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

March 30, 2017

Mr. Thomas A. Vehec
Site Vice President
NextEra Energy
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT TO REVISE TECHNICAL SPECIFICATIONS FUEL STORAGE REQUIREMENTS (CAC NO. MF7486)

Dear Mr. Vehec:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 299 to Renewed Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). The amendment consists of changes to the technical specifications (TSs) in response to your application dated March 15, 2016, as supplemented by letters dated September 21, 2016, and December 27, 2016.

The amendment revises the DAEC TS 4.3.1, "Fuel Storage, Criticality," and TS 4.3.3, "Fuel Storage, Capacity," to ensure that spent fuel pool maintains compliance with NRC sub-criticality requirements for the storage racks manufactured by Programmed and Remote Systems Corporation. The amendment also adds a new requirement in TS 5.5, "Program and Manuals," for a spent fuel pool neutron absorber monitoring program.

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

Sincerely,

Mahesh L. Chawla, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures:
1. Amendment No. 299 to
   License No. DPR-49
2. Safety Evaluation

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

NEXTERA ENERGY DUANE ARNOLD, LLC

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 299
License No. DPR-49

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

   A. The application for amendment by NextEra Energy Duane Arnold, LLC dated March 15, 2016, as supplemented by letters dated September 21, 2016, and December 27, 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-49 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 299, are hereby incorporated in the license. NextEra Energy Duane Arnold, LLC, shall operate the facility in accordance with the Technical Specifications.

3. The licensee shall update DAEC UFSAR to reflect the requirements for using CASMO-4 as part of the implementation of this license amendment. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License No. DPR-49 and Technical Specifications

Date of Issuance: March 30, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 299

RENEWED FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following page of Renewed Facility Operating License DPR-49 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
3	3

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

REMOVE	INSERT
4.0.2	4.0.2
4.0.3	4.0.3
5.0-18b	5.0-18b

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of 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:

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

NextEra Energy Duane Arnold, LLC is authorized to operate the Duane Arnold Energy Center at steady state reactor core power levels not in excess of 1912 megawatts (thermal).

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 299, are hereby incorporated in the license. NextEra Energy Duane Arnold, LLC shall operate the facility in accordance with the Technical Specifications.

(a) For Surveillance Requirements (SRs) whose acceptance criteria are modified, either directly or indirectly, by the increase in authorized maximum power level in 2.C.(1) above, in accordance with Amendment No. 243 to Facility Operating License DPR-49, those SRs are not required to be performed until their next scheduled performance, which is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment No. 243.

(b) Deleted.

(3) <u>Fire Protection Program</u>

NextEra Energy Duane Arnold, LLC shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated August 5, 2011 (and supplements dated October 14, 2011, April 23, 2012, May 23, 2012, July 9, 2012, October 15, 2012, January 11, 2013, February 12, 2013, March 6, 2013, May 1, 2013, May 29, 2013, two supplements dated July 2, 2013, and supplements dated August 5, 2013 and August 28, 2013) and as approved in the safety evaluation report dated September 10, 2013. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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. Fuel assemblies having the following limits for maximum k-infinity in the normal reactor core configuration at cold conditions and maximum lattice-average U-235 enrichment weight percent:

	k_{∞}	wt %
i) 7x7 and 8x8 pin arrays (Legacy Fuel Assemblies only; Holtec and PaR racks)	≤ 1.29	≤ 4.6
ii) 10x10 pin arrays (Holtec and PaR racks)	≤ 1.29	≤ 4.95

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in 9.1 of the UFSAR; and

c. A nominal 6.060 inches for HOLTEC designed and 6.625 inches for PaR designed center to center distance between fuel assemblies placed in the storage racks.

d. The Boral neutron absorber shall have a ^{10}B areal density greater than or equal to 0.0162 grams $^{10}B/cm^2$ with an uncertainty of 0.0012 grams $^{10}B/cm^2$.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum k-infinity of 1.31 in the normal reactor core configuration at cold conditions;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.90$ if dry, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

d. A nominal 6.625 inch center to center distance between fuel assemblies placed in storage racks.

(continued)

4.0 DESIGN FEATURES (continued)

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 831 ft. – 2 3/4 in.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2563 fuel assemblies in a vertical orientation, including no more than 152 fuel assemblies stored in the cask pit in accordance with UFSAR Section 9.1.

The new fuel storage vault is equipped with racks for storage of up to 110 fuel assemblies in a vertical orientation.

5.5 Programs and Manuals

5.5.14 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.5.15 Spent Fuel Pool Neutron Absorber Monitoring Program

This program provides routine monitoring and actions to ensure that the condition of Boral in the spent fuel pool racks is appropriately monitored to ensure that the Boral neutron attenuation capability described in the criticality safety analysis of UFSAR Section 9.1 is maintained. The program shall include the following:

a. Neutron attenuation in situ testing for the PaR racks shall be performed at a frequency of not more than 10 years, or more frequently based on observed trends or calculated projections of Boral degradation. The acceptance criterion for minimum Boral areal density will be that value assumed in the criticality safety analysis.

b. Neutron attenuation testing of a representative Boral coupon for the Holtec racks shall be performed at a frequency of not more than 6 years, or more frequently based on observed trends or calculated projections of Boral degradation. The acceptance criterion for minimum Boral density will be that value assumed in the criticality safety analysis.

c. Description of appropriate corrective actions for discovery on nonconforming Boral.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 299 TO FACILITY OPERATING LICENSE NO. DPR-49

NEXTERA ENERGY DUANE ARNOLD, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By letter dated March 15, 2016 (Reference 3), as supplemented by letters dated September 21, 2016 (Reference 2), and December 27, 2016 (Reference 1), NextEra Energy Duane Arnold, LLC (the licensee) submitted a license amendment request (LAR) to modify the technical specifications (TSs) at the Duane Arnold Energy Center (DAEC). The LAR proposed to revise TS 4.3.1, "Fuel Storage, Criticality," TS 4.3.3, "Fuel Storage, Capacity," and add a new requirement to TS 5.5, "Programs and Manuals," for a spent fuel pool (SFP) neutron absorber monitoring program. These modifications would change the minimum Boron-10 (^{10}B) areal density (AD) for the neutron absorbing material (NAM) to 0.0162 grams $^{10}B/cm$ (centimeter)^2 with an uncertainty of 0.0012 grams $^{10}B/cm^2$, reduce the SFP storage capacity limits, and add the NAM monitoring program to TS 5.5. This would ensure that the DAEC SFP maintains compliance with the U.S. Nuclear Regulatory Commission (NRC or Commission) sub-criticality requirements for the storage racks manufactured by Programmed and Remote Systems Corporation (PaR).

This LAR, and the proposed TS changes, will satisfy a licensee commitment submitted to resolve a non-sufficient TS issue identified in a letter dated February 27, 2014 (Reference 5). In essence, the licensee performed in-situ testing of the ^{10}B AD for the Boral installed in the PaR storage racks as a commitment for license renewal. The testing did not support the minimum ^{10}B AD assumed in the nuclear criticality safety (NCS) analysis of record (AOR). The licensee believes that the results are due to a relatively large uncertainty associated with the in-situ testing system, rather than any degradation leading to loss of ^{10}B AD. Nevertheless, in order to ensure that the in-situ testing results would be able to directly support the NCS AOR, the licensee committed to submitting a LAR to update their TS to reflect a new NCS AOR that would use a lower minimum ^{10}B AD.

Enclosures 4 and 6 to Reference 3 present non-proprietary and proprietary versions of the NCS analysis for the DAEC spent fuel storage racks. In this safety evaluation (SE), the NCS analysis is identified as Reference 4 (non-proprietary version). The report describes the methodology and analytical models used in the NCS analysis to show that the spent fuel storage racks maximum k-effective (k_{eff}) will be no greater than 0.95 when flooded with unborated water. Appendix A of Reference 4 also includes the benchmarking evaluation performed for the MCNP6 code package 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.

The DAEC SFP contains 12 PaR rack modules and 9 Holtec rack modules of varying sizes. Both types of rack modules contain Boral neutron-absorbing material that is credited to meet NRC subcriticality requirements, although the manufacturing specifications are different. The Holtec rack modules continue to be bounded by a NCS AOR from a previous license amendment (Reference 21), so they did not need to be reanalyzed. Consequently, the licensee elected to update the NCS AOR for the PaR rack modules and only address the interface between the PaR rack modules and Holtec rack modules, rather than doing a full reanalysis of the entire SFP. Therefore, the Reference 3 analysis focuses on the PaR rack modules with a brief Section addressing the existing interface between rack modules from different manufacturers.

The supplemental letters dated September 21, 2016, and December 27, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 5, 2016 (81 FR 43665).

2.0 REGULATORY EVALUATION

2.1 Regulatory Requirements

Section 50.36(c)(4) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that the plant TSs include "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."

Section 50.36(c)(5) of 10 CFR requires that the plant TSs include "Administrative controls. Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in § 50.4."

The licensee is requesting changes to TS 4.3.1, 4.3.3, and adding a new TS 5.5.15. All TS changes were reviewed to ensure that they were sufficient to meet the aforementioned 10 CFR 50.36 requirements.

Section 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."

Section 50.68 of 10 CFR, "Criticality Accident Requirements," which states that if the licensee does not credit soluble boron in its criticality analysis, the k-effective (k_{eff}) of the SFP storage racks must not exceed 0.95 at a 95 percent probability, 95 percent confidence level. The k_{eff} is defined as the effective neutron multiplication factor.

Paragraph 50.68(b)(4) of 10 CFR requires, in part, "If no credit for soluble boron is taken, the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95-percent probability, 95-percent confidence level, if flooded with unborated water."

The DAEC SFP NCS analysis does not take credit for soluble boron for normal operating conditions, so the other 50.68(b)(4) requirements do not apply. This evaluation only applies to fuel stored in SFPs and is not used to ensure compliance with any requirements other than criticality requirements, so the remaining 10 CFR 50.68(b) requirements were not explicitly addressed as part of this criticality safety analysis.

Appendix A of 10 CFR Part 50, Criterion 62, requires, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

The design basis for the DAEC SFP was evaluated in the DAEC updated final safety analysis report (UFSAR) (Section 3.1) against the General Design Criteria (GDC) in 10 CFR Part 50, Appendix A, as amended on July 7, 1971. At that time, Criterion 62 already existed in its current form so it continues to be applicable to DAEC. Demonstrating compliance with 10 CFR 50.68 is sufficient to show that this GDC is satisfied.

GDC 61, "Fuel Storage and Handling and Radioactivity Control," which states that "These systems shall be designed with a capability to permit appropriate periodic inspection and testing of components important to safety."

GDC 62, "Prevention of Criticality in Fuel Storage and Handling," which states that "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

NUREG-0800, "Standard Review Plan [SRP]," Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling." This NUREG provides guidance regarding the specific acceptance criteria and review procedures to ensure that the proposed changes satisfy the requirements in 10 CFR 50.68 and GDC 62.

NUREG-0800, "Standard Review Plan," Section 9.1.2, "New and Spent Fuel Storage." This NUREG provides guidance regarding the specific acceptance criteria and review procedures to ensure that the proposed changes satisfy the requirements in 10 CFR 50.68.

NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 2. This NUREG provides guidance regarding appropriate NAM monitoring programs for the Boral as the NAM.

NRC Interim Staff Guidance (ISG) DSS-ISG-[Division of Safety System Interim Staff Guidance] 2010-01, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," Revision 0, dated October 2011 (Reference 9). The purpose of this ISG is to provide updated guidance to the NRC staff reviewer for SFP nuclear criticality analyses and operations. This ISG is intended to reiterate existing guidance (i.e., SRP, Section 9.1.1), clarify ambiguity in existing guidance and identify lessons learned based on past submittals. As noted in Section 3.0, "Technical Evaluation," of the licensee's LAR, the new SFP criticality safety analysis for the DAEC meets the technical guidance in DSS-ISG-2010-01.

2.2 Regulatory Guidance

The DAEC UFSAR contains an evaluation of the design basis against 10 CFR Part 50, Appendix A, "General Design Criteria," in Section 3.1 "Conformance to AEC [Atomic Energy Commission] General Design Criteria [GDC] for Nuclear Power Plants." The AEC GDC were effective starting May 21, 1971, and were amended on July 7, 1971. The DAEC UFSAR, Revision 23, dated May 22, 2015, describes how the DAEC complies with the GDC.

Specific review criteria are contained in SRP, Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3, and SRP, Section 9.1.2, "New and Spent Fuel Storage," Revision 4.

The NRC staff issued a memorandum dated August 19, 1998 (Reference 22), 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 was 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 was to state staff positions on a number of strategies used in SFP NCS analyses at that time.

The DSS-ISG-2010-01 (Reference 9), 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 following the guidance described in SRP, NUREG-0800, Sections 9.1.1 and 9.1.2.

2.3 Method of Review

This SE involves a review of the data provided by the licensee to demonstrate that if their TS requirements, as amended by their proposed changes are satisfied, then compliance with the relevant NRC requirements for SFP sub-criticality will be assured. The review was performed consistent with Sections 9.1.1 of NUREG-0800 (Reference 16), and existing guidance on SFP NCS analyses captured by the Kopp Memo (Reference 22), and DSS-ISG-2010-01 (Reference 9). While Section 9.1.2 of NUREG-0800 is applicable, it does not concern itself directly with criticality safety considerations in fuel storage; therefore, Section 9.1.1 contains the primary SRP guidance for reviewing the proposed changes in the LAR.

2.4 Compliance with the Technical Specifications

The licensee is proposing changes to TSs 4.3.1.1.a and 4.3.3, as well as the addition of TSs 4.3.1.1.d and 5.5.15.

The proposed change to TS 4.3.1.1.a will update the K-infinity (k_{inf}) limit for fuel stored in the PaR SFP racks to a lower value as supported by the NCS analyses submitted with this LAR. The intent of the NCS analyses is to demonstrate that regulatory compliance is assured as long as all fuel loaded in the DAEC PaR SFP racks is of one of the analyzed fuel assembly designs, and satisfies the TS k_{inf} limit. The value is being reduced in order to resolve a non-conservative k_{inf} limit resulting from a lower than expected minimum measured value for the ^{10}B AD in the PaR racks. Section 3 of this safety evaluation (SE) contains the technical evaluation of the NCS analyses supporting this change.

The proposed change to TS 4.3.1.1.d and the addition of TS 5.5.15 support a minimum ^{10}B AD requirement for the PaR racks, as assumed in the NCS analyses evaluated in Section 3 of this SE.

Section 3.3 of this SE contains the technical evaluation of the adequacy of the proposed licensee actions to ensure that this requirement is met.

The proposed change to TS 4.3.3 is a reduction in the number of fuel assemblies allowed for storage in the SFP. A previous approved license amendment allowed DAEC to replace the rack modules in their SFP, freeing up space for storage of more fuel assemblies. The permitted storage capability at that time was revised to reflect the total storage capability envisioned at the end of the rack module replacement. However, the licensee elected not to replace all of the rack modules and has no plans to do so in the future. Therefore, the storage capacity is being reduced to reflect the actual storage capacity at the site. This change does not require any further technical evaluation since the storage capacity corresponds to a SFP rack module configuration that is already part of the current DAEC licensing basis and will be bounded by this proposed LAR.

3.0 TECHNICAL EVALUATION

3.1 Background

The licensee utilizes two different designs of spent fuel storage racks in the DAEC SFP. The racks are designed and manufactured by PaR and Holtec. The PaR SFP racks were approved for use in the DAEC SFP by the NRC in license amendment No. 45 in 1978 (Reference 25), and the Holtec SFP racks were approved for use in the DAEC SFP in license amendment No. 195 in 1994 (Reference 23). Both SFP rack designs contain Boral as the NAM. The current SFP criticality analysis that supports the LAR takes partial credit for the neutron absorbing properties of Boral in both rack designs.

During the license renewal process for DAEC (formerly known as FPL Energy), the NRC staff evaluated the licensee's aging management program to monitor the reduction of neutron-absorbing capacity and loss of material due to general corrosion in the SFP. The NRC staff had questioned the licensee's evaluation of a potential reduction in neutron-absorbing capacity in the SFP and eventually the licensee made a commitment (Commitment No. 48) in Appendix A to Section 18.4 of its license renewal application. Through this commitment, the licensee implemented a program to monitor the Boral in the SFP and also to conduct in-situ neutron attenuation testing of the PaR SFP racks prior to the period of extended operation.

To satisfy its license renewal commitment to conduct in-situ neutron attenuation testing on the PaR SFP racks, the licensee performed Boron-10 Areal Density Gage for Evaluating Racks (BADGER) testing from June 10 through 19, 2013. When the licensee received the results of this campaign on September 11, 2013, the licensee concluded that the results did not support the assumed ^{10}B minimum AD in the SFP criticality analysis of record.

On November 11, 2013, the licensee submitted a licensee event report (Reference 6) regarding the test results from their September 2013 BADGER campaign. These test results showed that the ^{10}B AD assumed in the licensee's AOR may not have been supported. The test results showed 11 of the 19 tested panels having a ^{10}B AD lower than the AD assumed in the licensee's criticality AOR. Because the test results did not support the ^{10}B AD assumed in the AOR, the licensee was in noncompliance with DAEC TSs 4.3.1.1(i) and 4.3.1.1(iii). The licensee implemented administrative controls to provide more rigorous controls on the SFP racks affected by this condition (i.e., the PaR SFP racks). Corrective actions for the licensee included the submittal of the LAR dated March 15, 2016 (Reference 3), to revise the TSs to address this condition.

3.2 SFP NCS Analysis Method

There is no comprehensive, NRC-approved generic methodology for performing NCS analyses for fuel storage and handling. The methods used for the NCS analysis for fuel in the DAEC SFP are described in the licensee's analysis (Reference 4). The computer code benchmarking analyses supporting use of Monte Carlo N-Particle (MCNP) 6 for this application are described in Appendix A of the licensee's analysis. Additional information describing the methods used is provided in the request for additional information (RAI) responses in the letter date December 27, 2016 (Reference 1). Several SFP analysis deficiencies were identified during the review but will be discussed below as sufficient margin is built into the analysis methodology to offset the deficiencies for 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 fuel currently stored in the SFP, but the findings of this SE do not generically extend to fuel that is not bounded by this NCS analysis.

3.2.1 Computational Methods

For the criticality calculation, the licensee used the MCNP code, with continuous energy group cross-Section data based on the Evaluated Nuclear Data File, Version 7.1 (ENDF/B-VII.1), neutron cross-Section library. MCNP is a state-of-the-art Monte Carlo criticality code developed and maintained by Los Alamos National Laboratory for use in performing reactor physics and criticality safety analyses for nuclear facilities and transportation/storage packages. The licensee used the most recent code version, MCNP6. The code and its accompanying nuclear data sets have been extensively validated by Lawrence Livermore National Laboratory for various neutron transport calculations, including criticality calculations. Therefore, the NRC staff finds the underlying neutron transport methodology to be acceptable but the code needs to be validated for specific applications. The staff review of the licensee's validation of MCNP6 for its SFP NCS application is discussed in Section 3.2.2 of this SE. In order to verify that the modeling inputs were appropriate for this application, the staff requested a sample MCNP6 input deck associated with a typical criticality calculation performed for this analysis. The licensee provided the input deck for the base case via an electronic portal, and the staff reviewed the deck and determined that the input parameters were appropriate for their intended purpose.

For the depletion calculation to determine the spent fuel isotopic compositions, the licensee used the two-dimensional (2-D) CASMO-4 computer code with a 70-group cross-Section library mainly derived from the ENDF/B-IV neutron cross Section library. In some cases, the ENDF data has been supplemented by other data sources. CASMO-4 is not used as part of the DAEC licensing basis documented in the UFSAR, instead the licensing basis uses CASMO-3. However, CASMO-4 has been approved by the NRC for depletion analysis with a wide range of boiling-water reactor (BWR) and pressurized-water reactor fuel assembly designs. The proposed TS limit does not directly state that CASMO-4 is to be used to determine the standard cold core geometry (SCCG) k_{inf} limit. Instead, the proposed TS limit references Section 9.1 of the UFSAR, which should discuss SCCG k_{inf} limit determination using CASMO-4 upon approval and subsequent implementation of the subject license amendment. However, Section 7.2 of Reference 4 supports the use of CASMO-4. The DAEC UFSAR identifies Toshiba General Electric BWR Lattice Analysis as their licensing code, so the NRC staff asked how the DAEC licensing basis documents and quality assurance (QA) program would ensure that the correct comparison is performed for future fuel. The licensee stated in their letter dated December 27, 2016 (Reference 1), that CASMO-4 had also been added to their QA program as part of the process to use it to support the SFP NCS calculations. In addition, the licensee will update their UFSAR to reflect the requirement that CASMO-4 be used in this comparison. The change to the UFSAR will be implemented at the same time as this license amendment, which satisfies the intent of ensuring that the methodology used to qualify future fuel is captured in the plant licensing basis documents.

The above computer codes, and the nuclear data sets with them, have been used in many NCS analyses and are state-of-the-art computer codes.

Typical NCS analyses must include uncertainties to cover the lack of validation of spent fuel compositions, the lack of validation for k_{eff} calculations of burned fuel systems containing minor actinides and fission products, and the use of any other short lived, volatile, and gaseous isotopes that are being modeled in the analyses. These depletion code related uncertainties are discussed in detail in Section 3.5.1 of this SE, since they affect the characterization of the number densities for the spent fuel lattices used in the criticality evaluation.

For all MCNP6 calculations, the licensee used reasonable values for the following calculational parameters: number of histories per cycle, number of cycles skipped before averaging, and total number of cycles. More importantly, the licensee confirmed that all calculations converged using appropriate checks. The staff found that the parameters were reasonable based on its review of general guidance provided in the supporting documentation for the MCNP6 code system and engineering judgment.

In the SFP criticality analysis report (Reference 4), the licensee stated that for all MCNP6 calculations the initial source is placed in the highest reactive area of the model. The NRC staff asked for clarification on the nature of the initial source distribution. The licensee explained in the letter dated December 27, 2016 (Reference 1) that for most of their calculations, they used multiple point sources that were distributed at different axial elevations and radial locations in the fuel assembly. The staff review of the sample MCNP input deck indicated that the point sources were sufficiently distributed to provide reasonable assurance that all regions that could potentially drive the k_{inf} calculation would be seeded with adequate numbers of neutrons. In response to RAIs (Reference 1), for the calculations modeling the interface and accident conditions, the licensee stated that a single point source was used at the location of maximum reactivity. In order to demonstrate that this was indeed the correct location to use, the licensee repeated the calculations with multiple point sources. The results showed that the difference in initial source did not lead to a difference in the calculated k_{inf}.

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

3.2.2 Computer Code Validation

The purpose of the criticality code validation is to ensure that appropriate code bias and bias uncertainty are determined for use in the criticality calculation. The ISG DSS-ISG-2010-01 references NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology" (Reference 18).

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

> In general, the critical experiments selected for inclusion in the validation must be representative of the types of materials, conditions, and operating parameters found in the actual operations to be modeled using the calculational method. A

sufficient number of experiments with varying experimental parameters should be selected for inclusion in the validation to ensure as wide an area of applicability as feasible and statistically significant results.

The NRC staff used NUREG/CR-6698 as guidance for review of the code validation methodology presented in the application. The basic elements of validation are outlined in NUREG/CR-6698, 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.

In the letter dated March 15, 2016, Enclosure 4, Appendix A (Reference 4), the licensee presented the results of the validation of MCNP6, performed by comparing calculated k_{eff} values with several different sets of critical configurations. A total of 131 fresh fuel critical configurations were included. The licensee used the International Handbook of Evaluated Criticality Safety Benchmark Experiments (IHECSBE) (Reference 7), to select appropriate critical benchmarks. The IHECSBE was prepared by a working group comprised of experienced criticality safety personnel from the United States, the United Kingdom, Japan, the Russian Federation, France, Hungary, Republic of Korea, Slovenia, Serbia, Kazakhstan, Israel, Spain, Brazil, Czech Republic, Poland, India, Canada, China, Sweden and Argentina. The handbook contains criticality safety benchmark specifications that have been derived from experiments that were performed at various nuclear critical facilities around the world. The benchmark specifications are intended for use by criticality safety engineers to validate calculational techniques used to establish minimum subcritical margins for operations with fissile material. Therefore, the NRC staff considers it an appropriate source of information for the critical experiment models, because it represents a consensus reference within the international criticality safety community for code benchmarks.

In the SFP criticality analysis report (Reference 4), the licensee identified the most important applicable operating conditions for the validation (e.g., fuel assembly materials and geometry, enrichment of fissile isotope, types of neutron absorbers, moderators and reflectors, and physical configurations). In Reference 4, the licensee compared the spectral parameters (e.g., energy of average lethargy causing fission (EALF), fuel/moderator ratio via the fuel pin pitch) between the benchmarks and the DAEC SFP conditions to demonstrate that the selected benchmarks are applicable. The licensee selected critical configurations that were similar to the DAEC SFP configuration, such that the Area of Applicability would be adequate to validate the MCNP6 code for this application. The staff's review of the selected benchmarks shows that the area of applicability is reasonably covered for fresh fuel by the critical benchmarks. There are other benchmarks that could have been used, but the staff has determined that the selected set is sufficiently large and representative to use for this code validation application.

In Reference 4, the licensee determined that it would be appropriate to treat all experiments as a single set and found that the data was not normally distributed. As a result, a non-parametric analysis was performed to identify the code bias and uncertainty. In order to ensure that the approach used was conservative, the data was also analyzed using the statistical approach for normal data and the most limiting values were used for the bias and uncertainty. However, the SFP configuration being analyzed primarily consists of fuel assemblies surrounded by plates containing ^{10}B , which is a strong neutron absorber. Many of the critical configurations have either weak neutron absorbing materials or nothing but water between the fuel rod clusters. As a result, the critical configurations containing strong neutron absorber plates should be evaluated as a group to verify that any trends specific to that configuration do not result in a more limiting bias or uncertainty. The NRC staff examined the data from the critical configurations in benchmark Nos. 9, 13, 14, 42, 62, and 65 with ^{10}B containing plates and determined that analysis of this set, without the other critical configurations, may change some of the observed trends. In particular, a small bias was identified at the lower end of the EALF range. However, this bias was smaller than the limiting bias from the licensee's benchmark analysis. Therefore, the staff found that the trend corrected bias and uncertainty values calculated using the full set of critical experiments is conservative and acceptable.

The ISG document for SFP NCS analyses, DSS-ISG-2010-01 (Reference 9), states in item IV.4.a.i that the critical experiments previously evaluated by the NRC in NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," (Reference 14) should be considered as part of the validation. The use of the HTC experiments is important to cover the actinide distribution of burned fuel, as well as to evaluate the criticality implications of different storage arrangements for burned fuel. This is especially important because the licensee did not select any of the critical benchmarks from the IHECSBE that contain plutonium. In Reference 4, the licensee eliminated all cases utilizing conditions that were not considered to be relevant to the licensee's criticality analysis (soluble boron, reflectors, neutron-absorbing plates other than Boral), a total of 53 cases. The analysis determined that the bias was less than that for the fresh fuel critical benchmarks, but the uncertainty was larger. For conservatism, the limiting bias from the fresh fuel critical benchmarks was used with the limiting uncertainty from the HTC benchmarks.

One potential shortcoming in the licensee's validation was in the area of considering burned fuel compositions in the presence of boron, which is a key characteristic of the design basis configuration used in the licensee's NCS analyses. The only case evaluated by the licensee that contained plutonium together with a significant quantity of boron was the single Boral case from Phase 3 of the HTC experiments. The licensee did evaluate 20 cases from the HTC experiments that included plutonium in the presence of gadolinium, which is also a very strong thermal neutron absorber and would be expected to capture a similar effect on the neutron spectrum compared to boron. Furthermore, the criticality evaluation is performed at the burnup at which the reactivity reaches a peak, which is relatively low (12-18 Gigawatt-day per metric ton of uranium (GWd/MTU)). Therefore, the isotopic composition of the fuel would be closer to the fresh fuel critical benchmark set. In light of the discussion above, and the fact that the findings from the HTC experiments are generally similar to the findings from the fresh fuel critical benchmarks, the staff finds that no significant impact to the final code bias or uncertainty would be expected with the addition of more benchmark cases including both plutonium and boron.

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

3.3 SFP and Fuel Storage Racks

3.3.1 SFP NAM Monitoring Program Description

In its application dated March 15, 2016 (Reference 3), the licensee described the Boral Surveillance Program. The licensee maintains a SFP NAM surveillance program at DAEC in order to monitor the condition of the Boral material in both SFP rack designs (Holtec and PaR). The SFP NAM surveillance program provides for monitoring of Boral in the SFP racks so that the assumption made for ^{10}B minimum areal density in the SFP criticality AOR is supported and so that any degradation of the NAM is detected and addressed.

3.3.2 PaR SFP Rack Surveillance Program

The PaR SFP racks contain Boral as the NAM, which is comprised of B_4C clad in an aluminum matrix core clad with 1100 series aluminum. It is sealed between two concentric square aluminum tubes that hold the Boral plates in place. The Boral has a nominal AD of 0.0250 grams $^{10}B/cm^2$, and a minimum AD of 0.0232 grams $^{10}B/cm^2$. The Boral in these racks have a thickness of .080 inches. The licensee uses in-situ neutron attenuation testing to monitor potential degradation of the Boral material.

The licensee credits Boral as the NAM in the PaR SFP racks for its SFP criticality analysis. The licensee maintains a Boral surveillance program in order to monitor the condition of the Boral material. This is meant to ensure that potential degradation of the NAM does not impact the intended safety function of the NAM, and the minimum ^{10}B AD of the NAM as assumed in the SFP AOR. This surveillance program is covered under the licensee's QA program. The surveillance program uses in-situ (i.e., BADGER) neutron attenuation testing to determine if there is any degradation of the Boral material because the licensee does not have any surveillance coupons in the PaR SFP racks. The licensee performed the first in-situ test in 2013, and has proposed to continue the tests on a frequency that does not exceed 10 years. In the SFP criticality AOR, the licensee states that the minimum ^{10}B AD for the NAM is 0.0162 grams $^{10}B/cm$ [centimeter]$^2 \pm$ 0.0012 grams $^{10}B/cm^2$. In the RAI responses dated September 21, 2016 (Reference 2), the licensee described how the measurement uncertainty from the BADGER in-situ measurement is accounted for in the minimum ^{10}B AD value that was chosen.

The licensee compares the results of the BADGER testing against the AD criteria to determine acceptable performance of the PaR SFP rack Boral panels. No action is required if all measured panels have a Boral areal density greater than the acceptance criteria. If any of the panels are below the minimum Boral areal density (0.0150 grams $^{10}B/cm^2$), then corrective actions are required (e.g., enter a condition report, prompt operability determination, re-perform SFP criticality analysis, etc.). In addition, the proposed TS 5.5.15, "Spent Fuel Pool Neutron Absorber Monitoring Program," states that testing intervals may be impacted based on licensee observed trends or calculated projections of potential degradation. The TS also states that if nonconforming Boral is discovered, the licensee will take the appropriate corrective actions.

3.3.3 Holtec SFP Rack Surveillance Program

The Holtec SFP racks are constructed from American Society of Mechanical Engineers 240-type 304 stainless steel and plate stock, and SA564. The NAM used in these racks is also Boral. The Boral is placed in pockets between the cell and outer sheathing plate of the racks. The Boral in these racks has a thickness of .070 inches. The licensee currently has a coupon testing program to monitor the condition of the Boral in the Holtec SFP racks.

In this LAR, the licensee indicated the information on the Holtec SFP rack surveillance program can be found in the DAEC license renewal supplement dated October 23, 2009 (Reference 13). Therefore, the staff used this material in its review of the LAR.

In its October 23, 2009 (Reference 13), supplement to its license renewal application, the licensee stated that testing is performed on the Boral coupons via the licensee's Boral surveillance program. The licensee uses neutron attenuation measurements to determine the presence of ^{10}B, thickness measurements to determine if the panels are bulging or swelling, visual examinations to detect pitting, blistering or other forms of potential degradation, length and width measurements, and weight and specific gravity determinations. The licensee uses the data gathered from these tests to evaluate any potential trends. The licensee also stated that the acceptance criteria for the coupons are a "Decrease of no more than 5% of Boron-10 content as determined by neutron attenuation," and "No increase in thickness at any point greater than 10% of the initial thickness at that point." In the licensee's LAR dated March 15, 2016, the licensee proposed a TS change that would make the acceptance criteria to "be that value assumed in the criticality analysis."

These coupons are placed in spent fuel cells such that the coupons receive a higher radiation dose than Boral panels in the Holtec SFP racks typically receive. In addition, the licensee stated that the Boral used in the Holtec SFP racks is made from the same production run as the Boral used in the test coupons. The coupons are mounted to closely simulate the actual conditions of the Boral used in the racks. Proposed TS 5.5.15 would require that a representative coupon is removed and tested at a frequency not to exceed 6 years. If nonconforming Boral is discovered, the licensee will take the appropriate corrective actions.

3.3.4 Staff Evaluation of the NAM Surveillance Programs

The NRC staff has reviewed the LAR as well as material related to the Holtec SFP racks submitted by the licensee during license renewal. The license renewal material was reviewed because in its LAR submittal dated March 15, 2016 (Reference 3), the licensee stated that additional details for the SFP neutron absorber monitoring program could be found in the licensee's October 23, 2009 (Reference 13) renewal supplement.

As part of this LAR, the licensee proposed the addition of TS 5.5.15 "Spent Fuel Pool Neutron Absorber Monitoring Program," in order to add the Boral surveillance program to the DAEC TSs. The proposed TS is meant to ensure that the condition of the Boral in the SFP racks will continue to support the minimum ^{10}B AD assumption in the criticality analysis, and that the NAM continues to perform its intended safety function. TS 5.5.15 proposes that "The acceptance criterion for minimum Boral areal density will be that value assumed in the criticality safety analysis." In addition, proposed TS 4.3.1.1.d, states that "The Boral neutron absorber shall have a ^{10}B areal density greater than or equal to 0.0162 grams $^{10}B/cm^2$ with an uncertainty of 0.0012 grams $^{10}B/cm^2$."

The minimum ^{10}B AD (0.0150 grams $^{10}B/cm^2$) established in the licensee's AOR, has been set as the acceptance criteria for testing on both the Holtec and PaR SFP racks. In addition, the licensee's surveillance program includes acceptance criteria of a 5% decrease in the ^{10}B minimum areal density for the Holtec coupons. The 5 percent decrease is calculated by comparing the test result to the previous test result. The NRC staff found the acceptance criteria acceptable as they provide reasonable assurance that the NAM can perform its intended safety function as described in the DAEC SFP criticality AOR, and because the minimum ^{10}B AD chosen by the licensee accounts for BADGER measurement uncertainty and potential NAM degradation. Proposed TS 5.5.15 would require that the licensee will calculate projections of Boral degradation and create trends from neutron attenuation testing data, which the staff determined was acceptable. In addition, the licensee will implement corrective actions if the ^{10}B minimum AD is found to be below the acceptance criteria for either rack design.

The NRC staff has reviewed the licensee's SFP NAM surveillance program described in TS 5.5.15 and determined that it provides reasonable assurance that the licensee will be able to detect and monitor indications of loss of material (i.e., ^{10}B) and a reduction in neutron-absorbing capacity, and thereby ensure the NAM performs its intended safety function. This is accomplished, in part, through in-situ neutron attenuation testing for the PaR SFP racks and through neutron attenuation testing on coupons for the Holtec SFP racks. In addition, the coupons for the Holtec SFP racks will also allow the licensee to monitor certain parameters (coupon dimensions, weight, specific gravity, and neutron absorption capacity). The staff found that these are acceptable methods to detect a potential loss of ^{10}B. The staff has also determined that the proposed testing intervals (10 year frequency for in-situ testing of PaR racks and at a frequency not to exceed 6 years for the coupon testing of Holtec racks) are consistent with NRC guidance in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 2, December 2010 (Reference 11).

The NRC staff has determined that the proposed TS 5.5.15 includes the requisite elements (i.e., neutron attenuation testing, test intervals, acceptance criteria, monitoring and trending of results, and the requirement to take corrective actions for nonconforming Boral) for an effective surveillance program to ensure the ^{10}B minimum areal density assumption in the SFP criticality analysis AOR continues to be met, and to ensure that the NAM continues to perform its safety function.

On this basis, the NRC staff has determined that the proposed language for the NAM surveillance program in TS 5.5.15 is acceptable.

3.3.5 SFP Water Temperature

The LAR states that the design basis calculations were run using the standard temperature for MCNP6 and its libraries, 300 K. However, the water density was set to 1.0 g/cm^3, which is the maximum density for water at atmospheric conditions and corresponds to a temperature of 4 °C (277 K). The impact on the moderator cross-sections of a 23 degree difference in temperature is not significant, and the impact on the number densities is captured by a conservatively high water density. Follow-up calculations were performed to verify that higher SFP temperatures (up to the maximum) did not result in a higher k_{eff} value.

The effect of the SFP water temperature was treated in a bounding manner, so the NRC staff finds the licensee's approach to be acceptable.

3.3.6 SFP Storage Rack Models

DAEC has multiple PaR rack modules of different sizes in the SFP, with water gaps between adjacent racks, installed in approximately half of the SFP on the left side. The Holtec racks are stored in the right half of the SFP, so there is an interface gap between the PaR and Holtec racks in the middle of the SFP. The licensee is only addressing the PaR racks in this LAR, though the interface between the two types of SFP storage racks is addressed. The PaR racks consist of a checkerboard arrangement of 'formed' cells which consist of a nested arrangement of two square aluminum tubes with Boral panels sandwiched between them on all four sides. The square tubes are welded to each other at the corners, resulting in a square array of 'formed' and 'resultant' cells. One simplification in the NCS analysis model utilized by the licensee was that the corners of the SFP storage cells were effectively modeled as four square tubes separated from each other by the thickness of the Boral cavity. In reality, a portion of this region would consist of welded aluminum instead of water. As a result, the physical geometry of the SFP storage rack cells has slightly more aluminum, and correspondingly, slightly less moderator than the modeled geometry. The licensee indicates that the reactivity impact can be estimated by use of the rack cell wall thickness manufacturing tolerance. Calculations by the staff estimate the change in material resulting from the rack cell wall thickness tolerance to be approximately three times the volume of the estimated quantity missing from the storage cell corners. Even though the missing material at the storage cell corners represents a more significant local loss of material, the volume change due to an increase in the cell wall thickness to the upper manufacturing tolerance is much larger, such that the reactivity impact due to the rack cell wall thickness manufacturing tolerance, 0.0015 Δk, is acceptable as a conservative estimate.

In the base criticality analysis, the licensee performed a series of calculations in which the lattice of interest was assumed to extend for the full active length of the fuel assembly stored in the PaR rack storage cells. The boundary conditions were set such that the rack storage cell repeated infinitely in the radial direction, and 12 inches of water was modeled at the top and bottom of the assembly. For normal cell-centered storage conditions, this is a reasonable representation of fuel stored in storage cells on the interior of the rack module. This is conservative for the interface between peripheral fuel storage cells and the SFP wall due to the fact that an infinite radial array boundary condition acts as a perfect reflector while the SFP wall will allow some neutron leakage. The interface between rack modules and other situations where non-uniform local geometries need to be captured are explicitly analyzed by the licensee in separate calculations. The 12 inches of water at the axial ends is sufficient to maximize the reflection of neutrons back into the fuel assembly, because any neutrons that travel further than 12 inches will generally not return and the structural material that would normally absorb some neutrons is not modeled. With the exception of the boron-10 (^{10}B) AD in the Boral, all other storage rack materials were modeled using typical compositions and densities. This is acceptable because the nuclear reaction cross sections for the other materials in the racks (aluminum, carbon, other isotopes of boron, etc.) are not significant. All geometry parameters were modeled at their nominal value and the manufacturing tolerances were addressed as uncertainties, as discussed in the next section.

An important parameter for criticality in the racks is the ^{10}B AD of the Boral NAM. The licensee's NCS analysis used 0.015 g[grams]/cm^2 as the ^{10}B AD of the Boral plates. The minimum certified ^{10}B AD for the Boral material was 0.0232 g/cm^2, but the licensee wanted to ensure that the B-10 areal density assumed in the NCS analysis was sufficiently low to accommodate measurement uncertainties in their in-situ testing to verify that the Boral continues to be capable of performing its safety function. This represents a bounding approach to treat the ^{10}B AD, which is acceptable as long as the licensee can continue to demonstrate that the Boral installed in their SFP racks has a minimum ^{10}B AD above the value assumed in the NCS analysis.

Boral consists of discrete boron carbide particles suspended in an aluminum matrix, so the potential exists for reduced neutron-absorbing efficiency due to self-shielding and neutron streaming effects. In the criticality safety analysis report (Reference 4), the licensee discusses this effect and states that the isotropic neutron flux in the SFP environment minimizes the reactivity impact of these effects, therefore, the effect was not explicitly evaluated. Recent studies found in the engineering literature show that there is some reduction in the neutron absorption effectiveness of this type of material compared to a homogeneous mixture with the same ^{10}B AD for SFP criticality calculations. This effect is generally on the order of 0.002-0.003 Δk for ^{10}B ADs near 0.02 g/cm^2 and boron carbide particle sizes comparable to that found in typical Boral (~100 μm). These parameters are close enough to the Boral material utilized at DAEC to use 0.003 Δk as a reasonable estimate of the potential non-conservatism for not explicitly treating the heterogeneity in the Boral core.

Boral has a known history of bulging and blistering due to exposure to the SFP environment, as discussed in Information Notices (INs) 1983-29 and 2009-26 (References 24 and 12). In order to account for the reactivity impact due to the displacement of moderator between SFP rack cells and the resulting reduction in the neutron absorption efficiency of the Boral plates, the licensee performed a calculation that assumed that all of the water between the SFP cell walls and the Boral panels was removed by blistering. The calculated reactivity increase was applied as a bias in the final k_{eff} calculation. However, NRC staff review of Section 9.1.2.2.2 of the DAEC UFSAR indicated that the PaR racks were manufactured in such a way that "[t]he outer can is formed into the inner can at the ends and totally seal welded to isolate the Boral from the pool water." This suggests that: (1) a void inside the poison cans is part of the normal configuration, and (2) the potential exists for bulging of the poison cans as discussed in prior operating experience associated with Boral installed in sealed, non-vented rack configurations. The staff requested verification of the installation method for the Boral described in the UFSAR, and the licensee confirmed that the UFSAR description was correct. In the letter dated December 27, 2016 (Reference 1), the licensee provided additional results of the calculations to address the reactivity impact of revising the NCS model to correctly model the PaR rack configuration and assess the potential reactivity impact due to bulging of the rack walls. The resulting change to the margin to the regulatory limit is included in the final margin rack-up analysis discussed in Section 3.7 of this SE.

Based on the supplemental information provided in response to RAIs (Reference 1), the NRC staff determined that the impact of correctly modeling the PaR rack configuration was straightforward, since it only required re-running of the base NCS calculation with no water modeled in the cavity between the Boral panels and the SFP rack walls. The potential reactivity impact due to bulging was addressed through analysis of a 20 x 20 rack model. The configurations analyzed included:

1. A base configuration with nominal dimensions and no water in the SFP rack wall cavities,
2. One cell with all walls fully bulged (i.e., moved so that the cell wall impinges upon the fuel),
3. 5 cells with all walls fully bulged (in the shape of a cross),
4. 9 cells with all walls fully bulged (in a 3 x 3 square array),
5. 1 cell out of every 3 x 3 square array of cells with all walls fully bulged, and
6. All cells with 10 percent bulging inward (~16 percent increase in void).

The licensee determined the resulting reactivity differences, then added the result of modeling the SFP rack wall cavities with no water and the most limiting result from the bulging calculations as a bias to the final k_{eff} rack-up, while also removing the bias for Boral blisters.

One question regarding the licensee's calculations is whether the bulging studies would be expected to bound the physical configuration of the licensee's SFP racks. Previous operating experience related to significant bulging in SFP racks have been identified based on difficulty in fuel assembly movement due to lack of clearance between the fuel assembly and the SFP rack walls. As described in IN 1983-29 (Reference 24), the bulging was caused by hydrogen gas produced when water entered the SFP rack wall cavity and reacted with the aluminum, most likely as a result of isolated cases of weld failure. In the previous cases, the severely bulged rack walls were identified in a relatively small number of cells. DAEC has seen no indications of bulging in their PaR racks to date, and their rack configuration encloses the Boral in relatively thick rack wall material (as opposed to the thin gage "wrapper" material used to enclose the Boral in other rack designs that have had severe bulging issues). In addition, any major bulging would be identified by the licensee as a result of challenges with fuel movements or BADGER probe movement as part of their periodic Boral monitoring program. As a result of the above discussion, the NRC staff concludes that the various bulging scenarios analyzed by the licensee, along with the available margin to the regulatory limit, provides reasonable assurance that any bulging that occurs without detection by the licensee would not result in a violation of the regulatory limit. If the licensee should identify severe bulging, they would be expected to enter it into their corrective action program and take appropriate action to evaluate the potential for bulging in other cells.

The licensee's treatment of the ^{10}B neutron-absorbing material in the Boral panels was found to be generally acceptable, but the NRC identified potential non-conservatisms that can be offset by available margin to the regulatory limit. NRC staff evaluated other relevant aspects of the storage rack modeling and found them to be modeled conservatively or using appropriate parameters with any uncertainties addressed, as discussed in the next section. As a result, the staff concluded that the storage rack modeling is acceptable.

3.3.7 SFP Storage Rack Models Manufacturing Tolerances, Uncertainties, and Deviations

The manufacturing tolerances of the storage racks 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 (referenced in DSS-ISG-2010-01) 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 racks. The licensee chose to utilize the former approach with the minimum ^{10}B AD. The remaining manufacturing tolerances were addressed using the latter approach.

The licensee's evaluation (Reference 4) of the tolerance variations included the following components: SFP cell pitch, SFP cell wall thickness, Boral thickness, and Boral panel widths. Calculations were performed using the same model as the one used for evaluating the fuel manufacturing tolerances, discussed in Section 3.4.4 of this SE. 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.

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 base calculations were performed assuming a cell-centered loading with all fuel assemblies oriented similarly. Further calculations were performed using a 30x30 array of cells to investigate the reactivity effect due to fuel assembly lean and/or rotation. Each calculation rotated all fuel assemblies such that the same corner for each fuel assembly pointed towards the center of the 30x30 array of cells, and one calculation was performed for each possible fuel assembly rotation. Once the most reactive corner was identified, additional calculations were run to determine the reactivity change due to movement of the most reactive corners of fuel assemblies towards the center of the 30x30 array (eccentric positioning), with and without fuel channels. This series of calculations was performed for two different gadolinium loading patterns that maximized the gadolinium in the corner nearest and furthest from the controlled corner, respectively. The maximum reactivity change from all calculations was identified and applied as a bias in the final estimation of k_{eff}. This approach is a reasonable way to bound all possible reactivity variations arising from: (1) radial variation in reactivity due to differences in gadolinium depletion at opposite corners, (2) fuel assembly rotation, and (3) possible proximity of fuel assemblies to each other, as constrained by the SFP rack walls.

Since the presence of a control blade during depletion can have a significant impact on the radial variation in reactivity for a fuel lattice, the NRC staff asked the licensee to explain what depletion conditions were used for these calculations. The licensee replied in letter dated December 27, 2016 (Reference 1), that they performed these calculations using both the design basis lattice and lattice design 20, and provided a radial pin exposure map corresponding to lattice design 20. The NRC staff confirmed that the radial pin exposure map exhibits a very high degree of radial heterogeneity corresponding to a minor change in reactivity for the analyzed configurations. Therefore, the licensee has captured the potential impact of very high heterogeneity in the radial exposure distribution on the rotation and eccentric positioning analyses.

The licensee discussed various configurations (Reference 4), that may arise as a result of normal SFP operations which includes fuel movement and storage of fuel in the fuel handling/ inspection/reconstitution platform. Fuel movement is bounded by the base calculation because the same lattice is assumed to exist along the entire length of the active fuel assembly, and only one fuel assembly can be moved at a time. While outside of the SFP rack cells, a single fuel assembly will be neutronically decoupled from the fuel stored in the racks. A similar reasoning process applies to the fuel handling/inspection/reconstitution platform, which is far enough from the SFP racks to be considered to be neutronically decoupled. However, fuel rods may be removed as part of reconstitution operations, so the licensee explicitly analyzed a model to determine the maximum reactivity increase due to removal of fuel rods. The maximum reactivity was more than 0.1 Δk less than the limiting k_{inf} from the base calculation, so the NRC staff does not believe that it was necessary to run additional sensitivities. In Reference 4, the licensee addressed the movement of reconstituted fuel back into the SFP cells by stating that any removed fuel rods must be replaced with stainless steel rods or natural uranium rods, which preserves the amount of moderator in the lattice while reducing the fissile material relative to the original lattice. Therefore, each configuration was found to be bounded by the infinite array used in the design basis calculations or explicitly analyzed to confirm that they did not result in a more limiting k_{eff} value.

As a result of evaluating the treatment of manufacturing tolerances, uncertainties, and other potential differences between the idealized storage rack model used in the NCS analyses and real-world storage racks, the 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 or bias (as appropriate).

3.3.8 SFP Storage Rack Interfaces

The DAEC SFP contains storage rack modules from two different manufacturers with different specifications (Reference 4). Consequently, when considering the PaR storage rack modules, two different interfaces between rack modules must be considered (i.e., the interface between PaR racks, and the interface between PaR racks and Holtec racks).

On the periphery of the PaR rack modules, the resultant cells are surrounded by formed cells on only three sides, so the fourth side is closed by aluminum with no Boral panel installed. As a result, two adjacent PaR rack modules may be aligned in such a way that two face adjacent peripheral cells across the interface do not have any Boral between the fuel assemblies. The licensee evaluated this condition by running two calculations: (1) a base calculation using a 20x20 storage array, and (2) an interface calculation that split the 20x20 storage array into two 20x10 storage arrays with the aluminum and moderator of the interface modeled under the assumption that the two adjacent PaR rack modules are touching. The fuel lattice used in all cells is the same limiting lattice as discussed in Section 3.4.3 of this SE. The interface calculation resulted in a significantly lower reactivity than the base calculation, therefore, the base infinite-array configuration is expected to bound the interface between adjacent PaR rack modules.

The interface between a PaR rack module and a Holtec rack module was analyzed by use of a 40x20 PaR rack model positioned next to a 40x20 Holtec rack model. Unlike the PaR-PaR rack module interface discussed in the previous paragraph, the casting on the outside of the modules was neglected, which is conservative due to the reduced spatial separation between fuel stored in the respective rack modules. However, the licensee indicates (Reference 4) that the density of the fuel pellet in the Holtec rack was reduced to obtain the same k_{inf} in both rack modules. Since the maximum SCCG k_{inf} for fuel stored in both types of storage racks is the same, this interface analysis needs to take into account the actual reactivity of the Holtec racks. The NRC staff asked the licensee to provide a technical justification for this reduction in the reactivity for the Holtec rack via manipulating the fuel density. In Reference 4, the licensee explained that simply modeling a full scale Holtec rack next to a PaR rack would result in the calculated k_{eff} being driven by the Holtec rack rather than the PaR rack. As a result, the calculated result would be more accurately representative of the interior of the Holtec racks rather than anything associated with the PaR racks. The NRC staff agrees with that reasoning, though the licensee's analysis report did not provide sufficient analysis of the interface to conservatively capture the full impact of the interface.

In order to provide further information to demonstrate that there was no reactivity increase due to the interface between the PaR racks and Holtec racks, the licensee provided two additional analyses in Reference 4. The first analysis reduced the ^{10}B AD for the Boral modeled in the PaR racks to increase the reactivity of the PaR rack model to be equal to the Holtec racks. A NCS calculation was then performed with the reduced ^{10}B PaR rack next to a Holtec rack. No reactivity increase was observed, which demonstrates that the difference in geometry for the two racks does not affect reactivity. In order to evaluate the effect of having the higher-reactivity Holtec rack cells adjacent to the peripheral cells of a PaR rack, the licensee ran a calculation with a large PaR rack model using reflective boundary conditions, with one row of cells replaced by Holtec cells. This simulates the "reflector" effect of having a higher reactivity fuel loading configuration adjacent to the periphery of a PaR rack. The licensee compared the resulting calculated k_{eff} to that calculated for an infinite array of PaR cells, which is equivalent to a large PaR rack model with reflective boundary conditions. Again, there was no reactivity increase, therefore, the staff is satisfied that the licensee has adequately demonstrated that the interface between the PaR and Holtec racks will not result in a significant reactivity increase compared to the base calculation.

The interface calculations were all performed assuming normal conditions for the rack modules. However, they did not consider accident conditions which may result in a significant increase in reactivity. Accident conditions such as a dropped fuel assembly or increase in SFP temperature do not change the SFP rack module configuration. Therefore, if the reactivity for the interface can be shown to be bounded by the base model, these accident conditions can be adequately analyzed using the base model. The same logic does not apply to accident conditions such as a missing Boral panel, because the possible locations for a missing Boral panel include configurations that are not captured by the base model. The NRC asked the licensee to describe why these accident conditions are not expected to become limiting. In Reference 4, the licensee did provide some analyses that demonstrate a small increase at the interface due to a missing PaR panel, however, the reactivity change was less than calculated for the missing Boral panel accident scenario. Furthermore, as discussed in Section 3.6 of this SE, the licensee does not have any reason to suspect that a missing Boral panel would be a potential scenario, and the NRC staff agrees. As a result, the staff does not find it necessary to consider the potential for a missing Boral panel on the periphery.

The possible reactivity impacts of the SFP storage rack interfaces was evaluated by the staff through more detailed modeling of the relevant geometries, and in all cases, the calculated k_{eff} was found to be less than the k_{inf} for the base infinite array used in development of the limiting rack k_{inf} value used in the final k_{eff} rack-up. Therefore, the staff concluded that any potential interface effects are bounded by the inherent conservatism of the base infinite array model.

3.4 Fuel Assembly Evaluations

In Reference 4, the licensee provided the SCCG k_{inf} value using CASMO-4 and the rack k_{inf} value using MCNP6 for all 10x10 lattices currently at DAEC. This includes lattices from GE (General Electric)12, GE14, and GNF (Global Nuclear Fuel)2 fuel assembly designs. Additional lattices were generated to produce more limiting data and perform sensitivity studies. The end result was a conservative engineering approximation of the limiting rack k_{inf} value corresponding with a given SCCG k_{inf} value. Based on this approximation, the licensee selected a SCCG k_{inf} limit to use in the TS and used the corresponding limiting rack k_{inf} value as the base rack k_{inf} value used in the final k_{eff} rack-up in the NCS analysis. A fuel lattice was generated that matched the SCCG k_{inf} limit for the purpose of determining the reactivity biases and uncertainties, as discussed in Section 3.4.3 of this SE. The following sections discusses the licensee's treatment of the fuel lattices used in the NCS analysis in further detail.

3.4.1 As-Manufactured Fuel Assemblies

The older 7x7 and 8x8 fuel assemblies stored in the DAEC SFP are dispositioned based on the fact that the maximum SCCG k_{inf} for these fuel lattices are significantly lower than the selected limit. According to the licensee as stated in Reference 4, these values are 1.190 and 1.246 for the 7x7 and 8x8 fuel, respectively. The NRC staff estimated corresponding limiting rack k_{inf} values of 0.8211 and 0.8597 by using the licensee's correlation discussed in the next section. This correlation was based solely on data from 10x10 fuel lattices, however, the values imply that the limiting rack k_{inf} corresponding to the maximum SCCG k_{inf} for the older fuel would need to be higher than expected relative to the 10x10 fuel by more than 0.03 Δk. This significant margin, in addition to the fact that modern fuel assembly designs have consistently been shown to be more limiting in SFP NCS analyses due to a combination of fuel assembly design details and legacy operating conditions, provides reasonable assurance that the older 7x7 and 8x8 fuel in the DAEC SFP is bounded by the 10x10 fuel. The licensee states that this disposition will only address the 7x7 and 8x8 fuel currently stored in the DAEC SFP, not future fuel. Based on its review of the licensee's submittal, the NRC staff finds this to be acceptable.

All 10x10 fuel lattices currently contained within fuel assemblies at DAEC were explicitly analyzed using one set of core operating conditions as evaluated in Section 3.5.3. The resulting SCCG k_{inf} values from CASMO-4 and corresponding rack k_{inf} values from MCNP6 are plotted in Figure 6.3 of Reference 4. The next Section will discuss the development of a limiting correlation between SCCG k_{inf} and rack k_{inf}, which the licensee uses to ensure that the SCCG k_{inf} limit is sufficient to bound future fuel assemblies. Figure 6.5 of Reference 4 shows this limiting correlation, and it clearly bounds all calculated values for the 10x10 fuel currently at DAEC. Based on its review of the licensee's analysis, the NRC staff agrees that the results of the NCS analysis will bound all 10x10 fuel currently at DAEC.

Some of the fuel in the DAEC SFP may have been modified from its original as-manufactured condition through reconstitution operations. The licensee was not clear in Reference 3 as to whether any reconstituted fuel was currently stored in the DAEC SFP, however, a generic justification was provided to address reconstituted fuel. The second to last paragraph in Section 3.4.7 of this SE documents the staff's considerations and acceptance of this justification.

Based on its review of the licensee's analysis and dispositioning of the as-manufactured fuel currently located at DAEC, the NRC staff finds that the correlation discussed in the next Section bounds all fuel currently stored in the DAEC SFP.

3.4.2 Future Fuel Assemblies

The licensee is proposing a SCCG k_{inf} limit to be incorporated into the DAEC TS to ensure that future fuel assemblies are bounded by the NCS analysis in Reference 3. In order to demonstrate that this limit will achieve its intended goal, the licensee must show that reasonable assurance exists that the MCNP6 rack k_{inf} for any fuel lattice meeting the SCCG k_{inf} limit will be bounded by the base rack k_{inf} value used in the Reference 3 NCS analysis. As shown in Figure 6.3 of Reference 4, the correlation between the SCCG k_{inf} and MCNP6 rack k_{inf} can vary significantly based on the lattice geometry (location of water holes and vanished fuel rods). In fact, the spread in MCNP6 rack k_{inf} calculated at the same SCCG k_{inf} can be as large as 0.03 Δk. Some additional variation, albeit less pronounced, comes from changes to the depletion conditions, most notably the void fraction used during depletion of the lattice.

Reference 4 shows that in order to determine a bounding base rack k_{inf} value to use, the licensee performed calculations for 21 additional GNF2 lattices that utilized various U-235 enrichments, gadolinium enrichments, and typical gadolinium loading patterns, in addition to all 10x10 lattices currently on site at DAEC. The additional lattices were designed to add data for the SCCG k_{inf} above the highest value for fuel currently stored at DAEC (1.267) so the evaluation would be valid up to and including the proposed SCCG k_{inf} limit of 1.29. The parameters associated with the additional lattices cover a broad range of possible parameters, from 4.4 to 5.0 weight percent U-235 enrichments, 2.0 to 10.0 weight percent gadolinium concentrations, and 5- to 20-gadolinium rod loading patterns. All currently planned future fuel assemblies are to be of the GNF2 design, and these ranges bound the expected nuclear design parameters for fuel expected to be near or over the SCCG k_{inf} limit of 1.29. As a result of its review of the information and discussion above, the staff concluded that the data for all current lattices plus the 21 additional GNF2 lattices is a reasonable sample size to determine a correlation between the SCCG k_{inf} and MCNP6 rack k_{inf} that would reasonably be expected to bound all current and future lattices.

The licensee did not analyze lattices associated with the regions in fuel assemblies corresponding to the upper plenum of the part length fuel rods. In a typical BWR fuel assembly, there are part length fuel rods that start at the bottom of the fuel assembly but end below the top of the active fuel length for the rest of the fuel rods. The part length fuel rods, like all fuel rods, is capped by a plenum region consisting of a mixture of gas, fuel rod cladding, and fuel rod structural material. This area can be represented by a lattice in which the relevant part length fuel rod locations are replaced by a mixture of void and cladding/structural material. Internally, the GNF2 fuel assembly is undermoderated in the SFP environment, as can be seen by the fact that the fuel manufacturing tolerance calculations show reactivity increases in the direction of a lower fuel to moderator ratio (e.g., greater fuel rod pitch or reduced cladding thickness). Therefore, the plenum region would be expected to be bounded by either the lattice above (when the fuel-to-moderator ratio is the dominant effect that may lead to a potential reactivity increase) or the lattice below (when the amount of fissile material is the dominant effect that may lead to a potential reactivity increase). This is consistent with what the NRC staff has observed in past calculations where the plenum lattices were explicitly analyzed.

The licensee chose to manually develop a correlation based on a ratio of the rack k_{inf} to SCCG k_{inf} of 0.69. This correlation was clearly limiting for all as-manufactured lattices, but to address future fuel lattices, the licensee needed to demonstrate that this correlation would bound any correlations based on the 95/95 threshold for the GNF2 data. One limitation of performing a 95/95 analysis is that the lattice geometry, depletion conditions, and gadolinium are known to have a potential systematic effect on the correlation between SCCG k_{inf} and MCNP6 rack k_{inf}. Therefore, the data cannot all be treated as a single group of independent data points. In Reference 4, the licensee addressed this by performing a 95/95 analysis on: (1) GNF2 data for lattices with no vanished fuel rods and limiting depletion conditions, and (2) GNF2 data for lattices with six vanished fuel rods and limiting depletion conditions, in addition to the GNF2 data set as a whole. The 95/95 limiting line from all three analyses was used to determine a limiting rack k_{inf} value based on the proposed SCCG k_{inf} limit of 1.29. The results showed that the rack k_{inf} value based on the manually developed correlation bounded all the other possible candidates. That value, 0.8901, was used as the base rack k_{inf} in the k_{eff} rack-up for comparison to the regulatory limit. Further studies were not performed to consider the possible impact of variations in gadolinium, but the data set contains a sufficient number of different gadolinium concentrations and loading patterns to be representative of typical variation for fuel that would be near the proposed SCCG k_{inf} limit.

An additional consideration in the qualification of future fuel lattices based on k_{inf} calculated by CASMO-4 is any bias or uncertainty in the CASMO-4 calculated k_{inf} values. The approach that the licensee used was to establish a bounding correlation between CASMO-4 calculated SCCG k_{inf} values and the MCNP6 calculated rack k_{inf} values that is valid at a better than 95/95 tolerance interval. As such, the code uncertainties are implicitly included in the correlation to ensure that for a given fuel lattice with a CASMO-4 SCCG k_{inf} of no more than 1.29, the MCNP6 rack k_{inf} will be no more than 0.8901. Otherwise, the CASMO-4 SCCG k_{inf} values are not used in the final NCS analysis, so no uncertainty is necessary for CASMO-4, though there is still a depletion uncertainty applied to account for uncertainty in the number densities transferred to MCNP6 from the CASMO-4 calculation (discussed in Section 3.5.1 of this SE).

Based on the results of the criticality analysis and as a result of the above discussion on how the licensee determined a rack k_{inf} value that can reasonably be expected to bound the rack k_{inf} for any GNF2 fuel lattice with a SCCG k_{inf} equal to or less than the proposed TS limit, the NRC staff finds the proposed SCCG k_{inf} limit to be acceptable for qualifying future GNF2 fuel assemblies for storage in the DAEC SFP.

3.4.3 'Limiting' Fuel Assembly Model

In order to evaluate the biases and uncertainties for the final k_{eff} rack-up, a lattice was needed that would be a reasonable representation of fuel near the TS limit. The licensee developed a GNF2 fuel lattice that had a CASMO-4 calculated SCCG k_{inf} of 1.29 (Reference 4), then imported the isotopics into a MCNP6 rack model to use as a basis for determining the biases and uncertainties. This lattice model was characterized as a "limiting lattice." Based on the results from the unperturbed case, the MCNP6 rack k_{inf} for this lattice appears to be on the low end of the range for GNF2 fuel at a SCCG k_{inf} of 1.29 from Figure 6.6 of Reference 4. However, the relatively small difference in this MCNP6 rack k_{inf} will not have a significant impact on the calculated reactivity increments due to perturbations. The MCNP6 rack k_{inf} for this GNF2 fuel is not directly used in the NCS analysis, rather, it is only used to compute the reactivity increments associated with the biases and uncertainties for the NCS analysis. Once the reactivity increments are determined, then they are applied as adders to the base rack k_{inf} as determined using the limiting correlation discussed in the previous section.

Based on the information presented in Reference 4, the NRC staff understands that it would be very difficult, if not impossible, to develop a fuel lattice that would be near the SCCG k_{inf} limit and the maximum rack k_{inf} used in the final k_{eff} rack-up due to the conservative nature of the correlation used to determine the latter value. However, for the purpose of determining the biases and uncertainties in the NCS analyses, the staff found the selected fuel lattice to be sufficiently representative of a fuel lattice corresponding to the maximum rack k_{inf}. Since this model is only used to determine reactivity increments, and its MCNP6 rack k_{inf} will not otherwise be used in the final k_{eff} rack-up, evaluation of the uncertainties and biases using this model is acceptable.

3.4.4 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 (Reference 22) 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 utilize the latter approach for the fuel assembly manufacturing tolerances.

Reference 4 shows that the licensee's evaluation of the tolerance variations included the following components: fuel pellet stack density, fuel pellet outer diameter, fuel cladding thickness, fuel pin pitch, water rod thickness, and channel thickness. The manufacturing tolerance uncertainty calculations were performed using the infinite array model in MCNP6, which combined the representative fuel lattice discussed in Section 3.4.3 with the limiting SFP water temperature and minimum ^{10}B AD. For each set of calculations associated with varying a specific parameter, the maximum reactivity increase was identified. The direction of variation in each parameter was verified to be consistent with general experience from similar SFP configurations: an increase in fuel material, a decrease in neutron absorber material, an increase in moderator within the fuel assembly, and a decrease in moderator between the fuel assembly and neutron absorber panels all typically result in a reactivity increase. These uncertainties were statistically combined with the other uncertainties, rounded up (which is conservative), and included in the final estimation of k_{eff}. This is consistent with past precedent for criticality analyses and consistent with guidance provided in the Kopp Memo, and thus is acceptable. However, the licensee neglected to evaluate the manufacturing tolerance for U-235 enrichment and gadolinium enrichment. The NRC staff reviewed NCS analyses submitted by 10 BWR licensees in the past decade that all utilized 10x10 BWR fuel. Some of the analyzed configurations are very similar to the analyzed DAEC SFP configuration, while others have no neutron absorber panels installed in the SFP walls. The staff conservatively determined that the highest identified values, 0.0045 Δk and 0.006 Δk, would be expected to bound the U-235 enrichment and gadolinium enrichment uncertainties, respectively, for the DAEC SFP configuration. When statistically combined with the total uncertainty calculated in the licensee's NCS analysis report, this results in an increase in reactivity of 0.00178 Δk, which can be accommodated by the available margin to the regulatory limit.

Irradiation-induced fuel assembly geometry changes were not considered in this analysis. The licensee states in Reference 4 that there should not be much fuel growth and creep at the relatively low burnups of peak reactivity. Typical fuel geometry changes that may impact reactivity include: (1) spacer growth, (2) fuel pellet growth, and (3) cladding thinning due to creep. The manufacturing tolerance calculation shows that a decrease in the internal fuel-to-moderator ratio for the GNF2 lattice would be expected to increase reactivity, so (2) would not have a positive reactivity impact due to the conservation of fissile material quantity while reducing moderator. Cladding thinning due to creep and fuel rod pitch increases due to spacer growth would have a reactivity impact. In Reference 4, the licensee provided an evaluation of an accident scenario related to a dropped fuel assembly where the fuel pins are modeled at the maximum possible pitch. The reactivity increase associated with that evaluation, 0.0017 Δk, may be considered as a very conservative assessment of the impact of spacer growth due to irradiation. No information was available to the NRC staff to quantify the impact of cladding thinning due to creep. However, the NRC staff would expect the maximum impact to be on the order of the cladding thickness manufacturing tolerance; as a conservative estimate, the staff will apply the reactivity impact due to a decrease in clad thickness associated with the manufacturing tolerance, 0.0039 Δk, as a bias. In reality, the average change in clad thickness is expected to be smaller and partially offset by the development of an oxide layer on the cladding surface (crud) which will also displace moderator.

The NRC staff evaluated the licensee's treatment of various fuel manufacturing tolerances and uncertainties, as well as geometry changes due to irradiation. The licensee's application of the reactivity impact from geometry changes due to irradiation was found to be incomplete, but the NRC staff was able to use available information to develop bounding values for the reactivity impacts that was covered by existing margin to the regulatory limit. The staff found that all other licensee evaluations were acceptable and in line with current guidance for SFP NCS analysis.

3.5 Spent Fuel Characterization

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

3.5.1 Depletion Uncertainty

In the Kopp Memo, the NRC staff provided its recommended method for evaluating depletion 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.

As stated in Reference 4, the licensee used this approach to address the uncertainty in the burned fuel compositions. CASMO-4 was used to determine the isotopic composition for the limiting fuel lattice described in Section 3.4.3 of Reference 4 at 25 GWd/MTU, which is beyond the peak reactivity exposure expected for any fuel assemblies qualified under the proposed SCCG k_{inf} limit. MCNP6 calculations were performed to determine the difference in rack k_{inf} between fuel using this isotopic composition and fresh fuel. While the licensee did not explicitly state as much, the magnitude of the calculated rack k_{inf} value makes it clear that the fresh fuel k_{inf} is for a lattice which does not include gadolinium. This uncertainty was statistically combined with the other uncertainties and included in the final estimation of k_{eff}.

DSS-ISG-2010-01 requires the uncertainty associated with minor actinides and fission products (including lumped fission products) be evaluated. Reference 4 states that NUREG/CR-7109 (Reference 8) is used as justification for use of a bias equal to 1.5 percent of the worth of the minor actinides and fission products to address this uncertainty. The staff verified that this is an appropriate application, except, as noted by the licensee, for the lumped fission products and any gaseous isotopes. The licensee stated in Reference 4 that while lumped fission products are used in CASMO-4 for depletion (and the impact on the other isotopic number densities relevant to reactivity would be captured through the depletion uncertainty), they are removed from the calculation used to determine the reactivity of the reference fuel lattices. This is conservative because removing credit for the lumped fission products means that the total neutron absorption cross Section is reduced, and consequently, the MCNP6 rack k_{inf} is increased.

In addition, one of the fission products credited in the analysis, xenon-131 (Xe-131), is a gas. The licensee did not reduce the number density of Xe-131 in the criticality analysis or otherwise account for the possibility of the gas escaping the fuel rods. However, Reference 4 documents a calculation that compared the worth for the Xe-131 to the worth for the lumped fission products, and indicated that the worth of the fission products was higher than the worth of the Xe-131. In effect, the licensee is bounding the Xe-131 uncertainty by not taking credit for lumped fission products. The NRC staff found this to be conservative, and, therefore, is acceptable.

The staff concluded that the licensee's evaluation of the uncertainty in the number densities due to the fuel depletion calculations and the subsequent application is consistent with NRC guidelines combined with the current state of knowledge in the SFP criticality field, and, therefore, is acceptable.

3.5.2 Axial Apportionment of the Burnup or Axial Burnup Profile

BWR peak k_{inf} analysis techniques usually use either 2-dimensional models or a 3-dimensional model with uniform axial burnup distributions. Generally, this is appropriate because the analysis is using the limiting lattice, at the burnup of its peak reactivity, to model the entire fuel assembly. If an analysis were to credit assembly burnup beyond the limiting peak reactivity burnup, then the analysis would have to consider the impact of non-uniform burnup shapes and how they affect the reactivity of lattices for different axial elevations.

The licensee chose to adopt the standard 3-D approach for dealing with the axial burnup distribution, which the staff found to be acceptable.

3.5.3 Burnup History/Core Operating Parameters

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

Factors that lead to a more thermal neutron energy distribution cause the gandolinium-155 and gandolinium-157 ($^{155\&157}Gd$) and fission products to deplete more quickly and reduce plutonium generation. This causes the peak reactivity condition to be reached earlier, achieving a higher in-rack k_{eff} value. Increased water density and decreased void fraction lead to a more thermal neutron energy distribution. Factors that lead to a less thermal neutron energy distribution cause the $^{155\&157}Gd$ and fission products to be depleted more slowly and result in increased plutonium generation. Decreased water density, increased void fraction, and control rod usage all result in neutron energy spectrum hardening.

DSS-ISG-2010-01 provides guidance that depletion simulations should be performed with parameters that maximize the reactivity of the depleted fuel assembly. DSS-ISG-2010-01 references NUREG/CR-6665 (Reference 19), which discusses the treatment of depletion parameters. For fuel and moderator temperatures, Reference 19 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 not always the limiting condition for peak reactivity analyses. The limiting lattice k_{inf} value is established as the maximum value for a given fuel lattice under all possible operating conditions. The higher moderation that occurs in the no-void condition results in a more rapid depletion of the 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 may not be large enough in BWRs to compete with the U-235 depletion that occurs prior to the later peak in $k_{inf.}$

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 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.

As discussed in Reference 4, the licensee performed a NCS analysis that depends on correlating the SCCG k_{inf} with the MCNP6 rack $k_{inf.}$ The first step prior to generating the data for this correlation was to run a series of sensitivity studies based on all lattices of the most reactive GNF2 fuel assembly currently located at DAEC. These studies were done to determine the set of depletion conditions that are expected to be the most limiting. The sensitivity studies included individually varying parameters associated with spacer modeling, control rod blade modeling, channel corner modeling, power density, and temperatures. Based on the sensitivity studies, the most limiting value for each parameter was used for subsequent calculations. The NRC staff verified with the licensee (Reference 1) that the k_{inf} values used in the evaluation of the sensitivity studies were all from the burnup of peak reactivity for each individual calculation, rather than a fixed burnup.

Based on Reference 4, the sensitivity studies were performed using controlled conditions and only considered the impact to the SCCG k_{inf}. The intent of the licensee's approach is to establish a limiting correlation between the SCCG k_{inf} and the MCNP6 rack k_{inf}, so it is not clear how the studies establish that the indicated limiting depletion conditions are also limiting for the correlation. In addition, the correlation captures data for both uncontrolled and controlled lattices, while the sensitivity studies appeared to have only been run using controlled conditions. The licensee clarified in the letter dated December 27, 2016 (Reference 1), that all sensitivity studies were run using both controlled and uncontrolled conditions. In addition, the licensee provided data for the same sensitivity cases when run using the rack model, in order to determine the reactivity impacts on the rack k_{inf}. Based on this data, the staff determined that three of the parameters used in the depletion analysis cases of all fuel lattices at DAEC were not conservative (outside the margin of calculational uncertainty) with respect to the rack k_{inf}: modeling the control blades as filled with water, increasing the control blade worth by 10 percent, and reducing the moderator temperature.

Using the data from References 4 and 1, the NRC staff calculated a summed reactivity impact of a 0.0028 Δk increase to the rack k_{inf} and a 0.0045 Δk decrease in the SCCG k_{inf} if these parameters were changed to be conservative for the rack k_{inf} rather than the SCCG k_{inf}. The actual reactivity impact is not necessarily additive, but this is a reasonable estimate for the purpose of conservatively determining the impact on the relationship between the SCCG and rack k_{inf}. Using the final limiting k_{inf} values provided in Table 6.8 in Reference 4, the NRC staff determined that this would lead to an approximately 7 percent increase in the ratio of the rack k_{inf} to the SCCG k_{inf}.

From Table 6.8 in Reference 4, the rack k_{inf} value determined using the selected ratio is 0.8901. Increasing this value by an amount consistent with the increase in the aforementioned ratio of rack k_{inf} to SCCG k_{inf} results in a new rack k_{inf} value of 0.8960, 0.0059 Δk higher than the value determined used the engineering approximation in Table 6.8 in Reference 4. This value is applied as a penalty in the offsetting margin analysis in Section 3.8. 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 from NUREG/CR-6665, so 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, and this value was also applied in the offsetting margin analysis in Section 3.8.

The depletions described in Reference 4, which were performed in order to develop a limiting correlation included a variety of different void fractions, control rod insertions, and gadolinium loading patterns. While the limiting correlation was selected as an engineering approximation to bound all data points calculated by the licensee, use of that correlation was justified in part by comparison to different selections for a limiting correlation based on 95/95 bounds on selected subsets of the data. The licensee recognized in Reference 4 that to some extent, the data points are not truly independent, since the correlation will depend in part on design details such as the location of water rods, vanished rods, and/or gadolinium-bearing rods. In order to address the first two dependent parameters, the licensee compared the engineering approximation to 95/95 bounds on subsets of the data that only considered GNF2 fuel assemblies and only specific axial lattices.

Some of the graphs and correlation bounds developed by the licensees in Reference 4 included data for multiple depletions of the same fuel lattice (different void fractions or control rod insertion). Since the final operating conditions for a fuel lattice are not known, the fuel lattice will be qualified based on its limiting maximum SCCG k_{inf} value over all possible depletion conditions in order to ensure that the TS limit is met regardless of the actual operating conditions that the fuel lattice experiences. If the data points calculated using the depletion conditions that result in the highest SCCG k_{inf} value tend to approach the correlation bound more closely than the other data points, then use of all data points may bias the 95/95 correlation bound in a slightly non-conservative direction. The licensee addressed this potential in Reference 4 by demonstrating that the engineering approximation bounded 95/95 correlations performed based only on the limiting SCCG k_{inf} data points from each unique nuclear lattice design in the GNF2 fuel included in the analysis.

The licensee treated all depletion parameters except for the void fraction and control rod insertion in a limiting manner. The void fraction and control rod insertion were varied sufficiently to cover the range of possibilities when developing the data used to validate the correlation between the SCCG k_{inf} and the MCNP6 rack k_{inf}. The licensee also used subsets of the data based on the depletion conditions which lead to the limiting SCCG k_{inf} for a given lattice to demonstrate that their limiting correlation remained valid at a 95/95 threshold, which accounts for the fact that fuel would only be qualified based on its maximum SCCG k_{inf} under all depletion conditions. Therefore, the NRC staff finds the core operating conditions used to determine the limiting correlation between the SCCG k_{inf} and the MCNP6 rack k_{inf} to be acceptable.

3.6 Integral Burnable Absorbers

As is typical for BWR plants, Reference 4 states that DAEC utilizes gadolinium poison to help control reactivity and peaking within fuel assemblies. The licensee performed a peak reactivity analysis which explicitly models the gadolinium in the fuel. The manufacturing tolerances associated with the gadolinium are addressed in Section 3.4.4 of this SE. The variation in gadolinium concentration and loading patterns may affect the correlation between the SCCG k_{inf} and MCNP6 rack k_{inf}, and this potential impact is discussed in Section 3.4.2. Therefore, the effects due to the gadolinium have been explicitly captured in the calculations and the associated manufacturing tolerances appropriately considered. No removable burnable absorbers are used, so there is no need for any further evaluation of the burnable absorbers.

Based on its review of the licensee's submittal, the staff has determined that the licensee's treatment of the integral burnable absorbers accounts for the reactivity impacts due to variations in gadolinium concentrations, loading patterns, and manufacturing tolerances. Therefore, the staff finds that the licensee has adequately captured the potential reactivity impacts and uncertainties due to the presence of burnable absorbers in fuel that is qualified for storage in the DAEC SFP racks.

3.7 Analysis of Abnormal Conditions

Section 6.10 of Reference 4 analysis presents the abnormal conditions considered in the analysis. The licensee considered the following abnormal conditions:

- Missing Boral panel in the interior of the rack (limiting condition)
- Boral storage racks being forced together
- Misplaced fuel assembly
- Dropped fuel assembly
- Loss of SFP cooling

Explicit calculations were performed for the missing Boral panel, misplaced fuel assembly, dropped fuel assembly, and loss of SFP cooling scenarios. The Boral storage racks were already considered to be in direct contact when evaluating the interface conditions, so no further analysis was performed to address this abnormal condition. The limiting scenario was found to be for the missing Boral panel, and the reactivity impact was treated as a bias.

One further consideration in the treatment of a missing Boral panel as an accident condition is the fact that the staff is not aware of any accidents which could plausibly cause a Boral panel to drop out of the poison can. If the missing Boral panel scenario is being used to address the possibility that a Boral panel was not installed during the rack manufacturing process, this would become part of the normal condition rather than an accident condition. Therefore, the NRC staff requested the licensee to clarify the intent for inclusion of a possible missing Boral panel. The licensee responded that their rack installation documentation did not indicate any possibility of a missing panel. Therefore, the staff considers use of the reactivity increase associated with the Boral panel to be a 0.0013 Δk conservatism in the calculation (i.e., the difference between the missing absorber panel calculation and the next most limiting accident scenario, that of a dropped fuel assembly).

Based on its review of the licensee's submittals, the staff has concluded that alll considered accident scenarios have either been explicitly analyzed or are treated in a bounding manner, therefore, the staff concludes that the reactivity increase due to accident conditions is appropriately applied in the determination of the final k_{eff} value for comparison to the regulatory limit.

3.8 Margin Analysis and Comparison with Remaining Uncertainties

The staff identified several potentially non-conservative assumptions during its review of this LAR. A bounding estimate of the reactivity impact for each assumption is listed in the below table, 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 below comparison, the staff concludes that the available margins offset the potential non-conservatisms.

	Estimated Reactivity Impact (Δk)
Potential Nonconservatisms	
Modeling of BORAL as a homogeneous material	0.0030
Simplification of SFP cell corner modeling	0.0015
Spacer growth due to irradiation	0.0017
Cladding thinning due to irradiation	0.0039
Non-conservatism in depletion parameters	0.0059
Use of a constant power history	0.0020
Lack of U-235 and Gd enrichment uncertainty evaluation	0.0018
Total reactivity impact of nonconservatisms	*0.0198*
Conservatisms	
Margin to regulatory limit	0.0199*
Missing Boral panel evaluation	0.0013

*Total reactivity impact of conservatisms***	*0.0212*

*From Reference 4 (substituting a dry rack wall cavity and bulging evaluation for the blister evaluation).
**Other potential conservatisms exist in the licensee's calculations, but the NRC staff did not attempt to quantify them due to the fact that the margin to the regulatory limit alone was sufficient to bound the identified non-conservatisms.

4.0 SUMMARY

The NRC staff review of the DAEC spent fuel storage racks NCS analysis, documented in Reference 4, identified some non-conservative items. Those items were evaluated against the margin to the regulatory limit and what the NRC considers an appropriate amount of margin attributable to conservatisms documented in the analyses. The identified conservatisms were sufficient to bound the non-conservative items, therefore, the NRC staff concludes that there is a reasonable assurance that the DAEC SFP fuel storage racks meet the applicable NCS regulatory requirements.

The NRC staff reviewed information from the licensee's Boral surveillance program and the proposed TS 5.5.15, and has determined that the surveillance program as described in TS 5.5.15 will provide reasonable assurance that the licensee will be able to detect degradation of the neutron absorbing material before its ability to perform its intended safety function is impacted. In addition, the staff has determined that the licensee has met the acceptance criteria in DSS-ISG-2010-01. On this basis, the staff concluded that the proposed changes to TS 4.3.1 and TS 4.3.3, and the proposed addition of TS 5.5.15 for a SFP neutron absorber monitoring program meet the applicable requirements of 10 CFR 50.68, and GDCs 61 and 62, and, are therefore acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that maybe released offsite, and that there is no significant increase in individual cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (81 FR 43665, dated July 5, 2016). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

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Date of issuance: March 30, 2017

SUBJECT: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT TO REVISE TECHNICAL SPECIFICATIONS FUEL STORAGE REQUIREMENTS (CAC NO. MF7486) DATED MARCH 30, 2017

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