4) of Title 10 of the *Code of Federal Regulations* (10 CFR) states that "[d]esign 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 a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section."

Paragraph 50.68(b)(1) of 10 CFR states that , "[p]lant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water."

Paragraph 50.68(b)(4) of 10 CFR states, in part, "[i]f no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95-percent probability, 95-percent confidence level, if flooded with unborated water."

The NRC staff notes that the general design criteria (GDC) in Appendix A of 10 CFR Part 50 were published after the issuance of the QCNPS construction permits. Section 3.0 of the QCNPS updated final safety analysis report (UFSAR) further describes how the design of the facility satisfies the intent of the GDC which were proposed in July 1967. A comparison between the proposed Atomic Energy Commission GDC in Section 3.0 of the QCNPS UFSAR, to which the facility was licensed, and the current GDC in 10 CFR 50 Appendix A indicates that the QCNPS-specific GDC 1, 2, 40, and 66 are similar to the current Appendix A GDC 1, 2, 4, and GDC 62, respectively. Based on this comparison, the NRC staff reviewed the license amendment request (LAR) submittal against the regulatory requirements of the GDC in Appendix A of 10 CFR Part 50, as indicated below.

The acceptance criteria are based on continued conformance with the requirements of the following regulations: (1) 10 CFR 50.55a, and GDC 1 of Appendix A to 10 CFR 50 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; and (3) GDC 4 as it relates to structures and components important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions and these structures and components being appropriately protected against dynamic effects, including the effects of missiles.

The requested GDC 62 states that, "[c]riticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." The licensee must limit the potential for criticality in the fuel handling building and storage system by physical systems or processes.

2.2 Review Guidance

Regulatory Guide (RG) 1.29, "Seismic Design Classification," provides guidance with respect to the requirements regarding structures, systems, and components (SSCs) which must be designed to withstand the effects of the loads resulting from a safe shutdown earthquake (SSE) and remain functional following an SSE; these are designated as Seismic Category I SSCs. Specifically, 1.I of Part C, "Regulatory Position," in RG 1.29 states, in part, that the spent fuel pool (SFP) structures, including the spent fuel racks, are designated as Seismic Category I.

Sections 9.1.1 "Criticality Safety of Fresh and Spent Fuel Storage and Handling," and 9.1.2 "New and Spent Fuel Storage," of NUREG-800, "Standard Review Plan for the review of Safety Analysis Reports for Nuclear Power Plants – LWR Edition" or the Standard Review Plan (SRP) provides a basis to evaluate proposed changes in scope and requirements in a licensee's licensing basis. Section 9.1.1, supports an evaluation of performance effectiveness of the neutron absorbing materials in the fresh and spent fuel racks. According to SRP 9.1.2, the review should ensure that there are no potential mechanisms that will (1) alter the dispersion of boron carbide (B4C) in the neutron attenuation panels, and/or (2) cause physical distortion of the tubes retaining the stored fuel assemblies.

The NRC staff issued a memorandum dated August 19, 1998, also known as the Kopp memo, containing guidance for performing the review of SFP nuclear criticality safety (NCS) analyses. This guidance supports the determination of compliance with GDC 62 and existing SRP sections 9.1.1 and 9.1.2. The principal objective of this guidance is to clarify and document staff positions that may have been incompletely or ambiguously stated in previously issued safety evaluations and other staff documents. A second purpose is to state staff positions on recently proposed storage configurations and characteristics in spent fuel rerack enrichment upgrade requests, for example, multiple-region spent fuel storage racks, checkerboard loading patterns for new and spent fuel storage, credit for burnup in the spent fuel to be stored, and credit for nonremovable poison inserts.

The Division of Safety System (DSS) Interim Staff Guidance (ISG), DSS-ISG-2010-01 (Reference 8), provides updated guidance to address the increased complexity of recent SFP nuclear criticality analyses and operations. The guidance is intended to reiterate existing guidance, clarify ambiguity in existing guidance, and identify lessons learned based on recent submittals. Similar to the Kopp memo, this guidance supports determining compliance with GDC 62 and existing SRP sections 9.1.1 and 9.1.2.

The ISG, DSS-ISG-2010-01, references NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology."

NUREG/CR-6698 states, in part, that:

> In general, the critical experiments selected for inclusion in the validation must be representative of the types of materials, conditions, and operating parameters found in the actual operations to be modeled using the calculational method. A sufficient number of experiments with varying experimental parameters should be selected for inclusion in the validation to ensure as wide an area of applicability as feasible and statistically significant results.

The NRC staff used NUREG/CR-6698 as guidance for review of the code validation methodology presented in the application. The basic elements of validation are outlined in NUREG/CR-6698, and include identification of operating conditions and parameter ranges to be validated, selection of critical benchmarks, modeling of benchmarks, statistical analysis of results, and determination of the area of applicability.

The regulatory guidance for design modifications of the SFP and storage racks are documented in the NRC Office of Technology (OT) Position Paper, "OT Position For Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, amended by an NRC letter dated January 18, 1979 (subsequently renamed NRC Generic Letters 78-11 and 79-04, respectively); and Section 3.8.4, "Other Seismic Category I Structures," including Appendix D to SRP Section 3.8.4, "Technical Position on Spent Fuel Racks."

3.0 TECHNICAL EVALUATION

Background

There are two SFPs at the QCNPS; one for each unit. Each unit's SFP has the capacity to hold up to 3,657 and 3,897 fuel assemblies (Units 1 and 2, respectively for a total of 7,554 fuel assemblies). Boraflex is currently in use in the QCNPS SFPs and is credited in the current licensing basis criticality analysis for the wet storage racks. Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks," documents a concern with the Boraflex neutron absorbing material used in SFPs. The degradation of Boraflex in the QCNPS SFPs has reduced its ability to perform its neutron absorption design function. The Boraflex degradation is continuing; thus prompting the need for this amendment request.

The QCNPS, Unit 1, SFP contains 19 high-density racks in nine different module sizes, and the QCNPS, Unit 2, SFP contains 20 high density racks in nine different module sizes. Outside the module sizes, the design of all racks used at QCNPS is the same. The racks contain Boraflex neutron absorbing material sandwiched between the 304 stainless steel walls between adjacent storage cells and a double layer with a flux trap between adjacent modules. However, due to historical degradation of Boraflex, QCNPS is submitted this LAR to remove credit for the Boraflex from the current licensing basis NCS analysis.

NETCO-SNAP-IN® rack inserts will be installed in each storage cell to serve the neutron-absorbing safety function credited in the NCS analysis. The rack inserts are made of the Rio-Tinto-Alcan, Inc. composite neutron absorbing material, which contains boron carbide particles distributed in metal. The rack inserts are designed to extend the full length of the storage rack cell, and to be folded in a ~90 degree angle and inserted in each SFP storage cell in a manner such that the insert material would cover two sides of the storage cell. The rack inserts are to be installed in the same orientation in every storage cell, so any adjacent storage cells will contain one panel of the composite neutron absorbing material between them.

The NETCO-SNAP-IN® neutron absorbing inserts are designed to be an integral part of the existing QCNPS spent fuel racks and are depicted graphically in Attachment 3a and 3b to the licensee's July 16, 2013, submittal. The NETCO-SNAP-IN® inserts are formed with a greater than 90-degree bend angle. This requires compression of the rack insert to install it into the SFP storage rack cell. After installation, the insert will conform to the 90 degree angle between adjacent spent fuel storage rack cell walls. When installed, the rack insert wings abut against the two adjacent faces of the SFP storage rack cell wall which retains the inserts in place.

3.1 SFP NCS Analysis Review

3.1.1 SFP NCS Analysis Method

Licensee Technical Reports

The July 16, 2013, submittal provided an NCS analysis and computer code benchmarking analysis to support demonstration of compliance with 10 CFR 50.68. On August 29, 2013, the NRC staff documented concerns with uncertainty in the areal density measurements and the need for additional detail related to a proposed methodology for analyzing future fuel assemblies. The licensee responded in a letter dated September 18, 2013, by withdrawing part of the proposed methodology and providing supplementary information. In response to additional NRC staff concerns, updated versions of the two analyses were submitted on January 22, 2014. The final analysis reports used in the review of this LAR were provided in Attachment 1 [Attachment 4 non-proprietary] to the supplement dated January 22, 2014, and Attachment 3 [Attachment 6 non-proprietary] to the supplement dated April 7, 2014.

Attachment 1 [4] to the January 22, 2014, supplement, HI-2104790, "Nuclear Group Computer Code Benchmark Calculations [- Non-proprietary Version]," Revision 1, presents the benchmarking evaluation performed for the MCNP5 Version 1.51 code used for the NCS analysis, to demonstrate the applicability of the code to geometries and compositions being analyzed and to determine the code bias and uncertainty.

Attachment 3 [6] to the April 7, 2014, supplement, HI-2125245 "Licensing Report for Quad Cities Criticality Analysis for Inserts [- Non-proprietary Version]," Revision 5, presents the NCS analysis for the QCNPS spent fuel storage racks. The report describes the methodology and analytical models used in the NCS analysis to show that the spent fuel storage racks maximum k-effective (k_{eff}) will be no greater than 0.95 when flooded with unborated water.

There is no comprehensive, NRC-approved generic methodology for performing NCS analyses for fuel storage and handling. The methods used for the NCS analysis for fuel in the QCNPS SFP are described in HI-2125245. The computer code benchmarking analyses supporting use of MCNP5-1.51 for this application are described in HI-2104790. These methods are discussed in the supplements dated January 22, April 7, and August 12, 2014. Consequently, the methodology is specific to this analysis and, without further revision, is not appropriate for other applications or use by other facilities or licensees. Certain configurations (most notably channeled fuel and reconstituted fuel) are not comprehensively addressed as part of the described methodology. This is acceptable for the limiting fuel currently stored in the SFP, but the findings of this SE do not generically extend to storage of channeled or reconstituted fuel.

3.1.1.1 Computational Methods

The NRC staff evaluated the computational method validation to verify that the validation study provided is thorough and uses benchmark critical experiments that are similar to the normal-conditions and credible abnormal-conditions models and to confirm that the k-effective (or k_{eff}) bias and bias uncertainty values are properly determined.

The licensee provided a detailed description of the validation study supporting the criticality analysis in Attachment 4 to the submittal dated July 16, 2013. The licensee used validation studies which involve modeling critical experiments that are similar to one or more of the criticality analysis cases. For the criticality calculation, the licensee used MCNP5 Version 1.51, with continuous energy cross-section data based on the Evaluated Nuclear Data File, Version 7 (ENDF/B-VII) neutron cross section library. MCNP5 is a state-of-the-art Monte Carlo criticality code developed and maintained by Los Alamos National Laboratory for use in performing reactor physics and criticality safety analyses for nuclear facilities and transportation/storage packages. The use of MCNP has previously been approved for use in SFP NCS analyses at Grand Gulf and Peach Bottom.

For the depletion calculation to determine the spent fuel isotopic compositions, the licensee used the two-dimensional (2-D) CASMO-4 Version 2.05.14 computer code with a 70-group cross-section library mainly derived from the ENDF neutron cross section library. In some cases, the ENDF data has been supplemented by other data sources. CASMO-4 has been approved by the NRC for depletion analysis with a wide range of boiling water reactor (BWR) and pressurized water reactor (PWR) fuel assembly designs.

The above computer codes, and the nuclear data sets with them, have been used in many NCS analyses and are industry standards. However, this is the first known use of CASMO-4 in the United States (U.S.) to perform depletion calculations for the Westinghouse SVEA-96 Optima2 fuel assembly design. This is further discussed in the section on bounding fuel assembly design (Section 3.1.3.1) and fuel assembly manufacturing tolerances and uncertainties (Section 3.1.3.2).

The spent fuel analysis includes a 5 percent depletion uncertainty factor to cover lack of validation of spent fuel compositions, including fission products, as well as an uncertainty to account for the lack of validation for k_{eff} calculations of burned fuel systems containing minor actinides and fission products. These uncertainties are statistically combined with other uncertainties in the calculation of the maximum k_{eff}. Recent work published in NUREG/CR-7109, April 2012 (ADAMS Accession No. ML12116A128) indicates that an uncertainty of about three percent of the minor actinide and fission product worth should be sufficient to conservatively bound biases that may be associated with calculating k_{eff} for systems with minor actinides and fission products. The value used by the licensee was higher, and therefore, conservative.

The licensee also indicated that several short lived, volatile, and gaseous isotopes are being modeled in the analyses that are not explicitly discussed in the analysis, assumptions, or otherwise evaluated. The licensee provided two studies. One demonstrated that the selected cooling time was conservative. The second study credited inputs similar to guidance provided in NUREG/CR-6487. As the radionuclide release rates should be similar, the NRC staff accepts this approach. However, it should be noted that the licensee was inappropriately combining negative reactivity values with positive reactivity values to obtain the final estimated bias. The negative reactivity values are artifacts of the calculation and are not statistically significant, so it is not appropriate to credit them to reduce the calculated bias. The staff determined the result of determining total bias with all negative reactivity values reset to zero would result in an insignificant increase in reactivity and is easily accommodated by the existing margin to the regulatory limit in the final calculation.

For all MCNP5 calculations, the licensee used reasonable values for the following calculational parameters: number of histories per cycle, number of cycles skipped before averaging, total number of cycles, and the initial source distribution. More importantly, the licensee confirmed that all calculations converged using appropriate checks.

Based on the pedigree of the computer codes and the methodology used by the licensee to address known uncertainties, the NRC staff finds that the computational methods implicit in the codes used for the NCS analyses are appropriate and applied correctly.

3.1.1.2 Computer Code Validation

The purpose of the criticality code validation is to ensure that appropriate code bias and bias uncertainty are determined for use in the criticality calculation. Since the NCS analysis credits fuel burnup, the NRC staff's review included validation of the computer codes and data used to calculate burned fuel compositions and the computer code (MCNP5-1.51) and data that utilize the burned fuel compositions to calculate k_{eff} for systems with burned fuel.

In Holtec Report, HI-2104790, the licensee performed the validation of MCNP5-1.51 by comparing calculated k_{eff} values with several different sets of critical configurations. A total of 532 critical configurations were included. The licensee determined that it would be appropriate to treat all experiments as a single set, but applied the distribution free statistical approach to determine the bias and bias uncertainty because the data was not normally distributed. In addition, the licensee evaluated separate sets of bias and bias uncertainty based on subsets of the critical experiments that exhibited specific storage characteristics. These separate bias/uncertainty sets were examined in Holtec Report, HI-2125245, to confirm that none of them were more limiting than the bias and bias uncertainty as determined based on all experiments, for the conditions expected in the SFP racks.

The licensee used three major sources of critical experiments to validate MCNP5-1.51 for SFP NCS analysis work. The first 156 critical experiments have previously been evaluated by the NRC in NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data." The use of the HTC experiments is important to cover the actinide distribution of burned fuel, as well as to evaluate the criticality implications of different storage arrangements for burned fuel.

The second source of critical configurations is the International Handbook of Evaluated Criticality Safety Benchmark Experiments (IHESBE). The IHECSBE contains criticality safety benchmark specifications that have been derived from experiments that were performed at various nuclear critical facilities around the world. The benchmark specifications are intended for use by criticality safety engineers to validate calculational techniques used to establish minimum subcritical margins for operations with fissile material.

The third source of critical experiments was a number of published benchmarks from various industry and U.S. researchers. The system compositions and geometry fall within a range of parameters consistent with SFP storage configurations. Review of prior NRC-approved applications of MCNP5 with the ENDF/B-VII data set shows that inclusion of this set of benchmarks leads to a larger (more conservative) bias uncertainty. As the licensee demonstrated the similarity of these critical experiments to this application, the NRC staff accepts the use of these benchmarks.

The licensee identified the applicable operating conditions for the validation (e.g., fuel assembly materials and geometry, enrichment of fissile isotope, fuel density, types of neutron absorbers, moderators and reflectors, rack material, and physical configurations). The licensee compared the spectral parameters (e.g., EALF, spectrum type) between the benchmarks and the QCNPS SFP conditions to demonstrate that the selected benchmarks are applicable. The NRC questioned whether the critical benchmarks and experiments addressed in HI-02104790 bounded the area of applicability for the fuel lattices being studied related to the geometry of the SVEA-9 Optima2 lattice. In Attachment 2 [5 – non-proprietary] to the supplement dated April 7, 2014, the licensee provided information which demonstrates that the benchmarks bound the unique features of the SVEA-96 Optima2 lattice not found in other BWR fuel assembly designs, most notably the water gap between quadrants of the fuel assembly.

Appropriate critical experiment data was not available to validate k_{eff} calculations crediting minor actinides or fission products. To address this validation deficiency, an uncertainty was applied. Recent work published in NUREG/CR-7109 indicates that an uncertainty of about 3 percent of the minor actinide and fission product worth should be sufficient to conservatively bound biases that may be associated with minor actinides and fission products. The uncertainty value used by the licensee was larger, therefore, the uncertainty value used by the licensee adequately covers the k_{eff} validation deficiencies.

The licensee's analysis has incorporated a "five percent of the reactivity decrement" uncertainty to cover lack of validation of fuel composition calculations. This uncertainty is calculated as 0.05 times the change in k_{eff} from the fresh fuel without gadolinia to the credited final fuel burnup at peak reactivity, including credit for residual gadolinia. The NRC staff's review determined that the licensee satisfactorily demonstrated that the statistical treatment used in this validation was appropriate for the conditions at QCNPS. Therefore, the staff finds that the uncertainty was calculated by the licensee and applied correctly.

Based on the NRC staff's review of the validation database and its applicability to the compositions, geometries, and methodologies used in the licensee's NCS analyses, the code validation was found to be acceptable and all identified biases and uncertainties were propagated appropriately.

3.1.2 SFP and Fuel Storage Racks

3.1.2.1 SFP Water Temperature

The SFP water temperature was also treated by the licensee in a bounding manner. The design basis calculations were run using the minimum SFP temperature, and follow-up calculations were performed to verify that the maximum SFP temperature did not result in a higher k_{eff} value. The calculations were performed using 293.6 degrees Kelvin (K) cross sections from MCNP, however, the expected variation in cross sections at 277 K (the minimum SFP temperature) is expected to be small. The licensee did some studies using the $S(\alpha,\beta)$ temperature adjustment card in MCNP to show that when adjusting the MCNP cross sections to account for a higher temperature the contribution to k_{eff} is relatively small, but this study could not be performed for the minimum temperature due to MCNP limitations. However, the results of the study support the expectation that this effect would have a minor impact on reactivity.

The NRC staff found that the licensee's use of SFP water temperature in the criticality analysis was treated in a bounding manner. As the approach used supports more reactivity in the modeled depleted fuel assembly, the NRC staff finds the licensee's approach to be acceptable.

3.1.2.2 SFP Storage Rack Models

The QCNPS has multiple racks of varying sizes in the SFP, with water gaps between adjacent racks. As indicated in Attachment 1 to the submittal, QCNPS intends to install NETCO-SNAP-IN® rack inserts in all locations intended for fuel assembly storage by December 31, 2014. In the criticality analysis, the licensee chose to use a bounding approach in which the most reactive fuel assembly lattice is identified as the design basis lattice. The SFP is then assumed to be fully loaded with this lattice at the exposure at which its in-rack reactivity reaches a maximum. In some of the accident configurations, 8x8 arrays are used or a 74x74 array is used to evaluate specific scenarios. Otherwise, the calculations are performed assuming an infinite array. In all cases, this ignores the presence of water gaps between storage rack modules. Increasing the local moderation would increase the worth of the boron in the rack inserts; however, the SVEA-96 Optima2 lattice is under-moderated and thus a higher water-to-fuel ratio may result in increased reactivity. Any potential difference in reactivity from increased moderation is expected to be much less than the margin to the regulatory limit in the final analysis, therefore the staff finds ignoring the water gaps between rack modules is acceptable. Using water in lieu of the Boraflex material that is no longer being credited is acceptable for similar reasons.

An important parameter for criticality in the racks is the Boron-10 (B-10) areal density of the NETCO-SNAP-IN® rack inserts. The licensee's NCS analysis used 0.0116 gm B-10/cm^2 as the areal density of the NETCO-SNAP-IN® rack inserts. The licensee's initial submittal indicated that the manufacturing acceptance criterion for its inserts was based on the lower tolerance limit of the 95 percent probability, 95 percent confidence (95/95) measurement threshold, and was equal to the B-10 areal density assumed in the NCS analysis. The NRC staff questioned this approach because the B-10 areal density used in the NCS analysis is typically a bounding minimum certified value rather than the minimum 95/95 threshold value.

In its supplement dated September 18, 2013, the licensee confirmed the use of the manufacturing acceptance criterion of the lower tolerance limit, at the 95/95 measurement threshold, set at the B-10 areal density assumed in the criticality analysis and as a result the licensee did not include restrictions on the placement of inserts from low density batches. The NRC staff was concerned that some batches might have an areal density less than 0.0116 B-10 g/cm^2. However, the licensee provided data in the supplement dated April 7, 2014, on the actual as-built areal density of the inserts that had been manufactured. The data indicates that it would require an additional reduction in the as-built B-10 areal density of about one sigma beyond the lower 95/95 measurement threshold to potentially result in an insert having an actual B-10 areal density below the acceptance criterion. Therefore, the staff finds the licensee demonstrated that there are no inserts with an as-built B-10 areal density less than 0.0116 B-10 gm/cm^2.

The aforementioned as-built data only accounts for the inserts that had been manufactured for QCNPS at the time that the information was provided to the NRC. Approximately 1,000 more inserts will be necessary to install an insert in every available SFP location at QCNPS. While the data provided by the licensee indicates that the manufacturing process is mostly producing inserts with B-10 areal densities well above the acceptance criterion, a possibility exists that inserts may be produced with lower areal densities than those already manufactured. The NRC staff estimated, based on prior studies, that a reduction in B-10 areal density from 0.0116 g/cm^2 to 0.0110 g/cm^2 for typical BWR SFP racks would result in an approximately 0.003 Δk increase in reactivity. The NRC staff recognizes that this small increase could be accommodated by existing margin to the regulatory limit in the analysis. Therefore, if the manufacturing process leads to inserts with lower 95/95 measurement thresholds that are at or very near the 0.0116 B-10 g/cm^2 acceptance criterion, there is a small possibility that the actual B-10 areal density could be slightly less than 0.0116 g/cm^2. The NRC staff expects the licensee to take action to ensure future inserts are procured consistent with the QCPNS Quality Assurance Program, such that future inserts remain within the criticality analysis assumptions and the regulatory limit is not exceeded.

The licensee's treatment of the minimum B-10 areal density in the NETCO-SNAP-IN® rack inserts was found to be potentially non-conservative based on the use of a lower tolerance limit in manufacturing acceptance criterion document. However, further evaluation indicates that sufficient margin to the regulatory limit exists to accommodate this non-conservatism. The NRC staff evaluated other relevant aspects of the storage rack modeling and found them to be modeled conservatively or using appropriate parameters with any uncertainties addressed, as discussed in the next section. Based on sufficient margin remaining to address the identified non-conservatisms in the licensee's rack model, the NRC staff finds the rack model acceptable for the specific conditions at QCNPS.

3.1.2.3 SFP Storage Rack Models Manufacturing Tolerances and Uncertainties

The manufacturing tolerances of the storage racks contribute to SFP reactivity. The NRC staff's review focused on whether the licensee's determination of the maximum k_{eff} considered either: (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study is expected to include all possible significant tolerance variations in the material and mechanical specifications of the racks. The licensee chose to utilize the former approach with the minimum B-10 areal density and the minimum thickness of the rack inserts. The remaining manufacturing tolerances were addressed using the latter approach.

The licensee's evaluation of the tolerance variations included the following components: SFP cell inner width, SFP cell pitch, SFP cell wall thickness, and rack insert widths. Calculations were performed using the design basis model in MCNP5, which combined the most limiting design basis lattice with the limiting depletion conditions, SFP water temperature, minimum rack insert B-10 areal density, and minimum rack insert thickness. For each set of calculations associated with varying a specific parameter, the maximum reactivity increase was identified. If there was no reactivity increase, then the contribution of this manufacturing tolerance to the uncertainty was considered to be zero. These uncertainties were statistically combined with the other uncertainties and included in the final estimation of k_{eff}. As the uncertainties were independent variations, the NRC finds this approach acceptable in support of the determining the maximum reactivity.

In addition to evaluating the manufacturing tolerances, the normal condition also includes many permutations of how fuel assemblies could be positioned in the SFP cells. The licensee performed the design basis calculations assuming a cell-centered loading with all fuel assemblies oriented similarly. Further calculations were performed using infinite arrays of multi-cell configurations. One series of calculations examined the impact of eccentric loading, where fuel assemblies are placed in different positions within the SFP cell with an effort to determine which positions would increase reactivity. Another series of calculations were performed to determine if the reactivity would increase if the most reactive quadrants of adjacent fuel assemblies in a 2x2 array were oriented towards each other. The maximum uncertainty was statistically combined with the other uncertainties and included in the final estimation of k_{eff}.

In Section 2.6.7 of HI-2125245, the NRC staff noted that the evaluation did not appear to address whether particular normal condition events were bounded by analyzed accident conditions events. In Attachment 1 to the supplement dated April 7, 2014, the licensee explained that a misoriented rack insert would not be considered a normal condition due to administrative controls which require verification of proper installed orientation. HI-2125245 also appeared to indicate that normal conditions of fuel movements, inspections and reconstitution operations were bounded by abnormal/accident scenarios. The licensee clarified in the same supplement that the various fuel configurations that are possible during fuel movements, inspections, and reconstitution operations are bounded by the accident scenario where a fuel assembly is mislocated between the SFP rack and the inspection platform while a fuel assembly is loaded for inspection. This is because any possible arrangement of fuel in the fuel movement machines would be independent of, and less reactive than, the arrangement of fuel in the aforementioned accident scenario. Therefore, even if both fuel arrangements happened at the same time, the accident scenario would still be limiting.

As a result of evaluating the treatment of manufacturing tolerances, uncertainties, and other potential differences between the idealized storage rack model used in the NCS analyses and real-world storage racks, the NRC staff has determined that all factors were appropriately considered in a bounding manner, or explicitly evaluated and any reactivity increase appropriately applied as an uncertainty.

3.1.2.4 SFP Storage Rack Interfaces

The QCNPS SFP racks are all of a single type, with the same rack inserts. The criticality analysis assumes an infinite array of the most limiting fuel lattice loaded into every SFP cell. Consequently, there are no interfaces between different regions to consider. As discussed in Section 3.1.2.2, the water gaps between racks are neglected. However, the rack inserts only cover the south and west walls of each cell. For the majority of SFP cells, this means that the north and east boundaries are shielded by rack inserts in the adjacent cells. There are cells on the periphery which do not contain adjacent cells to the north or to the east. A full 74 x 74 SFP model was executed to verify that the cell-centered and eccentric loadings do not lead to a reactivity increase due to the partial lack of neutron-absorbing material surrounding these cells. This was also considered in the evaluation of accident conditions discussed in Section 3.1.4.

The possible reactivity impacts of the SFP storage rack interfaces was evaluated by the licensee, and found to not result in a more limiting reactivity condition.

3.1.3 Fuel Assembly

3.1.3.1 Bounding Fuel Assembly Design

Section 2.3.1 of Holtec Report HI-2125245, provides information on the process of selecting the design basis lattice that is used to perform the final detailed NCS calculations. **[[**

]] Future fuel was not explicitly considered. The design basis fuel assembly selection was evaluated by a review of the analyses demonstrating that the fuel lattice used in the analysis is appropriate for the specific conditions identified at QCPNS.

[[

]]

The final step to determine the design basis lattice was to perform MCNP5 calculations on a group of lattices, including the most reactive GE14 lattice and all of the lattices from the highest-reactivity SVEA-96 Optima2 fuel assembly. All comparisons were based on peak reactivity and typical fuel assembly design parameters as presented in Tables 5.1(a) through 5.1(e) of HI2125245. Several additional nuances were also addressed through sensitivity studies, namely:

- Certain modeling simplifications in CASMO-4 for the SVEA-96 Optima2 fuel lattice, which are further discussed in Section 3.1.3.3.1 of this SE.
- While the SVEA-96 Optima2 fuel assembly cannot be removed from its fuel channel, the GE14 fuel assembly can be. Therefore, calculations were performed to demonstrate that the limiting GE14 fuel assembly will not become more limiting if it is de-channeled. These calculations included the potential impact of eccentric loading in the SFP rack since de-channeled fuel may have a tighter pitch in eccentric loading patterns than channeled fuel.

The design basis fuel assembly selection was performed using nominal core operating parameters as documented in Table 5.2(c) of HI-2125245. Further studies were performed to establish bounding parameters for use in the design basis calculations, henceforth referred to as "design basis core operating parameters." These studies are discussed in the next section, the results of which were also used in the additional sensitivity studies listed above.

The selection and modeling of the fuel lattice used in the NCS analyses was extensively reviewed by the staff, and found to be appropriate with the exception of one potential non-conservatism in the selection of the design basis lattice. This non-conservatism can be accommodated in the margin to the regulatory limit inherent in the final NCS analysis. While some of the methodologies used may not be appropriate for general application, they were verified to be appropriate for the specific applications utilized in the licensee's NCS analyses.

3.1.3.2 Fuel Assembly Manufacturing Tolerances and Uncertainties

The manufacturing tolerances of the storage racks and fuel assemblies contribute to SFP reactivity. The NRC staff review focused on whether the determination of the maximum k_{eff} consider either: (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant tolerance variations in the material and mechanical specifications of the fuel and racks. The licensee chose to utilize the latter approach for the fuel assembly manufacturing tolerances.

The licensee's evaluation of the tolerance variations included the following components: fuel enrichment, gadolinia loading, fuel pellet density, fuel pellet outer diameter, fuel cladding inner diameter, fuel cladding outer diameter, fuel pin pitch, fuel sub-bundle (quarter assembly) pitch, combined channel width dimensions, and combined channel wall thicknesses. Calculations were performed using the design basis model in MCNP5, which combined the most limiting design basis lattice with the limiting depletion conditions, SFP water temperature, minimum rack insert B-10 areal density, and minimum rack insert thickness. For each set of calculations associated with varying a specific parameter, the maximum reactivity increase was identified. If there was no reactivity increase, then the contribution of this manufacturing tolerance to the uncertainty was considered to be zero. These uncertainties were statistically combined with the other uncertainties and included in the final estimation of k_{eff}. As the uncertainties were independent variations, the NRC finds this approach acceptable in support of the determining the maximum reactivity.

[[

]]

The licensee only considered the design basis lattice (of the SVEA-96 Optima2 fuel assembly design) in the full evaluation of the biases and uncertainties due to manufacturing tolerances, [[ ]] and fuel assembly positioning within the SFP cell. The most reactive GE14 assembly was evaluated for a limited number of eccentric positioning scenarios, with the results documented in Table 7.2(b) of HI-2125245. The eccentric positioning bias is known to be unpredictable and, in some cases, quite large, so it was explicitly evaluated. Other biases and uncertainties were not evaluated for the GE14 fuel assembly design. Since other fuel assembly designs have different fuel-moderator ratios and other relevant characteristics, the reactivity impact of the biases and uncertainties could potentially be larger. In this case, the reactivity difference between the most reactive SVEA-96 Optima2 lattice at its limiting depletion conditions and the most reactive GE14 lattice at its limiting depletion conditions is more than 0.04 Δk, so the difference in biases and uncertainties is not likely to lead to a different criticality analysis that is more limiting.

Reconstituted fuel may be more reactive due to changes in the overall moderation. In Attachment 2 to the supplement dated April 7, 2014, the licensee addresses the reconstitution of legacy fuel, and indicated that any future reconstitutions would be explicitly evaluated and processed via 10 CFR 50.59. The staff finds the justification provided for reconstitutions of legacy fuel is acceptable, since the fuel that has undergone reconstitution is of such low reactivity that it is very unlikely that the reconstitution would result in a change in the limiting fuel assembly. However, the licensee did not provide any methodology that would be used in evaluating the reactivity change due to reconstitution, including what uncertainties, manufacturing tolerances, and/or biases would be incorporated. As a result, NRC staff could not perform an evaluation of any methodology planned for use to evaluate the criticality impact of future reconstitutions and this safety evaluation is not applicable to such scenarios.

The NRC staff evaluated the treatment of various fuel manufacturing tolerances, uncertainties, and other potential differences between the fuel lattice model and "real-world" fuel lattices likely to be found at QCNPS. Two specific situations, fuel channel bowing/bulging and fuel reconstitution were found to be treated in an acceptable manner for existing fuel in storage at QCNPS. As a result, the staff finds the uncertainty evaluation of the fuel lattice model to be acceptable for the purpose of approving the current LAR. However, a different methodological treatment of the two aforementioned situations may be necessary for future fuel lattices.

3.1.3.3 Spent Fuel Characterization

Characterization of fresh fuel is based primarily on uranium-235 enrichment, fuel rod gadolinia content and distribution, and various manufacturing tolerances. The manufacturing tolerances are typically manifested as uncertainties, as discussed above, or are bounded by values used in the analysis. These tolerances and bounding values would also carry through to the spent nuclear fuel, common industry practice has been to treat the uncertainties as unaffected by the fuel depletion. The characterization of spent nuclear fuel is more problematic. Its characterization is based on the specifics of its initial conditions and its operational history in the reactor. That characterization has three main areas: a burnup uncertainty, the axial and radial apportionment of the burnup, and the core operation that achieved that burnup.

3.1.3.3.1 Burnup Uncertainty

In the Kopp Memo, the NRC staff provided considerations for evaluating burnup uncertainty:

> A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.

The licensee used this approach to address the uncertainty in the burned fuel compositions. This uncertainty was statistically combined with the other uncertainties and included in the final estimation of k_{eff}.

The licensee used the 2-D CASMO-4 code to calculate the isotopic composition of the spent fuel as a function of fuel burnup, initial feed enrichment, and decay time. In a letter dated October 18, 1999 (Reference 24), the NRC staff approved CASMO-4 for BWR depletion calculations with AREVA and GNF fuel as part of the CASMO-4/MICROBURN-B2 methodology. However, CASMO-4 has not previously been approved by the NRC for use in a depletion methodology for the SVEA-96 Optima2 fuel assembly design. As discussed in Section 3.1.3.1, there are indications from the calculational results presented in HI-2125245 that a large bias exists for CASMO-4 relative to MCNP5. The uncertainty in the k_{eff} introduced by the depletion isotopic uncertainty was addressed by applying the 5 percent of the reactivity decrement from depletion as an uncertainty component in the determination of the maximum k_{eff}, consistent with NRC guidance. One of the implicit assumptions for developing such a rule-of-thumb is that the depletion methodology is consistent with past NRC-approved reload analysis practices.

One major consideration is that CASMO-4 uses several simplifications in the modeling of the SVEA-96 Optima2. Table 7.3 of HI-2125245 shows the results of studies performed to investigate the impact of these geometry modifications on the results by constructing MCNP5 models that match the CASMO-4 geometry as closely as possible. The results show that the MCNP5 k_{inf} results are conservative relative to CASMO-4**[[**

]] this type of significant difference in k_{inf} may impact the accuracy of the calculated isotopic changes arising from depletion. In addition, while many of the fuel assembly design features are similar to that used by GNF and AREVA fuel designs, some characteristics, such as the use of "water wings," result in a more heterogeneous fuel lattice that may require special modeling consideration to account for local spectral effects.

In Attachment 2 to the supplement dated August 12, 2014, the licensee provided additional calculations to demonstrate that CASMO-4 and MCNP5 would calculate similar k_{inf} values if similar inputs and geometry were used. In addition to modeling the same geometry with both codes, the licensee made two changes to the CASMO-4 and MCNP5 inputs to ensure that they are consistent in the isotopes included in the calculation. The results demonstrate good agreement between the codes (approximately one sigma), with the change in k_{inf} relative to the values from Tables 7.1(a) and 7.2(a) in HI-2125245 being consistent with expectations. This suggests that the apparent difference in k_{inf} calculated by the two codes is mostly due to the difference in treatment of specific isotopes. The calculational options are only used to determine how the isotopes are treated in the k_{inf} calculation in the rack geometry. The isotopes in question are treated normally during depletion, so the difference would not be expected to affect the isotopic changes as a result of depletion.

The NRC staff determined the licensee's evaluation of the uncertainty in the fuel depletion calculations and the subsequent application uses of a reactivity uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest, and the approved methodologies for performing depletion of nuclear fuel, and therefore, is acceptable.

3.1.3.3.2 Axial Apportionment of the Burnup or Axial Burnup Profile

The standard BWR peak k_{inf} analysis technique uses either 2-dimensional models or a 3-dimensional model with uniform axial burnup distributions. Generally, this is appropriate because the peak in limiting assembly reactivity occurs at lower burnups where the uniform axial burnup distribution is conservative. If one were to credit assembly burnup beyond the limiting peak reactivity burnup, at some assembly burnup value, the use of the uniform axial burnup would become non-conservative.

Based on the licensee's proposed approach for dealing with the axial burnup distribution being conservative, the NRC staff finds this acceptable.

3.1.3.3.3 Burnup History/Core Operating Parameters

The reactivity of light water reactor fuel varies with the conditions the fuel experiences in the reactor. This is particularly true for BWR fuel NCS analyses using the SCCG peak k_{inf} analysis method. As a result of the usage of Gd_2O_3 in fuel rods, fuel assembly reactivity increases as the gadolinium isotopes are depleted. The value of the in-rack k_{eff} at peak reactivity is affected by the reactor depletion parameters in several ways.

Factors that lead to a more thermal neutron energy distribution cause the $^{155\&157}Gd$ and fission products to deplete more quickly and reduce plutonium generation. This causes the peak reactivity condition to be reached earlier, achieving a higher in-rack k_{eff} value. Increased water density and decreased void fraction lead to a more thermal neutron energy distribution and to lower fuel rod temperatures due to improved fuel rod cooling.

Factors that lead to a less thermal neutron energy distribution cause the $^{155\&157}Gd$ and fission products to be depleted more slowly and result in increased Pu generation. Decreased water density, increased void fraction, and control rod usage all result in neutron energy spectrum hardening.

The ISG, DSS-ISG-2010-01, provides guidance that depletion simulations should be performed with parameters that maximize the reactivity of the depleted fuel assembly. The ISG, DSS-ISG-2010-01, references NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," which discusses the treatment of depletion parameters. For fuel and moderator temperatures, this document recommends using the maximum operating temperatures to maximize plutonium production. This recommendation is also applied to the moderator density for BWRs, but in practice, the high-void state is typically not the limiting condition for peak reactivity analyses. The limiting lattice k_{inf} value is established as the maximum value for a given fuel lattice under all possible operating conditions. The higher moderation that occurs in the no-void condition results in a more rapid depletion of the gadolinia, causing the k_{inf} to peak earlier and higher. A lower moderator density results in a harder neutron spectrum and increased Pu production, but this effect is generally not large enough in BWRs to compete with the U-235 depletion that occurs prior to the later peak in k_{inf}. The licensee performed calculations using the design basis lattice, as shown in Table 7.4 of HI-2125245, which confirms that the use of a moderator density corresponding to higher density water combined with the higher values for the fuel temperature and moderator temperature was limiting.

An estimate for the specific power and operating history of about 0.002 Δk_{eff} using operating histories is provided in NUREG/CR-6665. Based on the difficulty of reproducing a bounding or even a representative power operating history, the NUREG recommends using a constant power level and retaining sufficient margin to cover the potential effect of a more limiting power history. The licensee chose to perform the design basis calculations at a maximum value for the power density, and perform calculations (documented in Table 7.4 in HI-2125245) showing that a modest decrease in power density does not result in a reactivity increase. The moderator temperature cannot increase above saturation, and the impact on void fraction is already accounted for by the conservatively high void fraction used in the sensitivity studies on the core operating parameters. Consequently, no power history uncertainty is applied in the final design basis calculation. The licensee has not demonstrated that the use of a constant maximum power density bounds all possible operating histories. However, as the final margin to the regulatory limit is sufficiently large to accommodate the estimated 0.002 $\Delta k_{eff,}$ to account for the specific power and operating history, the NRC staff finds this approach acceptable.

The licensee considered the impact of rodded operation as it affects the reactivity of the discharged assembly by establishing two broad bounding conditions: rodded operation and unrodded operation. The evaluation of the rodded operation scenario is a somewhat artificial situation, because it does not account for the reduction in power density resulting from insertion of the adjacent control blade. However, the change in power density is small compared to the reactivity impact of the harder neutron spectrum due to the presence of the control blade. As such, the NRC staff finds this approach is acceptable, since a fuel assembly would be unlikely to be controlled during its entire depletion to peak reactivity. The results for the SVEA-96 Optima2 design basis lattice in Table 7.4 of HI-2125245 show that rodded operation results in a more limiting peak in-rack reactivity.

The licensee also noted that some of the values calculated for the most reactive GE14 lattice using MCNP5 and CASMO-4 using different core operating parameters did not result in expected trends. As a result, a complete series of calculations was performed to investigate the response of the limiting GE14 lattice to the core operating parameters. A different set of core operating parameters resulted in a more limiting in-rack k-infinity for the most reactive GE14 lattice. This difference was not enough to make this lattice more limiting than the design basis lattice. A similar investigation was not performed for other fuel assembly designs, which NRC staff finds to be acceptable because it is very unlikely that any such reactivity impacts would exceed the in-rack reactivity difference between the SVEA-96 Optima2 design basis lattice and the next most reactive fuel lattice that is not of the SVEA-96 Optima2 or GE14 design type (i.e., the most reactive GE 8x8 lattice).

All other core operating parameters were established at the values that are shown to maximize the reactivity of the design basis lattice in the rack configuration. This approach is conservative because it effectively applies multiple conflicting conditions at the same time. For example, it is not realistic to include high fuel temperature and control blade presence at the same time. Control blade insertion reduces assembly power level, thus reducing the fuel temperature. The final calculations were run using the limiting depletion conditions, so the variation due to depletion conditions was treated implicitly, rather than explicitly as a separate bias. The NRC staff finds that since this approach is conservative, it is acceptable.

3.1.3.3.4 Integral Burnable Absorbers

As is typical for BWR plants, QCNPS utilizes gadolinia poison to help control reactivity and peaking within fuel assemblies. The specific characteristics of the gadolinia poison loading (location of rods, gadolinia concentration, etc.) may affect the relationship between the SCCG k_{inf} for all SVEA-96 Optima2 lattices currently in use at QCNPS, along with a substantial number of leading potential candidates for limiting lattices as part of the design basis lattice selection. As a result, the various gadolinia loading patterns and their impact on the depletion has been explicitly captured in the calculations. No removable burnable absorbers are used, so there is no need for any further evaluation of the burnable absorbers (with the exception of the gadolinia loading uncertainty evaluated as part of the fuel tolerance calculations).

Based on the incorporation of gadolinia patterns and impacts in the depletion analysis, the NRC staff finds that the treatment of burnable absorbers is appropriate for the specific conditions at QCNPS.

3.1.4 Analysis of Abnormal Conditions

Section 2.6 of HI-2125245, presents the abnormal conditions considered in the analysis. The licensee considered the following abnormal conditions:

- SFP temperature exceeding the normal range
- Dropped fuel assembly
- Storage cell distortion
- Missing insert
- Misloading of a fuel assembly into a SFP cell with no insert
- Mislocated fuel assembly (fuel assembly positioned outside the storage rack)
- Mis-installment of an insert (incorrect orientation)
- Mechanical wear of the insert
- Rack movement

The licensee performed explicit calculations for high SFP temperature, fuel assembly misloading in a SFP cell with no insert, and mislocated fuel assembly (for all possible locations outside the SFP where the mislocated fuel assembly will be face-adjacent to other fuel assemblies). The licensee also performed a calculation to confirm that an empty cell without an insert did not result in any reactivity increases. The NRC staff notes that the calculation with an empty cell without an insert should technically be considered as part of the normal condition due to the presence of several unusable locations that will not have inserts. Since the presence of an empty cell with no insert would decrease the overall reactivity of the SFP, the staff finds that it is acceptable to ignore the potential presence of any such cell(s). The limiting condition was found to be when a fuel assembly is misloaded into a SFP cell that has no insert, with all fuel assemblies in the rack oriented eccentrically towards the unpoisoned part of the SFP rack. The same uncertainties and biases from the normal condition calculation were then applied to obtain the k_{eff} for the accident condition.

The licensee considered it to be unnecessary to perform calculations for the rest of the scenarios for the following reasons:

- Dropped fuel assembly – bounded by the mislocated fuel assembly scenario, because there is no way to orient a fuel assembly that results in higher neutronic coupling than the normal condition (in a SFP cell) or the postulated mislocated fuel assembly scenarios.
- Storage cell distortion – insignificant reactivity impact except insofar as it prevents a rack insert from being installed, in which case it is bounded by the missing insert calculations (with and without fuel loaded in the cell with no insert).
- Mis-installment of an insert – bounded by the missing insert calculations (with and without fuel loaded in the cell with no insert).
- Mechanical wear of the insert – bounded by the missing insert calculations (with and without fuel loaded in the cell with no insert).
- Rack movement – reductions in the gap size between racks is already accounted for by the fact that the calculations assume no gap exists; i.e., the entire SFP pool is treated as a single rack array.

As discussed in Attachment 1 to the April 7, 2014, supplement, the installation of all inserts in the correct orientation will be administratively verified after installation. Therefore, the NRC staff generally agrees with the first three explanations above. As such, incorrect orientation of inserts does not have to be analyzed as a normal condition. However, the staff does not agree with the use of accident conditions to bound mechanical wear of the insert. However, information provided by the licensee in Attachment 1 to the April 7, 2014, supplement indicates that observations to date of installation and insert performance shows that expected wear occurs in locations that do not impact the reactivity control function of the inserts. The licensee has a monitoring program in place to identify significant wear that may occur after long-term use of the inserts. The modeling approach, where the water gap between racks is neglected, was discussed in Section 3.1.2.2 and addresses the licensee's responsibility to ensure that the decision to neglect the water gap is not likely to impact the results substantially enough to result in violation of the regulatory limit.

3.1.5 Margin Analysis and Comparison with Remaining Uncertainties

This section provides evaluation of additional conservatism in the analysis and evaluation of items that may have been treated nonconservatively.

3.1.5.1 Potential Nonconservatisms

- [[

  ]]

- Neglecting water gaps between SFP racks
  Section 3.1.2.4 discusses the interfaces between SFP racks. No water gap is modeled at the interface between SFP racks, which may be slightly non-conservative because the SVEA-96 Optima2 fuel bundle is under-moderated and an increase in the moderator-to-fuel ratio may increase the local reactivity. Since the water gap is very small, the reactivity impact is not expected to be significant.

- Limited evaluation of power history effects
  Section 3.1.3.3.3 discusses information from NUREG/CR-6665 that suggests that determination of the limiting power history is not as simple as performing bounding calculations. However, the potential increased reactivity impact from a more comprehensive study is estimated to be no more than 0.002 Δk.

- Inclusion of several volatile and gaseous isotopes
  Section 3.1.1.1.explains that the licensee included several volatile and gaseous isotopes in the modeling of the fuel lattice. These isotopes may reduce the reactivity of the fuel lattice, but may escape the fuel rod during storage. In Attachment 1 to the April 7, 2014, supplement, the licensee provided an evaluation that demonstrated that escape of some of the credited gaseous and volatile isotopes results in a higher reactivity. However, the increase in reactivity is insignificant and well within the existing margin to the regulatory limit.

- Use of a 95/95 threshold as a bounding value
  The NCS analyses assume that the rack inserts have a B-10 areal density of 0.0116 g/cm^2. This value is also used as the manufacturing acceptance criterion, at a 95/95 measurement threshold. As such, the value assumed in the NCS analyses does not necessarily represent a minimum bounding value. A reduction in the B-10 areal density to 0.0110 g/cm^2 would be sufficient to provide reasonable assurance that there will not be areas in the SFP that potentially have co-located inserts below the assumed minimum value. NRC studies show that a reduction in the B-10 areal density from 0.0116 g/cm^2 to 0.0110 g/cm^2 results in an approximately 0.003 Δk increase in reactivity for a typical BWR rack analysis.

3.1.5.2 Potential Analysis Conservatisms

The analysis includes several aspects that add margin to the analysis.

These include the following:

- Margin to regulatory limit
  The criticality analyses demonstrate that under the bounding conditions discussed in this analysis and with all uncertainties and biases incorporated, there is still **[[ ]]** remaining margin to the regulatory limit.

- Large fission product validation uncertainty applied
  The fission product validation uncertainty applied was much higher than what has previously been demonstrated as adequate. This represents additional margin of approximately 0.001 Δk.

3.1.5.3 Conclusion on Analysis of Margins

Considering both the potential nonconservatisms identified in Section 3.1.5.1 and the conservatisms identified in Section 3.1.5.2, it is concluded that the available margins offset the potential nonconservatisms.

As discussed in Section 2.4 of Holtec Report HI-2125245, the licensee performed an evaluation based on using more realistic rather than conservative assumptions for the following: insert B_4C loading, insert thickness, amount of B-10 in boron, and lattice used for the length of the fuel assembly. The intent of this "margin evaluation" was to identify additional margin that could be used to offset potential effects not already considered in the criticality analysis model. This evaluation consisted of two additional calculations, one using nominal values for the insert specifications, and another using a more realistic axial distribution of lattices for the fuel assembly. The NRC staff finds the approach used to quantify the added margin was found to be deficient in both cases, but the added margin was not necessary to offset the potential nonconservatisms identified during review of the proposed request.

As a result of the above discussion, NRC staff does not accept the conclusions of the margin evaluation, but did not consider it to be necessary to find the proposed request acceptable.

3.1.6 Monitoring Program

As part of this review, NRC staff identified the monitoring program for the neutron-absorbing ability of the NETCO SNAP-IN® rack inserts as being essential to the NRC staff's ability to make a determination regarding the continued safe operation of the SFP. This was due to a lack of documented history on the failure mechanisms for the NETCO SNAP-IN® rack inserts. This program is intended to ensure that sufficient monitoring will be conducted to assess whether any degradation of the NETCO SNAP-IN® rack inserts occur. Should degradation occur compensatory measures are included to ensure that the QCNPS will continue to meet the criticality requirements in 10 CFR 50.68. Additionally a reporting requirement is included to ensure the NRC has sufficient information to assess whether the monitoring program indicates performance of the NETCO SNAP-IN® rack inserts outside of this approval.

In its submittal, the licensee indicates that coupons consisting of the same material as the rack inserts are cut and pre-characterized, then hung on a "coupon tree" in the SFP at the same time that the rack inserts are installed. Periodically, the licensee will remove some coupons to send to a laboratory for testing. An important parameter for nuclear criticality safety is the B-10 areal density, so the program identifies the frequency for monitoring, the requirement to have a qualified laboratory perform the testing, and the minimum criteria. In addition, the licensee committed to performing inspections of inserts installed in high-duty locations to look for localized service wear that may impact the insert's ability to perform its safety function; this type of effect would not be detected through coupon testing that is designed to capture the general effect of exposure to the SFP environment. The licensee's commitment includes the frequency and a basic requirement to ensure that this program is continued.

Finally, the licensee's commitment explicitly identifies that if there is a change in the measured boron-10 areal density for the coupons or any other abnormal indication, such as wear, then these results must be correlated to the inserts to determine the potential impact on the rest of the rack inserts installed in the SFP, and any inserts that are identified as potentially failing the acceptance criteria shall be taken out of service unless the licensee can positively demonstrate that the inserts are acceptable for use. This is an essential requirement to ensure that the licensee is not crediting any material that could potentially be unable to adequately perform its safety function. The licensee's commitment does not explicitly define the impact on allowed storage locations in the SFP, but at a minimum, the cell in which the out-of-service insert is installed would be expected to be excluded from permissible storage locations until it contains an in-service insert. HI-2125245 includes an analysis performed to evaluate a scenario where there is an empty cell in the SFP with no installed insert. The k_{inf} calculated was less than that for the design basis normal condition, so removing fuel from cells where out-of-service inserts are installed should preclude an adverse criticality impact. However, the licensee would be expected to evaluate the impact of taking multiple inserts out-of-service and take appropriate actions to ensure that the allowed storage configurations continue to remain within bounds of the criticality analysis.

The licensee submitted a monitoring program consistent with the above as commitments in the supplement dated November 11, 2014, acknowledging the NRC staff's intent to issue the commitments as a license condition. As discussed above, this program is intended to ensure that sufficient monitoring will be conducted to assess whether any degradation of the NETCO SNAP-IN® rack inserts occur. Should degradation occur compensatory measures are included to ensure that the QCNPS SFPs will continue to meet the criticality requirements in 10 CFR 50.68. Additionally a reporting requirement is included to ensure the NRC has sufficient information to assess whether the licensee's prescribed monitoring program indicates performance of the NETCO SNAP-IN® rack inserts outside of this approval. As these commitments are necessary to ensure protection of the fuel in the event of degradation of the inserts, Paragraphs 3.BB for Unit 1 and 3.AA for Unit 2 will be added to the QCNPS licenses and amended to reflect the following:

1. Ensure that coupon evaluations of Boron-10 areal densities are performed by a qualified laboratory;

2. Ensure that insert evaluations are performed to verify that any service wear is within expected parameters;

3. Ensure that the evaluations are performed at intervals not to exceed four years for

coupon Boron-10 areal density, and 10 years for insert service wear;

4. Ensure that if any inserts are identified as potentially failing the minimum certified Boron-10 areal density criterion, based on correlation of the coupon evaluation or insert service wear evaluation results to inserts, or other abnormal indications, EGC will take affected inserts out of service until it can be positively demonstrated that the minimum certified Boron-10 areal density criterion (0.0116 g/cm^2}) is met for each insert; and,

5. Submit a report to the NRC, within 90 days following completion of evaluations associated with Item 4 above, that describes the testing results, assessments performed, and interim and long-term corrective actions for abnormal indications.

3.1.7 Conclusions

The NRC staff review of the spent fuel storage racks NCS analysis, documented in HI-2125245, identified some nonconservative items. Those items were evaluated against the margin to the regulatory limit and what the NRC considers an appropriate amount of margin attributable to conservatisms documented in the analyses.

[[

]] However, the margin to the regulatory limit alone was determined sufficient to cover this potential non-conservatism. Other non-conservatisms are minor and easily accommodated by the conservatisms in the calculations. The licensee performed an evaluation to identify potential added margin, which was found by the staff to be deficient but was not necessary to find the proposed request acceptable.

3.2 Rio Tinto Alcan Composite Coupon Surveillance Program

The QCNPS criticality analysis currently relies on Boraflex as a neutron absorbing material. Degradation of Boraflex material results from irradiation of the silicon polymer matrix in the spent fuel pool environment. Because of the ongoing degradation of the Boraflex, the licensee relies on a monitoring program to ensure sufficient neutron absorbing capabilities of the material. The licensee's long-term solution to the degradation of the Boraflex is the proposed use of NETCO-SNAP-IN® rack inserts constructed of Rio Tinto Alcan, Inc. composite material with 17 volume percent boron carbide. The minimum certified areal density of the Rio Tinto Alcan, Inc. composite material is 0.0116 g/cm^2. Once the proposed inserts have been installed the licensee will no longer need to credit the Boraflex for neutron attenuation in the criticality analysis and can therefore discontinue the Boraflex monitoring program. The Rio Tinto Alcan, Inc. composite material uses AA1100 alloy as a metal matrix to retain boron carbide. This is the same alloy that is used in Boral neutron absorber materials. Unlike Boral, the Rio Tinto Alcan, Inc. composite material is manufactured by mixing boron carbide powder into molten aluminum. The composite material is then formed into a billet and hot-rolled into sheets that will form the final NETCO-SNAP- IN® inserts. The resulting material is a fully dense homogenous mixture of boron carbide particles embedded in AA1100 series aluminum. The manufacturing process is similar to that of Metamic® material, which the NRC staff has approved for plant specific use in several SFPs at operating reactors.

3.2.1 Program Description

In its submittal dated July 16, 2013, the licensee has provided a Rio Tinto Alcan Coupon Surveillance Program which consists primarily of monitoring the physical properties of the absorber material by performing periodic physical inspection and neutron attenuation testing to confirm the ability of the material to perform its intended function. The purpose of the licensee's coupon surveillance program is to ensure the physical and chemical properties of the Rio Tinto Alcan composite material behave in a similar manner as that described in NETCO's simulated service performance and qualification testing of the material. The coupon program will monitor how the neutron absorber material properties change over time under the radiation, chemical, and thermal environment found in the SFP. This surveillance program will provide a means to detect any significant degradation in a timely manner that will allow for implementation of corrective actions prior to the material losing its ability to perform its intended function.

3.2.2 Long-Term Surveillance Program

The long-term surveillance program will use coupon samples, made from the same material as used for rack construction, to monitor the performance of the neutron absorbing material. A total of 96 coupons will be suspended from a mounting tree that is placed in the SFP at the time of the initial rack insert installation. Each coupon will be examined prior to insertion in the SFP to determine an initial condition that will be compared to the coupons condition when it is removed from the pool after exposure. The coupons will be removed from the SFP, examined, and compared to the initial condition, on a prescribed schedule. These coupons will not be re-inserted into the SFP after being removed for inspection. In addition to dimensional and weight measurements, areal density testing will be performed to ensure that the boron content of the material is sufficient to perform its neutron attenuation function. Based on the proposed sampling schedule there are a sufficient number of coupons to allow for over 80 years of surveillance. The licensee stated that the coupon tree will remain in the pool as long as the spent fuel storage racks continue to be used.

The coupon tree includes bent coupons to simulate stresses on the actual inserts, and bi-metallic coupons to simulate galvanic corrosion that may occur with stainless steel, inconel, and zircalloy in the SFP as shown in Table 3.9-2 to Attachment 1 of the July 16, 2013, submittal. The acceptance criteria for the general and galvanic coupons are shown in Table 3.9-3 of Attachment 1 to the August 12, 2014, supplement.

In addition to the acceptance criteria in Table 3.9-3, the licensee also has acceptance criteria for stress relaxation as shown in Table 3.9-4 to Attachment 1 of the July 16, 2013, submittal. The "Bend" coupons that are installed in the SFP will be pulled and evaluated for stress relaxation. The frequency at which they are pulled is in Table 3.9-5 of Attachment 1 to the submittal.

In addition to the coupon inspections described above, the licensee will also perform camera aided visual examinations on two rack inserts at the same frequency as the general coupon inspection schedule. The visual examinations will monitor for physical deformities such as blistering, pitting, cracking, and flaking. The licensee stated that the examinations will pay special attention to any edge or corner defects in the rack inserts. The licensee will also select an insert to be removed and inspected every 10 years to "verify the inserts have sustained uniform wear over their service life." This will consist of visual inspections and measurements of thickness along the length of the insert, and retention force. The acceptance criteria for these inspections are listed in Table 3.9-7 to Attachment 1 of the submittal. After the inspection, the insert will not be reinstalled.

The program takes into account galvanic corrosion, bend stresses, stress relaxation, service wear, and general corrosion. These are the only known possible mechanisms that may impact the NETCO-SNAP- IN® insert capability to attenuate neutrons. The tests proposed on the coupons are intended to ensure that any degradation mechanism postulated to date would be detected. The frequency of testing and the acceptance criteria of the coupons are intended to provide confidence that any postulated degradation mechanism would be detected in a timely manner. Last, as the licensee is going to inspect and test the rack inserts, this ensures that the licensee will be confirming the results from the coupons testing with the results of the in-service racks inserts. The NRC staff has evaluated the long-term surveillance program and finds that if implemented as described, there is reasonable assurance that the program will detect degradation of the NETCO-SNAP- IN® inserts before it will have an impact on the criticality analysis.

3.2.3 Fast Start Coupon Surveillance

In addition to the long-term surveillance program at LaSalle Unit 2 SFP, the licensee indicates that it has already implemented a "Fast Start" coupon surveillance program. This program consists of 24 coupons on a string (connected with stainless steel chain) that is suspended inside of a spent fuel storage rack cell and surrounded in all adjacent cells with freshly discharged fuel. The intent is to expose the Fast Start coupons to the maximum temperature and gamma irradiation. Two of the coupons will be removed approximately every 6 months and sent to a qualified laboratory for testing, inspection, and comparison to their pre-installation condition. The Fast Start coupons will provide early performance data on the Rio Tinto Alcan composite since the coupon string has been installed prior to rack insert installation. As of January 2013, the results of this program reported in Attachment 11 to the submittal dated July 16, 2013 are that "there was essentially no change in the Rio-Tinto Alcan composite coupons from their pre-use characterization values." The results at LaSalle are intended to be an indicator for QCNPS as to whether there would be any "unanticipated insert material performance" because the pool chemistries are similar.

The NRC staff has reviewed the January 2013 results of the "Fast Start" program. The licensee has demonstrated that the SFP chemistries at QCPNS are sufficiently similar to LaSalle to inform the licensee should some "unanticipated insert material performance" occur. However, the NRC expects that QCNPS maintain their own plant specific surveillance program to ensure that any plant specific SFP conditions that could affect the NETCO-SNAP- IN® Rack Inserts would be detected, addressed, and reported to the NRC staff in a timely manner.

Based on its review of the licensee's coupon sampling program and material qualification tests, the NRC staff concludes that the NETCO-SNAP- IN® rack inserts made from Rio Tinto Alcan composite neutron absorber are compatible with the environment of the QCNPS SFP. Also, the staff finds the proposed surveillance program, which includes visual, physical and confirmatory tests, is capable of detecting potential degradation of the rack insert material that could impair its neutron attenuation capability. Therefore, the NRC staff concludes that the use of Rio Tinto Alcan composite as a neutron absorber rack inserts at QCNPs is acceptable.

3.3 Interim Boraflex Credit

The Boraflex monitoring program directly impacts the validity of the criticality analysis for the period of time prior to the insert installation. Until all of the inserts are installed in the QCNPS FP racks, the interim criticality analysis will still rely on Boraflex credit for subcriticality. The licensee stated that the inserts will be in place no later than December 31, 2014. Until the inserts are all in place, the licensee will be using the existing Boraflex monitoring program to monitor and mitigate the Boraflex degradation.

3.4 Seismic and Structural Integrity

The licensee in Section 3.1.3 of Attachment 1 to the submittal stated that the NETCO-SNAP-IN® inserts are intended to become an integral part of the existing QCNPS spent fuel racks. As such, a portion of the NRC staff's review of the seismic and structural aspects of the LAR focused on determining whether the licensee adequately demonstrated that the inserts have been adequately designed to withstand the loads induced by a safe-shutdown earthquake (SSE) and remain functional (i.e., Seismic Category I).

The NRC staff's review focused on three primary areas: 1) the seismic and structural integrity of the NETCO-SNAP-IN® inserts proposed for installation at the QCNPS; 2) the potential impact of the installation of the inserts on the existing SFP storage racks at the QCNPS, including an assessment of the effects of insert installation on mechanical fuel handling accidents; and 3) the potential impact of the installation of the inserts on the QCNPS SFP structure itself, including the SFP walls, slab and stainless steel liner.

3.4.1 Neutron Absorbing Inserts

The NRC staff's assessment of the structural and seismic aspects of the NETCO-SNAP-IN® neutron absorbing inserts focused on the performance and integrity of the inserts during normal and abnormal loading conditions. Given that the inserts will become an integral part of the existing QCNPS spent fuel racks and thus classified as Seismic Category I, much of the staff's review focused on the ability of the inserts to maintain their intended configuration and continue performing their intended safety function under design basis seismic loading conditions (i.e., SSE). The licensee's approach to demonstrate acceptable structural and seismic performance of the insert was based on a combination of analysis and testing. Additionally, the licensee plans to monitor the structural performance of the inserts continuously using surveillance coupons throughout the life of the inserts.

3.4.1.1 Evaluations and Testing

The licensee performed analytical and confirmatory analyses to evaluate the stresses induced in the inserts during installation. During the licensee's clean pool testing activities, the licensee established a minimum retention force greater than 200 pounds-force (lbf), and the expected installation force less than 1000 lbf and a maximum drag force of 50 lbf. The licensee also measured the insertion, drag and retention forces. In addition, the licensee provided a supplemental discussion regarding the demonstration program, which was undertaken using full-scale test cells fabricated according to the QCNPS spent fuel rack specifications.
The demonstration program results show that adequate retention force is maintained by the inserts such that they remain in place during normal (i.e., fuel handling) and abnormal (i.e., design basis seismic event) loading conditions. While the insert does not appreciably

reduce the mechanical clearance within a rack cell due to its small thickness, the licensee's demonstration program included an assessment of the mechanical design criteria to ensure that any reduced clearance would not have an effect on the normal fuel handling activities undertaken at the QCNPS. The NRC staff notes that for the normal loading conditions during fuel handling activities, the licensee also relies on administrative controls for situations where a fuel assembly may be dragging on an insert due to a warping or bowing of the insert. This is conveyed in Section 3.4.1 of the July 16, 2013, submittal, where the licensee stated that the hoist load cell will recognize the increased load resulting from the resistance induced by a fuel assembly dragging an insert. Additionally, the licensee noted that cells where a fuel assembly is unable to be inserted due to a lack of mechanical clearance will not be used and the fuel assembly will be de-channeled and stored.

In the April 7, 2014, supplement, the licensee stated, in part, that the QCNPS has a peak vertical acceleration of less than 1.0g. Using this value to perform the calculation for the SSE seismic load shows an uplift force that is less than the weight of the inserts. The calculation determines that retention force is not required during a seismic event to prevent the insert from moving in a vertical direction. The weight of the insert itself is enough to prevent this from occurring. Therefore, a conservative stress relaxation rate of 58.5 percent over 20 years in the insert material, assuming a minimum retention force of 100 lbf during insert installation, is acceptable with regards to satisfying the SSE conditions simply because the weight of the insert alone satisfies this condition.

The licensee uses the conservative static coefficient method in its analysis. Under this method the peak acceleration (PA) is multiplied by the static coefficient factor of 1.5 to take into account the effects of both multi-frequency excitation and multimode response. The dead weight (DW) of the insert is multiplied by the SSE acceleration and static coefficient factor.

The vertical seismic design spectra for the QCNPS were taken as a constant value of .08g for operating basis earthquakes and 0.16g for SSEs based on the design basis calculation that determined the seismic design criteria for the QCNPS, Units 1 and 2, Reactor and Turbine buildings.

$$SSE_{acc} = PA = 0.16_g$$

Peak loads in the (+) vertical direction for faulted (SSE) vertical load is calculated as follows:

$$SSE_{load} = (1.5 \times PA \times DW) - DW$$

$$SSE_{load} = (1.5 \times 0.16_g \times 16.83lbs) - 16.83lbs \qquad SSE_{load} = -12.791lbsf$$

The licensee provided the weights and SSE_{load} results for each of the NETCO-SNAP-IN® insert designs on page 12 of Attachment 1 to the April 7, 2014, supplement. Due to the fact that the calculated SSE_{load} is below 0 for every insert design, it can be concluded that during an SSE seismic event, there is insufficient seismic acceleration to cause the NETCO-SNAP-IN® insert to come up and out of the spent fuel storage rack.

3.4.1.2 Staff Evaluation

The NRC staff has reviewed the licensee's assessment of the seismic and structural aspects of the NETCO-SNAP-IN® inserts and finds the licensee's assessment acceptable. This is based on the following testing and assessment implemented by the licensee under normal loading conditions and under abnormal conditions. The staff notes that under normal loading conditions, the licensee performed stress analyses of the insert and undertook successful insertion and drag testing to demonstrate adequate performance of the insert during installation and fuel handling activities. The NRC staff also finds that during an SSE seismic event, there is insufficient seismic acceleration to cause the NETCO-SNAP-IN® insert to come up and out of the spent fuel storage rack. Based on the above considerations, the staff concludes that the inserts should maintain their configuration as integral parts of the QCNPS spent fuel racks and be able to perform their intended safety functions under normal and abnormal loading conditions.

3.5 Spent Fuel Storage Racks

In the July 16, 2013, submittal, the licensee stated that a structural analysis has been performed to show that the in-service loads on the NETCO-SNAP-IN® rack insert during normal and seismic conditions are insufficient to cause an operational failure of the rack insert.

The rack insert has a pre-installed angle of greater than 90 degrees. After installation, the insert will be at approximately 90 degrees. The stress on the structure of the existing SFP storage racks due to the force exerted from the rack insert has been evaluated. As demonstrated in the clean pool test and the on-site demonstration test, installation of the insert will not damage the existing SFP storage rack structural integrity or the rack inserts itself. In Section 3.4.3 of Attachment 1 to the July 16, 2013, submittal, the licensee stated that, the only force that is applied to the racks is through the NETCO-SNAP-IN® rack insert. The yield stress of the aluminum-boron carbide composite material is less than the yield stress of the SFP storage rack material (i.e., stainless steel); therefore, the applied stress on the SFP storage rack is significantly less than the allowable stress for stainless steel SFP storage racks and, therefore, will not damage the existing racks.

The external stresses to the spent fuel storage rack wall during the insert installation were calculated and determined to be negligible. The normal force which prevents the insert from moving under seismic loading induces a shear stress along the contact region of the fuel rack. The shear stress ratio resulting from these normal forces was calculated to be below one percent.

The potential for impact between the spent fuel elements and the structure of the spent fuel rack was considered. Because the rack inserts are installed between the fuel element and the rack, they reduce the distance that would be traveled before an impact would occur. The reduction in travel distance would result in a lower velocity at impact and reduced impact forces. Therefore, the stresses on a fuel element due to an impact with an insert would be significantly lower than an equivalent impact with the spent fuel rack structure.

In the April 17, 2014, supplement, the licensee stated that the SFP racks in their inception were analyzed with stress limits in accordance with Section III, Subsection NF, of the ASME Code 1980 Edition. Load cases which include the combination of deadweight, thermal and seismic were analyzed for both normal operating and abnormal conditions. Additionally, the SFP racks were analyzed for displacement and tipping.

The installation of the inserts superimposed a new load onto the SFP racks. The SFP racks have been analyzed for this condition, and the results of this analysis confirmed that the SFP racks are well within the original acceptance criteria. With the inserts installed, the resulting maximum stress ratio is 86 percent of allowable.

The dead load analysis documented that each insert will be pre-loaded during installation by compressing it from its initial greater than 90 degree bend angle to its installed approximate 90 degree bend angle. The analysis, for bending, showed that the maximum internal stresses developed in the inserts are less than the ultimate strength of the material used to manufacture the insert and is thus acceptable. In addition, the rack inserts are installed between the fuel element and the rack; they reduce the distance that would be traveled before an impact would occur. The reduction in travel distance would result in a lower velocity at impact and reduced impact forces. Therefore, the stresses on a fuel element due to an impact with an insert would be significantly lower than an equivalent impact with the spent fuel rack structure. Therefore, the NRC staff finds that the structural analysis demonstrates that the in-service loads on the NETCO-SNAP-IN® rack insert during normal and seismic conditions are insufficient to cause an operational failure of the rack insert.

Installation of the NETCO-SNAP-IN® rack inserts does not result in a significant increase in the probability of an accident previously analyzed because there are no changes in the manner in which spent fuel is handled, moved, or stored in the rack cells. The probability that a fuel assembly would be dropped is unchanged by the installation of the NETCO-SNAP-IN® rack inserts. These events involve failures of administrative controls, human performance, and equipment failures that are unaffected by the presence or absence of Boraflex and the rack inserts. Subsequently, there is no effect on the design basis deadweight and seismic loads. The NRC staff also notes that there is no expected increase in thermal loads. As such, the staff concludes that the design criteria stipulated by the OT Position Paper and stress limits prescribed by the ASME Code will continue to be satisfied following installation of the NETCO-SNAP-IN® inserts.

3.6 Spent Fuel Pool Structure

The NRC staff's review focused on determining whether the addition of the NETCO-SNAP-IN® neutron absorbing inserts affects the overall structural integrity of the SFP structure, including the walls, slab and liner. The high-density spent fuel storage racks at the QCNPS provide a capacity of 7,554 storage locations in the QCNPS SFPs. In the July 16, 2013, submittal, the licensee stated that a calculation was performed to determine the effects of the installation of the inserts on the Reactor Building structure, SFP, and fuel pool liner. These calculations document that sufficient margin exists and the qualification of the QCNPS Reactor Building, SFP and fuel pool liner is not compromised. Additionally, the force exerted to the insert by a fuel bundle during a seismic event will have minimal effects and the integrity of the inserts will not be compromised. The governing load combinations considered in the design of the reinforced concrete and structural steel of the SFP structures are outlined in Section 3.8.4 of the QCNPS UFSAR.

3.6.1 Staff Evaluation

The NRC staff finds the licensee's justification regarding SFP structures acceptable because the proposed change does not increase the capacity of the Unit 1 and Unit 2 SFPs beyond the current capacity of 3,657 and 3,897 fuel assemblies respectively or total of 7,554 fuel assemblies.

The licensee has previously evaluated the SFP structures for loads which are much larger than those which will be present following installation of the inserts and found those lager loads to be acceptable. Additionally, given that there is no increase in the maximum heat load experienced by the QCNPS SFP, there should be no effect on the thermal loads assumed in the current analyses of record for the QCNPS SFP structures. Combined, these results are in no impact on the SFP structure design basis analyses. Therefore, the NRC staff finds that the licensee has satisfactorily demonstrated that the structural integrity of the SFPs should not affected by the installation of the NETCO-SNAP-IN® inserts.

3.7 Proposed Changes to Technical Specifications

Paragraph 50.36(c)(4) of 10 CFR requires those features such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety be included in the TSs. The licensee has proposed two new TS requirements, TS 4.3.1.1.c and 4.3.1.1.d. The former requirement defines a maximum calculated k_{inf} limit (i.e., before biases and uncertainties are applied) of 0.8991 for fuel assemblies modeled at 4 degrees Celsius in the normal SFP in-rack configuration.

The final proposed TS wording was provided in the August 12, 2014, supplement. This value was updated relative to the July 16, 2013, submittal to change the k_{inf} limit to the maximum k_{inf} corresponding to the design basis lattice in HI-2125245. The latter requirement being added requires that all SFP cells used for fuel storage must have rack inserts installed that meet the minimum B-10 areal density specification used in the criticality analyses. These two design features ensure that the allowed SFP storage configurations are bounded by the NCS analysis.

Based on the proposed language reflecting changes necessary to ensure that the maximum fuel assembly reactivity will not exceed 0.95, at a 95/95 confidence level, if flooded with unborated water, the NRC staff finds the proposed changes acceptable.

3.8 Commitments

As discussed in the Attachment to the supplement dated November 11, 2014, EGC made a commitment to revise the UFSAR to add a description of the Composite Surveillance Program. The NRC staff finds the addition of this commitment as an implementation condition of the amendment to reflect the staff's reliance of the satisfactory implementation of the Composite Surveillance Program as discussed in Section 3.2. As such, the NRC staff has added the following as a condition of implementation of the amendment to ensure that the UFSAR is revised:

> Implementation of the amendment shall include revision of the Updated Final Safety Analysis Report as described in the Attachment to the licensee's letter dated November 11, 2014.

Following incorporation of the Composite Surveillance Program into the UFSAR, future changes to the program will be made under the provisions of 10 CFR 50.59.

3.9 Technical Conclusion

While the methodology evaluated by NRC staff in the NCS analyses submitted for review was found to be acceptable, the methodology did omit details on the appropriate approach to use in evaluation of fuel channel bowing/bulging or reconstituted fuel. The licensee provided a satisfactory explanation for not including these conditions in the current analysis, but declined to provide a detailed explanation of how these conditions might be modeled (including potential conservatisms/non-conservatisms, uncertainties, and biases). Therefore, the findings in this safety evaluation do not extend to such conditions.

Because of the need to evaluate offsetting effects in the licensee's analysis and the lack of a defined methodology for addressing fuel channel bow/bulge and fuel reconstitution, this analysis constitutes a methodology any change to which would be a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses, and thus require prior NRC review and approval. The margin evaluation performed by the licensee as part of the criticality analysis was found to have significant deficiencies, but was not necessary to make the safety determination documented in this SE. However, due to the deficiencies, the calculated margins cannot be credited for any potential non-conservatisms identified in the future.

The NRC staff concludes that the licensee provided an adequate technical justification which shows that the inserts, as an integral part of the existing spent fuel racks, will maintain their intended configuration during normal and abnormal loading conditions, such that they will be able to perform their intended safety function under these conditions. This justification was based on a combination of analysis and testing, which are further substantiated by the licensee's plan to monitor for stress relaxation in the inserts over the lifespan of the inserts. Additionally, the NRC staff notes that no increase in maximum thermal loads is expected with the installation of the inserts. As such, the design bases acceptance criteria related to the existing racks and SFP structures will remain satisfied following installation of the inserts.

Based on these considerations, the NRC staff concludes that there is reasonable assurance that the structural integrity of SSCs affected by the proposed change at the QCNPS will be adequately maintained following installation of the NETCO-SNAP-IN® inserts, such that all affected SSCs, including the inserts, will be able to perform their intended safety functions. The staff concludes that the licensee has demonstrated that the intent of the aforementioned regulatory requirements, related to the seismic and structural aspects of the request, will continue to be satisfied following installation of the NETCO-SNAP-IN® inserts.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to installation or use of a facility's components 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 dated July 8, 2014 (79 FR 38577). 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.

6.0 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) there is reasonable assurance that 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.

7.0 REFERENCES

1. Exelon Generation Company, LLC letter RS-13-148, David M Gullott, Manager, Licensing, Exelon Generation Company, LLC to USNRC document control desk, re: "License Amendment Request – Use of Neutron Absorbing Inserts in Units 1 and 2 Spent Fuel Pool Storage Racks," July 16, 2013 (ADAMS Accession No. ML13199A032).

2. Exelon Generation Company, LLC letter RS-13-228, Patrick R Simpson, Manager, Licensing, Exelon Generation Company, LLC to USNRC document control desk, re: "Supplemental Information Regarding License Amendment Request Associated with Use of Neutron Absorbing Inserts in Spent Fuel Pool Storage Racks," September 18, 2013 (ADAMS Accession No. ML13261A518).

3. Exelon Generation Company, LLC letter RS-14-039, Patrick R Simpson, Manager, Licensing, Exelon Generation Company, LLC to USNRC document control desk, re: "Additional Information Regarding License Amendment Request Associated with Use of Neutron Absorbing Inserts in Spent Fuel Pool Storage Racks," January 22, 2014 (ADAMS Accession No. ML14024A496).

4. Exelon Generation Company, LLC letter RS-14-100, Patrick R Simpson, Manager, Licensing, Exelon Generation Company, LLC to USNRC document control desk, re: "Additional Information Regarding License Amendment Request Associated with Use of Neutron Absorbing Inserts in Spent Fuel Pool Storage Racks," April 7, 2014 (ADAMS Accession No. ML14101A213).

5. Exelon Generation Company, LLC letter RS-14-229, Patrick R Simpson, Manager, Licensing, Exelon Generation Company, LLC to USNRC document control desk, re: "Additional Information Regarding License Amendment Request Associated with Use of Neutron Absorbing Inserts in Spent Fuel Pool Storage Racks," August 12, 2014 (ADAMS Accession No. ML14224A445).

6. Exelon Generation Company, LLC letter RS-14-329, Patrick R Simpson, Manager, Licensing, Exelon Generation Company, LLC to USNRC document control desk, re: "Additional Information Regarding License Amendment Request Associated with Use of Neutron Absorbing Inserts in Spent Fuel Pool Storage Racks," November 11, 2014 (ADAMS Accession No. ML14317A753).

7. NRC Memorandum from L. Kopp to T. Collins, Guidance on the Regulatory Requirements

for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998 (ADAMS Accession No. ML003728001).

8. NRC Division of Safety Systems Interim Staff Guidance DSS-ISG-2010-01, Rev. 0, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," October 13, 2011 (ADAMS Accession No. ML110620086).

9. J.M. Scaglione, D.E. Mueller, J.C. Wagner, and W.J. Marshall, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Criticality (k_{eff}) Predictions," NUREG/CR-7109 (ORNL/TM-2011/514), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, April 2012 (ADAMS Accession No. ML12116A128).

10. D.E. Mueller, K.R. Elam, and P.B. Fox, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," NUREG/CR-6979, ORNL/TM-2007/083, U.S. Nuclear Regulatory Commission, Lawrence Livermore National Laboratory, November 1996.

11. B. L Anderson, R. W. Carlson, L. E. Fischer, "Containment Analysis for Type B Packages Used to Transport Various Contents," NUREG/CR-6487, UCRL-ID-124822, U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, September 2008.

12. J. C. Dean, R.W. Tayloe, Jr., "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," NUREG/CR-6698, U.S. Nuclear Regulatory Commission, Science Applications International Corporation, January 2001.

13. C. V. Parks, M. D. DeHart, J. C. Wagner, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," NUREG/CR-6665, ORNL/TM-1999/303, U. S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, February 2000.

14. *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, NEA/NSC/DOC(95)03, NEA Nuclear Science Committee, September 2012.

15. NRC letter from B. Mozafari, Project Manager, Plant Licensing Branch III-2, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation to M. J. Pacilio, Senior Vice President, Exelon Generation Company, LLC, "Quad Cities Nuclear Power Station, Units 1 and 2 – Acceptance Review – Unacceptable with Opportunity to Supplement (TAC Nos. MF2489 and MF2490)," August 29, 2013 (ADAMS Accession No. ML13241A510).

16. NRC e-mail from B. Mozafari, Project Manager, Plant Licensing Branch III-2, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation to K. Nicely, Exelon Generation Company, LLC, "Request for Additional Information for Use of Neutron Absorbing Inserts in Spent Fuel Pool Storage Racks (TAC Nos. MF2489 and MF2490)," February 28, 2014 (ADAMS Accession No. ML14059A156).

17. NRC e-mail from B. Mozafari, Project Manager, Plant Licensing Branch III-2, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation to K. Nicely, Exelon Generation Company, LLC, "Fw: Draft Request for Additional Information for Use of Neutron Absorbing Inserts in Spent Fuel Pool Storage Racks (TAC Nos. MF2489 and MF2490)," February 24, 2014 (ADAMS Accession No. ML14066A135).

18. NRC e-mail from B. Mozafari, Project Manager, Plant Licensing Branch III-2, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation to K. Nicely, Exelon Generation Company, LLC, "Draft RAI from TAC Nos.: MF2489 and MF2490 Review on Neutron Absorbing Inserts," February 24, 2014 (ADAMS Accession No. ML14066A136).

19. NRC e-mail from B. Mozafari, Project Manager, Plant Licensing Branch III-2, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation to K. Nicely, Exelon Generation Company, LLC, "FW: Draft RAI from TAC Nos.: MF2489 AND MF2490 review on NEUTRON ABSORBING INSERTS," June 23, 2014 (ADAMS Accession No. ML14203A613).

20. NRC letter from A. Wang, Project Manager, Plant Licensing Branch IV, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation to Vice President, Entergy Operations, Inc., Grand Gulf Nuclear Station, "Issuance of Amendment Re: Changes to the Nuclear Criticality Safety Analysis (TAC No. ME7111)," September 25, 2013 (ADAMS Accession No. ML13261A264).

21. NRC letter from R. B. Ennis, Senior Project Manager, Plant Licensing Branch I-2, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation to M. J. Pacilio, President and Chief Nuclear Officer, Exelon Nuclear, "Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments Re: Use of Neutron Absorbing Inserts in Spent Fuel Pool Storage Racks (TAC Nos. ME7538 and ME7539)," May 21, 2013 (ADAMS Accession No. ML13114A929).

22. NRC letter from J. Stang, Senior Project Manager, Plant Licensing Branch II-2, Division of Operating Reactor Licensing, Office of Nuclear Reactor Regulation to P. Gillespie, Site Vice President, Oconee Nuclear Station, "Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding the Use of CASMO-4/SIMULATE-3 Methodology for Reactor Cores Containing Gadolinia Bearing Fuel (TAC Nos. ME4646, ME4647, and ME4648)," August 2, 2011 (ADAMS Accession No. ML101580106).

23. NRC letter from J. N. Donohew, Senior Project Manager, Section 2, Project Directorate IV and Decommissioning, Division of Licensing Project Management, Office of Nuclear Reactor Regulation to G. R. Overbeck, Senior Vice President, Nuclear, Arizona Public Service Company, "Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 – Issuance of Amendments on CASMO-4/SIMULATE-3 (TAC Nos. MA9279, MA9280, and MA9281," March 20, 2001 (ADAMS Accession No. ML010860187).

24. Siemens Document No. EMF-2158(NP)(A), Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," October 1999.

Principal Contributors: K. Wood, SRXB
S. Krepel, SRXB
M. Yoder, ESGB
D. Hoang, EMCB

Date of issuance: December 31, 2014

3. Ensure that the evaluations are performed at intervals not to exceed four years for coupon Boron-10 areal density, and 10 years for insert service wear;

4. Ensure that if any inserts are identified as potentially failing the minimum certified Boron-10 areal density criterion, based on correlation of the coupon evaluation or insert service wear evaluation results to inserts, or other abnormal indications, EGC will take affected inserts out of service until it can be positively demonstrated that the minimum certified Boron-10 areal density criterion (0.0116 g/cm^2) is met for each insert; and

5. Submit a report to the NRC, within 90 days following completion of evaluations associated with Item 4 above, that describes the testing results, assessments performed, and interim and long-term corrective actions for abnormal indications.

As discussed in the Attachment to the supplement dated November 11, 2014, EGC made a commitment to revise the Updated Final Safety Analysis Report to add a description of the Composite Surveillance Program. The addition of this commitment as an implementation condition of the amendment was discussed with Mr. Ken Nicely of your staff on December 10, 2014.

A copy of the related SE is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/
Brenda Mozafari, Senior Project Manager
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-254 and 50-265

Enclosures:
1. Amendment No. 253 to DPR-29
2. Amendment No. 248 to DPR-30
3. Safety Evaluation (Non-Proprietary)
4. Safety Evaluation (Proprietary)

cc w/encls 1, 2, and 3

Distribution via Listserv

DISTRIBUTION:
NONPUBLIC LPL3-2 R/F
RidsNrrDorl Resource
RidsNrrPMQuadCities Resource
RidsAcrsAcnw_MailCTR Resource
EWong, NMSS
RidsNrrDssStsb Resource
RidsNrrDssSrxb Resource
RidsRgn3MailCenter Resource
SKrepel, NRR
DHoang, NMSS
RidsNrrDorlLpl3-2 Resource
RidsNrrDeEsgb Resource
RidsNrrDeEmcb Resource
KWood, NRR
MYoder, NRR

ADAMS Accession Nos. Package: ML14346A309, Non-Proprietary Amendment: ML14346A306 * by memo

OFFICE	LPLIII-2/PM	LPLIII-2/PM	LPLIII-2/LA	ESGB/BC*
NAME	EBrown (BMozafari for)	BMozafari	SRorher (*MHenderson for*)	GKulesa
DATE	12/31/14	12/31/14	12/31/14	04/16/14
OFFICE	EMCB/BC*	SRXB/BC*	OGC	LPLIII-2/BC
NAME	YLi(A)	CJackson	MSpencer	TTate
DATE	07/30/14	12/02/14	12/23/14	12/31/14

OFFICIAL RECORD COPY