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

April 29, 2016

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS TO AMEND RENEWED FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25 TO SUPPORT USE OF A NEW NUCLEAR CRITICALITY SAFETY ANALYSIS METHODOLOGY (CAC NOS. MF5734 AND MF5735)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (NRC or Commission) has issued the enclosed Amendment No. 249 to Renewed Facility Operating License No. DPR-19 and Amendment No. 242 to Renewed Facility Operating License No. DPR-25 for Dresden Nuclear Power Station, Units 2 and 3 (DNPS). The amendments are in response to your application dated December 30, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14364A100), supplemented by letters dated May 8, July 30, and October 15, 2015, and February 8, 2016 (ADAMS Accession Nos. ML15128A305, ML15215A336, ML15288A160, and ML16039A110, respectively). The amendments allow revision to technical specifications (TSs) in support of a new nuclear criticality safety analysis methodology, use of a new fuel assembly design to store AREVA ATRIUM 10XM fuel in the DNPS spent fuel pools (SFPs), and addition of a new TS 4.3.1.1c criticality parameter related to the maximum in-rack infinite k-effective (k_{inf}) limit for fuel assemblies allowed to be stored in the SFP racks.

As discussed in its supplemental letter dated February 8, 2016, the licensee made a commitment to meet several performance objectives related to the DNPS SFP BORAL coupon surveillance program. The NRC has determined that proper implementation of these commitments is essential to ensure that the BORAL material in the DNPS Unit 2 and 3 SFPs will perform as intended. The intent of these commitments is to ensure that any loss or reduction of neutron-absorbing capacity will be promptly detected, and that confirmatory testing is performed to ensure that the minimum B-10 areal density continues to be met for all panels installed. Therefore, the DNPS licenses have been amended to reflect the following requirements for the DNPS SFPs:

Upon implementation of amendments 249 and 242, Exelon Generation Company, LLC (EGC) shall have established and will maintain a coupon surveillance program to assess the degree of degradation, if any, of the neutron-absorbing material. In accordance with this program the licensee shall:

1. Ensure that coupon measurements of B-10 areal density are performed by a qualified laboratory;

2. Ensure that the coupons are removed for evaluation every 10 years;

3. Ensure that should any coupon be identified as failing the minimum certified B-10 areal density criterion based on coupon test results, EGC will perform in-situ testing to confirm that the minimum B-10 areal density (0.02 g/cm2) is met for the BORAL panels installed in the DNPS spent fuel pools, and;

4. Submit a report to the NRC within 90 days following the completion of evaluations associated with Item 3 above. The report shall include; a description of the testing results, the assessments performed, and the interim and long-term corrective actions for abnormal indications.

The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

EA Brown

Eva A. Brown, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

Enclosures:
1. Amendment No. 249 to DPR-19
2. Amendment No. 242 to DPR-25
3. Safety Evaluation

(Enclosure 1)

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 249
Renewed License No. DPR-19

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 December 30, 2014, as supplemented by letters dated May 8, July 30, and October 15, 2015, and February 8, 2016, 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 2.C.(2) of Renewed Facility Operating License No. DPR-19 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 249, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the DNPS Unit 2 Renewed Facility Operation License No. DPR-19 is amended as indicated in the attachment to this license amendment, paragraph 2.C.(21) is hereby amended to read as follows:

(21) Upon implementation of Amendment No. 249 the licensee shall adhere to the following requirements as part of the DNPS unit 2 spent fuel pool coupon surveillance program to ensure that the B-10 areal density of the BORAL remains at or above its minimum credited value and that the regulatory requirement to maintain the Technical Specification value of $k_{eff} \leq 0.95$ continues to be met:

1. Ensure that coupon measurements of B-10 areal density are performed by a qualified laboratory;

2. Ensure that the coupons are removed for evaluation every 10 years;

3. Ensure that should any coupon be identified as failing the minimum certified B-10 areal density criterion based on coupon test results, EGC will perform in-situ testing to confirm that the minimum B-10 areal density (0.02 g/cm2) is met for the BORAL panels installed in the DNPS spent fuel pools, and;

4. Submit a report to the NRC within 90 days following the completion of evaluations associated with Item 3 above. The report shall include; a description of the testing results, the assessments performed, and the interim and long-term corrective actions for abnormal indications.

4. Any planned fuel reconstitution at DNPS that is not bounded by the criticality analysis methodology described in the Holtec International (HI) HI-2146153 analysis will require explicit NRC approval.

5. 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 Attachment 1 to the licensee's letter dated December 30, 2014.

FOR THE NUCLEAR REGULATORY COMMISSION

Justin C. Poole, Acting Branch Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: April 29, 2016

(Enclosure 2)

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 242
Renewed License No. DPR-25

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 December 30, 2014, as supplemented by letters dated May 8, July 30, and October 15, 2015, and February 8, 2016, 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-25 is hereby amended to read as follows:

    B. Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 242, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the DNPS Unit 3 Renewed Facility Operation License No. DPR-25 is amended as indicated in the attachment to this license amendment, paragraph 3.DD is hereby amended to read as follows:

    DD. Upon implementation of Amendment No. 242 the licensee shall adhere to the following requirements as part of the DNPS unit 3 spent fuel pool coupon surveillance program to ensure that the B-10 areal density of the BORAL remains at or above its minimum credited value and that the regulatory requirement to maintain the Technical Specification value of $k_{eff} \leq 0.95$ continues to be met:

    (1) Ensure that coupon measurements of B-10 areal density are performed by a qualified laboratory;

    (2) Ensure that the coupons are removed for evaluation every 10 years;

    (3) Ensure that should any coupon be identified as failing the minimum certified B-10 areal density criterion based on coupon test results, EGC will perform in-situ testing to confirm that the minimum B-10 areal density (0.02 g/cm^2) is met for the BORAL panels installed in the DNPS spent fuel pools; and,

    (4) Submit a report to the NRC within 90 days following the completion of evaluations associated with Item 3 above. The report shall include; a description of the testing results, the assessments performed, and the interim and long-term corrective actions for abnormal indications.

4. Any planned fuel reconstitution at DNPS that is not bounded by the criticality analysis methodology described in the Holtec International (HI) HI-2146153 analysis will require explicit NRC approval.

5. 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 Attachment 1 to the licensee's letter dated December 30, 2014.

FOR THE NUCLEAR REGULATORY COMMISSION

Justin C. Poole, Acting Branch Chief
Plant Licensing Branch LPL III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: April 29, 2016

ATTACHMENT TO LICENSE AMENDMENT NOS. 249 AND 242

RENEWED FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

DOCKET NOS. 50-237 AND 50-249

Replace the following pages of the Renewed Facility Operating License Nos. DPR-19 and DPR-25 with the attached revised pages. The revised pages are identified by amendment numbers and marginal lines indicating the areas of change.

REMOVE	INSERT
License DPR-19	License DPR-19
Page 3	Page 3
Page 9	Page 9
	Page 9A
License DPR-25	License DPR-25
Page 4	Page 4
	Page 10A

Replace the following page of Appendix A, Technical Specifications, with the attached revised page. The revised page is identified by amendment number and marginal line indicating the area of change.

REMOVE	INSERT
Page 4.0-2	Page 4.0-2

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

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

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

(5) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct special nuclear materials as may be produced by the operation of the facility.

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

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2957 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 249, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Operation in the coastdown mode is permitted to 40% power.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.14.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.

(21) Upon implementation of Amendment No. 249 the licensee shall adhere to the following requirements as part of the DNPS unit 2 spent fuel pool coupon surveillance program to ensure that the B-10 areal density of the BORAL remains at or above its minimum credited value and that the regulatory requirement to maintain the Technical Specification value of $k_{eff} \leq 0.95$ continues to be met:

1. Ensure that coupon measurements of B-10 areal density are performed by a qualified laboratory;

2. Ensure that the coupons are removed for evaluation every 10 years;

3. Ensure that should any coupon be identified as failing the minimum certified B-10 areal density criterion based on coupon test results, EGC will perform in-situ testing to confirm that the minimum B-10 areal density (0.02 g/cm^2) is met for the BORAL panels installed in the DNPS spent fuel pools; and,

4. Submit a report to the NRC within 90 days following the completion of evaluations associated with Item 3 above. The report shall include; a description of the testing results, the assessments performed, and the interim and long-term corrective actions for abnormal indications.

D. The facility has been granted certain exemptions from the requirements of Section III.G of Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." This section relates to fire protection features for ensuring the systems and associated circuits used to achieve and maintain safe shutdown are free of fire damage. These exemptions were granted and sent to the licensee in letters dated February 2, 1983, September 28, 1987, July 6, 1989, and August 15, 1989.

In addition, the facility has been granted certain exemptions from Sections II and III of Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." This section contains leakage test requirements, schedules and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted and sent to the licensee in a letter dated June 25, 1982.

These exemptions granted pursuant to 10 CFR 50.12 are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. 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 March 22,1978 with supplements dated December 2, 1980, and February 12, 1981; January 19, 1983; July 17, 1987; September 28, 1987; and January 5, 1989, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

F. 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.

f. Surveillance Requirement 4.9.A.10 - Diesel Storage Tank Cleaning (Unit 3 and Unit 2/3 only)

Each of the above Surveillance Requirements shall be successfully demonstrated prior to entering into MODE 2 on the first plant startup following the fourteenth refueling outage (D3R14).

3. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state power levels not in excess of 2957 megawatts (thermal), except that the licensee shall not operate the facility at power levels in excess of five (5) megawatts (thermal), until satisfactory completion of modifications and final testing of the station output transformer, the auto-depressurization interlock, and the feedwater system, as described in the licensee's telegrams; dated February 26, 1971, have been verified in writing by the Commission.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 242, are hereby incorporated into this renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. Restrictions

Operation in the coast down mode is permitted to 40% power.

DD. Upon implementation of Amendment No. 242 the licensee shall adhere to the following requirements as part of the DNPS unit 3 spent fuel pool coupon surveillance program to ensure that the B-10 areal density of the BORAL remains at or above its minimum credited value and that the regulatory requirement to maintain the Technical Specification value of $k_{eff} \leq 0.95$ continues to be met:

(1) Ensure that coupon measurements of B-10 areal density are performed by a qualified laboratory;

(2) Ensure that the coupons are removed for evaluation every 10 years;

(3) Ensure that should any coupon be identified as failing the minimum certified B-10 areal density criterion based on coupon test results, EGC will perform in-situ testing to confirm that the minimum B-10 areal density (0.02 g/cm^2) is met for the BORAL panels installed in the DNPS spent fuel pools; and,

(4) Submit a report to the NRC within 90 days following the completion of evaluations associated with Item 3 above. The report shall include; a description of the testing results, the assessments performed, and the interim and long-term corrective actions for abnormal indications.

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; and

b. A nominal 6.30 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 0.8895 as determined at 39.2°F in the normal spent fuel pool in-rack configuration.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 589 ft 2.5 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 3537 fuel assemblies.

ENCLOSURE 3

~~(PROPRIETARY)~~

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO LICENSE AMENDMENT REQUEST TO CHANGE

SPENT FUEL POOL CRITICALITY SAFETY ANALYSIS

EXELON GENERATION COMPANY, LLC

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO LICENSE AMENDMENT REQUEST TO CHANGE

SPENT FUEL POOL CRITICALITY SAFETY ANALYSIS

EXELON GENERATION COMPANY, LLC

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By application dated December 30, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14364A100), supplemented by letters dated May 8, July 30, and October 15, 2015, and February 8, 2016 (ADAMS Accession Nos. ML15128A305, ML15215A336, ML15288A160, and ML16039A110, respectively), Exelon Generation Company, LLC (EGC, the licensee) submitted a license amendment request (LAR) to amend Facility Operating License Nos. DPR-19 and DPR-25 to revise the Dresden Nuclear Power Station (DNPS), Units 2 and 3, technical specifications (TSs) in support of a new nuclear criticality safety (NCS) analysis methodology, evaluation of a new fuel assembly design to store AREVA ATRIUM 10XM fuel in the DNPS spent fuel pools (SFPs), and addition of a new TS 4.3.1.1c criticality parameter related to the maximum in-rack infinite k-effective (k_{inf}) limit for fuel assemblies allowed to be stored in the DNPS SFP racks. The following evaluation presents the results of the U.S. Nuclear Regulatory Commission (NRC or Commission) staff review of the licensee's submittal.

The licensee's letter dated December 30, 2014, provided, among other things, an NCS analysis and computer code benchmarking analysis that demonstrates compliance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.68. The licensee's supplemental letter dated July 30, 2015, included updated versions of the NCS analysis as a result of NRC staff concerns associated with the proprietary marking of submitted information.

The supplements dated October 15, 2015, and February 8, 2016, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration dated November 5, 2015 (80 FR 68573).

The final analysis reports used in the review of this LAR are Holtec International (HI) reports HI-2104790, Revision 1 (benchmarking analysis), and HI-2146153, Revision 2 (criticality analysis). These reports were provided as attachments to the licensee's aforementioned letters dated December 30, 2014, and July 30, 2015, respectively. HI-2146153 presents the NCS analysis for the DNPS SFP racks. The report describes the methodology and analytical models used in the NCS analysis to show that the SFP rack maximum k-effective (k_{eff}) will be no greater than 0.95 when flooded with unborated water.

HI-2104790 presents the benchmarking evaluation performed for the Monte Carlo N-Particle, Version 1.51 (MCNP5-1.51), 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.

2.0 REGULATORY EVALUATION

2.1 Regulatory Requirements

Paragraph 50.36(c)(4) of 10 CFR requires, "Design features. Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have 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 requires, "Plant 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."

The credited neutron absorber material (NAM) installed in DNPS SFP racks ensures that the effective multiplication factor (k_{eff}) does not exceed the values and assumptions used in the NCS analyses of record and other licensing basis documents. Neutron absorber materials utilized in SFP racks exposed to treated water or treated borated water may be susceptible to reduction of neutron-absorbing capacity, changes in dimension that increase k_{eff}, and loss of material. A monitoring program is implemented to ensure that degradation of the neutron-absorbing material used in the SFPs, which could compromise the NCS analysis, will be detected. Paragraph 50.68(b)(4) of 10 CFR requires, in part, that:

> [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.

Consistent with the design of DNPS, the SFP NCS analysis does not take credit for soluble boron for normal operating conditions.

Part 50 of 10 CFR, Appendix A, General Design Criterion (GDC)-62, "Prevention of Criticality in Fuel Storage and Handling," requires, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." Specific review criteria are contained in Standard Review Plan (SRP) Section 9.1.2, "New and Spent Fuel Storage," Revision 4.

2.2 Regulatory Guidance

The NRC staff issued a memorandum dated August 19, 1998 (ADAMS Accession No. ML003728001), also known as the "Kopp memo," containing staff guidance for performing the review of SFP NCS analyses. This guidance supports determining 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 non-removable poison inserts.

The NRC Division of Safety Systems (DSS) Interim Staff Guidance (ISG) DSS-ISG-2010-01 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 DSS-ISG-2010-01 references NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology."

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

> [i]n 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, including 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."

2.3 Method of Review

The NRC staff's review included the benchmarking analyses in HI-2104790 and the NCS analyses (HI-2146153) for the DNPS SFP racks. The related TS change proposed in the licensee's December 30, 2014, submittal is addressed below. The NCS analysis analyzes the existing SFP racks with BORAL neutron absorbing material for use to store fuel assemblies of the ATRIUM 10XM design. The review was performed consistent with NRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3, dated March 2007.

2.4 Proposed Technical Specification Change

Paragraph 50.36(c)(3) of 10 CFR states that surveillance requirements are "requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

No changes to the existing TS 4.3.1.1.a or 4.3.1.1.b were proposed in the licensee's submittal. The proposed changes in this license amendment request include a new TS requirement, 4.3.1.1.c. This proposed change will read as follows:

> c. The combination of U-235 enrichment and gadolinia loading shall be limited to ensure fuel assemblies have a maximum k-infinity of 0.8895 as determined at 39.2 °F [degrees Fahrenheit] in the normal spent fuel pool in-rack configuration.

The new TS limit defines a maximum calculated k_{inf} limit (i.e., before biases and uncertainties are applied) of 0.8895 for fuel assemblies modeled at 39.2 °F in the normal SFP in-rack configuration. The in-rack k_{inf} limit is an effective limiting specification because it accounts for the principal fuel assembly reactivity drivers of U-235 enrichment and gadolinium loading. Enrichment and gadolinium loading can vary from assembly design to assembly design. However, compliance with the in-rack k_{inf} limit in the proposed TS 4.3.1.1.c ensures peak in-rack reactivity does not exceed the design basis supporting the TS limit. Using the in-rack k_{inf} limit ensures that the SFP criticality analysis remains bounding. This is the same protection offered by the in-core k_{inf} limit proposed in the Standard Technical Specifications.

This design feature ensures that all fuel assemblies loaded in the SFP are bounded by the NCS analysis submitted as part of this LAR. The licensee stated in the submittal that the Updated Final Safety Analysis Report (UFSAR) will be revised, upon implementation of the approved amendment, as part of the EGC configuration control process.

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

3.0 TECHNICAL EVALUATION

Background

The Unit 2 SFP contains 33 high-density fuel storage racks in two different module sizes for a total capacity of 3537 designated storage locations. The Unit 3 SFP contains an identical set of SFP racks made of type 304 stainless steel. Other than the module sizes, the design of all SFP racks used at DNPS is the same. The SFP racks contain BORAL neutron absorbing material affixed to a type 304 stainless steel box. Type 304 stainless steel sheathing secures each BORAL panel to each face of the box. Multiple boxes are then welded together in a checkerboard pattern to form the storage racks, including filler plates along the side of the SFP racks, ensuring that all adjacent SFP rack cells are separated by at least one BORAL panel. The Units 2 and 3 SFP racks were installed in the early 1980s, and will continue to be credited to meet the NRC sub criticality requirement.

3.1 SFP NCS Analysis Review

3.1.1 SFP NCS Analysis Methodology

There is no comprehensive, NRC-approved generic methodology for performing NCS analyses for nuclear fuel storage and handling. The methods used for the NCS analysis for fuel in the DNPS SFPs are described in HI-2146153. The computer code benchmarking analyses supporting use of MCNP5-1.51 for this application are described in HI-2104790. Some SFP analysis deficiencies were identified during the review, and will be discussed below; sufficient margin is built into the analysis methodology to offset the deficiencies for the existing fuel. Consequently, the methodology is specific to this analysis and, without further revision, is not appropriate for other applications. This is acceptable for the limiting fuel currently stored in the SFPs. Certain configurations, such as reconstituted fuel, are not comprehensively addressed as part of the described methodology. The NRC staff's review of reconstituted fuel is addressed in Section 3.1.3.2 below.

3.1.1.1 Computational Methods

For the criticality calculation, the licensee used MCNP5-1.51 with continuous energy cross-section data based on the Evaluated Nuclear Data File, Version 7, (ENDF/B-VII) neutron cross section library. MCNP5-1.51 is a state-of-the-art Monte Carlo criticality code developed and maintained by Los Alamos National Laboratory (LANL) for use in performing reactor physics and criticality safety analyses for nuclear facilities and transportation/storage packages. The code and its accompanying nuclear data sets have been extensively validated by LANL for various neutron transport calculations, including criticality calculations. The NRC staff finds the underlying neutron transport methodology to be acceptable, but the code needs to be validated for specific applications. The NRC staff review of the licensee's validation of MCNP for its SFP NCS application is discussed in Section 3.1.1.2. For the depletion calculation, used to determine the spent fuel isotopic compositions, the licensee used the two-dimensional (2-D) CASMO-4 (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 was approved by the NRC in a letter dated October 18, 1999 (ADAMS Accession No. ML031290328) for depletion analysis with a wide range of boiling-water reactor and pressurized-water reactor 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. Furthermore, the NRC has previously approved the use of CASMO-4 with the ATRIUM 10 fuel assembly design for reload physics analysis purposes at other plants.

The spent fuel analysis includes a 5 percent depletion worth uncertainty factor to cover lack of validation of spent fuel compositions, including fission products, as discussed in Section 3.1.3.3.1. An additional uncertainty is necessary 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, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses-Criticality k_{eff} Predictions," dated April 2012, indicates that a bias of 1.5 percent of the minor actinide and fission product worth should be sufficient to conservatively bound the uncertainty that may be associated with calculating k_{eff} for systems with minor actinides and fission products. The licensee calculated an uncertainty and statistically combined it with the other uncertainties. The resulting k_{eff} is higher than it would have been if a 1.5-percent-of-worth bias was applied, and therefore, is conservative.

The licensee also described its treatment of short lived, volatile, and gaseous isotopes. **[[**

]] With respect to the short lived isotopes, the licensee provided a study that demonstrated that use of a cooling time of zero hours was conservative. Therefore, the licensee's treatment of the short lived, volatile, and gaseous isotopes is conservative and acceptable.

For all MCNP5-1.51 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 acceptable.

3.1.1.2 Computer Code Validation

Since the NCS analysis credits fuel burnup, it is necessary to consider validation of the computer codes and data used to calculate burned fuel compositions and the computer code and data that use the burned fuel compositions to calculate k_{eff} for systems with burned fuel.

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.

DSS-ISG-2010-01 references NUREG/CR-6698, "Guide for Validation of NCS Calculational Methodology," dated January 2001.

The NRC staff used NUREG/CR-6698 as guidance for review of the criticality code validation methodology presented in the licensee's submittal. The basic elements of validation are outlined in NUREG/CR-6698, including 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.

HI-2104790 addressed the validation of MCNP5-1.51 by comparing calculated k_{eff} values with several different sets of critical configurations. A total of **[[ ]]** 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.

The NRC staff had previously reviewed the validation report as part of the evaluation of a different LAR and found it to be acceptable. Consequently, the validation report was not re-evaluated as part of the review of this LAR. However, the licensee was still required to demonstrate how the validation report's findings were applied for this specific analysis. 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, SFP rack material, and physical configurations).

The licensee compared the spectral parameters (e.g., energy of average lethargy causing fission, spectrum type between the benchmarks, and DNPS SFP conditions) to demonstrate that the selected benchmarks are applicable. The separate bias/uncertainty sets discussed in the previous paragraph were examined in Holtec Report HI-2146153 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. 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. NUREG/CR-7109 indicates that a bias of 1.5 percent of the minor actinide and fission product worth should be sufficient to conservatively bound uncertainties that may be associated with minor actinides and fission products.

The licensee used an uncertainty value, but the reactivity impact due to the licensee's methodology was larger than a 1.5-percent-of-worth bias would be.

Therefore, the uncertainty value used by the licensee adequately covers the k_{eff} validation deficiencies. Consistent with the guidance provided in the Kopp Memo and DSS-ISG-2010-01, the analysis has incorporated a "5 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 gadolinium to the credited final fuel burnup at peak reactivity, including credit for residual gadolinium. This uncertainty was calculated by the licensee and applied correctly.

Based on the staff's review of the validation database and its applicability to the compositions, geometries, and methodologies used in the licensee's NCS analyses, the staff determined that the code validation was 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 treated 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 Kelvin (K) cross sections from MCNP5-1.51, however, the 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 MCNP5-1.51 with different temperatures to show that the reactivity effect due to adjustment of the MCNP5-1.51 cross sections to account for a higher temperature is relatively small, but this study could not be performed for the minimum temperature due to MCNP5-1.51 limitations. However, the results of the study support the expectation that this effect would have a minor impact on reactivity.

The NRC staff determined that the SFP water temperature was used in the criticality analysis 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

Both units have multiple SFP fuel racks of two different sizes in the SFP, with water gaps between adjacent SFP racks. 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 or eccentrically positioned fuel configurations, 8x8 arrays, 16x16, or 80x80, arrays are used to evaluate specific scenarios. Otherwise, the calculations are performed assuming a 2x2 array with periodic boundary conditions. In all cases, this ignores the presence of water gaps between SFP rack modules. The potential for any water gaps would occur at the interface, therefore, the reactivity considerations associated with water gaps are discussed in Section 3.1.5.

According to the DNPS UFSAR, the SFP racks are constructed by welding together a checkerboard arrangement of square tubes. The UFSAR describes each square tube as consisting of BORAL panels on each side enclosed by a thick outer tube and a thinner inner tube, where both tubes are stainless steel. In the NCS analysis documented in HI-2146153, the SFP cells appear to be modeled as a thicker inner tube with the BORAL panels secured to the outside via a sheathing plate welded to the inner tube. This configuration results in a gap between the corners of the square tube when the correct cell pitch value is used.

Based on the diagrams in HI-2146153, this gap appears to be modeled with water. This would be an acceptably conservative approach in that the essential geometry for neutron transport through the BORAL is captured, while the gaps at the corners of the square tubes removes an absorption mechanism for neutrons traveling through that region. However, HI-2146153 is not very clear about the details of the square tube modeling and how they differ from the physical configuration, therefore, the NRC requested clarification. In response, the licensee clarified that the UFSAR wording regarding the tube specifications was inaccurate and the issue was entered into the licensee's corrective action program. The licensee also confirmed that the criticality analysis models described in HI-2146153 are accurate and represent the actual DNPS SFP rack configuration. An important parameter for criticality in the SFP racks is the boron-10 (B-10) areal density of the BORAL panels installed in the walls. The licensee's NCS analysis used 0.02 B-10 grams per square centimeter (g/cm^2) as the areal density of the BORAL material. No information was given about the basis for this value or how the licensee determined that the areal density of the BORAL material is sufficient to ensure that the regulatory limit is met at a 95 percent probability, 95 percent confidence (95/95) level. The final margin to the regulatory limit in the NCS analysis is not very large, so reasonable assurance is needed that the probability of collocated BORAL panels with an areal density below 0.02 B-10 g/cm^2 is acceptably low. The NRC staff requested additional information from the licensee to determine if the areal density value used in the NCS analysis was appropriately bounding for local conditions in the SFP. In response, the licensee stated that the assumed B-10 areal density of 0.02 g/cm^2 is the manufactured minimum certified areal density. This is sufficient to provide reasonable assurance that all BORAL panels in the DNPS SFP will have a B-10 areal density of 0.02 g/cm^2.

Holtec Report HI-2146153 shows that the BORAL panel is modeled as a homogeneous material. In reality, BORAL is fabricated from a mixture of aluminum and boron carbide powders along with aluminum cladding. The boron carbide remains suspended in particulate form in the BORAL panels, thus, the B-10 distribution is granular rather than homogeneous. Recent studies show that there is some reduction in the neutron absorption effectiveness of this type of material compared to a homogeneous mixture with the same B-10 areal density for SFP criticality calculations. This effect is generally attributed to neutron "streaming" and self-shielding, that has generally been shown in studies to be on the order of 0.002-0.003 delta-k-effective (Δk_{eff}) for B-10 areal densities near 0.02 g/cm^2 and boron carbide particle sizes comparable to that found in typical BORAL (~100 μm). HI-2146153 did not include any information regarding this effect, therefore the NRC staff requested that the licensee discuss the potential impact on the effectiveness of the BORAL neutron absorbing function. In response, the licensee stated that:

> [i]n the SFP environment, neutrons of a continuous, mostly thermal energy spectrum are impinging on the neutron absorber from all different directions. Because of this condition, it is acceptable to treat materials such as BORAL as ideal absorbers.

The NRC staff finds that this argument is unsubstantiated and is not based on any quantitative evaluation. Consequently, the NRC staff assigns a 0.003 Δk penalty, based on the aforementioned studies of the impact of the non-homogeneous BORAL core material, to account for a lack of sufficient technical basis. This penalty is included in the Section 3.1.5 discussion.

One more modeling assumption that may affect reactivity is the assumption that the BORAL material remains the same as when it was initially installed. Operating experience has shown that BORAL has, in many cases, exhibited blistering. In the licensee's submittal, the licensee indicated that blistering of up to 1 inch in diameter was found on two out of thirteen coupons tested since installation. Any blisters on the BORAL installed in the SFP racks will displace water. The SFP rack manufacturing tolerance evaluation results indicate that a decrease in moderator between adjacent fuel assemblies will lead to an increase in reactivity, most likely due to a reduction in BORAL absorption effectiveness in the presence of a harder neutron spectrum. This effect was not addressed by the licensee, therefore, the NRC staff requested that the licensee discuss the potential reactivity impact due to any blistering that may exist for its BORAL material. In response, the licensee included a discussion of the low neutron interaction probability in the water gap between the BORAL panel and the stainless steel sheathing by comparing the mean free path of neutrons in water to the nominal water gap thickness. In its review of the licensee's submittal, the NRC staff determined that the licensee appears to be off by an order of magnitude in its estimation of the mean free path of neutrons in water, however, the underlying point remains valid. The NRC staff agrees that there is a low neutron interaction probability in the water gap.

The licensee states that "BORAL blisters are not expected to have any meaningful impact on the neutron spectrum and thus reduce the absorption capability of the BORAL panel." The licensee also justified its position by performing an additional calculation where the design basis model is changed so that the entire water gap between the BORAL panel and the stainless steel sheathing has been voided along the entire panel width and length simulating the worst possible blister that does not cause SFP rack deformation. The licensee determined the result of this calculation was not statistically different from the design basis calculation k_{eff} result thus confirming their position that BORAL blisters are not expected to reduce the absorption capability of the BORAL panel. Finally, the licensee stated that there is no evidence of any blister initiated deformation of the stainless steel sheathing within the SFP rack cells since there have been no reported issues with inserting or removing fuel from the cells in the DNPS SFPs. Based on the licensee's description of the physical phenomenon, confirmation the description through quantitative analysis, SFP rack design, current BORAL blistering data from examined coupons, and no signs of BORAL blister-initiated SFP rack deformation, the NRC staff finds the treatment of BORAL blisters for the DNPS SFPs to be acceptable. However, it will be necessary for the licensee to reconsider its BORAL blistering analysis if subsequent surveillance coupon data invalidates the analysis assumptions. Reconsideration of the BORAL blistering analysis would be part of any operability determination performed, and potentially part of long-term corrective actions.

The NRC staff evaluated other relevant aspects of the SFP rack modeling and determined them to be modeled conservatively or using parameters with uncertainties appropriately addressed, as discussed in the next section. As a result, the NRC staff finds the SFP rack modeling acceptable for the specific conditions at DNPS.

3.1.2.3 SFP Storage Rack Models Manufacturing Tolerances and Uncertainties

The manufacturing tolerances of the SFP racks contribute to SFP reactivity. DSS-ISG-2010-01 does not explicitly discuss the approach to be used in determining manufacturing tolerances, however, past practice has been consistent with the Kopp Memo that determination of the maximum k_{eff} should 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 SFP racks. The licensee chose to utilize the former approach with the minimum B-10 areal density and the minimum thickness of the BORAL panels. 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 BORAL panel width. Calculations were performed using the design basis model in MCNP5-1.51, which combined the most limiting design basis lattice with the limiting depletion condition set, SFP water temperature, minimum BORAL B-10 areal density, and minimum BORAL 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}. This is consistent with past precedent for criticality analyses and with the guidance provided in the Kopp Memo, and thus is acceptable.

The assumption made for the SFP cell wall thickness tolerance was that it was 10 percent of the SFP cell wall thickness value. A review of data from previously submitted SFP NCS analyses associated with NRC-approved LARs from the past decade shows that a tolerance of 10 percent would bound most tolerances used for BWR SFP storage racks, including all SFP racks similar to the ones installed at DNPS. Based on this, the assumptions used for the unknown SFP cell manufacturing tolerances are acceptable.

The tolerance on the sheathing thickness was not evaluated. An increase in the sheathing thickness would be expected to have a similar effect to an increase in the SFP cell wall thickness. If the reactivity effect of an increase in the sheathing thickness is exactly the same as that of the SFP cell wall thickness, then the result would be a 0.00026 Δk increase in reactivity. This increase can be accommodated by the available margin to the regulatory limit. 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 design basis calculations were performed assuming a cell-centered loading with all fuel assemblies oriented similarly. Further calculations were performed using infinite arrays of 2x2 and 8x8 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 and without channels, in an effort to determine which positions may 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 positive reactivity difference, if any, for the eccentric positioning eccentric rotation configurations was applied as a bias, and the applicable calculational uncertainties were statistically combined with the other uncertainties and included in the final estimation of k_{eff}. The NRC staff noted that the reference k_{inf} value used in the eccentric positioning and rotation evaluations was slightly

lower than the value used elsewhere in the NCS analysis. However, this would tend to increase the calculated bias, and thus, is conservative.

As a result of evaluating the treatment of manufacturing tolerances, uncertainties, and other potential differences between the idealized SFP rack model used in the NCS analyses and real-world SFP 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 and the rack model tolerances and uncertainties are therefore acceptable.

3.1.2.4 SFP Storage Rack Interfaces

The DNPS SFP racks are all of a single type, with the same neutron-absorbing materials. The criticality analysis assumes an infinite array of the most limiting fuel lattice loaded into every SFP cell. As discussed in Section 3.1.5, the water gaps between SFP racks are neglected. The SFP rack modules are manufactured by welding together a checkerboard arrangement of square tubes containing BORAL, every other cell along the periphery of the SFP rack module is surrounded by BORAL-containing square tubes on only three sides. The fourth side, along the SFP rack module edge, is covered by a steel filler plate. As a result, it is possible for two SFP racks to be adjacent to each other in such a way that the gaps in the BORAL line up, leaving only steel filler plates between two fuel assemblies. This was evaluated using a 16x16 model where every other BORAL panel is removed from every 8th column. This model was evaluated for both cell-centered and eccentric loading configurations, with the maximum reactivity difference applied as a bias in determining the maximum k_{eff}. The existence of a water gap between SFP racks would only be applicable at the SFP rack interface. For situations where the fuel lattice is under-moderated, it is possible that additional moderator between adjacent fuel assemblies (such as the existence of a water gap between SFP racks) may result in higher reactivity. However, the results from the interface calculations indicate that the eccentric loading configuration (where the fuel assemblies on either side of the interface are closer together) is more conservative, therefore, including the water gap would be expected to reduce the reactivity. This conclusion is supported by multiple uncertainty analyses that indicate that reducing the amount of moderator and distance between fuel assemblies will result in higher reactivities. Therefore, reasonable assurance exists that the water gap has been conservatively modeled in the determination of the interface bias.

Another interface that may exist is along the edge of the SFP rack modules adjacent to the SFP walls. The water and concrete act as reflectors, returning a certain percentage of neutrons back into the SFP cells without BORAL installed on the side facing the SFP wall. However, this configuration is bounded by the above interface evaluation between two SFP rack modules. The symmetry of the interface between two SFP rack modules, with no gap between the modules, is such that the configuration emulates a perfect reflection of neutrons back into the SFP cell for each wall with no BORAL installed. At the SFP pool wall, the reflection will be imperfect and some neutrons are expected to be lost due to leakage.

Therefore, the evaluation of the interface between SFP rack modules performed by the licensee is sufficient to bound all possible interfaces that may exist in the SFP. The possible reactivity impacts of the SFP rack interfaces was evaluated, and the maximum positive reactivity difference was applied as a bias and is therefore acceptable.

3.1.3 Fuel Assembly

3.1.3.1 Bounding Fuel Assembly Design

Section 2.3.1 of HI-2146153 provides information on the process of selecting the design basis lattice that is used to perform the final detailed NCS calculations. This was essentially a three-part process that used different methods to progressively narrow the field of candidate lattices. The licensee considered all current and legacy fuel assembly designs, including 7x7, 8x8, 9x9, and 10x10 fuel from Exxon, GE/GNF, AREVA, and Westinghouse. The licensee intends to transition to the AREVA ATRIUM 10XM fuel assembly design in the near future, so lattices from fuel assemblies planned for future storage in the SFP were included as candidate lattices. The design basis fuel assembly selection was evaluated consistent with Section IV.1 of DSS-ISG-2010-01.

First, all legacy fuel assemblies with an average lattice enrichment of less than about **[[ ]]** were removed from consideration. By "legacy fuel assemblies," the licensee means all fuel stored in the DNPS SFP that is not of the Westinghouse SVEA-96 Optima2 fuel assembly design currently used at DNPS, or of the AREVA ATRIUM 10XM fuel assembly design planned for future use. This is acceptable because the final design basis lattice selected has an average enrichment of **[[ ]]**, and the NRC staff experience with SFP NCS analyses has been that the more modern fuel designs tend to be more limiting.

The likelihood of any legacy fuel assembly proving to be more reactive in the SFP environment to a degree sufficient to overcome a 0.65 percent U-235 difference in enrichment is extremely low. The remaining lattices, including all SVEA-96 Optima2 and ATRIUM 10XM lattices and a few legacy lattices with an average enrichment of more than **[[ ]]**, move to the next step of the screening process.

The second step of the screening calculation is discussed in Section 2.3.1.2 of HI-2146153, where all unique lattices (with the exception of the natural uranium blankets, which are extremely unlikely to be limiting) of the candidate fuel assemblies identified in the prior step were evaluated with CASMO-4. Each lattice was depleted separately in a 2-D CASMO-4 calculation using periodic boundary conditions (consistent with typical 2-D depletion methodologies). Four different sets of core operating parameters are used for each lattice to ensure that the impact of the most important parameters on each lattice is considered, as described in HI-2146153 Table 5.2(c). At points near the peak reactivity for the fuel lattice, the model (including isotopic compositions) was re-initialized in the SFP rack environment to calculate its rack k_{inf}. This approach ensures that the peak rack k_{inf} value will be identified for each lattice, by clearly defining the k_{inf} trend and its peak in the rack environment (which will generally occur near, but not always exactly at, the exposure of peak reactivity in the core). In this step, CASMO-4 is being used for screening purposes, not for the final criticality Analysis of Record (AOR). The licensee did not perform any code validation for CASMO-4, though a proprietary report performed by Studsvik was referenced that performed comparisons for different fuel assembly designs. This report was not made available to the NRC staff for evaluation, therefore the NRC staff was not able to develop any conclusions regarding the accuracy of CASMO-4. CASMO-4 has been previously evaluated by the NRC staff for use in core reload physics calculations and found to be acceptable for use with most fuel assembly designs, including Global Nuclear Fuel (GNF) and AREVA fuel. CASMO-4 has not been evaluated for SFP rack geometries, but the material compositions and geometries are similar

enough that the NRC staff agrees that use of relative reactivity values from CASMO-4 would be an acceptable screening method, as long as the screening criterion is appropriate.

The "screen-out" criterion used is to only perform further evaluation for lattices that are calculated as having an in-rack k_{inf} of greater than 0.85 when using CASMO-4. The primary consideration for whether this criterion is acceptable is whether the k_{inf} threshold is set sufficiently low, such that reasonable assurance exists that variability in the bias of CASMO-4 compared to MCNP5-1.51 will not change the final selection of the design basis lattice. In this situation, the difference between the criterion (k_{inf} = 0.85) and the maximum k_{inf} value calculated for all lattices is larger than the range of variability observed in the bias of CASMO-4 relative to MCNP5-1.51. This conclusion was based on the results provided in Holtec Report HI-2146153 Appendices A and B. Therefore, limiting follow-up calculations to the subset of lattices that meet this criterion is acceptable.

The final step to determine the most limiting lattice is to perform MCNP5-1.51 calculations on the group of lattices identified in the previous step. All comparisons were based on peak reactivity and typical fuel assembly design parameters as presented in HI-2146153, Tables 5.1(a) through 5.1(h).

Once the limiting lattice was identified, the licensee increased the enrichment of all fuel rods to the maximum allowed enrichment of **[[** **]]**, and reduced the number of gadolinium-containing pins by **[[** **.]]** This resulted in a large reactivity increase, with the presumed purpose being to allow for qualification of future fuel that is more reactive than the fuel currently being considered. Future fuel will be evaluated in the SFP rack configuration, as per the TS requirement, so any impacts from how the nuclear composition affects the burnup characteristics of the fuel will be explicitly considered. Therefore, the approach of using a design basis fuel lattice with a nuclear composition that is significantly different (and more reactive) than existing lattices is acceptable. The resulting lattice was used in all design basis calculations. The bounding parameter set was confirmed for use in the design basis calculations, henceforth referred to as "design basis core operating parameters." These studies are discussed below in Section 3.1.3.2.

The selection and modeling of the fuel lattice used in the NCS analyses was reviewed by the NRC staff, and found to be appropriate. While some of the specific assumptions 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 and are therefore acceptable.

3.1.3.2 Fuel Assembly Manufacturing Tolerances and Uncertainties

The manufacturing tolerances of the storage racks and fuel assemblies contribute to SFP reactivity. DSS-ISG-2010-01 does not explicitly discuss the approach to be used in determining manufacturing tolerances, but past practice has been consistent with the Kopp Memo that determination of the maximum k_{eff} should 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 use the latter approach for the fuel assembly manufacturing tolerances.

The licensee's evaluation of the tolerance variations included the following components: fuel enrichment, gadolinium loading, fuel pellet density, fuel pellet outer diameter, fuel cladding thickness, fuel pin pitch, and channel wall thicknesses. Calculations were performed using the design basis model in MCNP5-1.51, which combined the most limiting design basis lattice with the limiting depletion condition set, limiting SFP water temperature, minimum BORAL B-10 areal density, and minimum BORAL 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}. The NRC staff's evaluation of the licensee's methodology to adequately analyze uncertainties, associated with this submittal, is consistent with other criticality analyses completed. Additionally, the licensee's analysis is consistent with the guidance as provided in the Kopp Memo and is therefore acceptable.

The licensee considered various irradiation-caused fuel assembly geometry changes, including fuel rod growth, cladding creep, crud buildup, channel bulging/bowing, and fuel rod bow. The fuel rod growth, cladding creep, and crud buildup are collectively dispositioned by indicating that significant changes in the fuel/cladding geometry due to these physical processes are not expected at the low burnups [[ ]] of the design basis lattice at peak reactivity.

The potential fuel rod bow and channel bulging/bowing was evaluated by performing calculations based on information provided in Holtec Report HI-2146153 Table 5.1(h).

The licensee only considered the ATRIUM 10XM design in the full evaluation of the biases and uncertainties due to manufacturing tolerances, fuel assembly geometry changes due to irradiation, and fuel assembly positioning within the SFP cell. The tolerances for other fuel assembly designs were not evaluated. The reactivity difference between the design basis lattice and the most limiting fuel lattice currently being used or stored at DNPS is 0.03 Δk, which is much more than any potential variation in the manufacturing tolerances resulting from use of a different fuel assembly design. Therefore, explicit calculation of the manufacturing tolerances for other fuel assembly designs is not necessary.

Section 2.3.16 of HI-2146153 states that all reconstituted fuel currently at DNPS did not meet the initial screening criterion applied at the beginning of the design basis lattice identification process. Therefore, the reactivity of these fuel lattices is sufficiently low such that they did not need to be evaluated as potentially limiting lattices even though reconstituted fuel may be more reactive due to changes in the overall moderation. This justification provided for existing reconstitutions of legacy fuel is acceptable. 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.

In the October 15, 2015, response, the licensee indicated that if fuel reconstitution is performed in the future, a criticality analysis will be performed to check if HI-2146153 remains bounding. The licensee indicated that this review will include an explicit criticality analysis of the reconstituted configuration, the results of which will be directly compared to the NCS AOR. Fuel reconstitution procedures typically involve removal of fuel rod(s) and/or replacement of fuel rod(s) with stainless steel rod(s). Since the area of applicability defined by the criticality code validation would cover typical fuel reconstitution, the NRC staff finds that any atypical planned fuel reconstitution or planned reconstitution that is determined by the licensee not to be

bounded by the criticality analysis methodology as defined in HI-2146153, will require explicit NRC approval in accordance with regulatory requirements, as discussed in Section 3.1.5.

The NRC staff reviewed the evaluation and treatment of reconstituted fuel within the constraints of HI-2146153. The NRC staff finds that in general the disposition of any fuel reconstitution as discussed in HI-214653 is acceptable provided that the reconstituted fuel remains bounded by the HI-214653 analysis.

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 DNPS. 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 DNPS. As a result, the staff finds the uncertainty evaluation of the fuel lattice model to be acceptable for the purpose of approving this LAR.

3.1.3.3 Spent Fuel Characterization

Characterization of fresh fuel is based primarily on U-235 enrichment, fuel rod gadolinium 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 fuel; common industry practice has been to treat the uncertainties as unaffected by the fuel depletion. Spent nuclear fuel is characterized on the basis of its conditions and operational history in the reactor. That characterization has three main areas: (1) a burnup uncertainty, (2) the axial apportionment of the burnup, and (3) the core operation that achieved that burnup.

3.1.3.3.1 Burnup Uncertainty

In the Kopp Memo, the NRC staff provided its recommended method 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 CASMO-4 to calculate the isotopic composition of the spent fuel as a function of fuel burnup, initial feed enrichment, and decay time. The NRC staff has approved CASMO-4 for BWR depletion calculations with AREVA and GNF fuel in a letter dated October 31, 1999 (ADAMS Accession No. ML031290328) as part of the CASMO-4 methodology including the ATRIUM 10 fuel design. The reactivity uncertainty was determined based on comparison of the reference case at peak reactivity with a fresh, no gadolinium calculation.

The NRC staff determined that the licensee's evaluation of the uncertainty in the fuel depletion calculations uses a reactivity uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest, calculated using 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 a 2-D model or a 3-dimensional (3-D) 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.

The licensee chose to adopt the standard approach for dealing with the axial burnup distribution using a 3-D model with the limiting lattice modeled along the entire active fuel length. Twelve inches of water were modeled above and below the fuel to act as an axial reflector. Review of the DNPS SFP configuration as described in the UFSAR shows that there is at least that much flooded space above and below the active fuel length. Neglecting the structural materials is conservative because the stainless steel structural material will absorb some neutrons rather than reflecting them back into the active fuel. Twelve inches is sufficiently large compared to the diffusion length for water at room temperature such that the reflection of neutrons back into the active fuel is comparable to that of an infinite thickness of water above and below the active fuel.

Since the licensee applied a conservative approach in the use of standard peak reactivity analysis methods for modeling the axial burnup distribution, the NRC staff finds this approach 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 peak k_{inf} analysis method. As a result of using Gd_2O_3 [gadolinium-oxide] in fuel rods, fuel assembly reactivity initially 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 gadolinium-155 and gadolinium-157 isotopes (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 Gd and fission products to be depleted more slowly and result in increased plutonium generation from neutron energy spectrum hardening. Decreased water density, increased void fraction, and control rod usage all result in neutron energy spectrum hardening.

The NRC DSS Interim Staff Guidance (DSS-ISG-2010-01) provides guidance that depletion simulations should be performed with parameters that maximize the reactivity of the depleted fuel assembly. This guidance includes 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, NUREG/CR-6665 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 an elevated depletion rate of the gadolinium causing the k_{inf} to peak earlier and higher. A lower moderator density results in a harder neutron spectrum and increased plutonium 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}.

Typically, sensitivity studies are performed where each core operating parameter is varied to determine a bounding value that will maximize the impact to the reactivity of the design basis lattice in the SFP. The reactivity impact of specific core operating parameters may vary for different fuel lattice designs, therefore this should also be considered in the design basis lattice selection. The licensee considered the impact of core operating parameters on the design basis lattice selection by using four different sets of core operating parameters in the screening calculations that were selected to consider the most likely candidates for limiting conditions. The licensee took into consideration prior experience with NCS analyses and how the final set of core operating parameters selected compares to the other three; in this regard the licensee's consideration of how the core operating parameters may impact the design basis lattice selection was acceptable. However, the licensee did not do specific sensitivity studies that vary individual core operating parameters to verify that the selected value would result in the most positive impact to the SFP reactivity. The NRC staff requested the licensee to justify the lack of parameter-specific sensitivity studies. The licensee's response confirmed that bounding core operating parameters were selected in determining the design basis lattice selection by performing additional sensitivity studies, therefore the staff finds that the selected core operating parameters for the design basis lattice are acceptable.

NUREG/CR-6665 does not have a specific recommendation for specific power and operating history. NUREG/CR-6665 estimated this effect to be about 0.002 Δk_{eff} using the operating histories it considered. Based on the difficulty of reproducing a bounding or even a representative power operating history, NUREG/CR-6665 merely 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 to perform calculations 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 did not demonstrate that the use of a constant maximum power density bounds all possible operating histories. However, the final margin to the regulatory limit is sufficiently large to accommodate the estimated 0.002 Δk_{eff} from NUREG/CR-6665, therefore the NRC staff does not consider it to

be necessary to perform a more detailed sensitivity study or otherwise justify the lack of a power history uncertainty.

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, this approach is acceptable, especially since a fuel assembly would be unlikely to be controlled during its entire depletion to peak reactivity. The results show that unrodded operation results in a more limiting peak in-rack reactivity. The effect of rodded operation was evaluated using one control rod design, but the results are not expected to differ for other control rod designs due to the similar impact on the neutron flux during core operation.

All other core operating parameters were established at the values that are shown to maximize the reactivity of the design basis lattice in the SFP rack configuration. This approach is an appropriate modeling simplification because the limiting conditions will typically only exist in a shorter portion of high-power fuel assemblies, rather than the entire axial length of the fuel. 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. Since this approach is conservative, the staff finds it is acceptable.

3.1.3.3.4 Integral Burnable Absorbers

As is typical for BWR plants, DNPS utilizes gadolinium poison to help control reactivity and peaking within fuel assemblies. The specific characteristics of the gadolinium poison loading (location of rods, gadolinium concentration, etc.) may affect the relationship between the Standard Cold Core Geometry k_{inf} and the in-rack k_{inf}. The licensee chose to perform an evaluation of the in-rack k_{inf} for all lattices considered in this NCS analysis. As a result, the various gadolinium 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 gadolinium loading uncertainty evaluated as part of the fuel tolerance calculations).

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

3.1.4 Analysis of Abnormal Conditions

Section 2.3.15 of HI-2146153 presents the abnormal conditions considered in the analysis. The licensee considered the following abnormal conditions:

- SFP temperature exceeding the normal range
- Dropped fuel assembly
- SFP rack movement
- Mislocated fuel assembly (fuel assembly positioned outside the storage SFP rack)

Explicit calculations were performed for high SFP temperature, a vertically dropped fuel assembly into a SFP cell with breach of the SFP rack bottom, a missing BORAL panel, and a mislocated fuel assembly (for all possible locations outside the SFP where the mislocated fuel assembly will be near fuel stored in the SFP racks or in the fuel preparation machine). The limiting condition was found to be when a fuel assembly is [[.]] 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:

- Horizontally dropped fuel assembly - fuel would be separated from other fuel by 12 inches by the SFP rack; this configuration is not considered to be more reactive than the mislocated fuel assembly assessments.

- SFP rack movement - reductions in the gap size between SFP 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 SFP rack array.

With respect to the scenarios that were not explicitly evaluated, the NRC staff agrees that the horizontally dropped fuel assembly configuration is not likely to be limiting relative to the mislocated fuel assembly scenario. The most limiting mislocated fuel assembly scenario has a fuel assembly almost directly facing [[.]] This configuration is surrounded, for the most part, by fuel. By contrast, the horizontal dropped fuel assembly would be expected to exhibit weak neutronic coupling with the fuel in the SFP due to the distance, and most of the neutrons associated with the assembly would be lost to leakage. The modeling approach in which the water gap between SFP racks is neglected, was discussed in Section 3.1.5 and includes the NRC staff's assessment that the decision to neglect the water gap appears to be conservative.

Use of the normal condition biases and uncertainties for accident conditions is common practice, however, in this situation, the margin to the regulatory limit is not very large. The reactivity impact due to the biases and uncertainties may increase somewhat with increasing overall reactivity of the configuration. An additional consideration is that a bias due to SFP rack interfaces was analyzed and applied to both the normal and accident scenarios. However, the geometry associated with the SFP rack interface was not included in the mislocated fuel assembly analysis. Review of the DNPS UFSAR suggests that the postulated location for mislocated fuel is adjacent to multiple interfaces between SFP racks. The combination of different reactivity impacts may be multiplicative rather than additive. Therefore, the NRC staff requested that the licensee address these possibilities. The licensee's response in its supplemental letter dated October 15, 2015, included calculated biases and uncertainties specifically for the bounding accident case, using the bounding accident case geometry, and compared the combined bias and uncertainty to that in HI-2146153 based on normal conditions.

The bias and uncertainty analysis shows that use of the normal condition biases and uncertainties relative to the bounding accident condition is conservative by approximately [[ ]] $\Delta k_{eff,}$ therefore the NRC staff finds that the use of the normal condition biases and

uncertainties for accident conditions is acceptable. Given that the licensee performed a thorough evaluation of all potential accident conditions, as discussed above, and considered all possible reactivity impacts, the NRC staff finds the calculation of the k_{inf} for accident conditions to be acceptable.

3.1.5 Margin Analysis and Comparison with Remaining Uncertainties.

Several potentially non-conservative assumptions were identified as part of the NRC staff review of this LAR. A bounding estimate of the reactivity impact for each assumption is listed in the table below, based on NRC staff calculations or studies. In addition, any extra margins to the regulatory limit identified during the review of the NCS analysis are listed. Based on the comparison (presented below), the NRC staff concludes that the available margins offset the potential non-conservatisms.

	Estimated Reactivity Impact (Δk)
Non-conservatisms	
Modeling of BORAL as a homogeneous material	0.003
Neglecting water gaps between SFP racks	~0.0*
Limited evaluation of power history effects	0.002
Total reactivity impact of non-conservatisms	0.005
Conservatisms	
Margin to regulatory limit	0.0131
Large fission product validation uncertainty applied	0.001
Total reactivity impact of conservatisms	0.0141

*This assumption is conservative for the ATRIUM 10XM design basis fuel lattice used in this analysis. This may not be true for other fuel lattice designs that are under-moderated, but the reactivity impact would not be expected to be significant due to the very small size of the water gap.

The NRC staff review of the DNPS spent fuel storage SFP racks NCS analysis, documented in HI-2146153, identified some non-conservative items. Those items were evaluated against the margin to the regulatory limit and what the NRC staff considers as an appropriate amount of margin attributable to conservatisms documented in the analyses. Non-conservatisms are minor and easily accommodated by the conservatisms in the calculations.

While the methodology evaluated by the NRC staff in the analyses submitted for review was found to be acceptable, the methodology did omit details on the appropriate approach to use in evaluation of 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).

As previously stated, the NRC staff finds fuel reconstitution at DNPS to be acceptable. In accordance with regulatory requirements, any result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses, requires explicit NRC approval prior to implementation. Any planned fuel reconstitution at DNPS that is not bounded by the criticality analysis methodology described in the HI-2146153 analysis will require explicit NRC approval. To ensure that any unbounded

conditions from this methodology are appropriately analyzed, the NRC is amending the DNPS license to reflect this requirement.

3.1.6 SFP Neutron Absorber Monitoring Program

The current DNPS SFP high density fuel racks have been in place since the early 1980s. During this timeframe, the licensee has used a SFP corrosion sampling program, also referred to as a BORAL coupon surveillance program, to monitor the condition of the BORAL in the SFP racks. This surveillance program is described in the DNPS UFSAR Section 9.1.2.3.1. In the UFSAR, the licensee states that the intent of the coupon surveillance program is to ensure that any loss of NAM and/or swelling of the storage tubes will be detected. Additionally, the license states that a SFP B-10 loading analysis will be performed should the coupon surveillance program indicate any reduction in the NAM below 0.02g/cm^2, B-10. The B-10 areal density of the NAM must remain equal to or greater than the minimum areal density described in the NCS analysis of record and that no other degradation mechanism impacts the ability of the NAM to perform its design function.

The DNPS coupon surveillance program tests BORAL coupons for indications of corrosion and reduction of neutron absorption capacity. As part of the DNPS license renewal, the coupon testing is performed on a 10-year frequency based in part on NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 2, dated December 2010, the results of earlier coupon testing, and the number of coupons remaining in the pool. Since the coupons are not returned to the spent fuel pool after testing, the licensee has indicated that there are a total of three BORAL coupons remaining. The NRC staff finds the interval for inspection and testing acceptable because the frequency is based on the condition of the neutron-absorbing material and is determined and justified with plant-specific operating experience, not to exceed 10 years. In addition, the testing interval is consistent with NRC guidance found NUREG-1801, via periodic examination.

Following the removal of a coupon from a SFP, it is sent to a qualified laboratory for testing. Prior to the test, the coupon jacket is removed to expose the BORAL specimen. The specimen is then micro-photographed and weighed. It is then cleaned and dried in a three stage process. Following drying, the specimen is micro-photographed a second time at which point length, height, and thickness measurements are taken. After the second microphotograph, the specimen is visually examined for pitting. If pitting is observed, the depth of the pit is measured. Subsequent to performing visual inspections and measuring dimensional and weight parameters, the licensee performs an evaluation of the B-10 areal density. The B-10 evaluation is performed to confirm whether the areal density is at least 0.02 g/cm^2, B-10. The NCS analysis of record credits a minimum certified areal density of 0.02g/cm^2, B-10 for the BORAL material. The NRC staff finds the performance of coupon testing acceptable because it includes measurements of B-10 areal density and of geometric changes in the material. Measuring areal density and geometric changes provides information on whether there is a loss of material and/or reduction in the neutron-absorbing capacity. The licensee states that the behavior of the coupon specimen is used to judge the performance of the NAM in the SFP racks. The coupons in the DNPS SFPs are currently placed in a location where they are exposed to ambient conditions that are similar to that of an average BORAL plate in the SFP storage racks. The NRC staff has determined that the licensee's coupon surveillance program is essential for the detection of any NAM degradation, over time. The physical condition of the NAM, as observed from sampled coupons, can be directly related to the performance of the neutron absorber in the SFP storage racks.

The licensee stated that 13 BORAL coupons have been tested since installation, two of which have been identified as having blisters. The largest blister was 1 inch in diameter. Other coupons have been observed with corrosion pits. The licensee performed neutron attenuation testing on the sampled coupons and determined that the neutron attenuation properties were not impacted. Based on these results, the licensee has determined that the blisters and corrosion pits, identified to date, have no significant impact on the neutron absorbing capability. Furthermore, it was reported that all coupons tested have an areal density of 0.03g/cm^2, B-10 or greater which exceeds the minimum certified areal density for the DNPS SFP racks of 0.02g/cm^2, B-10, a value credited in the NCS analysis of record. Degradation or deformation of the credited neutron-absorbing material may reduce safety margin and potentially challenge the sub-criticality requirement. As controlled by TSs, the NRC staff finds the approach of using the areal density value analyzed in the NCS analysis of record in evaluating test results to be acceptable, because the sub-criticality margin requirement for the spent fuel pool is maintained.

The NRC staff has reviewed the information provided by the licensee regarding the DNPS operating experience and the test results. The NRC staff determined that there is reasonable assurance that the B-10 areal density of the NAM is consistent with assumptions found in the NCS analysis of record.

3.1.6.1 License Condition

The NRC staff identified concerns with the NCS modelling and assumed uncertainties as well as the effect these may have to ensure that the TS value of $k_{eff} < 0.95$ continues to be met. In its supplemental letter dated October 15, 2015, the licensee provided details regarding additional margin in some areas of the NCS model and committed in Attachment 1 to a BORAL coupon surveillance program to confirm that the Boron-10 (B-10) areal density of the NAM remains equal to or greater than the minimum areal density described in the NCS analysis of record. The B-10 areal density serves as a significant input into the determination of the TS limiting k_{eff}. The NRC staff reviewed the proposed coupon surveillance program and was concerned that should it be determined that a coupon has failed, the cause of the failure and the ability to identify the extent of condition of the failure mechanism within the SFP was not well defined. The NRC staff would expect that any testing would increase the sample size to include a sufficient number of panels to make an engineering determination used to verify that all SFP panels meet or exceed the minimum B-10 areal density.

In its supplemental letter dated October 15, 2015, the licensee submitted responses to NRC staff requests for information which included several regulatory commitments associated with its coupon surveillance program. In its letter dated February 8, 2016, the licensee superceded one of these regulatory commitments to clarify the test conditions. This clarification reads as follows:

3. Ensure that should any coupon be identified as failing the minimum certified B-10 areal density criterion based on coupon test results, EGC will perform in-situ testing of a statistically representative sample of BORAL panels to confirm that the minimum B-10 areal density (0.02 g/cm2) is met for all panels installed in the DNPS spent fuel pools; and,

Upon further review, the NRC staff determined the licensee's revised commitment could result in over-sampling of the SFP panels upon identification of a failed coupon. To allow the licensee

to perform an effective coupon surveillance program while ensuring the design objectives to meet (or exceed) the minimum areal density are being met, the following license conditions, taken from the October 15, 2015, commitments as supplemented, are hereby established;

1. Ensure that coupon measurements of B-10 areal density are performed by a qualified laboratory;

2. Ensure that the coupons are removed for evaluation every 10 years;

3. Ensure that should any coupon be identified as failing the minimum certified B-10 areal density criterion based on coupon test results, EGC will perform in-situ testing to confirm that the minimum B-10 areal density (0.02 g/cm2) is met for the BORAL panels installed in the DNPS spent fuel pools, and;

4. Submit a report to the NRC within 90 days following the completion of evaluations associated with Item 3 above. The report shall include a description of the testing results, the assessments performed, and the interim and long-term corrective actions for abnormal indications.

The NRC staff has concluded that adherence to these license conditions should ensure that loss of material and reduction in neutron-absorbing capacity is promptly detected. Additionally, the implementation of these license conditions provides reasonable assurance that the regulatory requirements of 10 CFR 50.36(c)(4) "Design features," the TS limit requiring the licensee to maintain a $k_{eff} \leq 0.95$, will be met.

3.1.7 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 reconstituted fuel. The licensee's submittal did not include a detailed explanation of how this condition might be modeled (including potential conservatisms/non-conservatisms, uncertainties, and biases). Therefore, the findings in this safety evaluation do not extend to any unbounded conditions associated with reconstituted fuel.

Because of the need to evaluate offsetting effects in the licensee's analysis and the lack of a defined methodology for addressing fuel reconstitution, this analysis constitutes a methodology for which any unbounded change 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.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration (*Federal Register Notice*; 80 FR 68573, dated November 5, 2015) and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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) 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.

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Date of Issuance: April 29, 2016

Upon implementation of amendments 249 and 242, Exelon Generation Company, LLC (EGC) shall have established and will maintain a coupon surveillance program to assess the degree of degradation, if any, of the neutron-absorbing material. In accordance with this program the licensee shall:

1. Ensure that coupon measurements of B-10 areal density are performed by a qualified laboratory;

2. Ensure that the coupons are removed for evaluation every 10 years;

3. Ensure that should any coupon be identified as failing the minimum certified B-10 areal density criterion based on coupon test results, EGC will perform in-situ testing to confirm that the minimum B-10 areal density (0.02 g/cm2) is met for the BORAL panels installed in the DNPS spent fuel pools, and;

4. Submit a report to the NRC within 90 days following the completion of evaluations associated with Item 3 above. The report shall include; a description of the testing results, the assessments performed, and the interim and long-term corrective actions for abnormal indications.

The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/
Eva A. Brown, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

Enclosures:
1. Amendment No. 249 to DPR-19
2. Amendment No. 242 to DPR-25
3. Safety Evaluation

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Letter to the Bryan C. Hanson from Eva Brown dated April 29, 2016

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS TO AMEND RENEWED FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25 TO SUPPORT USE OF A NEW NUCLEAR CRITICALITY SAFETY ANALYSIS METHODOLOGY (CAC NOS. MF5734 AND MF5735)

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