ENCLOSURE 3 CONTAINS PROPRIETARY INFORMATION
WITHHOLD FROM PUBLIC DISCLOSURE IN ACCORDANCE WITH 10 CFR 2.390

Xcel Energy®

Monticello Nuclear Generating Plant
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Monticello, MN 55362

October 30, 2012

L-MT-12-076
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Monticello Nuclear Generating Plant
Docket 50-263
Renewed License No. DPR-22

Subject: License Amendment Request for Fuel Storage Changes

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, hereby requests an amendment to the renewed operating license for Monticello Nuclear Generating Plant (MNGP). Specifically, NSPM proposes to revise Technical Specification (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 the legacy fuel types in addition to the new AREVA ATRIUM™ 10XM fuel design.

The proposed TS changes will also correct a non-conservatism in the criticality analysis of the New Fuel Vault (NFV). This non-conservative TS has been addressed in the MNGP Corrective Action Program, and safety has been ensured through interim administrative restrictions that are more restrictive than the current TS. The proposed License Amendment Request (LAR) is being submitted to address the need for timely amendments to supersede the current non-conservative TS in accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety". The proposed TS changes would prohibit the use of the NFV.

Enclosure 1 to this letter provides the evaluation of the proposed TS changes and their supporting justifications, including a no significant hazards determination. Enclosure 2 provides the existing TS pages marked-up to show the proposed changes.

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A copy of AREVA Report ANP-3113(P), Revision 0, Monticello Nuclear Plant Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM™ 10XM Fuel, Revision 0 is provided in Enclosure 3. This report is referenced in the evaluation provided in Enclosure 1. ANP-3113(P), Revision 0 contains information that AREVA considers to be proprietary as defined by 10 CFR 2.390. AREVA, as the owner of the proprietary information, has executed an affidavit provided in Enclosure 4, which identifies that the enclosed proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. AREVA requests that the enclosed proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390. A non-proprietary version of ANP-3113(P), Revision 0 is provided in Enclosure 5.

NSPM has determined that the information for the proposed amendment does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment meets the categorical exclusion requirements of 10 CFR 51.22(c)(9) and an environmental impact assessment need not be prepared.

A copy of this submittal, including the Determination of No Significant Hazards Consideration, without Enclosures 2 through 5, is being forwarded to the designated State of Minnesota official pursuant to 10 CFR 50.91(b)(1).

NSPM requests approval of this proposed amendment by September 30, 2014. Whereas this license amendment request (LAR) covers a fuel assembly design that is not yet licensed for core power operation, it is submitted in advance of an LAR for the AREVA ATRIUM 10XM fuel transition because the lead-time for spent fuel criticality amendment reviews is expected to require two years. Once approved, the amendment will be implemented within 60 days.

If there are any questions or if additional information is needed, please contact Glenn Adams at 612-330-6777.

<u>Summary of Commitments</u>

This letter makes no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: October 30, 2012

Mark A. Schimmel
Site Vice-President
Monticello Nuclear Generating Plant
Northern States Power Company-Minnesota

Enclosures (5)

cc: Regional Administrator, Region III, USNRC
Project Manager, Monticello Nuclear Generating Plant, USNRC
Resident Inspector, Monticello Nuclear Generating Plant, USNRC
Minnesota Department of Commerce (w/o enclosures 2-5)

ENCLOSURE 1

Evaluation of the Proposed Change

License Amendment Request for Fuel Storage Changes

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, hereby requests an amendment to the renewed operating license for Monticello Nuclear Generating Plant (MNGP). Specifically, NSPM proposes to revise Technical Specification (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 the legacy fuel types in addition to the new AREVA ATRIUM 10XM fuel design.

2.0 DETAILED DESCRIPTION

The proposed changes to the TS are as follows:

2.1 Proposed Change to TS 4.3.1, "Fuel Storage Criticality"

The proposed change to TS Design Feature 4.3.1.1.a will revise the parameter and value associated with the nuclear fuel neutron multiplication factor (k) that correlates with the Spent Fuel Pool (SFP) storage rack k-effective (k-eff or k_{eff}) criterion of 0.95. The reactor parameter will be changed from an in-core k-infinity (k_{inf}) to an in-rack k-infinity parameter, and the value will be changed based on the proposed criticality safety analysis (CSA).

The proposed change to TS Design Feature 4.3.1.1.b will include the low-density storage rack in this TS to align its criticality criterion ($k_{eff} \leq 0.95$) with the regulation (10 CFR 50.68(b)) and the other storage racks.

The proposed change to TS Design Feature 4.3.1.1.c will delete the requirement in its entirety. The low-density rack will be included in the $k_{eff} \leq 0.95$ criterion described above (4.3.1.1.b).

The proposed change to TS Design Feature 4.3.1.1.d will involve an administrative change to renumber the requirement to 4.3.1.1.c. It will also involve a substantive change to remove reference to the "8 x 8 high density storage rack", which is being removed from TS and is not analyzed in the criticality safety analysis. Further, this proposed revision will change the value given for the minimum required gap between the high-density and low-density rack.

The proposed change to TS Design Feature 4.3.1.2 will eliminate from TS all criticality criteria for the New Fuel Vault (NFV) and replace them with one TS statement to prohibit the use of the NFV for fuel storage.

2.2 Proposed Change to TS 4.3.3, "Fuel Storage Capacity"

The proposed change to TS Design Feature 4.3.3 will revise the stated value of fuel storage capacity from 2301 fuel assemblies to 2217 fuel assemblies.

2.3 Other Proposed Changes to the Current Licensing Basis

In addition to the proposed TS changes discussed above, this LAR also proposes the following changes to the current licensing bases for which NRC approval is requested:

- The proposed amendment will change the evaluation methodology used for fuel storage criticality safety analysis to that described in Enclosure 3.

3.0 TECHNICAL EVALUATION

3.1 Design Description

Design information applicable to the proposed amendment includes description of the MNGP fuel handling and storage facilities such as the New Fuel Vault (NFV), Spent Fuel Pool (SFP) and the handling apparatus used to place fuel in those facilities. Also applicable to the proposed amendment is a description of the nuclear fuel designs that are placed in these facilities. These designs include the legacy nuclear fuel assemblies as well as the design of the new AREVA ATRIUM 10XM fuel assembly. These descriptions are limited to the extent necessary to support the proposed fuel storage amendment. The proposed amendment does not seek approval of core operation with AREVA ATRIUM 10XM fuel.

Currently, MNGP provides storage facilities for new unirradiated fuel assemblies and wet storage of irradiated / spent fuel assemblies. The current safety function of the New Fuel Vault (NFV) is to maintain the new fuel assemblies in a safe and subcritical array during all postulated storage conditions, including "optimum moderation". The safety function of the spent fuel pool (SFP) and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions.

<u>NFV Design and Operation - Current</u>

The NFV is a reinforced-concrete Class I structure, accessible only through top hatches. Racks in the vault can hold a maximum of 150 fuel bundles in an upright position. There is an open drain in the floor of the vault.

As described in the USAR (Section 10.2.1), and as subject to the NSPM corrective action program, the NFV does not currently meet the subcriticality criteria prescribed in TS 4.3.1.2.d with respect to "optimum moderator conditions."

Thereby, the TS were determined to be non-conservative in the Corrective Action Program and interim administrative restrictions were established to prohibit loading new fuel in the NFV. These administrative restrictions are manifest in the fuel loading procedures discussed below.

The current designs of the NFV and associated handling areas (Reactor Refueling Floor – 1027' Elevation) provide no permanent physical barriers specifically designed to prevent placement of a new fuel assembly in those associated handling areas. Safe handling and placement of new fuel assemblies is controlled by qualified operators and fuel handling procedures that specify safe locations for new fuel and safe travel load paths between those locations.

Loading the NFV is described in USAR Section 10.2.1 to include the following basic steps: (1) movement of new fuel bundles in shipping containers to the refueling floor, (2) removal of the bundles from these shipping containers and placement in qualified facilities, and (3) inspection of new fuel and installation of fuel channels. Plant procedures for handling new fuel provide more specificity and make no provision to place new fuel in the NFV, but rather, procedures specify that new fuel be placed in the SFP after inspection and channel installation is completed.

NFV Design and Operation - Proposed

The proposed amendments involve no physical modifications to the NFV because operator qualifications and procedural controls will continue to be adequate means to preclude the placement of new fuel in a potentially critical array. This position is supported by the following:

1. The general areas of the Reactor Refueling Floor do not provide a stable landing for a fuel assembly (i.e., a centering dimple and lateral supports) and such uncontrolled placement is not authorized by procedure.
2. The NFV would provide a stable landing for a new fuel assembly, but access to a cell is impeded by the hatch cover, and access is not authorized by procedure.
3. Procedures will continue to specify the safe locations for new fuel after removal from shipping containers, and the SFP will continue to be the location for new fuel assemblies following inspection and channel installation.
4. Qualified operators are required to follow procedures such that placement in an unauthorized location would constitute a violation of procedures and would constitute an incident.
5. The Double Contingency Principle, which has been a regulatory basis for nuclear fuel storage criticality analyses, states that two unlikely independent and concurrent incidents or postulated accidents are beyond the scope and need not be analyzed (Reference 6.3).

Thus, it would take multiple procedural violations to place new fuel assemblies in sufficient number and in such close proximity so as to create a potentially critical configuration. Therefore, consistent with the current licensing basis and the Double Contingency Principle, no design changes are proposed to install permanent and immovable blocks to plant facilities where new fuel assemblies might possibly be unloaded.

SFP Design and Operation - Current

The spent fuel storage pool has been designed to withstand earthquake loadings as a Class I structure. It is a reinforced concrete structure, completely lined with seam-welded, stainless steel plates welded to reinforcing members embedded in concrete.

The spent fuel pool has a design capacity of 2217 fuel assemblies consisting of 13 High Density Fuel Storage System (HDFSS) modules and one 2x10 low-density rack. The HDFSS modules are composed of rectangular fuel storage tubes that are arranged in a 13x13 array; each tube fabricated by forming an inner and outer sheet of stainless steel sandwiching a core of borated aluminum (Boral). Boral is a neutron absorber that helps maintain the high-density fuel array in a subcritical condition. The low-density storage rack design is aluminum construction with a cell pitch sufficient to maintain fuel subcriticality without the need for any neutron absorber material.

Each fuel tube in the HDFSS is vented top and bottom to provide a positive flow path for any gases formed by the interaction of rack materials (primarily Boral) in the water environment. The module design, materials, and fabrication are in accordance with the requirements of ASME Code, Section III. These free-standing modules meet the seismic requirements for Class I equipment. Analysis has confirmed that the frictional forces between the module support and the pool floor liner, and the low seismic overturning moment of the modules, make them stable under all conditions of storage. Calculated horizontal displacement of the modules during an earthquake is less than the nominal spacing, assuring no interaction between modules resulting from a safe shutdown earthquake.

The fuel pool structure is capable of supporting the HDFSS racks plus the loading associated with one control blade rack and two 2x10 low-density fuel racks (one of which was never installed), for a total of 2237 fuel assemblies. Although the spent fuel storage pool has been structurally analyzed for 2237 fuel assemblies, the current installed fuel storage rack configuration provides for only 2217 storage slots. Thus, the current configuration is bounded by the current structural analysis.

In 2007, NRC approved MNGP license amendment 150 to revise TS 4.3.1.1.d and 4.3.3 to allow storage of an additional 64 fuel assemblies associated with an 8 x 8 high density storage rack. This amendment was a contingency measure taken prior to the Independent Spent Fuel Storage Installation (ISFSI) becoming operational; however, this rack was never installed. This amendment raised the total SFP fuel storage capacity to 2301 fuel assemblies.

The MNGP spent fuel storage rack designs offer sufficient spacing and neutron poison between assemblies so that complex loading patterns are not required. MNGP TS only require that fuel assemblies satisfy a certain in-core k-infinity criterion prior to placement in the SFP. Once qualified, the fuel assembly may be placed in any approved storage location. There are no special loading patterns, no minimum burnup requirements, no neutron absorber inserts, and no requirements to leave empty storage cells between assemblies. Thus, any fuel assembly that qualifies for SFP storage may be placed in any approved storage cell, such that a criticality event caused by a fuel assembly misplacement is not a credible accident at MNGP.

The spent fuel storage racks are designed and installed so that it is impossible to insert assemblies between high-density rack modules; however, there is space between the low-density rack and high-density racks as well as space between the SFP walls and peripheral storage racks for a fuel assembly. Nevertheless, placement of a fuel assembly outside of an approved storage location would be a procedural violation. Accordingly, misplacement of a single fuel assembly is considered an accident condition worthy of analysis, but multiple misplacements exceed the Double Contingency Principle and are not analyzed. As discussed previously, the Double Contingency Principle states that two unlikely independent and concurrent incidents or postulated accidents are beyond the scope and need not be analyzed (Reference 6.3).

The neutron absorber capability of the installed Boral is stated in terms of the boron areal density, and is ensured through compliance with the MNGP Aging Management Program (AMP), described later.

Fuel stored in the spent fuel storage pool is covered with sufficient water for radiation shielding. There is sufficient water depth above the fuel to provide decontamination of releases from damaged fuel resulting from a fuel handling accident.

A refueling platform, equipped with a refueling grapple and two 1/2-ton auxiliary hoists is provided.

Further discussion of the NFV, SFP, and handling system design is provided in USAR Section 10.2.1 and Enclosure 3.

<u>SFP Design and Operation - Proposed</u>

The proposed amendments involve no physical modifications to the SFP, storage racks, or to any other system, structure, or component. However, in the proposed amendment, the MNGP design and licensing basis will recognize that un-installed storage racks will not be included. These un-installed racks are one 8x8 high-density storage rack and one 2x10 low-density rack.

The proposed CSA puts no additional burden on the SFP storage racks, nor does it take any additional credit for the neutron absorbing capability of the installed Boral. The proposed CSA assumes the same value of boron areal density as that assumed in the analysis of record. Therefore, the Boral Aging Management Program is not affected by the proposed amendment.

With respect to implementation, the proposed amendment will not significantly affect how fuel is handled in the SFP, nor will it affect how fuel assemblies are qualified. The existing spent fuel inventory will not have to be relocated in order to implement the proposed TS. Furthermore, the new ATRIUM 10XM fuel type will not have any loading restrictions different from those of the legacy fuel types.

Fuel Design

To support future operations, the ATRIUM 10XM fuel assembly is designed to be compatible with the MNGP reactor core and co-resident legacy fuel. Thereby, the ATRIUM 10XM fuel assembly is constructed of similar materials within a spatial envelope that is similar to the currently-licensed fuel design (GE14) and other legacy fuel types.

NSPM's 10 CFR 50 Appendix B Design Control Program requirements ensure that proper review and evaluation is performed on the effects of the change in fuel type on fuel storage facilities. For example, the following implementation activities have been identified in the course of the design control program's plant impact review:

- Fuel dimensional compatibility calculation. This calculation demonstrates the dimensional compatibility of the ATRIUM 10XM design to the co-resident GE14 assembly, including parameters such as the outside envelope clearance for spent fuel pool rack inner dimension, and clearance of the assembly handling bail to the MNGP fuel handling grapple (J-hook) dimensions.
- Fuel dimensional compatibility calculation. Of particular importance to the CSA, this calculation shows that the ATRIUM 10XM design provides an elevation for the start of the fuel column (active fuel region) that is within 0.1 inch of the current fuel type and therefore compatible with the storage rack designs.
- Performance of an in situ fit-up functional test to demonstrate that an ATRIUM 10XM upper tie plate mates acceptably with the MNGP fuel handling grapples.
- Material compatibility review to confirm that the exposed materials of the ATRIUM 10XM are compatible with the spent fuel pool material and chemistry requirements.
- Direct measurement of the physical gap between the low-density storage rack and the closest high-density storage rack to confirm it is greater than the 12-inch gap prescribed in the proposed TS.

Fuel design parameters important to the spent fuel criticality safety analysis are described further in Enclosure 3.

3.2 Current Licensing Basis

Herein, the term "k" (or the neutron multiplication factor) is used to describe the ratio of the neutrons produced to those lost. A ratio of 1.0 indicates that the system is perfectly balanced with the same number of neutrons being produced to those lost (through absorption or other means such as leakage outside the system). This is known as a "critical" or self-sustaining system.

The term k-infinity (k_{inf}) is used to describe the calculated ratio for an infinite multiplying medium, i.e., the lattice (or lattices) repeat infinitely in all directions. Because the assumed medium is infinite, no neutrons can be lost to leakage because there are no outer boundaries to allow leakage. On the other hand, a k-effective (k_{eff}) term is used to describe a finite system in which there are clear outer boundaries from which neutrons can leak.

The criticality analysis provided in Enclosure 3 necessarily contains a potentially confusing combination of k_{inf} and k_{eff} values. The individual lattices are evaluated as an infinite medium, and hence the results are represented as k_{inf} values. The racks are evaluated in a combination of infinite and finite geometries; therefore, they can be reported differently depending on the specific model. For example, a finite rack was used for rack interface calculations, and results are reported in terms of k_{eff}.

In general, a k_{inf} result will bound the actual configuration because the real system is finite and therefore will lose neutrons to leakage. For example, Enclosure 3 Table 6.1 demonstrates that the infinite boundary representation (k_{inf} result) of the 13x13 storage rack is more reactive than the more realistic representation (k_{eff} result) that accounts for leakage above and below the storage rack arrays. Consequently, it is conservative to use a k_{inf} result in the determination of a 95/95 k_{eff}.

At a regulatory level, 10 CFR 50.68(a) requires licensees to select one of two options to satisfy criticality accident requirements: (1) 10 CFR 70.24, or (2) 10 CFR 50.68(b). Historically and currently, NSPM has chosen to adopt 10 CFR 50.68(b), Criticality Accident Requirements for MNGP. These criteria are represented in current TS 4.3.1.1 and TS 4.3.1.2.

USAR Section 10.2.1.1 and TS 4.3.1 describe the design bases for the fuel storage systems. Design bases applicable to the current criticality safety analysis (CSA) are summarized below.

The basic design criterion associated with the storage of new fuel and irradiated (spent) fuel is that the effective multiplication factor (k-effective) of fuel stored under normal conditions will be less than 1.0 with sufficient regulatory margin as listed below:

k-eff	New Fuel Vault conditions:
< 0.90	If dry
< 0.95	If flooded with unborated water
≤ 0.98	Under optimum moderated conditions. As noted in USAR 10.2.1.2, a review of optimum moderated conditions concluded that new fuel may not meet this value of k-effective and administrative controls were established to prohibit loading the NFV due to this criterion.

k-eff	SFP Low-Density Rack conditions:
≤ 0.90	If flooded with unborated water. Note that MNGP complies with 10 CFR 50.68(b)(4), which only requires a k-effective ≤ 0.95 for the spent fuel storage racks. Heretofore, as a matter of convenience, the value of 0.90 has been retained for the low-density rack because criticality safety analyses continued to support that result (≤ 0.90) and there was no compelling interest to relax the acceptance criterion.

k-eff	SFP High-Density Rack conditions:
≤ 0.95	If flooded with unborated water

Fundamental to the underlying criticality analyses is maintaining the allowances for uncertainties that ensure the statistical confidence level prescribed by the regulation (10 CFR 50.68), which is 95 percent probability at a 95 percent confidence level. These allowances for uncertainties are described in USAR Section 10.2.1.

Permanent design features of the storage racks such as cell-to-cell pitch help ensure the subcriticality criteria are met as long as fuel assemblies meet k-infinity maximum reactivity requirements (described below). Other important storage rack design features are the integrity of the installed neutron poison (Boral) and the physical gap between the low-density and high-density racks. Boral integrity is monitored in accordance with the MNGP AMP described and approved by NUREG-1865, dated October 2006 (Reference 6.6). That NUREG states that the program will manage the aging effects caused by corrosion, cracking, erosion, fouling, fretting, or thermal exposure. NRC Staff found the AMP acceptable because it conforms to the recommended Generic Aging Lessons Learned (GALL) Report AMP.

Integral with the permanent design features are the limits on the nuclear fuel's maximum infinite neutron multiplication factor (k-infinity), which are described in TS 4.3.1 as follows:

- Spent fuel storage. Fuel assemblies in the normal in-core configuration shall have a maximum k-infinity of 1.33.

- New fuel storage. Fuel assemblies in the normal in-core configuration shall have a maximum k-infinity of 1.31.

These k-infinity values represent the only TS limitations on fuel assembly design with respect to criticality safety, and once satisfied, allow for fuel placement in the respective rack irrespective of initial enrichment or burnup. Once these k-infinity values are satisfied and confirmed through the core design process by nuclear analysis engineers, the other spent fuel pool design features (rack geometry, coolant temperature, neutron poison areal density) will ensure the current k-effective criteria are met.

The current licensing basis employs no special spent fuel storage rack loading restrictions that limit the placement of fuel based on its design or operational history (or that of its neighboring fuel).

The design bases of the NFV, SFP, and fuel handling systems are further described in the USAR Section 10.2.1.

3.3 Justification for the Proposed Changes

3.3.1 Justification for Technical Specification Changes

The proposed change to TS Design Feature 4.3.1.1.a will revise the parameter and value associated with the nuclear fuel neutron multiplication factor (k) that correlates with the Spent Fuel Pool (SFP) storage rack k-effective (k_{eff}) criterion of 0.95. The parameter will be changed from an in-core k-infinity (k_{inf}) to an in-rack k-infinity parameter to be more representative of the actual storage configuration, and the value will be changed based on the proposed criticality safety analysis (CSA). Revising this parameter and the value will not change the fundamental requirement to meet the k-effective value that maintains a subcritical condition in the storage racks with margins prescribed by the respective TS and regulation (10 CFR 50.68(b)).

When looking at the differences between the current TS in-core k_{inf} limits (1.31 and 1.33) and the proposed TS in-rack value (0.8825), it is important to realize that the in-core value is defined as an uncontrolled in-core k-infinity. That current value is extraordinarily high because it is assuming an infinite array of lattices at their peak reactivity with no control blades present. Conversely, the in-rack values provided for the limiting rack include the neutron absorber in the Boral racks. The pitch and moderator conditions have impact as well, but the primary difference is due to the impact of the neutron poison.

The actual value of interest for storage in the SFP is the reactivity in the in-rack geometry. While the in-core k-infinity can provide an estimate of this reactivity, it may not be as representative because the bias between the core and rack conditions is not always constant. For example, the top and bottom lattice geometries have different biases due to differences in the fuel/moderator ratios.

The proposed change to TS Design Feature 4.3.1.1.b will include the low-density storage rack in this TS to align its criticality criterion ($k_{eff} \leq 0.95$) with the regulation (10 CFR 50.68(b)) and the other storage racks. This legacy criterion ($k_{eff} \leq 0.90$) had been preserved for the low-density rack only for convenience. During the licensing of the high-density storage rack (circa 1977), there was no impetus to reanalyze the low-density rack considering that the legacy criterion (≤ 0.90) was more conservative than the revised criterion (≤ 0.95).

The proposed change to TS Design Feature 4.3.1.1.c will delete the requirement in its entirety. The low-density rack will be included in the $k_{eff} \leq 0.95$ criterion described above (4.3.1.1.b).

The proposed change to TS Design Feature 4.3.1.1.d will involve an administrative change to renumber the requirement to 4.3.1.1.c. It will also involve a substantive change to remove reference to the never-installed "8 x 8 high density storage rack", which is being removed from TS and is not analyzed in the updated criticality safety analysis. Further, this proposed revision will change the value given for the minimum required gap between the high-density and low-density rack. The new value of 12 inches provides sufficient distance between racks so as to neutronically decouple them as described in the criticality safety analysis.

The proposed change to TS Design Feature 4.3.1.2 will eliminate from TS all criticality criteria for the New Fuel Vault (NFV) and replace them with one TS statement to prohibit the use of the NFV for fuel storage. This prohibition is appropriate because the CSA does not support new fuel in the NFV.

The proposed change to TS Design Feature 4.3.3 will revise the stated value of fuel storage capacity from 2301 fuel assemblies to 2217 fuel assemblies. This change reconciles the value by eliminating two permanent storage facilities: (1) the 64 storage locations associated with the 8x8 fuel high-density storage rack that was never installed, and (2) the 20 storage locations associated with a 2x10 low-density storage rack that was never installed. This amendment is justified because the reduction in storage capacity represents a net reduction in physical challenges to the SFP structures and its auxiliary systems (e.g., cooling) as discussed in section 3.4 below.

3.3.2 Justification for Criticality Safety Analysis Methods

The analysis methods are described in the CSA (Enclosures 3 and 5) and justified based on the following:

- key analytical codes used in the analysis are topically approved (as described in Section 5 of the CSA),
- inputs are conservatively selected and applied (as described in Section 2 of the CSA),
- biases and uncertainties are conservatively applied (as described in Section 7 of the CSA), and
- results satisfy the regulatory criteria of 10 CFR 50.68(b) (as described in Section 2 of the CSA).

Furthermore, as discussed in Section 3 of the CSA (Enclosures 3 and 5), the analysis was performed to meet the regulatory expectations established in NRC Interim Staff Guidance DSS-ISG-2010-01 (Reference 6.1). Therein, each element of the Draft ISG is addressed.

3.4 Associated Evaluations

Insofar as the proposed TS changes relate predominantly to fuel storage criticality, most of the changes are justified and described in the enclosed CSA (Enclosures 3 and 5), as discussed above. Other elements of the transition to ATRIUM 10XM fuel are being reviewed and justified in accordance with the NSPM 10 CFR 50 Appendix B Design Control Program and 10 CFR 50.59, and are not included herein. Transition to ATRIUM 10XM fuel for core operation is not within the scope of this proposed amendment and is not evaluated herein.

Some elements of the TS changes related to spent fuel storage capacity are reviewed and justified in the sections below to the extent necessary to justify the change to TS 4.3.3 (reduction in storage capacity) and to support the No Significant Hazards Consideration (NSHC) which is presented in Section 4.3 below.

3.4.1 Spent Fuel Heat Removal - Bulk

With respect to the bulk heat load of the spent fuel pool, the proposed amendment introduces two changes of potential significance: (1) a reduced SFP storage capacity, and (2) new fuel type. NSPM has evaluated these effects and summarizes them below, concluding that there is no negative effect on the spent fuel pool cooling performance.

The reduction in the spent fuel inventory (by 84 assemblies) would inherently serve to reduce the bulk heat load on the SFP for the design basis cases that assume all racks are filled.

The decay heat generated by a fuel assembly following shutdown is primarily dependent on its operating power history prior to shutdown. AREVA experience and analysis has concluded that decay heat generated by different fuel types is very similar when the fuel types are operated under the same conditions (i.e., reactor power and exposure). Based on this experience, the decay heat generated by ATRIUM 10XM fuel will not be significantly different than the decay heat generated by the legacy fuel designs that are the basis for MNGP analyses. Furthermore, preliminary core design evaluation with ATRIUM 10XM fuel has shown that there is nothing significantly different in the fuel design to warrant consideration of a reload batch size that is greater than that currently considered. Further, even if the discharge batch size were increased by the transition to the ATRIUM 10XM fuel type, the MNGP SFP cooling system licensing basis would accommodate such a change because it allows for a reload-specific calculation of SFP heat load to ensure SFP temperature criteria are met (Reference USAR 10.2.2.3).

3.4.2 Spent Fuel Heat Removal - Local

With respect to the capabilities of the SFP storage rack system and the resident fuel assembly to remove its own heat load, the proposed amendment introduces only one change of significance: a new fuel type. Evaluation has shown no negative effect associated with this fuel change.

Fuel assemblies stored in the SFP are covered by water during all storage conditions. During normal design basis conditions, the bulk water temperature is maintained below the licensing limit of 140°F. Even during accident conditions that may involve bulk boiling at 212°F, the height of water above the SFP racks will help ensure that the state of water entering the active fuel region from the spent fuel pooling cooling system will be subcooled liquid. Thus, under all design basis conditions, the decay heat generated by a fuel assembly is transferred to the subcooled water and a natural convection flow is promoted as the less-dense hot water rises out the top of the storage cell and is replaced by cooler water in the bottom of the assembly. Inherently, as the assumed heat load of an assembly increases, so does the convective heat flow within the flow channel. Also, for any given heat load, fuel assemblies of comparable design (heat transfer area and flow resistance) will develop comparable natural convection flow rates.

ATRIUM 10XM and GE14 fuel assemblies (the current operating fuel) are thermal-hydraulically compatible during reactor operating conditions. When operating at the same core operating conditions, an ATRIUM 10XM assembly will experience slightly more flow than a co-resident GE14 assembly, mostly due to its reduced flow resistance. Based on the geometric similarities and the similar hydraulic performance of ATRIUM 10XM and GE14 assemblies, it is a reasonable judgment that the ATRIUM 10XM fuel assembly will develop natural convection flow similar to or greater than the GE14 assembly in the spent fuel pool racks. Because the flow, the decay heat, and the surface area for heat transfer from the fuel rods are similar for GE14 and ATRIUM 10XM fuel, ATRIUM 10XM fuel assemblies will have adequate cooling when loaded in the SFP.

3.4.3 Spent Fuel Pool Storage Rack Seismic Response

NSPM has reviewed the SFP storage rack seismic analyses and determined that the small difference in fuel assembly mass and stiffness that may be associated with introduction of the ATRIUM 10XM fuel assembly will have negligible effect on the results of those seismic analyses.

Although the ATRIUM 10XM fuel channel design is stiffer than the legacy designs, the difference in stiffness does not appear as a parameter in the rack seismic analyses. For rack sliding, overturning, and for a dropped fuel assembly, the fuel is characterized by mass. In the case of the rack structural analysis, the fuel is also considered an added mass to the structure. Because of the variety of possible configurations (e.g., partially filled racks, with or without fuel channel), differences in the channeled fuel assembly stiffness are not significant to the overall results.

Based on the review, the introduction of the ATRIUM-10XM fuel does not significantly change the dynamic response in the storage racks for the postulated seismic loads. In addition, the fuel is able to withstand the postulated loads in the racks. Therefore, storage racks are qualified with regard to seismic loads for the ATRIUM-10XM fuel design, and the post-earthquake displacements are not expected to increase.

As stated in the USAR, the high-density storage racks are freestanding, and their seismic horizontal displacement is calculated to be less than the nominal spacing between racks (1-7/8 inches). With respect to criticality, the updated criticality analysis takes no credit for gaps between high-density rack modules, so any seismically-induced movement between high-density racks that puts them in closer proximity is bounded by the analysis. Section 7.6 of Enclosure 3 provides further discussion of the seismic evaluation.

Also evaluated was the effect of an earthquake on the 12-inch gap that is assumed in the criticality analysis to separate the low-density rack and the nearest high-density rack. The low-density rack is bolted to the SFP floor, so it is not postulated to shift. Thus, the only possible displacement (1-7/8 inches as discussed above) would be due to the high-density rack. The existing gap between the fixed-location low-density rack and the closest high-density rack is sufficiently large so as not to infringe on the 12-inch requirement of the CSA, even after applying the maximum calculated displacement in the direction of the low-density rack.

3.4.4 Fuel Handling Reliability and the Fuel Handling Accident

To ensure continued reliability of the MNGP fuel handling systems with the ATRIUM 10XM fuel design and acceptable consequences for a postulated drop of this fuel assembly design, NSPM has evaluated the effects of this design including: (1) dimensional compatibility of the ATRIUM 10XM fuel design with existing lifting systems to demonstrate fuel handling system reliability, (2) calculated quantity of fuel rod damage following the postulated drop of an ATRIUM 10XM fuel assembly, and (3) calculated radiological source term of ATRIUM 10XM fuel rod gases.

AREVA has performed a dimensional compatibility evaluation that concludes the following:

- ATRIUM 10XM maximum outer dimensions are compatible with the minimum inner dimensions of the SFP storage racks.
- ATRIUM 10XM lifting bail is compatible to the lifting system grapples and provides an acceptable margin of safety for live loads and dead loads associated with fuel handling operations.

Further pre-operational fitup tests will ensure the compatibility of fuel handling systems to reliably handle the ATRIUM 10XM fuel assemblies without any increased risk of a drop accident.

Notwithstanding the expected reliability of ATRIUM 10XM fuel handling, AREVA calculated the extent of fuel rod damage that could mechanistically occur during the postulated fuel assembly drop accident at MNGP. In that analysis, the ATRIUM 10XM assembly is found to be 12 pounds heavier than the MNGP GE14 assembly, but actually lighter than the GE14 assembly that is assumed in the generic General Electric drop analysis-of-record. The AREVA analysis estimates 162 fuel rod failures in the ATRIUM 10XM, which is fewer failures than that calculated for the comparable 10x10 fuel assembly described in the MNGP USAR. Thus, the mechanistic failure of ATRIUM 10XM fuel rods is shown to be bounded by that calculated in the analysis of record.

After having shown that the analysis-of-record bounds the ATRIUM 10XM design with respect to fuel rod damage, the last step is to show that the radiological source term of the ATRIUM 10XM fuel design is less than or not significantly greater than that of the analysis-of-record. To this end, AREVA calculated an ATRIUM 10XM source term submersion and inhalation dose terms that were comparable to that of the analysis-of-record, with an overall conclusion was that the overall accident dose from a

Fuel Handling Accident (FHA) would be lower for the ATRIUM 10XM fuel assembly drop.

3.5 Conclusion

The proposed amendments are adequately justified and the criticality safety analysis adequately supports the storage configurations including consideration of the ATRIUM 10XM fuel design. The analysis concludes that subcriticality licensing criteria are met with due consideration of the regulatory expectations of NRC Interim Staff Guidance.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The NRC's acceptance criteria for spent fuel storage systems are based on 10 CFR 50 Appendix A GDC-62, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations. Regulations also specify design criteria for fuel storage systems in 10 CFR 50.68, which have been described in Section 3.2 of this enclosure.

The applicable Monticello principal design criteria predate the general design criteria listed in 10 CFR 50, Appendix A. The Monticello principal design criteria are listed in USAR Section 1.2, "Principal Design Criteria." In 1967, the Atomic Energy Commission (AEC) published for public comment a revised set of proposed General Design Criteria (Federal Register 32FR10213, July 11, 1967). Although not explicitly licensed to the AEC proposed General Design Criteria published in 1967, Northern States Power Company (NSP), the predecessor to NSPM, performed a comparative evaluation of the design basis of the Monticello, Unit 1, with the AEC proposed General Design Criteria of 1967. The Monticello USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria," contains this comparative evaluation. USAR Appendix E provides a comparative evaluation with each of the groups of criteria sent out in the July 1967 AEC release.

While Monticello is not generally licensed to the 10 CFR 50 Appendix B GDC or the 1967 AEC proposed General Design Criteria, a comparison of the current GDC to the applicable AEC proposed General Design Criteria can be made. For the 10 CFR 50 Appendix B GDC-62, the Monticello comparative evaluation of the comparable 1967 AEC proposed GDC (referred to as "draft GDC") is contained in Monticello USAR Appendix E, as follows:

> Criterion 66, Prevention of Fuel Storage Criticality (Category B) - Criticality in new and spent storage shall be prevented by physical

systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality and to provide sufficient cooling for spent fuel. The new fuel storage vault racks (located inside the secondary containment reactor building) are top entry, and are designed to prevent an accidental critical array, even in the event the vault becomes flooded. Vault drainage is provided to prevent possible water collection. The handling and storage of spent fuel, which takes place entirely within the reactor building (which provides containment), is done in the spent fuel storage pool. The pool has provisions to maintain water clarity, temperature control, and instrumentation to monitor water level. Water depth in the pool will be such as to provide sufficient shielding for normal reactor building occupancy by operating personnel. The storage racks in which spent fuel assemblies are placed are designed and arranged to ensure subcriticality in the storage pool. The spent fuel pool cooling and demineralizer system is designed to maintain the pool water temperature (decay heat removal) to control water clarity (safe fuel movement), and to reduce water radioactivity.

4.2 Precedent

In August 2010, NRC issued Interim Staff Guidance DSS-ISG-2010-01, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools (Reference 6.1) to rebaseline NRC's expectations for spent fuel criticality analysis. That guidance was intended to reiterate existing guidance, clarify ambiguity in existing guidance, and identify lessons learned based on recent submittals. Further, Section 5 (entitled Miscellaneous) of the ISG included a specific caution for applying precedent. The expectations of the ISG were further reinforced in subsequent NRC Information Notice 2011-03 (Reference 6.2).

Based on the new NRC baseline guidance and the caution for use of precedence on this topic, little precedent is applicable to this LAR. Thus, only two precedents were identified as applicable. These are the only precedents representing comparable changes and comparable methods associated with boiling water reactor (BWR) fuel designs:

- In 2007, Brunswick Steam Electric Plant Unit 1 and 2 amendments (Reference 6.5) were approved for the AREVA ATRIUM-10 fuel design description and criticality methods. The amendments permitted the storage of ATRIUM-10 fuel in the SFP and new fuel storage racks. This precedent is applicable because it used AREVA criticality methods from which the MNGP CSA is derived, and because it addressed a comparable 10x10 ATRIUM-10 fuel design. The precedent differs from the NSPM amendment request in two significant respects: (1) NSPM does not propose to change the description of fuel in the reactor core (TS 4.2.1) because this amendment specifically excludes any evaluation of fuel for core operation, and (2) the proposed changes do not request complete elimination of the in-core k-infinity criteria (as was approved for Brunswick), but rather, the proposed change involves replacing the in-core k-infinity criteria with a single in-rack k-infinity criterion.

- In 2011, LaSalle County Station (LSCS) Unit 1 and 2 amendments (Reference 6.7) approved the AREVA criticality analysis methods and revised technical specifications to include the use of neutron absorber inserts in the spent fuel storage racks. The MNGP proposed amendments differ from these LSCS amendments in that MNGP involves no Boraflex and no rack inserts. However, as discussed in Enclosure 3 (Table 3.1) to this letter, the approach taken in the MNGP spent fuel pool criticality safety analysis is similar to the LaSalle analysis. Some changes were incorporated in the MNGP analysis to directly address NRC concerns identified in the NRC safety evaluation (Reference 6.7) and to provide closer compliance to the Interim Staff Guidance (Reference 6.1).

4.3 Significant Hazards Consideration

Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, hereby requests an amendment to the renewed operating license for Monticello Nuclear Generating Plant (MNGP). Specifically, NSPM proposes to revise Technical Specification (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 the legacy fuel types in addition to the new AREVA ATRIUM 10XM fuel design.

NSPM has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) as discussed below:

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

   Response: No

   The proposed amendment does not change the fuel handling processes, fuel storage racks, decay heat generation rate, or the SFP cooling and cleanup system. The proposed amendment was evaluated for impact on the following previously-evaluated events and accidents: (1) fuel handling accident (FHA), (2) fuel assembly misloading, (3) seismically-induced movement of spent fuel storage racks, and (4) loss of spent fuel pool cooling.

   Whereas fuel handing procedures will not be changed materially for the new fuel type or the revised criticality methods, the probability of a FHA is not increased because the implementation of the proposed amendment will employ the same equipment and procedures to handle fuel assemblies that are currently used. Therefore, the proposed amendment does not increase the probability for occurrence of a FHA. In that the proposed amendment does not increase the mechanistic damage to a fuel assembly or the radiological source term of any fuel assembly, the amendment would not increase the radiological consequences of a FHA. With regard to the potential criticality consequences of a dropped assembly coming to rest adjacent to a storage rack or on top of a storage rack, the results are bounded by the current analysis involving a potential missing neutron poison plate in the storage rack. The fuel configuration caused by a dropped assembly resting on top of loaded storage racks is inherently bounded by the assembly misloaded in the storage rack because the misloaded assembly is in closer proximity to other assemblies along its entire fuel length.

Operation in accordance with the proposed amendment will not change the probability of a fuel assembly misloading because fuel movement will continue to be controlled by approved fuel selection and fuel handling procedures. The consequences of a fuel misloading event (fuel assembly loaded into an unapproved location) are not changed because the reactivity analysis demonstrates that the same subcriticality criteria and requirements continue to be met for the worst-case fuel misloading event.

Operation in accordance with the proposed amendment will not change the probability of occurrence of a seismic event, which is considered an Act of God. Also, the consequences of a seismic event are not changed because the proposed amendment involves no significant change to the types of material stored in SFP storage racks or their mass. In this manner, the forcing functions for seismic excitation and the resulting forces are not changed. Also, particular to criticality, the supporting criticality analysis takes no credit for gaps between high-density rack modules so any seismically-induced movement between high-density racks that puts them in closer proximity would not result in an unanalyzed condition with consequences worse than those analyzed. Also, the small displacement of the high-density rack closest to the fixed location of the low-density rack will not put those racks in a closer proximity than that analyzed. In summary, the proposed amendment will not increase the probability or consequence of a seismic event.

Operation in accordance with the proposed amendment will not change the probability of a loss of spent fuel pool cooling because the changes in fuel criticality limits and introduction of the ATRIUM 10XM fuel design have no bearing on the systems, structures, and components involved in initiating such an event. The proposed amendment does not change the heat load imposed by spent fuel assemblies nor does it change the flow paths in the spent fuel pool. Therefore, the accident consequences are not increased for the proposed amendment.

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

2. **Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No

   The proposed amendments involve no new SFP loading configurations for current and legacy fuel designs of the nuclear plant. The proposed amendments do not change or modify the fuel handling processes, fuel storage racks, decay heat generation rate, or the spent fuel pool cooling and cleanup system. Further, the new fuel type does not introduce any incompatible materials to the spent fuel pool environment.

   As such, the proposed changes introduce no new material interactions, man-machine interfaces, or processes that could create the potential for an accident of a new or different type.

   Operation with the proposed amendment will not create a new or different kind of accident because fuel movement will continue to be controlled by approved fuel handling procedures. There are no changes in the criteria or design requirements pertaining to fuel storage safety, including subcriticality requirements, and analyses demonstrate that the proposed storage arrays meet these requirements and criteria with adequate margins. Thus, the proposed storage arrays cannot cause a new or different kind of accident.

3. **Do the proposed changes involve a significant reduction in a margin of safety?**

   Response: No

   The proposed amendment was evaluated for its effect on current margins of safety for criticality. Although the amendment involves changing the subcriticality acceptance limit for the low-density storage rack from a value of 0.90 to 0.95, the margin of safety for subcriticality is not significantly reduced in that the limit is consistent with that of the other storage racks and the regulation described by 10 CFR 50.68 (b)(4). The new criticality analysis confirms that operation in accordance with the proposed amendment continues to meet the required subcriticality margin. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Therefore, based on the above, NSPM has concluded that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment of an operating license for a facility requires no environmental assessment if the operation of the facility in accordance with the proposed amendment does not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (3) result in a significant increase in individual or cumulative occupational radiation exposure. NSPM has reviewed this LAR and determined that the proposed 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 needs to be prepared in connection with the issuance of this amendment. The basis for this determination follows.

1. As demonstrated in the 10 CFR 50.92 evaluation, the proposed amendment does not involve a significant hazards consideration.
2. The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite. Implementation of the proposed project involves no new physical activity: loading procedures and the quantity of fuel handling operations do not change. Thereby, implementing the new TS is not expected to generate any solid, gaseous, or liquid effluent that would not otherwise be generated in the course of routine spent fuel pool operations over its lifetime.
3. The proposed amendment does not result in an increase in individual or cumulative occupational radiation exposure. Implementation of the proposed amendment will not involve a campaign of fuel movements nor will it involve any increase in the amount or frequency of fuel handling operations. In addition, the radiological source term from the ATRIUM 10XM is not expected to be any different from that of the legacy fuel types irradiated to licensed power levels.

6.0 REFERENCES

6.1 Final Interim Staff Guidance DSS-ISG-2010-01, Revision 0, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools, dated August 10, 2010 (ADAMS Accession No. ML110620086).
6.2 NRC Information Notice 2011-03, Nonconservative Criticality Safety Analyses for Fuel Storage, dated February 16, 2011 (ADAMS Accession No. ML103090055)
6.3 NRC Memorandum, Kopp to Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", dated August 19, 1998. The "Kopp Letter" (ADAMS Accession No. ML003728001).
6.4 Carolina Power & Light letter to NRC, Brunswick Steam Electric Plant, Units 1 and 2, Request for License Amendments Regarding Fuel Design and Storage Requirements for AREVA NP Fuel, dated January 22, 2007 (ADAMS Accession No. ML070300372).
6.5 NRC letter to Carolina Power & Light, Brunswick Steam Electric Plant, Units 1 and 2 - Issuance of Amendment Storage of AREVA NP Fuel, dated November 27, 2007 (ADAMS Accession No. ML073310552).
6.6 NUREG-1865, Safety Evaluation Report Related to the License Renewal of the Monticello Nuclear Generating Plant, published October 2006 (ADAMS Accession No. ML063050414).
6.7 NRC letter to Exelon Nuclear, LaSalle County Station, Units 1 and 2, Issuance of Amendments Concerning Spent Fuel Neutron Absorbers, dated January 28, 2011 (ADAMS Accession No. ML110250051).

Enclosure 2

Marked-Up Technical Specification Pages

2 pages follow

4.0-1
4.0-2

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 ~~1.33~~ 0.8825 in the normal ~~reactor core~~ spent fuel pool storage rack configuration ~~at cold conditions~~;

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. ~~keff ≤ 0.90 for original fuel rack if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.2.1 of the USAR; and~~

~~d.~~ 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 8 x 8 high density storage rack,~~ a nominal 6.625 inch center to center distance between fuel assemblies placed in the original storage rack, and a ~~two~~ inch gap between the high density racks and the original rack. [12-]

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

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

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

~~b. k_{eff} < 0.90 if dry;~~

~~c. k_{eff} < 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.2.1 of the USAR;~~

~~d. k_{eff} ≤ 0.98 under optimum moderator conditions, which includes an allowance for uncertainties as described in Section 10.2.1 of the USAR; and~~

~~e. A minimum 6.5 inch center to center distance between fuel assemblies placed in storage racks within a row and a minimum 10 inch center to center distance between fuel assemblies placed in storage racks between rows.~~

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 ~~2301~~ fuel assemblies. [2217]

Enclosure 4

AREVA Affidavit

3 pages follow

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the report ANP-3113(P), Revision 0, "Monticello Nuclear Plant Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM™ 10XM Fuel," dated August 2012 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 23rd day of August, 2012.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/14/2016

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-14-16
STATE OF WASHINGTON