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

October 24, 2014

Karen D. Fili
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company - Minnesota
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT TO REVISE THE TECHNICAL SPECIFICATIONS TO SUPPORT FUEL STORAGE SYSTEM CHANGES (TAC NO. ME9893)

Dear Mrs. Fili:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 182 to Renewed Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant (MNGP). The amendment consists of changes to the technical specifications (TSs) in response to your application dated October 30, 2012, as supplemented on May 16, 2013, June 7, 2013, March 13, 2014, and May 30, 2014.

The amendment revises MNGP TS 4.3.1, "Fuel Storage Criticality," and TS 4.3.3, "Fuel Storage Capacity," to reflect fuel storage system changes; revises the criticality safety analysis that addresses legacy fuel types, in addition to the planned use of AREVA ATRIUM™ 10XM fuel design; and adds a new TS 5.5.14, "Spent Fuel Pool Boral Monitoring Program," for assuring that the spent fuel pool storage rack neutron absorber material (Boral) continues to meet the minimum requirements assumed in the criticality safety analysis.

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

Sincerely,

Terry A. Beltz

Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 182 to DPR-22
2. Safety Evaluation

cc w/encls: Distribution via ListServ

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

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 182
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company - Minnesota (NSPM, the licensee), dated October 30, 2012, as supplemented on May 16, 2013, June 7, 2013, March 13, 2014, and May 30, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Renewed Facility Operating License No. DPR-22 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 182, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

Enclosure 1

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-22 and
Technical Specifications

Date of Issuance: October 24, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 182

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following page of Renewed Facility Operating License No. DPR-22 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
4.0-1	4.0-1
4.0-2	4.0-2
5.5-12	5.5-12
---	5.5-13

2. Pursuant to the Act and 10 CFR Part 70, NSPM to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operations, as described in the Final Safety Analysis Report, as supplemented and amended, and the licensee's filings dated August 16, 1974 (those portions dealing with handling of reactor fuel);

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

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

5. Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess, but not separate, such byproduct and special nuclear material as may be produced by operation of the facility.

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

1. Maximum Power Level

   NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1775 megawatts (thermal).

2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 182, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. Physical Protection

   NSPM shall implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search

4.0 DESIGN FEATURES

4.1 Site Location

4.1.1 Site and Exclusion Area Boundaries

The site area and exclusion area boundaries are as shown in Chapter 15, Figure ND-95208 of the USAR.

4.1.2 Low Population Zone

The low population zone is all the land within a 1 mile radius circle as shown in Chapter 15, Figure ND-95208 of the USAR.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 484 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material and water rods. Some fuel rods may consist of a Zircalloy base and a zirconium inner liner. Fuel assemblies shall be limited to those fuel designs that have been analyzed with NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 121 cruciform shaped control rod assemblies. The control material shall be boron carbide or hafnium metal as approved by the NRC.

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 a maximum k-infinity of 0.8825 in the normal spent fuel pool storage rack configuration;

b. $k_{eff} \leq 0.95$ for high density fuel racks and low density fuel racks if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.2.1 of the USAR;

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

c. A nominal 6.563 inch center to center distance between fuel assemblies placed in the 13 x 13 high density storage racks, a nominal 6.625 inch center to center distance between fuel assemblies placed in the original storage rack, and a 12-inch gap between the high density racks and the original rack.

4.3.1.2 The new fuel vault shall not be used for fuel storage. The new fuel shall be stored in the spent fuel storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 1003 ft 7.25 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2217 fuel assemblies.

5.5 Programs and Manuals

5.5.13 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filtration (CREF) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventative maintenance.

c. Requirements for (i) determining the unfiltered air in-leakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one subsystem of the CREF System, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air in-leakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air in-leakage measured by the testing described in paragraph c. The unfiltered air in-leakage limit for radiological challenges is the in-leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air in-leakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered in-leakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5 Programs and Manuals

5.5.14 Spent Fuel Pool Boral Monitoring Program

The 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 USAR Section 10.2.1 is maintained. The program shall include the following:

a. Periodic physical examination of representative Boral coupons or in situ storage racks at a frequency defined by observed trends or calculated projections of Boral degradation. The measurement will be performed to ensure that average thickness of the coupon (or average thickness of a representative area of the in situ storage rack) does not exceed the nominal design thickness of the coupon (or storage rack) plus the 0.055-inch dimension assumed for the analyzed blister.

b. Neutron attenuation testing of a representative Boral coupon or in situ storage rack shall be performed prior to December 31, 2015, and thereafter 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 boron areal density will be that value assumed in the criticality safety analysis (0.013 gm/cm^2).

c. Description of appropriate corrective actions for discovery of 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. 182 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY - MINNESOTA

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By application dated October 30, 2012 (Reference 1), as supplemented by letters dated May 16, 2013; June 7, 2013; and March 13, 2014 (References 2, 3, and 4, respectively), Northern States Power Company – Minnesota (NSPM, the licensee), doing business as Xcel Energy, Inc., requested changes to the technical specifications (TSs) for the Monticello Nuclear Generating Plant (MNGP). Specifically, NSPM requested to revise TS 4.3.1, "Fuel Storage Criticality," and TS 4.3.3, "Fuel Storage Capacity," to reflect fuel storage system changes and a revised criticality safety analysis that addresses legacy fuel types, in addition to the planned use of AREVA ATRIUM™ 10XM fuel design.

Additionally, in a letter dated May 30, 2014 (Reference 5), NSPM proposed new TS 5.5.14 to incorporate a Spent Fuel Pool (SFP) Boral Monitoring Program in its Technical Specifications.

The proposed change to TS 4.3.1.1.a would revise the parameter and value associated with the nuclear fuel neutron multiplication factor that correlates with the SFP storage rack k-effective criterion of 0.95, based on the analysis completed in AREVA Report ANP-3113(P). The licensee submitted AREVA Report ANP-3113(P), documenting the MNGP criticality analysis.

The proposed change to TS 4.3.1.1.b would include the low-density fuel storage racks to align it with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.68, "Criticality accident requirements," and the other high-density fuel storage racks.

The proposed change to TS 4.3.1.1.c would delete the requirement due to the low-density fuel storage racks being included in the proposed TS 4.3.1.1.b.

The proposed change to TS 4.3.1.1.d involves an administrative change, as well as removing a reference to the 8x8 high-density fuel storage racks that was approved by the U.S. Nuclear Regulatory Commission (NRC) in 2007 by Amendment No. 150, but was never installed in the

SFP. The proposed change would also change the minimum required gap specified between the high-density to low-density racks from 2 inches to 12 inches.

The proposed change to TS 4.3.1.2 would eliminate the criticality criteria for the new fuel vault (NFV) and prohibit use of the NFV for fuel storage. The proposed license amendment request also corrects non-conservatism in the criticality analysis for the NFV. The non-conservative TS has been addressed by the licensee's Corrective Action Program and, as part of its corrective actions and as previously discussed above, the licensee will prohibit use of the NFV. As such, all new reactor fuel assemblies will be stored directly in the SFP.

The proposed change to TS 4.3.3 would revise the fuel storage capacity from 2301 fuel assemblies to 2217 fuel assemblies.

The proposed license amendment request would add a new TS 5.5.14, "Spent Fuel Pool Boral Monitoring Program," to assure that the SFP storage rack neutron absorber material (Boral) continues to meet the minimum requirements assumed in the criticality safety analysis. The neutron absorber program will monitor the integrity of the Boral neutron absorber to ensure that it maintains a minimum areal density of 0.013 g/cm^2.

In Enclosure 5 of the Reference 1, the licensee submitted a non-proprietary version of AREVA Report ANP-3113(P) to document the MNGP criticality analysis.

The licensee's supplemental letters dated May 16, 2013, June 7, 2013, and March 13, 2014, 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 June 11, 2013 (78 FR 35063). A revised no significant hazards consideration determination was published in the *Federal Register* on June 24, 2014 (79 FR 35805), to consider the aspects of the new Boral monitoring program provided in the supplemental letter dated May 30, 2014.

2.0 REGULATORY EVALUATION

The regulation in 10 CFR 50.36(c)(4) requires that design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section.

The regulation in 10 CFR 50.68(b)(1) requires that plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

The regulation in 10 CFR 50.68(b)(4) requires, in part, that if credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The regulation in 10 CFR Part 50, Appendix A, General Design Criterion 62, "Prevention of criticality in fuel storage and handling," 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.

The NRC staff issued an internal memorandum on August 19, 1998, containing guidance for performing SFP criticality analysis reviews. The memorandum is known as the "Kopp Letter," after the author. The Kopp Letter provides guidance on salient aspects of a criticality analysis. The guidance is germane to both boiling-water reactors and pressurized-water reactors, and to borated and unborated conditions. The staff used the Kopp Letter as guidance for the review of the current analysis.

On September 29, 2011, the NRC staff in the Division of Safety Systems (DSS) issued Interim Staff Guidance (ISG) DSS-ISG-2010-1 (Reference 6). This ISG provides updated review guidance to the NRC staff for addressing the increased complexity of recent SFP licensing actions. The NRC staff used ISG DSS-ISG-2010-1 for its review of this license amendment request.

Finally, the NRC staff utilized guidance provided Section 9.1.2, "New and Spent Fuel Storage," of NUREG-0800, Standard Review Plan (SRP)," to address the capability of the spent fuel storage facilities to maintain the fuel in a safe and subcritical array during all anticipated operating and accident conditions, and to determine that no potential mechanisms will alter the dispersion of boron carbide (B_4C) in the Boral panels and/or cause physical distortion of the tubes retaining the stored fuel assemblies.

3.0 TECHNICAL EVALUATION

There is currently no generically-approved methodology for performing SFP criticality analysis. Therefore, a licensee must submit a plant-specific SFP criticality analysis that includes technically supported margins. The NRC staff reviewed the MNGP analysis to assure that the assumptions made, both stated and unstated, were technically substantiated. The NRC staff reviewed the application and supplemental information to determine whether the submittal provides reasonable assurance that the regulatory requirements will be met. As discussed below, the NRC staff finds that the licensee has provided sufficient technical information to support completion of its review of the license amendment request.

3.1 Depletion Analysis

3.1.1 Selection of Bounding Fuel Assembly

Section 2.0 and Appendix B of AREVA Report ANP-3113(P) provide information on fuel assembly selection. The licensee analyzed current, future, and all legacy fuel assembly designs used, or expected to be used, at MNGP to establish the limiting fuel assembly design. MNGP has loaded a number of different product lines, starting with GE 7x7, GE 8x8 (GE7 through GE10), GE 9x9 (GE11), and GE 10x10 (GE12 and GE14) fuel.

The licensee performed a reactivity comparison and concluded that the ATRIUM 10XM fuel assembly design is the limiting fuel assembly design and an acceptable reference fuel assembly

design for subsequent criticality analyses. The ATRIUM 10XM fuel assembly is a 10x10 fuel rod array with an internal square water channel offset in the center of the assembly, which takes the place of nine fuel rod locations. The assembly contains part-length fuel rods and is defined as two uranium (U)-235 enrichment/gadolinia concentration zones separated by the ATRIUM 10XM geometry transition at [ ] inches. The bottom enrichment/gadolinia zone extends up to this transition boundary and contains [ ] fuel rods. The top enrichment/gadolinia zone extends from this geometric transition to the top of the fuel assembly and contains [ ] fuel rods.

Fuel may be stored in the SFP provided that the lattices are bounded by the reference lattice. The fuel should meet one of two criteria. The first criteria is that for all the fuel height, the maximum lattice average enrichment may be no more than 4.7 percent U-235, and there must be a minimum number of eight gadolinium rods. For the top enrichment/gadolinia zone, the minimum percent gadolinia is 3.5 percent. For the bottom enrichment/gadolinia zone, the minimum percent gadolinia is 3.919 percent. These gadolinia rods cannot be loaded on the perimeter of the lattice or adjacent to the water channel. Alternatively, the fuel can meet the second criterion which states that, for a lattice average enrichment of less than 5 percent U-235, the maximum in-rack infinite k-effective cannot exceed 0.8825 at any point during its lifetime. Once all the biases and uncertainties are taken into account, as described below, the maximum k-effective results in a value of 0.928.

The method used by the licensee to determine the limiting assembly follows the guidance set forth in DSS-ISG-2010-1. Therefore, the NRC staff finds that the licensee's approach for selection of a bounding assembly used to determine limiting in-rack reactivity for the MNGP SFP is acceptable.

3.1.2 Reactor Parameters

Section 6.4 of ANP-3113(P) discusses the reactivity effects in the SFP of fuel temperature, moderator temperature, void history, power density, and reactivity depletion during rodded operation. The conclusion of the licensee's analysis is that moderator void is the most significant parameter for its depletion analysis.

The licensee used sensitivity studies to determine the effect of fuel temperature, and the temperature was varied plus or minus 100 degrees Fahrenheit (°F) relative to the base depletion temperature. The results of this study are depicted on Table 6.4. There was no significant change (less than 0.0001 Δk) in reactivity when the temperature was varied and, therefore, it does not have a significant effect in the analysis.

The power density used in the fuel depletion analysis is based on the core rated power per unit volume, and values of 50 percent, 100 percent, and 150 percent were analyzed to determine its effect on reactivity. The results are depicted on Table 6.5, and indicate that there is a small effect on in-rack lattice reactivity over very large changes in depletion power density. However, this effect is less significant than the effect from the change in moderator void.

The moderator temperature used for in-core depletion is the temperature at saturated conditions corresponding to rated dome pressure, and the parameter used in the analysis is the moderator density/void fraction. The analysis was performed at varying void histories for the bottom and top lattices. Table 6.4 and Table B.3 indicate the results of the void history sensitivity analysis

for both the bounding legacy lattice as well as the ATRIUM 10XM fuel. For all of the lattices, the k-effective from the most limiting void fraction for each top and bottom lattice was used in the SFP in-rack criticality analysis.

Based on the above, the NRC staff finds the above consideration of reactor parameters used to determine limiting in-rack reactor for the MNGP SFP to be acceptable.

3.1.3 Rodded Operation

The licensee evaluated the impacts of rodded operation on the fuel assembly reactivity during reactor operation. Rodded operation can increase fuel assembly reactivity due to the power gradient imposed by the inserted control rod which can reduce local power densities and the amount of U-235 depletion, while increasing the amount of plutonium production. There are four factors that the licensee has evaluated that may mitigate the amount of controlled depletion due to rodded operation: 1) not all available control rods are used during a sequence; 2) control rods are typically not fully inserted; 3) bundles in peripheral and near-peripheral core locations are usually not controlled, and 4) bundles that are at a reduced power during the controlled time period reduce their accumulated burnup while in the controlled state. The effect was evaluated to be for a typical bundle, as it will experience controlled depletion for only a fraction of its time in the reactor core.

A sensitivity study was conducted to show the impacts of in-core controlled depletion on the peak in-rack k-infinity, where both the power density and the depth of the inserted blade was varied for the new ATRIUM 10XM reference bounding lattices, as well as the limiting legacy fuel lattices. Table 6.6 indicates the results of the sensitivity study for both controlled and uncontrolled depletions. The use of uncontrolled depletion at rated power conditions is shown to bound depletion at controlled conditions for the ATRIUM 10XM reference bounding lattices; therefore, fuel with uncontrolled depletion at rated power is used to characterize the SFP in-rack reactivity in the subsequent criticality analysis.

Based on the above, the NRC staff finds that rodded operation is appropriately accounted for in the MNGP SFP criticality analysis.

3.1.4 Depletion Uncertainty

The licensee uses the CASMO-4 code when conditions require fuel and gadolinia depletion. CASMO-4 is a multi-group, two-dimensional transport theory code with a rack geometry option that allows typical storage geometries to be defined on an infinite lattice basis. CASMO -4 was used to perform in-core isotopic depletion at characteristic void history levels for both bottom and top lattices; perform in-rack k-infinite assessments to identify the most reactive lattices; define the reactivity equivalence at the beginning of life (REBOL) lattices with fresh fuel and no gadolinia; confirm subsequent KENO V.a base case criticality calculations; and evaluate the depletion component in the manufacturing uncertainty for gadolinia content.

Appendix D of ANP-3113(P) describes the depletion uncertainty applied in the criticality analysis. The depletion uncertainty is derived from the AREVA licensing topical report EMF-2158(P). The REBOL lattices are based upon a uniform enrichment distribution, which increases the boiling water reactor (BWR) lattice reactivity because the low-enriched rods in the

corners of the lattice are replaced with rods at an average enrichment level. The distribution was determined to be more reactive than the ATRIUM 10XM design by up to 0.004 Δk.

To verify that the depletion uncertainty applied by the EMF-2158(P) method is bounding based on the guidance in ISG DSS-ISG-2010-1, the licensee calculated five percent of the reactivity difference from beginning of life (without gadolinia) to peak reactivity, and compared the value to the value obtained via EMF-2158(P); the five percent burn up reactivity decrement can be found in Table D.4 for the limiting lattices. The depletion uncertainty based on the five percent of the reactivity decrement was calculated to be 0.0055 Δk for the bounding lattices, and this value is bounded by a 0.010 Δk bias term applied during the REBOL lattice calculation. This bias term is used in the final calculation of k-effective to account for depletion uncertainty.

Based on the above, the NRC staff finds that depletion uncertainty is appropriately accounted for in the MNGP SFP criticality analysis.

3.1.5 Burnable Absorbers

Section 7.1 and Appendix B of ANP-3113(P) discuss the use of burnable absorbers in the fuel. There are eight gadolinia rods in the reference bounding lattices, and in the limiting rack model (see Section 3.2.1) the location of these rods was selected to maximize the reactivity of the lattices. This arrangement was determined to be more reactive by 0.0020 Δk, and bounds the lattice designs. As stated above, the licensee required that new fuel must have at least eight gadolinia rods with a minimum percent gadolinia of 3.5 percent for the top enrichment/gadolinia zone, and 3.919 percent for the bottom enrichment/gadolinia zone.

Based on the above, the NRC staff finds that the burnable absorber requirements are conservative and appropriate as they pertain to the MNGP SFP criticality analysis.

3.2 Criticality Analysis

The criticality analysis is based upon using an ATRIUM 10XM reference bounding assembly as described above, which was depleted to establish the lifetime maximum k-infinity. The results are dependent on the lattice geometry, the U-235 enrichment level, and the gadolinia concentration. There is no axial burn-up profile assumption in the analysis due to the low level of burnup in the lattices.

3.2.1 Rack Model

The MNGP SFP provides the capability of storing a maximum of 2217 fuel assemblies. Twenty (20) fuel assemblies can be stored in an aluminum I-beam rack (the original rack), and 2197 assemblies can be stored in 13x13 high density Boral storage rack modules.

The licensee modeled three different rack configurations for the high-density racks in order to determine the impacts on in-rack reactivity. The first model is a simplified single cell model where the rack has an arrangement of staggered or alternating Boral tubes, and uses an array of two tube cells and two non-tube cells for explicit modeling. The Boral plate is modeled as Boron-10 (B-10) only, and the location of the Boral is shifted between the storage cells so that half of the actual thickness is assigned to each cell wall. The k-effective from this simplified

model is compared to the k-effective from a model which includes carbon and aluminum in the Boral material specification. Average storage cell pitch and water gap values are also used in these rack models.

The second model is an explicit storage cell model where the arrangement is composed of a 2x2 array with two Boral tube storage cells and two open or non-tube cells. The Boral plate is modeled using the nominal width and the average assembly cell pitch. As with the single-cell model, k-effective values are compared between a B-10-only Boral model and a model that also includes carbon and aluminum in the Boral material specification.

The third model is an explicit rack model where the explicit storage cell model is expanded to a 13x13 array with tube cells in each corner. The nominal rack-to-rack water gap is modeled and the stainless steel closure plates are approximated for non-tube cells along the perimeter of the rack.

Each cell is assumed to contain an assembly composed of 3.38 weight percent (wt%) U-235 (top) and 3.21 wt% U-235 (bottom), uniformly-enriched REBOL lattices without gadolinia, which are the most reactive lattices.

The results of the modeling are depicted in Table 6.1. The B-10 only, explicit 2x2 model with periodic boundary conditions has the largest in-rack reactivity and is therefore chosen to characterize the high-density storage rack model, which is 0.0150 Δk more reactive than the result from the more realistic storage rack model.

The licensee also modeled the original low-density rack. The results are provided in Table 6.3, which indicate that the low-density rack model is bounded by the 2x2 B-10 only model.

Composition of Boral

The Boral is imbedded in stainless steel in a high density rack module. The Boral core is made up of a central segment composed of a dispersion of boron carbide in aluminum. This segment is clad on both sides with aluminum. The stainless steel container tubes are closure welded with vent holes to prevent hydrogen gas buildup. The minimum B-10 areal density of 0.013 grams per square centimeter (g/cm^2) is used when modeling Boral. To account for the potential from significant Boral blistering in the future, a uniform 0.055-inch void region has been used to model the Boral in the analysis, which increases reactivity by 0.004 Δk. In its May 30, 2014, supplement, the licensee submitted a program to monitor the neutron absorber and ensure that the material remains in an operable condition in the SFP.

Interface Requirements

The only interface that is present is that between the original low-density rack and the high-density Boral racks. The racks are neutronically isolated by twelve or more inches of water and, therefore, are independent of each other. Consequently, the interface between the racks was not evaluated for this analysis.

Based on the above, the NRC staff finds that the licensee's modeling of the MNGP SFP storage racks is acceptable.

3.2.2 Normal Conditions

Fuel movements required to load and unload assemblies into allowable storage locations within the SFP are considered normal operation. The allowed storage locations include the SFP storage racks and the fuel preparation machines (FPM). As discussed above, the reference fuel storage array model assumes an infinite array in both radial and axial dimensions using a periodic boundary condition, and assumes that each storage location contains the bounding fuel design. The model bounds the case of suspending a single bundle over the rack, as well as raising and lowering the fuel into a cell during normal fuel operation.

There are two FPMs that allow for the storage of a single fuel assembly within each machine. The FPMs are neutronically isolated from each other. However, the licensee analyzed a condition where an assembly suspended from the refuel bridge can be brought into close proximity to an assembly stored in the FPM. A reactivity optimization search was performed using different assembly spacing and orientations, with and without channels, as well as varying pool water temperature from 4 degrees Celsius (°C) to 60°C, with the maximum result being calculated with unchanneled assemblies at 4°C, which is bounded by the limiting rack model.

Since it is possible that fuel assemblies could be positioned eccentrically in the storage rack, the licensee evaluated how fuel assembly orientation could affect the in-rack reactivity. The ATRIUM 10XM fuel assembly is asymmetric in nature, so the assembly being rotated 90, 180, and 270 degrees from the base case was evaluated. The 90-degree rotated case was the most limiting with a reactivity increase of 0.0010 Δk, which was included in the final k-effective reactivity calculations. The ability for the assembly to be offset from the cell-centered position was also evaluated. Between one and four assemblies were moved relative to one another within their cells, and there was no statistically significant increase relative to the centered position; therefore, this was not considered in the analysis.

The licensee's analysis evaluated the following uncertainty components by varying each parameter individually and combining the maximum positive reactivity effect of each component in quadrature: fuel pellet density, fuel pellet outer diameter, fuel rod pitch, enrichment, channel growth, channel thickness, pellet void volume, cell wall thickness, gadolinium concentration, gadolinium pellet density, fuel cladding inner diameter, fuel cladding outer diameter, eccentric fuel positioning, and storage cell pitch. A bounding pellet density, minimum Boral panel thickness, and minimum B-10 areal density were used in all k-effective calculations.

3.2.3 Abnormal Conditions

Section 7.6 of ANP-3113(P) provides information on abnormal and accident conditions that are considered in the analysis. The licensee considered the following abnormal conditions:

- Missing Boral plate in the interior of the rack

- Boral storage racks being forced together (rack movement)

- Misloaded bundle scenarios
  - o Assembly misloaded between the pool wall and storage rack adjacent to an open cell with no Boral
  - o Assembly misloaded into the corner region adjacent to 3 racks
  - o Assembly misloaded between the FPM adjacent to an open cell with no Boral
  - o Assembly misloaded in the space between the original rack and the Boral rack
- Dropped assembly lying horizontally across the top of the spent fuel pool
- Loss of SFP cooling

There are no special loading patterns, minimum burnup requirements, neutron absorber inserts, or requirements to leave empty storage cells between assemblies. Therefore, these conditions were not considered in analyzing the following abnormal conditions.

The licensee determined that the limiting abnormal condition is the single Boral plate missing from an interior storage rack location. The NRC staff does not consider a missing Boral plate an abnormal condition; rather, the staff considers this condition a design flaw in the storage rack. As part of this analysis, the moderator temperature and the assembly position were varied to find the worst case scenario. The most limiting condition occurs at 4°C with one assembly moved to the edge of the storage cell and the adjacent assembly moved half the distance to the edge of the cell. This accident condition added 0.0060 Δk of reactivity, which was used in the calculation of the final k-effective.

Rack movement, in which two or more Boral racks could be forced together during a seismic event, was also evaluated. To model this event, the spacing between the racks was reduced from two or more inches to less than a half inch. This accident condition added about 0.0050 Δk of reactivity, which is bounded by the missing Boral plate scenario. The NRC staff considers this to be the limiting abnormal condition.

Various misloaded assembly accident conditions were analyzed. The first misload accident was an assembly that was placed at the edge of a Boral storage rack adjacent to an assembly in an open storage cell. This condition increases reactivity by less than 0.0010 Δk. The second misload accident analyzed was an assembly placed where three racks meet together, and this condition produced no significant reactivity increase. The third accident was an assembly placed in the FPM, while a second assembly is moved between the FPM and the fuel storage rack. This results in a reactivity increase of less than 0.0010 Δk. The last misload accident analyzed is an assembly placed in the water gap between the high-density Boral rack and the original low-density storage rack, optimized for the worst-case positioning. This scenario produced a reactivity increase of less than 0.0010 Δk. All of these misloading accidents are bounded by the rack movement scenario.

A fuel assembly dropped vertically onto an assembly in the storage rack was also analyzed. This particular accident can cause deformation to the dropped assembly, as well as any other

assemblies that are contacted during the drop. The fuel deformation was found to have negligible impact on the in-rack reactivity. There is also a minimal reactivity increase when the assembly comes to rest in a horizontal or inclined position on top of the storage rack because the dropped assembly is neutronically isolated from the fuel in the storage cells. The fuel assembly drop cases are bounded by the rack movement scenario.

For the high-density Boral rack model, the limiting moderator temperature is 4°C. Therefore, a loss of SFP cooling does not increase the reactivity of the Boral racks. The original low-density rack was analyzed at the limiting temperature/void condition that correlates to the applicable pressure at 30 to 40 feet below the surface of the pool, which increases the reactivity of the original rack by 0.0080 Δk. This increase does not affect the adjacent Boral rack because there is sufficient water between the original rack and the Boral racks to neutronically isolate it from the original rack. The in-rack reactivity for the original rack under this accident condition is bounded by the rack movement event for the Boral rack due to the original rack's reactivity being significantly lower that the Boral rack reactivity.

Based on the above, the NRC staff finds that the licensee's evaluation of the abnormal conditions in the MNGP SFP is acceptable.

3.2.4 Criticality 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. ISG DSS-ISG-2010-1 references NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology."

NUREG/CR-6698 states, in part, that 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 reviewing the code validation methodology presented in the licensee's application. NUREG/CR-6698 outlines the basic elements of validation, including identification of operating conditions and parameter ranges to be validated, selection of critical benchmarks, modeling of benchmarks, statistical analysis of results, and determination of the area of applicability.

The analysis methodology uses the KENO V.a Monte Carlo code, which is part of the SCALE 4.4a Modular Code System. The SCALE driver module CSAS25 uses the ENDF/B-V 44 energy group data library. It also uses module BONAMI-2 and NITAWL to perform spatial and energy self-shielding adjustments of the cross sections for use in KENO V.a. The SCALE 4.4a code system is used to calculate Dancoff coefficients, to calculate k-effective results, and to evaluate accident conditions, alternate loading conditions, and manufacturing tolerance conditions.

The sources for the critical configurations are the International Handbook of Evaluated Criticality Safety Benchmark Experiments (IHECSBE, 2009), NUREG/CR-6361, "Criticality Benchmark

Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," and NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data."

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 the IHECSBE to be an appropriate source of information for the critical experiment models. Critical experiments from NUREG/CR-6361 contain important features such as soluble boron and poisoned fuel rods. Use of the HTC experiments documented in NUREG/CR-6979 is important to cover the actinide distribution of burned fuel.

The licensee used 68 critical experiments that included configurations performed with lattices of uranium dioxide (UO_2) fuel rods in water having various enrichments and moderating ratios. Since the licensee used a fresh fuel equivalent calculation, only critical experiments for fresh fuel were included in the benchmark data set.

The licensee considered the use of HTC critical experiments in its criticality analysis. Twenty-three (23) of the 26 cases from the HTC experiments were added to the 68 critical experiments used, and the results analyzed in Appendix C of ANP-3113(P). It was found that a more conservative result was achieved when the HTC experiments were not included in the analysis.

The licensee identified the applicable operating conditions for its validation, and compared the spectral parameters between the benchmarks and the SFP conditions to demonstrate that the selected benchmarks are applicable. The licensee performed a trend analysis and identified an additional bias for those parameters with a statistically significant trend as shown in Appendix C of ANP-3113(P).

Based on the above, the NRC staff finds that the information supporting the code validation is acceptable.

3.3 Boral Surveillance Program

The licensee's Boral Monitoring Program consists primarily of monitoring the physical properties of the neutron absorber material Boral by performing periodic dimensional and visual checks to confirm the material's physical properties and neutron attenuation testing to confirm the neutron absorption capabilities of the material are being maintained.

3.3.1 Program Description

The purpose of the licensee's Boral Monitoring Program is to characterize certain properties of the Boral to assess the capability of the Boral panels in the racks to continue to perform their intended function. The program will monitor how the Boral absorber material properties change over time under the radiation, chemical, and thermal environment found in the SFP by

monitoring the condition of Boral coupons that are representative of the Boral found in the spent fuel storage racks.

Each coupon sampled in the monitoring program is approximately 6 inches square and has a ring welded to it to allow it to be suspended from a stainless steel chain attached to a top-mounting piece designed to fit over the peripheral edge of a high density fuel storage rack. The licensee's original Boral program consisted of seven sets of coupons, 21 coupons in total. All but one coupon set, coupon set number seven, has been removed from service during prior testing. A sample set consists of three coupons: two coupons encased in the stainless steel "sandwich" array of the high-density fuel storage tubes, and a third, bare, coupon that consists of Boral without a stainless steel encasement. For the two coupons in the set that are encased in stainless steel, one coupon has two holes drilled in the corners to simulate the first four high density fuel storage racks installed in the spent fuel pool. This is the "A" sample coupon of the set. The second coupon, or "B" sample coupon, of the set has four holes drilled in the corners to simulate the remaining nine storage racks that were installed. The "C" sample coupon of the set is the bare Boral coupon.

The licensee will continue to use the remaining coupon set to monitor for degradation, deformation, or reduction in neutron attenuation capability of the Boral in the SFP racks. The licensee stated that the tests will be non-intrusive such that the integrity of the coupon set as a representative surrogate for the rack's Boral will not be compromised. The tests will include measurements of the coupon's physical dimensions (including thickness), visual examination of the coupons, weight measurements, and neutron attenuation testing. The remaining coupon set will be removed from the SFP for up to eight weeks during the course of each testing campaign. The next neutron attenuation test for the remaining coupon set will be performed prior to December 31, 2015. Subsequent retesting of the remaining coupon set will be performed on an interval not to exceed 10 years.

3.3.2 Monitoring Changes in the Physical Properties and Testing of Coupons

On an interval not to exceed 10 years, the remaining coupon set will be removed from the SFP pool, dried, and the following measurements will be performed. The coupons will not be baked.

1. Measurement of coupon dimensions (including thickness)

   Each coupon in the set will be measured for overall dimensions and thickness and compared to a baseline. The measurement data will be recorded for trending purposes and evaluation. Any significant trend in thickness changes will be evaluated in accordance with the Corrective Action Program. Dimensional measurements will be taken at locations that are representative of the Boral panels in the SFP (i.e., over the stainless steel cladding and Boral) using a calibrated micrometer and/or caliper.

2. Visual examination of coupon surface for blemishes and blisters

   The exposed surfaces of each coupon in the set will be visually examined for blemishes and blisters. Blemishes such as corrosion pits could be an indication of material loss that could lead to reduced neutron attenuation. The character of any blister formation will be recorded and reconciled with that assumed in the criticality analysis. Any significant

blemishes will be characterized and evaluated in accordance with the Corrective Action Program.

3. Weight

   Each coupon in the set will be weighed and compared to the nominal coupon weight and any previous measurement. Weight of the coupon will be trended and used to help identify any material loss that might be occurring under the stainless steel cladding. Any significant trend in weight reduction will be evaluated in accordance with the Corrective Action Program to determine if there is any impact on the neutron attenuation capability of the material.

4. Neutron Attenuation Testing

   Each coupon in the set will be subjected to neutron attenuation testing at a qualified laboratory to determine the boron-10 areal density. Any coupon that does not meet the acceptance criterion below would be evaluated in accordance with the Corrective Action Program to determine if there is any impact on the SFP rack neutron attenuation capability.

The licensee's acceptance criteria are as follows:

- Thickness measurement of the stainless steel coupons is not to exceed 0.055 inch plus nominal cladding thickness.
- B-10 minimum areal density of 0.013g/cm^2

The acceptance criterion for B-10 areal density represents the minimum areal density assumed in the criticality analysis of record. For coupon thickness, the value 0.055 inch comes from an assumed value in the criticality analysis of record for modeling a blistering condition. The licensee stated that assuming a uniform void with a 0.055 inch height bounds the condition of having blisters, with a 0.125 inch height and a 1.25 inch spherical cross section, packed edge to edge covering one side of a Boral plate. The licensee chose this blister size based on the largest typical blister size identified by the manufacturer. Because there is no space between the Boral material and the surrounding stainless steel cladding of the SFP storage racks (or the Boral coupons), any blister formation would be detectable through thickness measurements as the blister would push out on the stainless steel cladding as it formed. Therefore, a measurement exceeding 0.055 inches plus nominal cladding thickness will be a good indication that there is blistering of the Boral taking place. Coupon weight will tend to vary based on how dry the licensee is able to get the coupon before it is tested; therefore, there is no acceptance criterion for weight.

In its May 30, 2014, letter (Reference 5), the licensee proposed the following addition as Section 5.5.14 of its Technical Specifications:

5.5.14 Spent Fuel Pool Boral Monitoring Program

The 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 USAR Section 10.2.1 is maintained. The program shall include the following:

a. Periodic physical examination of representative Boral coupons or in situ storage racks at a frequency defined by observed trends or calculated projections of Boral degradation. The measurement will be performed to ensure that average thickness of the coupon (or average thickness of a representative area of the in situ storage rack) does not exceed the nominal design thickness of the coupon (or storage rack) plus the 0.055- inch dimension assumed for the analyzed blister.

b. Neutron attenuation testing of a representative Boral coupon or in situ storage rack shall be performed prior to December 31, 2015, and thereafter 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 boron areal density will be that value assumed in the criticality safety analysis (0.013 gm/cm2).

c. Description of appropriate corrective actions for discovery of non-conforming Boral.

The Boral monitoring program proposed by the licensee is similar to other Boral programs approved for use by the NRC staff at other facilities. The staff finds that the proposed tests, frequency of testing, and acceptance criteria provide reasonable assurance that the licensee will be able to identify material property changes in the Boral before significant degradation occurs. The staff has reasonable assurance that the licensee will evaluate any coupons that do not meet the above acceptance criteria or any significant trends in coupon testing in accordance with the site's Corrective Action Program.

3.4 Conclusions

The NRC staff evaluated the submittal against the criteria for the SFP conditions. The NRC staff reviewed the analysis to assure that the assumptions and analytical technique used are adequately substantiated to conclude at a 95 percent probability, 95 percent confidence level, that the regulatory requirements will be met.

The licensee demonstrated in its original application and supplemental information that the methodologies used in the criticality analysis follow the guidelines set forth in DSS-ISG-2010-1 and the appropriate NUREGs. Therefore, the NRC staff finds reasonable assurance that the proposed TS changes are acceptable.

Based on its review of the enhancements to the licensee's Boral monitoring program, the NRC staff concludes that the physical condition of the Boral neutron absorber will be adequately monitored in the SFP. The staff also finds that the proposed Boral monitoring program, as

described by the licensee, provides reasonable assurance of detecting potential degradation of the Boral material that could impair its neutron absorption capability in the SFP racks.

Therefore, the NRC staff concludes that the proposed Boral monitoring program is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the use of facility components located within the restricted area as defined in 10 CFR Part 20 or changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding as published in the *Federal Register* on June 11, 2013 (78 FR 35063), and June 24, 2014 (79 FR 35805). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) 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.

7.0 REFERENCES

1. Letter from M. Schimmel, NSPM, to the U.S. NRC Document Control Desk (DCD), "License Amendment Request for Fuel Storage Change," dated October 30, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML123070544).

2. Letter from M. Schimmel, NSPM, to the U.S. NRC DCD, "License Amendment Request for Fuel Storage Changes, Supplement to Respond to NRC Staff Requests for Additional Information (TAC ME9893)," dated May 16, 2013 (ADAMS Accession No. ML13136A145).

3. Letter from M. Schimmel, NSPM, to the U.S. NRC DCD, "License Amendment Request for Fuel Storage Changes, Supplement to Respond to NRC Staff Requests for Additional Information (TAC ME9893)," dated June 7, 2013 (ADAMS Accession No. ML13158A269).

4. Letter from K. Fili, NSPM, to the U.S. NRC DCD, "License Amendment Request for Fuel Storage Changes, Supplement to Propose a Spent Fuel Pool Boral Monitoring Program Technical Specification (TAC ME9893)," dated March 13, 2014 (ADAMS Accession No. ML14072A390).

5. Letter from K. Fili, NSPM, to the U.S. NRC DCD, "License Amendment Request for Fuel Storage Changes, Supplement to Propose a Spent Fuel Pool Boral Monitoring Program Technical Specification (TAC ME9893)," dated May 30, 2014 (ADAMS Accession No. ML14150A271).

6. DSS-ISG-2010-1, Revision 0, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," dated September 2011 (ADAMS Accession No. ML110620086).

7. Code of Federal Regulations, Title 10, Part 50, Section 68, "Criticality Accident Requirements."

8. L.I. Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," dated August 1998.

Principal Contributors: D. Cunanan, NRR
K. Wood, NRR
M. Yoder, NRR

Date of issuance: October 24, 2014

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

Sincerely,

/RA/

Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 182 to DPR-22
2. Safety Evaluation

cc w/encls: Distribution via ListServ

DISTRIBUTION:
PUBLIC
LPL3-1 r/f
RidsNrrPMMonticello Resource
RidsNrrLAMHenderson Resource
RidsAcrsAcnw_MailCTR Resource
RidsNrrDorlLpl3-1 Resource
RidsRgn3MailCenter Resource
RidsNrrDorlDpr Resource
RidsNrrDeEsgb Resource
RidsNrrDssSrxb Resource
RidsNrrDssStsb Resource
MYoder, NRR
DCunanan, NRR
KWood, NRR

ADAMS Accession No. ML15160A207 (redacted)
ADAMS Accession No.: ML14197A020 (nonpublic)

* SE transmitted by memo dated June 30, 2014
** SE transmitted by memo dated October 3, 2014

OFFICE	LPL3-1/PM	LPL3-1/LA	ESGB/BC *	SRXB/BC **
NAME	TBeltz	MHenderson	GKulsea	UShoop
DATE	10/07/14	10/08/14	06/30/14	10/03/14
OFFICE	STSB/BC(A)	OGC - NLO	LPL3-1/BC	LPL3-1/PM
NAME	RElliott (MChernoff for)	STurk	DPelton	TBeltz
DATE	10/15/14	10/21/14	10/22/14	10/24/14

OFFICIAL RECORD COPY