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

June 23, 2016

Mr. David A. Heacock
President and Chief Nuclear Officer
Dominion Nuclear
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION CHANGES FOR SPENT FUEL STORAGE (TAC NO. MF0435)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 327 to Renewed Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2 (MPS2), in response to your application dated December 17, 2012, as supplemented on February 25, 2013, May 28, 2013, July 21, 2015, December 18, 2015, and June 1, 2016.

The amendment revises the MPS2 Technical Specifications (TSs) to reflect the results and constraints of a new criticality safety analysis for fuel assembly storage in the MPS2 fuel storage racks. Specifically, the amendment revises TS 1.39 "Storage Pattern," TS 3.9.18, "Spent Fuel Pool - Storage," TS 3.9.19, "Spent Fuel Pool - Storage Patterns," TS 5.3.1 "Fuel Assemblies," TS 5.6.1, "Criticality," and TS 5.6.3, "Capacity." The amendment will implement the following items associated with fuel storage at MPS2: (1) allow removal of Boraflex credit; (2) eliminate reactivity credit for Boraflex panels in current regions A and B of the spent fuel pool; (3) revise allowed storage patterns for fuel assemblies in the spent fuel pool to meet effective neutron multiplication factor (K_{eff}) requirements under normal and accident conditions; (4) revise alphanumeric designation of spent fuel regions from Regions A, B, and C to Regions 1, 2, 3, and 4 to clearly distinguish from existing designations; (5) allow use of control element assemblies as well as borated stainless steel poison rodlets in Region 3; and (6) eliminate requirement to use spent fuel rack-cell blocking devices.

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

Sincerely,

RGuzman

Richard V. Guzman, Sr. Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 327 to DPR-65
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-336

MILLSTONE POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 327
Renewed License No. DPR-65

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

   A. The application for amendment by Dominion Nuclear Connecticut, Inc. (the licensee) dated December 17, 2012, as supplemented on February 25, 2013, May 28, 2013, July 21, 2015, December 18, 2015, and June 1, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-65 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 327 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Travis L. Tate

Travis L. Tate, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: June 23, 2016.

ATTACHMENT TO LICENSE AMENDMENT NO. 327

RENEWED FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3

Replace the following pages of the 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
1-8	1-8
3/4 9-21	3/4 9-21
3/4 9-22	3/4 9-22
	3/4 9-22a
3/4 9-23	3/4 9-23
	3/4 9-23a
	3/4 9-23b
	3/4 9-23c
	3/4 9-23d
3/4 9-24	3/4 9-24
3/4 9-25	3/4 9-25
3/4 9-25a	3/4 9-25a
3/4 9-26	3/4 9-26
5-4	5-4
5-5	5-5
5-5a	5-5a

Connecticut, in accordance with the procedures and limitations set forth in this renewed operating license;

(2) Pursuant to the Act and 10 CFR Part 70, 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 operation, as described in the Final Safety Analysis Report, as supplemented and amended;

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

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

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

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and 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

The licensee is authorized to operate the facility at steady-state reactor core power levels not in excess of 2700 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 327 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

VENTING

1.35 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

MEMBER(S) OF THE PUBLIC

1.36 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

The term "REAL MEMBER OF THE PUBLIC" means an individual who is exposed to existing dose pathways at one particular location.

SITE BOUNDARY

1.37 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

UNRESTRICTED AREA

1.38 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the SITE BOUNDARY used for residential quarters or industrial, commercial institutional and/or recreational purposes.

STORAGE PATTERN

1.39 A STORAGE PATTERN designates acceptable fuel assembly storage in a 2 x 2 storage array (4 spent fuel rack storage locations) within Regions 1, 2, and 4 of the spent fuel racks. Each 2 x 2 storage array includes at least one location in which storage is NOT permitted (fuel or non-fuel).

1.40 A NON-STANDARD FUEL CONFIGURATION is an object containing fuel that does not conform to the standard fuel configuration. The standard fuel configuration is a 14 x 14 array of fuel rods (or fuel rods replaced by un-enriched fuel rods or stainless steel rods) with five (5) guide tubes that occupy four lattice pitch locations each. Fuel in any other array is a "Non-standard Fuel Configuration." Reconstituted fuel in which one or more fuel rods have been replaced by either un-enriched fuel rods or stainless steel rods is considered to be a standard fuel configuration.

REFUELING OPERATIONS

SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.17 The boron concentration in the spent fuel pool shall be greater than or equal to 2100 parts per million (ppm).

APPLICABILITY: Whenever any fuel assembly or Non-standard Fuel Configuration is stored in the spent fuel pool.

ACTION:

With the boron concentration less than 2100 ppm, suspend the movement of all fuel assemblies, Non-standard Fuel Configurations, and shielded casks, and immediately initiate action to restore the spent fuel pool boron concentration to within its limit.

The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.17 Verify that the boron concentration is greater than or equal to 2100 ppm every 7 days, and within 24 hours prior to the initial movement of a fuel assembly or Non-standard Fuel Configuration in the Spent Fuel Pool, or shielded cask over the cask laydown area.

LIMITING CONDITION FOR OPERATION

3.9.18 The following spent fuel pool storage requirement will be met:

(a) Region 1 fuel assemblies have a maximum initial planar average enrichment of 4.85 weight percent of U-235. A fuel rod shall have a maximum enrichment of 5.0 weight percent of U-235. No burnup credit is required.

(b) Region 2 has two types of storage locations:

(1) The combination of initial planar average enrichment and burnup of a fuel assembly stored in Region 2 Type 2A shall be within the acceptable burnup domain of Figure 3.9-1A.

(2) The combination of initial planar average enrichment and burnup of a fuel assembly stored in Region 2 Type 2B shall be within the acceptable burnup domain of Figure 3.9-1B.

(c) Fuel assemblies stored in Region 3 shall contain either Borated Stainless Steel Poison Rodlets or a full length, full strength Control Element Assembly:

(1) The combination of initial planar average enrichment and burnup of a fuel assembly containing Borated Stainless Steel Poison Rodlets stored in Region 3 shall be within the acceptable burnup domain of Figure 3.9-1C. The Borated Stainless Steel Poison Rodlets shall be installed in the assembly's center guide tube and in two diagonally opposite guide tubes.

(2) The combination of initial planar average enrichment and burnup of a fuel assembly containing a full length, full strength Control Element Assembly stored in Region 3 shall be within the acceptable burnup domain of Figure 3.9-1D.*

(d) The combination of initial planar average enrichment and burnup of a fuel assembly stored in Region 4 shall be within the acceptable burnup domain of Figure 3.9-1E.

* Full-length, reduced-strength Control Element Assemblies and part-length Control Element Assemblies shall NOT be used in Region 3.

LIMITING CONDITION FOR OPERATION

(e) Each Non-standard Fuel Configuration must have a separate criticality analysis to determine where it can be stored in the spent fuel racks. The analysis may qualify storage in one or multiple Regions, and may or may not require Borated Stainless Steel Poison Rodlets or a full length, full strength Control Element Assembly if stored in Region 3.

APPLICABILITY: Whenever any fuel assembly or Non-standard Fuel Configuration is stored in the spent fuel pool.

ACTION:

Immediately initiate action to move the non-complying fuel assembly or Non-standard Fuel Configuration to an acceptable location.

The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.18 Prior to storing a fuel assembly in the spent fuel racks, verify by administrative means the initial planar average enrichment and burnup of the fuel assembly is in accordance with the acceptable specifications for that Storage Region. Prior to storing a Non-standard Fuel Configuration in the spent fuel racks, verify by administrative means the Non-standard Fuel Configuration is qualified for that Storage Region.

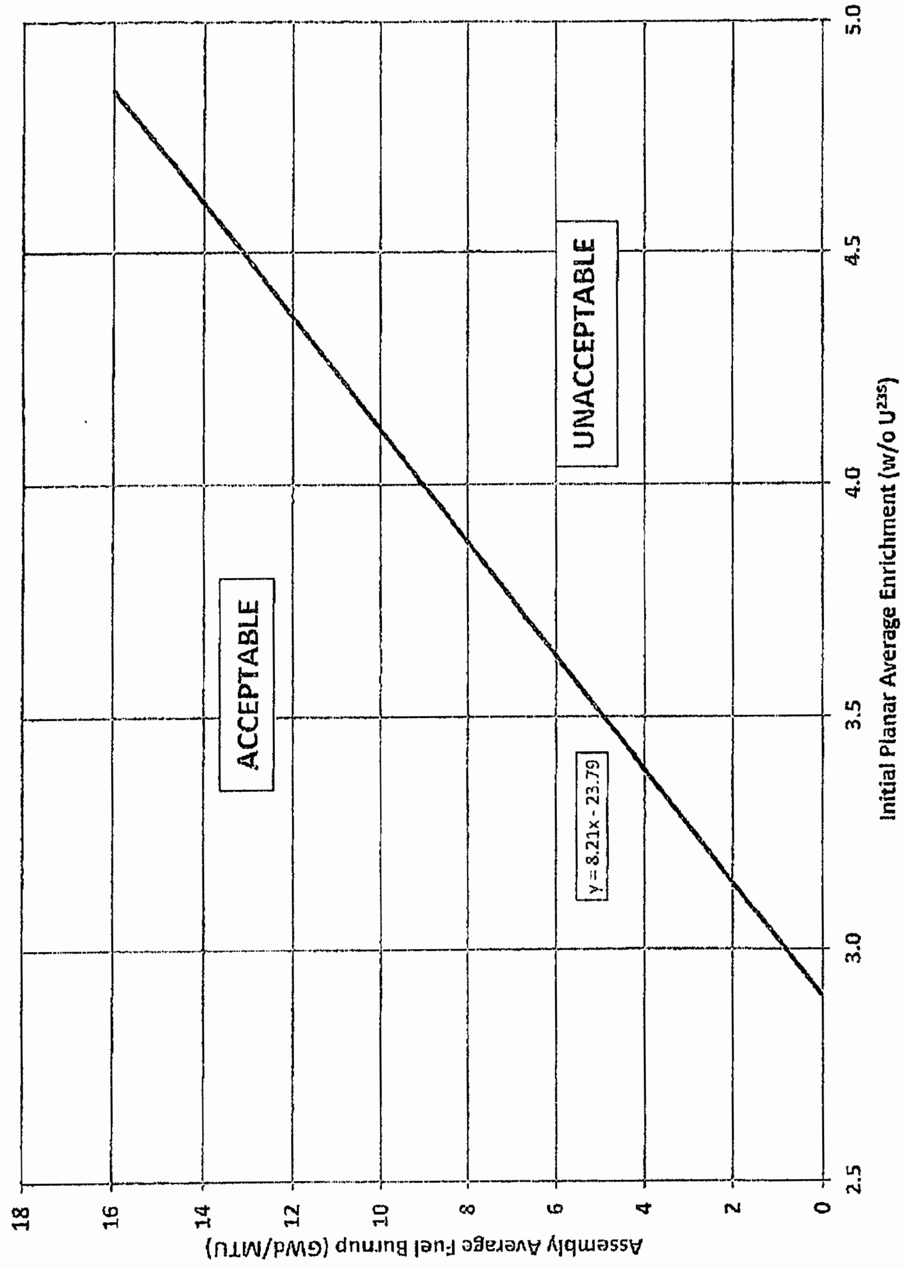


FIGURE 3.9-1A MINIMUM REQUIRED AVERAGE FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2A

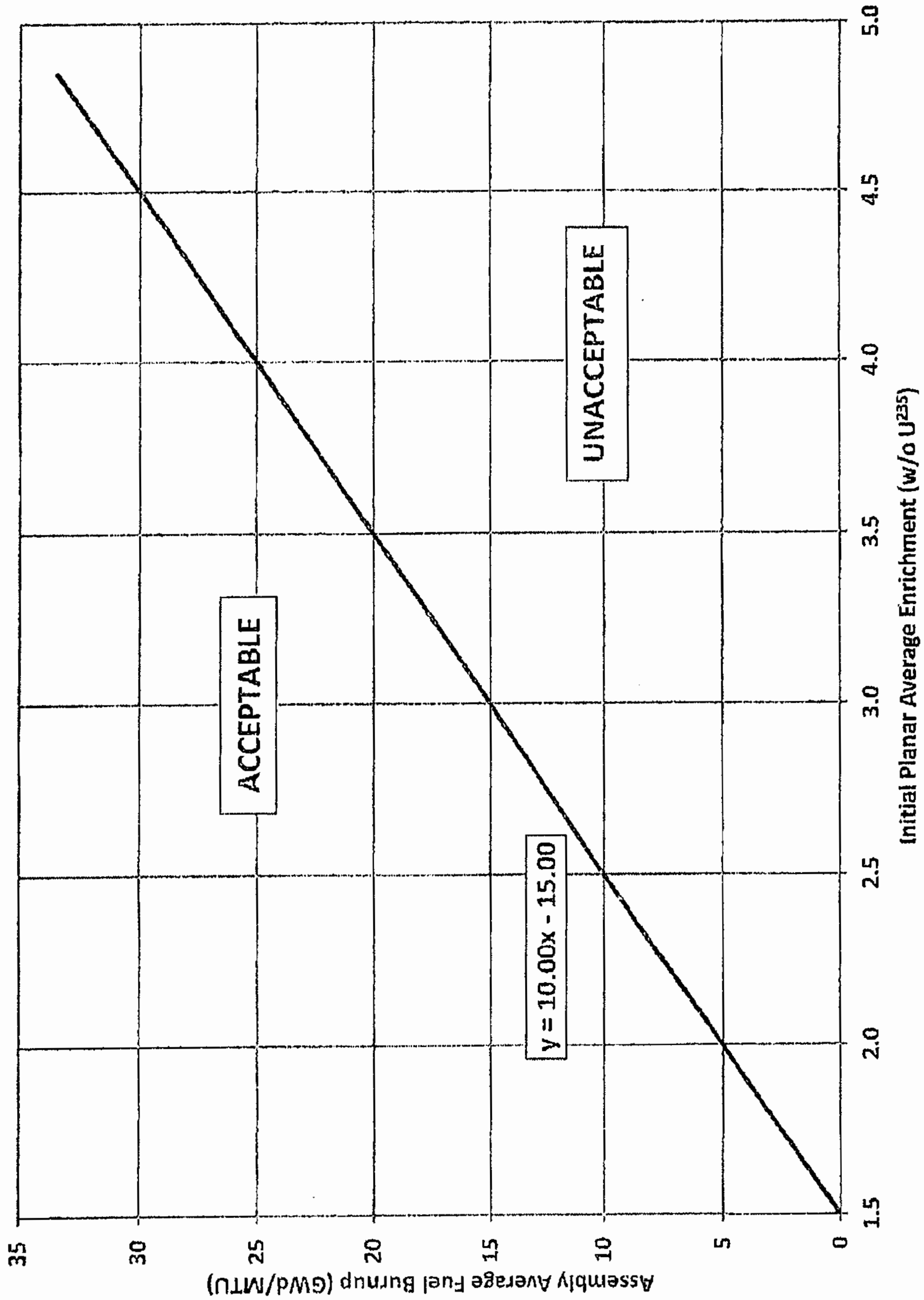


FIGURE 3.9-1B MINIMUM REQUIRED AVERAGE FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2B

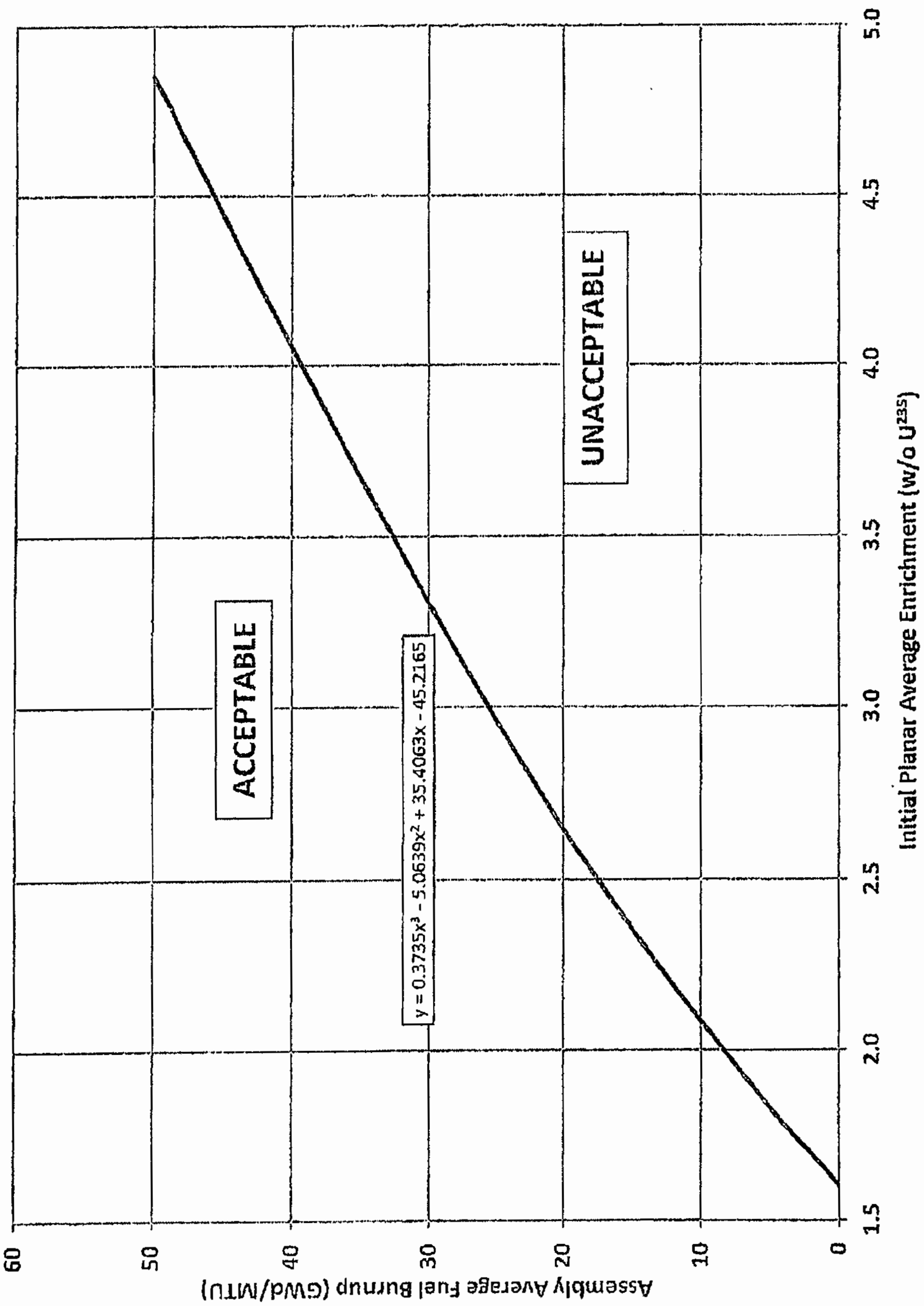


Figure 3.9-1C MINIMUM REQUIRED AVERAGE FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 3
(with insertion of 3 Borated Stainless Steel Poison Rodlets)

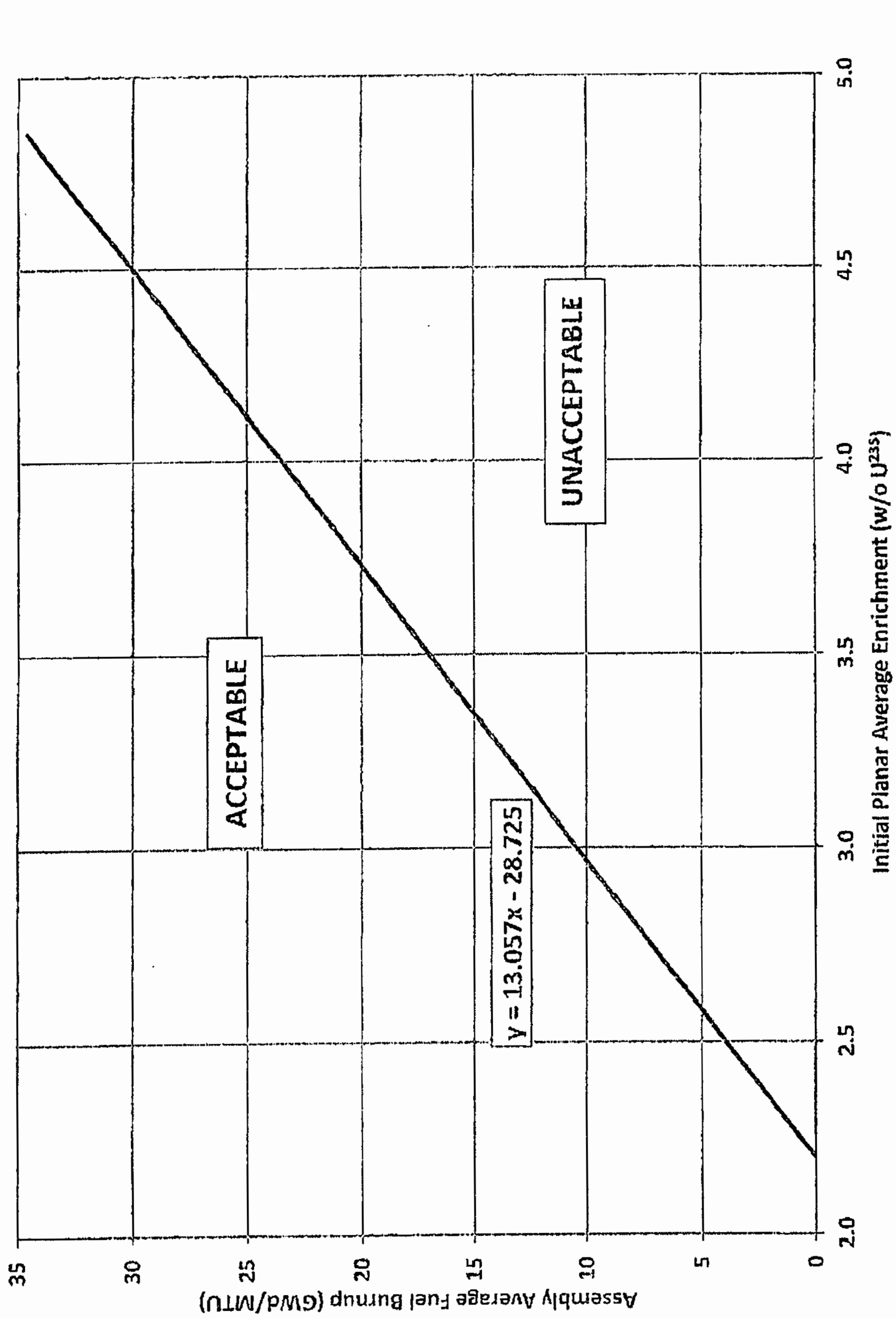


Figure 3.9-1D MINIMUM REQUIRED AVERAGE FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 3 (with insertion of a full length, full strength Control Element Assembly)

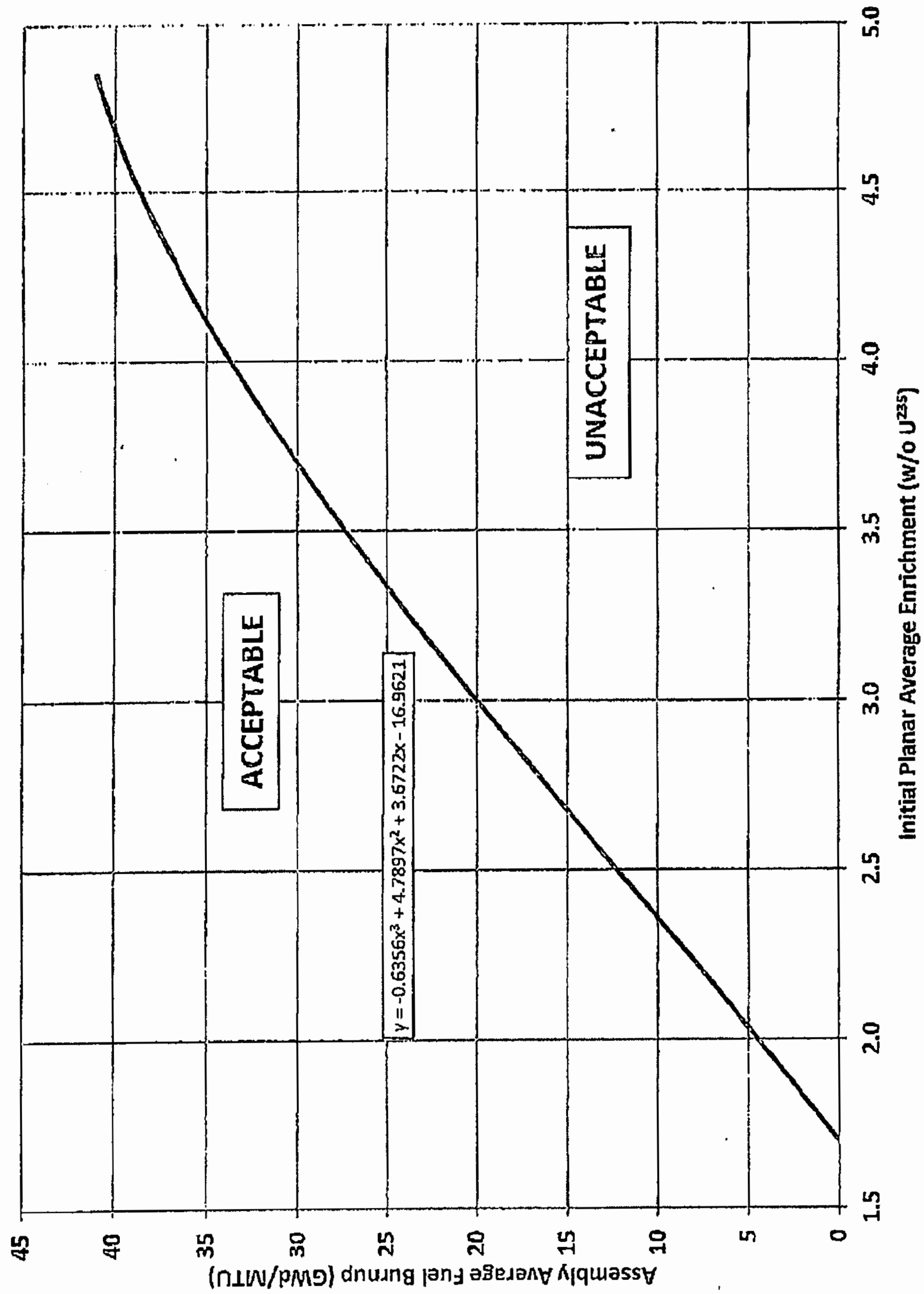
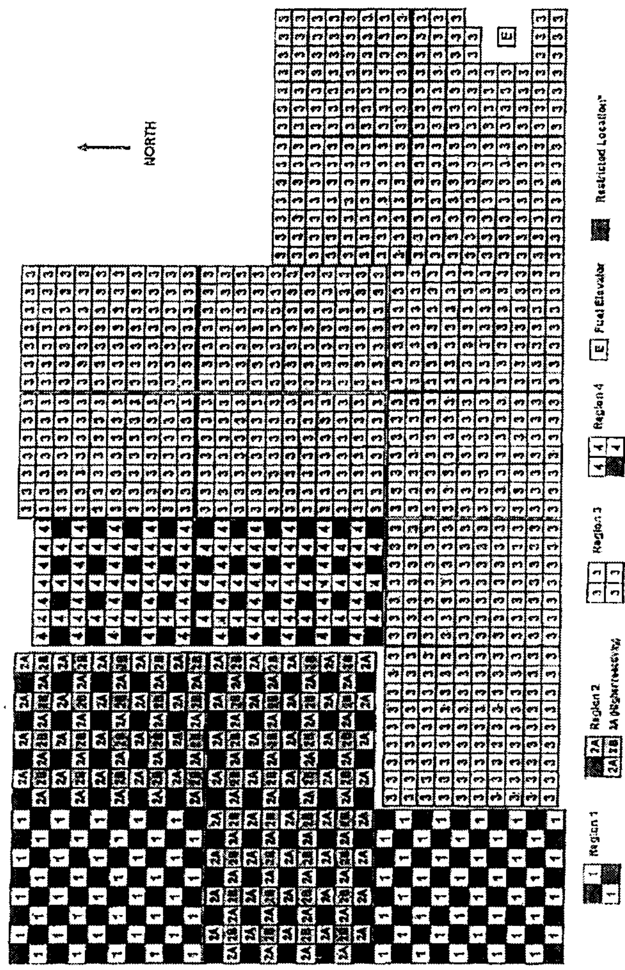


FIGURE 3.9-1E MINIMUM REQUIRED AVERAGE FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 4

SPENT FUEL POOL ARRANGEMENT
FIGURE 3.9-2
(NOT TO SCALE)



* A Restricted Location shall remain empty. No fuel assembly, no Non-standard Fuel Configuration, no non-fuel component, nor any hardware/material of any kind may be stored in a Restricted Location.

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SPENT FUEL POOL - STORAGE RESTRICTIONS

LIMITING CONDITION FOR OPERATION

3.9.19 The following spent fuel pool storage restrictions will be met:

(1) Restricted Locations shall remain empty. No fuel assembly, no Non-standard Fuel Configuration, no non-fuel component, nor any hardware/material of any kind may be stored in a Restricted Location (shown in Figure 3.9-2).

(2) Fuel assemblies and Non-standard Fuel Configurations shall NOT be stored in Region 1 and 2 storage locations in which the Boraflex panel box has been removed. It is permissible to store non-fuel components in non-restricted locations with or without a Boraflex panel box. *

APPLICABILITY: Fuel assemblies, Non-standard Fuel Configurations, or non-fuel components in the spent fuel pool.

ACTION:

Take immediate action to comply with either 3.9.19(1) or (2).

The provisions of specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.19 Verify that 3.9.19 is satisfied prior to storing fuel assemblies, Non-standard Fuel Configurations, or non-fuel components in the spent fuel racks.

* Note that Region 1 and 2 spent fuel pool rack storage locations contain removable Boraflex panel boxes which house the Boraflex panels. The Boraflex panel boxes were manufactured as an integral part of the original spent fuel pool racks and as such are NOT stored components in SFP rack storage locations. Criticality analysis has shown that the Restricted Locations are acceptable with or without the Boraflex panel boxes.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing 176 rods. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum initial planar average enrichment of 4.85 weight percent of U-235. A fuel rod shall have a maximum enrichment of 5.0 weight percent of U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 control element assemblies. The control element assemblies shall be designed and maintained in accordance with the design provisions contained in Section 3.0 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.4 DELETED

5.5 DELETED

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a) The new fuel (dry) storage racks are designed and shall be maintained with sufficient center to center distance between assemblies to ensure a $K_{eff} \leq .95$. The maximum initial planar average fuel assembly enrichment to be stored in these racks is 4.85 weight percent U-235. The maximum initial fuel rod enrichment to be stored in these racks is 5.0 weight percent of U-235.

b) The spent fuel storage racks are designed and shall be maintained with fuel assemblies having a maximum initial planar average enrichment of 4.85 weight percent U-235. The maximum initial fuel rod enrichment to be stored in these racks is 5.0 weight percent U-235.

c) The spent fuel storage racks are designed and shall be maintained with $K_{eff} < 1.00$ if fully flooded with unborated water, which includes an allowance for uncertainties and biases.

d) The spent fuel storage racks are designed and shall be maintained with $K_{eff} \leq .95$ if fully flooded with water borated to 600 ppm, which includes an allowance for uncertainties and biases.

e) Region 1 of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Region 1 contains the Restricted Locations, shown in Figure 3.9-2. Fuel having an initial planar average enrichment of 4.85 weight percent U-235 may be stored in available locations.

f) Region 2 of the spent fuel storage pool is designed and shall be maintained with a nominal 9.8 inch center to center distance between storage locations. Region 2 contains Type 2A and Type 2B storage locations as well as the Restricted Locations shown in Figure 3.9-2. Fuel assemblies stored in this region must comply with Figure 3.9-1A or Figure 3.9-1B. Fuel assemblies utilizing Figure 3.9-1A must be stored in the Region 2 Type 2A storage locations, and fuel assemblies utilizing Figure 3.9-1B must be stored in the Region 2 Type 2B storage locations.

g) Region 3 of the spent fuel storage pool is designed and shall be maintained with a nominal 9.0 inch center to center distance between storage locations. Fuel assemblies stored in this region must comply with Figure 3.9-1C or Figure 3.9-1D. Additionally, fuel assemblies utilizing Figure 3.9-1C require that Borated Stainless Steel Poison Rodlets be inserted in the fuel assembly's center guide tube and in two diagonally opposite guide tubes. The poison rods are solid nominal 0.87 inch O.D. borated stainless steel, with a nominal boron content of 2.0 weight percent boron. Finally, fuel assemblies utilizing Figure 3.9-1D require that a full length, full strength Control Element Assembly be inserted in the fuel assembly (full-length, reduced-strength Control Element Assemblies and part-length Control Element Assemblies shall NOT be used in Region 3).

h) Region 4 of the spent fuel storage pool is designed and shall be maintained with a nominal 9.0 inch center to center distance between storage locations. Region 4 contains Restricted Locations as shown in Figure 3.9-2. Fuel assemblies stored in this region must comply with Figure 3.9-1E.

i) Each region of the spent fuel storage pool is designed to permit storage of Non-standard Fuel Configurations, except for the Restricted Locations. Each of the Non-standard Fuel Configurations must have a separate criticality analysis which may allow storage in one or multiple Regions, and which may or may not require Borated Stainless Steel Poison Rodlets or a full length, full strength Control Element Assembly if stored in Region 3.

j) Regions 1 and 2 spent fuel racks are equipped with boxes that contain the Boraflex panels which may be removed from both non-restricted and Restricted Locations. Fuel assemblies and Non-standard Fuel Configurations shall NOT be placed in storage locations in which the Boraflex panel box has been removed (however, it is permissible to store non-fuel components in a location in which the Boraflex panel box has been removed as long as the location is NOT a Restricted Location).

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 22'6".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with the number of storage locations (including Restricted Locations) limited to no more than 160 storage locations in Region 1, 224 storage locations in Region 2, 822 storage locations in Region 3, and 140 storage locations in Region 4 for a total of 1346 storage locations.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 327

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By application dated December 17, 2012 (Reference 1), as supplemented by letters dated February 25, 2013 (Reference 2), May 28, 2013 (Reference 3), July 21, 2015 (Reference 4), December 18, 2015 (Reference 5), and June 1, 2016 (Reference 6), Dominion Nuclear Connecticut, Inc. (the licensee) requested to amend Renewed Facility Operating License No. DPR-65 and revise the Millstone Power Station Unit 2 (MPS2) Technical Specifications (TSs) to reflect the results and constraints of a new criticality safety analysis for fuel assembly storage in the MPS2 fuel storage racks. Specifically, the amendment revises TS 1.39 "Storage Pattern," TS 3.9.18, "Spent Fuel Pool - Storage," TS 3.9.19, "Spent Fuel Pool - Storage Pattern," TS 5.3.1 "Fuel Assemblies," TS 5.6.1, "Criticality," and TS 5.6.3, "Capacity."

The amendment will implement the following items associated with fuel storage at MPS2: (1) allow removal of Boraflex credit; (2) eliminate reactivity credit for Boraflex panels in current regions A and B of the spent fuel pool (SFP); (3) revise allowed storage patterns for fuel assemblies in the spent fuel pool to meet effective neutron multiplication factor (K_{eff}) requirements under normal and accident conditions; (4) revise alphanumeric designation of spent fuel regions from Regions A, B, and C to Regions 1, 2, 3, and 4 to clearly distinguish from existing designations; (5) allow use of control element assemblies (CEA) as well as borated stainless steel poison rodlets in Region 3; and (6) eliminate requirement to use spent fuel rack-cell blocking devices.

The supplemental letters dated May 28, 2013, July 21, 2015, December 18, 2015, and June 1, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 11, 2013 (78 FR 35060).

2.0 REGULATORY EVALUATION

In accordance with the licensee's amendment request, the regulatory requirements and guidance which the NRC staff considered in assessing the proposed TS change are as follows:

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, Criterion 62, "Prevention of criticality in fuel storage and handling," requires that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Per 10 CFR 50.68(a), each holder of an operating license shall comply with either 10 CFR 70.24 or the requirements in 10 CFR 50.68(b). The licensee has elected to meet 10 CFR 50.68(b). Accordingly, and as relevant to this license amendment request, the licensee must comply with the following 50.68(b) requirements:

> (1) 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.
>
> (2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.
>
> (3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.
>
> (4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. 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 categories of items required to be in the TSs are provided in 10 CFR 50.36(c). As required by 10 CFR 50.36(c)(4), the TSs will include design features which 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 10 CFR 50.36.

In addition to the above citied regulations, the NRC staff also reviewed the proposed license amendment request (LAR) against the guidance of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," and Section 9.1.2, "Spent Fuel Storage," to ensure that there are no potential mechanisms that will: (1) alter the dispersion of boron-10 (B-10) in the borated stainless steel rodlets, and/or (2) cause physical distortion of the tubes retaining the stored fuel assemblies.

3.0 TECHNICAL EVALUATION

3.1 Background

The proposed amendment requested TS revisions to support removal of criticality analysis credit for Boraflex neutron absorber panels for the MPS2 SFP. The request also sought to revise the TSs for the New Fuel Storage Racks (NFSR). The following evaluation presents the results of the NRC's review of the nuclear criticality safety (NCS) analysis, which was provided as Attachment 4 to the licensee's December 17, 2012, application, and updated through the subsequent aforementioned supplemental letters. The criticality safety basis is thus composed of both the original criticality analysis and the licensee's supplemental letters in response to NRC's requests for additional information (RAI) dated June 11, 2014 (Reference 7).

Additionally, the NRC performed an audit at Dominion Innsbrook Technical Center in Glen Allen, VA on February 23 and 24, 2015, to clarify understanding of the NRC RAIs and to discuss the bases for proposed approaches for addressing the RAIs.

The licensee's NCS analysis, as supplemented through RAI responses, describes the methodology and analytical models used to show that the SFP storage rack maximum K_{eff} will be less than 1.0 when flooded with unborated water for normal conditions, and less than or equal to 0.95 when flooded with borated water for normal and credible accident conditions at a 95-percent probability, 95-percent confidence level. For the NFSR, the analysis shows that NFSR rack maximum K_{eff} will be no greater than 0.95 when the NFSR is flooded with unborated water at a 95-percent probability, 95-percent confidence level and will be no greater than 0.98 if the NFSR is flooded with low density water (i.e., at optimum moderation conditions) at a 95-percent probability, 95-percent confidence level.

The MPS2 NFSR NCS analysis demonstrated that, consistent with the requirements of 10 CFR 50.68, the maximum K_{eff} at full water density was less than 0.95, at a 95-percent probability/95-percent confidence level and that the maximum K_{eff} at optimum water density, around 4 percent of full density, was less than 0.98, at a 95-percent probability, 95-percent confidence level.

The MPS2 SFP storage racks are of two designs. One design contained the neutron absorbing material Boraflex. These racks were constructed by welding steel cans together at their corners and welding additional steel plates and corners to complete the perimeter of each rack module. Storage locations in the resulting rack module have a nominal 9.8 inch cell center-to-center spacing. The design includes removable Boraflex boxes. The new analysis and proposed TSs require that the Boraflex boxes be installed in all Region 1 and 2 locations where fuel is stored. The new analysis takes no credit for any potential residual Boraflex in the Boraflex boxes. This design is being designated Region 1 and Region 2. The second design does not include any neutron absorbing material or removable boxes of any kind. This design is of similar construction but is sized such that storage locations have 9.0 inch cell center-to-center spacing. This design is designated as Region 3 and Region 4.

For Region 1, new or used fuel with a maximum planar average initial enrichment no greater than 4.85 wt (weight) percent (%) uranium-235 (U-235) may be stored in a repeating checkerboard arrangement, with required empty cells being referred to as "restricted" locations. Physical blocking devices are not required for restricted cells.

For Region 2, fuel is stored in a 3-out-of-4 repeating arrangement requiring one empty cell and one lower-reactivity assembly (Region 2B) in each 2x2 array of storage cells. The other two locations may contain fuel with somewhat higher reactivity (Region 2A). Burnup credit loading curves are used to qualify fuel assemblies as acceptable for storage as Region 2A or 2B.

For Region 3, fuel assemblies must have either three borated stainless steel (BSS) rods or a CEA installed in the guide tubes of each fuel assembly. Burnup credit loading curves are used to qualify fuel assemblies as acceptable for storage as Region 3 with BSS rods or Region 3 with a CEA.

For Region 4, fuel is stored in a 3-out-of-4 repeating arrangement requiring one empty cell and three fuel assemblies meeting the specified burnup and enrichment requirements.

In general, post-irradiation decay time is not credited in the analysis. However, some decay time was credited for old fuel in the justification for the 3.5% burnup record uncertainty included in the uncertainty analysis for Regions 2, 3 and 4.

New fuel assemblies may be stored in what are normally dry conditions in the MPS2 NFSR. The proposed revisions to the NFSR TSs to limit the "maximum initial planar average enrichment" rather than the "maximum nominal average fuel assembly enrichment."

No credit for fixed or integral burnable absorbers is taken in the fresh or spent fuel storage racks.

3.2 Proposed Change

3.2.1 NCS analyses and Fuel Storage Requirements

There are several proposed TS changes that either impact NCS analyses or implement changes in fuel storage requirements.

The proposed TSs for the NFSR would limit the maximum initial planar average fuel enrichment to 4.85 wt % U-235. The current TSs limit the maximum nominal average fuel assembly enrichment to 4.85 wt % U-235. The proposed TSs more accurately reflect the nuclear criticality safety analysis that was performed to support the TS and eliminate a potential ambiguity.

The proposed TS significantly revise the organization and storage requirements for the SFP. The current MPS2 TS divide the SFP into three regions; Region A, B, and C. The proposed TS divide the SFP into four regions; 1, 2, 3, and 4 with new storage requirements for each Region. The proposed revision to TS Figure 3.9-2 shows the location of each Region. The proposed revision to TS Figure 3.9-2 also shows the location of each Restricted Location. A Restricted Location is a cell where no fuel assembly, non-standard fuel configuration, non-fuel component, or any hardware/material of any kind may be stored.

The proposed Region 1 TS requires fuel, having a maximum planar initial enrichment no greater than 4.85 wt %, to be in stored in Boraflex boxes in a 2-out-of-4 checkerboard arrangement with Restricted Locations. Any residual Boraflex in these boxes would not be credited. The Boraflex boxes must be installed in cells that contain fuel, but may be removed from cells that do not contain fuel.

The proposed Region 2 TS requires that fuel be stored in a 3-out-of-4 arrangement where each repeating 2x2 array of storage locations includes one Restricted Location; one cell with lower reactivity (Region 2B) and two cells with higher reactivity (Region 2A). Separate burnup credit limit curves are provided to qualify fuel for storage as Region 2A or 2B. Any residual Boraflex in these boxes would not be credited. The Boraflex boxes must be installed in cells that contain fuel, but may be removed from cells that do not contain fuel.

The proposed Region 3 TS requires that fuel be stored in a 4-out-of-4 arrangement where each repeating 2x2 array of storage locations includes assemblies that have either three BSS rods with 2 wt % boron, or a full length, full strength CEA installed in each assembly. Separate burnup credit limit curves are provided to qualify fuel for storage in Region 3 with BSS rods or a CEA. The staff has identified implementation activities that the licensee has committed to (Reference 6) that are associated with fuel storage in Region 3 (see Section 3.9).

The proposed Region 4 TS requires that fuel be stored in a 3-out-of-4 arrangement in which each repeating 2x2 array of storage locations includes one Restricted Location. A burnup credit limit curve is provided to qualify fuel for storage in Region 4.

Figure 3.9-2 in the proposed TS provides a graphical representation showing the locations of each region and the required empty or "restricted" locations. As is noted on the figure, storage of fuel or non-fuel components in restricted locations is not permitted.

The proposed TSs also include provisions requiring separate criticality analyses for each non-standard fuel configuration. A definition for "non-standard fuel configuration (NSFC)" is provided in Section 1.40 of the proposed TSs. Attachment 4 to Reference 4 only includes analysis for the current inventory of NSFCs. Therefore, the scope of this license amendment and the NRCs review is limited to the licensee's current NSFC inventory.

Proposed TS 3.9.18 (e) indicates that each NSFC must have a separate criticality analysis and that storage requirements are specified in the analysis. Based on the limited scope of this license amendment, any NSFC that is not bounded by the inventory currently analyzed would be a departure from a method of evaluation described in the updated final safety analysis report (UFSAR) used in establishing the design bases or in the safety analyses and requires prior NRC approval.

3.2.2. Elimination of Boraflex Credit

Shrinkage, gap formation, and dissolution of the Boraflex poison material in the spent fuel racks is a phenomenon addressed in generic communications from the NRC. Specifically, in Generic Letter (GL) 96-04, the NRC staff requested that all licensees of power reactors with installed racks containing Boraflex provide an assessment of the physical condition of the Boraflex and state whether the subcritical margin of 5-percent can be maintained for the racks in unborated water. In addition, the licensees were requested to submit a description of any proposed actions to monitor or confirm that this sub criticality margin can be maintained for the lifetime of the storage racks and to describe any corrective actions in the event that it cannot be maintained. The descriptions of the GL 96-04 commitments for MPS2 are provided in a letter from Connecticut Yankee Atomic Power Company to the NRC dated October 24, 1996 (Reference 8).

The licensee's LAR supplement dated February 25, 2013, described the current Boraflex Program and current results. The current MPS2 Boraflex program monitors Boraflex degradation destructive examination which includes a visual examination, gap measurements, and thickness, density, hardness and B-10 areal density testing. The licensee stated that the most current results from the program are within the assumptions of the criticality analysis and were analyzed to maintain compliance until September 2015. In its letter dated December 18, 2015 (Reference 5), the licensee provided sufficient technical justification to provide assurance that the criticality analysis of record remains valid beyond September 2015 and that the performance of a Boraflex inspection is unnecessary prior to the expected approval date of the December 17, 2012, LAR. The licensee also stated upon approval of this LAR, Boraflex will no longer be credited, and the need for the Boraflex monitoring program would be eliminated. The licensee's approach is conservative and therefore, acceptable by the staff.

3.3 Method of Review

This safety evaluation (SE) involves a review of the licensee's NCS analyses for the MPS2 NFSR and the SFP which was provided as Attachment 4 to the December 17, 2012, license amendment request and updated through the subsequent supplements. The review was performed consistent with Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," and Section 9.1.2, "Spent Fuel Storage," of NUREG-0800.

The NRC staff also used an internal memorandum dated August 19, 1998, containing guidance for performing the review of SFP NCS analysis, hereafter referred to as the 'Kopp Memo' (Reference 9). While the Kopp Memo does not specify a methodology, it does provide some guidance on the more salient aspects of an NCS analysis, including computer code validation.

The guidance is germane to boiling-water reactors and pressurized-water reactors (PWRs), for both borated and unborated fuel storage pools. The Kopp Memo has been used during NRC review of virtually every light-water reactor SFP NCS analysis since, including this MPS2 analysis.

The NRC staff also used interim staff guidance document DSS-ISG-2010-01, (Reference 10) dated September 2010, for review of SFP criticality analyses. The guidance in DSS-ISG-2010-01 is used by the NRC staff to review nuclear criticality safety analyses for the storage of new and spent nuclear fuel as they apply to: (i) future applications for construction and/or operating licenses; and (ii) future applications for license amendments and requests for exemptions from compliance with applicable requirements, that are approved after the date of this ISG [interim staff guidance].

3.4 SFP NCS Analysis Review

3.4.1 SFP NCS Analysis Method

There is no generic or standard NRC-approved methodology for performing NCS analyses for fuel storage and handling. The methods used for the NCS analysis for fuel in the MPS2 SFP are described in the criticality analysis which was provided as Attachment 4 to the December 17, 2012, application, and updated through the subsequent supplements. Some SFP criticality analysis potential non-conservatisms were identified during the review, but as will be discussed below, sufficient margin is built into the analysis methodology to offset the potential non-conservatisms. Consequently, the methodology is specific to this analysis and is not appropriate for other applications.

3.4.1.1 Computational Methods

The MPS2 NCS analysis considers the decrease in fuel reactivity typically seen in PWRs as the fuel is depleted during reactor operation for three of the four newly defined regions. This approach is frequently used in PWR NCS analyses and is sometimes referred to as burnup credit (BUC). BUC NCS analysis requires a two-step process. The first step relates to depletion where a computer code simulates the reactor operation to calculate the changes in the fuel composition of the fuel assembly. The second step is a modeling of the depleted fuel assembly in the SFP storage racks and the determination of the system K_{eff}. The validation of the computer codes in each step is a significant portion of the analysis. Since the MPS2 NCS analysis credits fuel burnup, it is necessary for the NRC to consider validation of the computer codes and data used to calculate burned fuel compositions, and the computer code and data that utilize the burned fuel compositions to calculate K_{eff} for systems with burned fuel.

For the depletion step, BUC NCS analyses typically involve use of a computer code approved by the NRC for the purposes of performing reactor core simulation analyses. Those computer codes have an NRC SE governing their use, including any necessary limitations and conditions. Additionally, those NRC-approved codes are being used by numerous licensees to perform reactor core analysis, thereby providing a feedback mechanism should significant differences be observed between reactor core analyses and actual reactor core performance. MPS2 used the T5-DEPL depletion sequence from SCALE 6.0 to perform its depletion step. Prior to this LAR,

T5-DEPL had not been used in a SFP NCS licensing application submitted to the NRC and was not previously reviewed and approved by the NRC in such an application. Therefore, the applicability of previous guidance associated with SFP NCS analyses needed to be established for the use of T5-DEPL as a depletion code for this specific analysis.

3.4.1.1.1 Depletion Computer Code Validation

In the NRC staff's RAI letter dated June 11, 2014 (Reference 7), the staff identified concerns (RAIs 9 and 11) associated with computer code validation that required resolution in order to establish a basis for use of the SCALE 6.0 T5-DEPL sequence for burned fuel compositions. During the staff's review, the following concerns were identified:

- Adequacy of convergence of neutron fluxes calculated by KENO V.a
- Adequacy of the validation methodology

The analysis and methodology used in the licensee's application is unique to MPS2 SFP and, therefore, is solely applicable to MPS2. For example, the licensee demonstrated an acceptable use of the SCALE TRITON depletion sequence for this application. However, there is no regulatory guidance on the use of the SCALE TRITON depletion sequence and the NRC staff would consider any change in the way SCALE TRITON depletion sequence was used in this analysis a deviation from the approved methodology.

Flux convergence is a concern for the NRC because the deterministic codes historically used for this analysis typically iterate on neutron flux to ensure that the maximum flux difference between iterations is acceptably small. In response to the RAIs, the licensee identified that the KENO V.a code calculates flux using a track length estimator (see Section F11.6.6 of Reference 11) and does not perform checks for maximum flux difference. To check convergence, confirmatory calculations were run by the licensee with many more neutron histories. In these calculations, the computed burned fuel compositions did not change appreciably with many more neutron histories. Thus, the staff finds the licensee provided reasonable assurance that the calculated assembly average neutron fluxes are adequately converged for the specific model it used. The MPS2 burned fuel compositions were calculated using a relatively thin slice of an assembly; and burned fuel compositions were calculated on an assembly average basis. The staff notes that assembly slice average burned fuel compositions have been used and accepted in many prior burned fuel analyses, and are therefore, acceptable.

In its original MPS2 criticality analysis, the licensee used a modified version of the depletion validation methods described in NUREG/CR-7108 (Reference 12). In response to RAIs regarding the depletion validation method, the licensee altered its approach to use the depletion uncertainty method described in the Kopp Memo (response to RAI 11 in Reference 4). The Kopp Memo recommends "in the absence of any other determination of the depletion uncertainty," an uncertainty equal to "5% of the reactivity decrement" associated with fuel burnup.[1]

[1] This uncertainty is calculated as 0.05 times the change in K_{eff} from the fresh fuel to the credited final fuel burnup.

Since using the Kopp Memo recommendation for SCALE 6.0 T5-DEPL calculations has not been used before in SFP burnup credit analyses, the licensee performed additional fuel depletion calculations using industry standard CASMO4 and CASMO5 codes to show that, for the MPS2 calculations, equal or higher in-rack K_{eff} values were generated using the T5-DEPL sequence as compared with using CASMO4 or CASMO5. The staff finds that since use of the T5-DEPL sequence produces consistently higher in-rack reactivity than CASMO4 or CASMO5, application of the Kopp Memo depletion uncertainty guidance is acceptable for this analysis. The staff notes that the work done for MPS2 is too narrow to claim broad applicability to spent nuclear fuel at other plants; however, the use of SCALE 6.0 T5-DEPL sequence with the Kopp Memo (uncertainty equal to 5% of the reactivity decrement) is considered acceptable for the MPS2 amendment request.

The T5-DEPL sequence also uses ORIGEN-S to calculate the burned fuel compositions. ORIGEN-S has been used in many criticality analyses, is considered to be an industry standard, and is considered acceptable by the staff.

The staff finds that the licensee's analyses provide reasonable assurance that previous guidance regarding the depletion validation is applicable to this MPS2 LAR. Consistent with the guidance provided in the Kopp Memo, the licensee's analysis has incorporated an uncertainty equal to 5% of the reactivity decrement to cover lack of validation of fuel composition calculations. This uncertainty was calculated by the licensee and applied correctly.

3.4.1.1.2 SFP K_{eff} Computer Code Validation

The study used to support validation of K_{eff} calculations using the SCALE 6.0 CSAS5 sequence is documented in Appendix A of Attachment 4 to Reference 1 and in the responses provided by the licensee for RAI 42. The validation set includes 253 critical configurations that included 66 of the French Haut Taux de Combustion (HTC) critical experiments (Reference 13), and 187 low enrichment uranium fuel pin experiments from 25 different evaluations published in the International Handbook of Evaluated Criticality Safety Benchmark Experiments (IHECSBE) (Reference 14). During review of the LAR, the NRC identified a concern (RAI 42) that the trending analysis did not evaluate trends in the calculated bias and bias uncertainty associated with variation of experiment temperature and plutonium content.

In the response to Part a. of RAI 42, the licensee modeled 17 critical configurations from LEU-COMP-THERM-046, having temperatures varying from 14 to 85 degrees Celsius (°C). The results were then used to develop a temperature dependent bias and uncertainty. The licensee reported that the bias sensitivity was -1.70 percent millirho per degree Celsius (pcm/°C) at the 95-percent upper confidence interval limit. The NRC staff performed an inverse variance weighted linear least squares fit of the data reported in the RAI response, calculating that the change in reactivity with temperature to be -1.14 +/- 0.61 pcm/°C. Since the licensee reported the same -1.14 pcm/°C best estimate value, the NRC concluded that the licensee did not use a 2-standard-deviation adder and that the 95-percent upper confidence interval limit value should have been -2.35 pcm/°C. Applying the 0.65 pcm/°C difference over the range from 20°C to 100°C introduced a non-conservatism of approximately 51 pcm. However, based upon engineering judgement, the staff finds that there is more than enough margin to cover up to 51 pcm of non-conservatism. Therefore, the staff finds the licensee's model acceptable.

Part b. of RAI 42 noted that the set of mixed uranium and plutonium oxide (MOX) critical experiments used in the original validation was not adequately diverse, since the validation included only experiments from the HTC critical experiment series. In response, the licensee modeled an additional 63 MOX critical configurations from the IHECSBE. The mean K_{eff} for these MOX experiments was 0.0015 delta-k-effective (Δk) higher than the mean for the low enrichment uranium (LEU) experiments. Since the average K_{eff} was higher for the MOX experiments, the licensee determined that it is conservative to exclude the MOX experiment results from the bias and 95/95 bias uncertainty calculation. The licensee's evaluation did not include consideration of the spread of the MOX experiment results. Based on the NRC staff's evaluation, it is unlikely that including the MOX experiments would cause the 95/95 bias uncertainty to increase by 50%. However, even if inclusion of the MOX experiments increase the uncertainty by 50% (increasing the validation 95/95 bias uncertainty from 520 to 780 pcm), the largest impact on the total uncertainty and on maximum K_{eff} values would be about 150 pcm. The staff finds that there is more than enough margin to cover up to 150 pcm of non-conservatism. Therefore, the staff finds the licensee's model acceptable.

Combining the potential non-conservatisms associated with the responses to RAI 42, the staff finds there is reasonable assurance that the total impact to K_{eff} computer code validation is no greater than 200 pcm. Based on the above, the staff finds the licensee's SFP K_{eff} computer code validation to be acceptable.

3.4.2 SFP and Fuel Storage Racks

3.4.2.1 SFP Water Temperature

NRC guidance provided in the Kopp Memo states the NCS analysis should be done at the temperature corresponding to the highest reactivity. The licensee's analysis was performed with water temperatures of 32 degrees Fahrenheit (°F), 67 °F (base case), 110 °F, and 210 °F. The water densities used were adjusted consistent with the water temperatures being modeled. The effects of water temperature variation on the maximum K_{eff} value were included as bias terms for all regions both with and without soluble boron. Therefore, the staff finds that the water temperature and density was handled appropriately in the licensee's criticality analysis.

3.4.2.2 SFP Storage Rack Models

Two fuel storage rack variations are used in the MPS2 SFP. Both were manufactured by fabricating steel boxes that were then welded together at their corners, creating an arrangement of storage cells inside fabricated boxes and between boxes. The Region 1 and 2 boxes are sized such that they have a nominal storage cell center-to-center spacing of 9.8 inches. The Region 3 and 4 boxes are sized such that they have nominal storage cell center-to-center spacing of 9.0 inches.

The Region 1 and 2 rack modules have removable "Boraflex boxes". In the Boraflex box model, the Boraflex sheet is replaced with water. The licensee reviewed the design drawings to confirm that the Boraflex boxes have vent holes. The presence of the vent holes provides reasonable assurance that gas is not accumulating in the Boraflex boxes. The Region 3 and 4

rack modules do not use Boraflex boxes. Therefore, the NRC staff finds that the design basis models used to model the Region 1, 2, 3 and 4 racks are appropriate and conservative.

3.4.2.3 SFP Storage Rack Models Manufacturing Tolerances and Uncertainties

The nominal dimensions of cell center-to-center spacing, box walls, sheathing and Boraflex boxes were used in design basis calculations. To provide estimates for uncertainties associated with rack manufacturing tolerances and uncertainties, the licensee performed sensitivity calculations for each region, for each of the permitted storage configurations, as a function of burnup, and both with and without soluble boron in the pool. The staff's review finds that the uncertainties were conservatively estimated by the licensee, including proper inclusion of the Monte Carlo uncertainties associated with the CSAS5 calculations used to calculate the sensitivities.

3.4.3 Fuel Assembly

3.4.3.1 Bounding Fuel Assembly Design

The fuel assemblies used at MPS2 are all Combustion Engineering (CE) 14x14 or similar assemblies manufactured by other suppliers. As documented in Section 3.1.1 of the licensee's criticality analysis, all past and present MPS2 fuel designs were evaluated over the enrichment, burnup, soluble boron range and in both SFP fuel storage rack designs. The design used in the analysis had reactivity that was higher than or statistically equal to the reactivity calculated for all other MPS2 fuel designs.

In some of the older CE 14x14 assemblies, fuel rods were replaced with rods containing B_4C. More recently, the fuel assemblies have included some fuel rods with Gd_2O_3 mixed in with the UO_2 in the fuel pellets. The U-235 enrichment is lower for fuel pellets mixed with Gd_2O_3 compared to the U-235 mass maximum planar average enrichment for the fuel pellets that do not have Gd_2O_3.

In both new and old fuel designs, the initial uranium enrichment may be varied from pin-to-pin to control fuel assembly power peaking factors. The fuel assembly design model used in the MPS2 SFP NCS analysis does not include any B_4C or Gd_2O_3 rods and utilizes the maximum planar average enrichment for all fuel pins. This increases the amount of U-235 present relative to the actual fuel assemblies. The licensee performed sensitivity calculations to estimate the bias introduced by using the planar average enrichment rather than the actual pin-dependent enrichment variations. The sensitivity studies indicated that use of the planar average enrichment for some Region 3 and 4 configurations may be slightly non-conservative. Consequently, the licensee included a 0.001 Δk bias in the calculation of the maximum K_{eff} values for Regions 3 and 4. The NRC staff finds that the inclusion of the Δk bias in the calculation is appropriate and conservative, and is therefore, considered acceptable.

Some fuel assemblies stored at MPS2 include reduced enrichment blankets on the ends of the fuel assemblies. The licensee's analysis conservatively assumed all fuel was non-blanketed and used conservative non-blanketed assembly axial burnup profiles, which are discussed in more detail below.

As is noted in Section 6.1.1.1 of the licensee's criticality analysis, the fuel assembly model included the lower tie plate, lower end plugs, active fuel length, upper plenum, and upper tie plate. Justification was provided in the RAI responses (Reference 4) to support modeling simplifications wherein the upper and lower tie plates were homogenized with the collocated water.

The bounding model used for K_{eff} calculations did not include fuel assembly grids. Instead, sensitivity calculations were performed both with and without soluble boron to quantify the impact of this simplification. Bias terms were calculated and, where appropriate, incorporated into the calculation of the total bias and uncertainty. Based upon engineering judgement, the staff finds this approach acceptable.

3.4.3.2 Fuel Assembly Manufacturing Tolerances and Uncertainties

The analysis, revised as documented in the RAI responses, used a standard approach for quantifying the uncertainty in K_{eff} associated with the fuel assembly manufacturing tolerances and uncertainties. Sensitivity calculations were performed by the licensee for various initial enrichment/final burnup points, in each rack and storage configuration, with and without soluble boron, and in both normal and accident conditions. The NRC staff finds the uncertainty analysis performed by the licensee is thorough, follows a standard approach and is, therefore, considered acceptable.

3.4.3.3 Spent Fuel Characterization

Characterization of fresh fuel is based primarily on U-235 enrichment and various manufacturing tolerances. The manufacturing tolerances are typically manifested as uncertainties, as discussed above, or are bounded by values used in the analysis. These tolerances and bounding values would also apply to the spent nuclear fuel. Common industry practice has been to treat the uncertainties as unaffected by the fuel depletion. The characterization of spent nuclear fuel is complex. Its characterization is based on the specifics of its initial conditions and its operational history in the reactor. That characterization has three main areas: depletion uncertainty, the axial and radial apportionment of the burnup, and the core operation that achieved that burnup. These characteristics are evaluated in the following sections.

3.4.3.3.1 Depletion Uncertainty

As initially discussed in Section 3.4.1.1.1 the licensee initially used a modified version of the guidance in NUREG/CR-7108 (Reference 12) to determine a depletion uncertainty. In response to an RAI, the licensee altered its approach to conform with the guidance in the Kopp Memo. The staff finds the licensee demonstrated that within the confines of this analysis, applying the guidance in the Kopp Memo was acceptable.

3.4.3.3.2 Axial Apportionment of the Burnup or Axial Burnup Profile

Another important aspect of fuel characterization is the selection of the axial burnup profile. At the beginning of life, a PWR fuel assembly will be exposed to a near-cosine axial-shaped flux,

which will deplete fuel near the axial center at a greater rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission-product buildup that occurs near the center. Near the fuel assembly ends, burnup is suppressed due to neutron leakage. If a uniform axial burnup profile is assumed, then the burnup at the ends is over predicted. Analysis discussed in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis" (Reference 15), has shown that, at assembly burnups above about 10 to 20 gigawatt-days per metric ton of uranium (GWd/MTU), the use of a uniform axial burnup profile results in an under prediction of K_{eff}; generally the under prediction becomes larger as burnup increases. This is what is known as the "end effect." Proper selection of the axial burnup profile is necessary to ensure K_{eff} is not under-predicted due to the end effect.

Consistent with the guidance provided in DSS-ISG-2010-01, the MPS2 SFP criticality analysis used the bounding axial burnup profiles from NUREG/CR-6801. While a significant fraction of the MPS2 fuel has axial blankets, the analysis treated all fuel as non-blanketed, effectively adding more highly enriched fuel to the end of each assembly. The staff finds this is an acceptable approach to conservatively address the end effect.

However, the licensee's initial analysis did not include uniform axial burnup profiles. This was a concern for the NRC because NUREG/CR-6801 looked at axial burnup shapes that were expected to maximize the end effect and did not explore the impact of more uniform shapes at lower burnups. Common practice is to do an analysis with both bounding axially-varying shapes and with axially uniform profiles, using whichever result yields higher K_{eff} values. This is also important for mixed fresh and higher burnup systems, under both normal and accident conditions, because the fission density in the fresh fuel will peak near the axial center of the assembly. In RAI 13, the staff identified a concern with the licensee's initial analysis in that NUREG/CR-6801 shapes may not be bounding due to axial mismatch of fission density profiles.

The licensee's response to RAI 13 notes that the uniform axial burnup shapes were added to the analysis and staff review determined that they were appropriately considered. Consequently, the staff finds that the treatment of axial burnup distribution by the licensee is acceptable.

3.4.3.3.3 Planar Burnup Distribution

Due to the neutron flux gradients in the reactor core, assemblies can show a radially tilted burnup distribution (i.e., differences in burnup between portions or quadrants of the cross section of the assembly). The MPS2 analysis did not consider the effect of planar burnup distribution on reactivity.

The impact of radial burnup gradients may be estimated by comparing the distribution of radial burnup tilt information provided in Figure 3-4 of DOE/RW-0496 (Reference 16) with information on the sensitivity of K_{eff} to radial burnup tilt provided in Section 6.1.2 of NUREG/CR-6800 (Reference 17). From DOE/RW-0496, the maximum quadrant deviation from assembly average burnup had been observed to be less than 25 percent at low assembly average burnups (burnup < 20 GWd/MTU) and was observed to decrease with burnup, generally being less than 10 percent at burnups above 20 GWd/MTU. Combining these radial tilt bounding estimates with

the K_{eff} sensitivity information provided in NUREG/CR-6800, the NRC staff's review of these radial burnup tilts indicate that K_{eff} could increase by as much as 0.002 Δk. Based on the above information, the staff finds that its potential impact is small and it is conservative to consider this effect as a bias. The staff finds this reactivity effect is accommodated within the analysis margins.

3.4.3.3.4 Burnup History/Core Operating Parameters

NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," (Reference 18) provides some discussion on the treatment of depletion analysis parameters that determine how the burnup was achieved. While NUREG/CR-6665 is focused on NCS analysis in storage and transportation casks, the basic principles with respect to the depletion analysis apply generically, since the phenomena occur in the reactor as the fuel is being depleted. The results have some applicability to MPS2 NCS analyses. The basic strategy for this type of analysis is to select parameters that maximize the Doppler broadening/spectral hardening of the neutron field resulting in maximum plutonium-239/241 production. NUREG/CR-6665 discusses six parameters affecting the depletion analysis: fuel temperature, moderator temperature, soluble boron, specific power and operating history, fixed burnable poisons, and integral burnable poisons. While the mechanism for each is different, the effect is similar: Doppler broadening/spectral hardening of the neutron field resulting in increased plutonium-239/241 production. NUREG/CR-6665 provides an estimate of the reactivity worth of these parameters. The largest effect appears to be due to moderator temperature. NUREG/CR-6665 approximates the moderator temperature effect, in an infinite lattice of high burnup fuel, to be 90 pcm per degree Kelvin (°K). Thus, a 10 °F change in moderator temperature used in the depletion analysis would result in 0.005 Δk. The effects of each core operating parameter typically have a burnup or time dependency.

For fuel and moderator temperatures, NUREG/CR-6665 recommends using the maximum operating temperatures to maximize plutonium-239/241 production. For fuel and moderator temperatures, the MPS2 analysis used a licensed core design model, PRISM, to calculate moderator and fuel temperatures based on conservative estimates of assembly power (the details of this analysis is found in section 3.1.3.2 of the licensee's criticality analysis and in its response to RAI 14 (Reference 4)). Therefore, based upon the information above, the NRC staff finds the moderator and fuel temperatures used by the licensee are acceptable.

For boron concentration, NUREG/CR-6665 recommends using a conservatively high cycle-average boron concentration. The licensee's analysis used a cycle average soluble boron concentration of 800 ppm for all cycles. This value is 100 ppm higher than the highest cycle average soluble boron concentration seen in Cycles 1 through 21. As a result, the NRC staff finds the boron concentration used in the licensee's analysis is acceptable.

The staff has identified implementation activities that the licensee has committed to (Reference 6) that are associated with confirmation of burnup history and core operating parameters (see Section 3.9).

3.4.3.3.5 Integral and Fixed Burnable Absorbers

In the past, MPS2 used two types of integral burnable absorbers. B_4C burnable absorber rods replaced fuel rods in some assemblies. Modeling these assemblies as though the absorber rods were fuel rods is considered conservative and acceptable by the staff. MPS2 has also used fuel rods in which gadolinia (i.e., Gd_2O_3) is mixed in with the UO_2 during manufacturing and is present in some axial zones in some fuel rods in some assemblies. The MPS2 analysis models all fuel as non-gadolinia poisoned fuel. The staff finds this approach is consistent with conclusions provided in NUREG/CR-6760 (Reference 19) and with some more recent studies supporting a previously approved LAR (ADAMS Accession No. ML12139A198) and is acceptable.

The licensee stated in its LAR that it has never used fixed burnable absorbers. The staff has identified implementation activities that the licensee has committed to (Reference 6) that are associated with confirmation of the use of integral and fixed burnable absorbers (see Section 3.9).

3.4.3.3.6 Control Element Assembly Usage

If CEAs are present in assemblies for significant amounts of time in the reactor, the associated spectral hardening can increase plutonium generation, leading to higher fuel reactivity for the same burnup. The MPS2 analysis identified the bounding maximum MPS2 control rod history based on historical operating data and simulated this small insertion for all fuel. The staff finds this is a conservative approach because most fuel saw less or no control rod exposure. Therefore, the staff finds this modeling of the effects of CEA usage is acceptable.

The staff has identified implementation activities that the licensee has committed to (Reference 6) that are associated with confirmation CEA usage (see Section 3.9).

3.4.3.3.7 Credited Nuclides

In response to RAI 15, the licensee provided a list of isotopes used in the MPS2 analysis and revised the analysis to reflect fission gas releases that may occur under severe accident conditions. Release fractions used are shown in RAI-15 Table 15.4. The staff finds these release fractions are conservative for SFP criticality analysis because the SFP accident conditions are not nearly as severe as those considered in the source reference. Therefore, the staff finds that the licensee's accident release fractions used are acceptable.

3.4.4 Non-Standard Fuel Configurations/Reconstituted Fuel

RAIs 1, 3, 4, 5 and 16 requested information on description of and analysis covering what were identified as NSFCs. In response to RAI 3, standard fuel is defined as a 14x14 array of fuel pins with five guide tubes. Standard fuel may also include reconstituted fuel in which one or more fuel pins have been replaced by either non-enriched fuel pins or stainless steel pins. Any other fuel is a NSFC. A definition for NSFC is provided in Section 1.40 of the proposed TS. The NSFCs include three consolidated fuel storage boxes (CFSBs) that contain fuel rods from two spent fuel assemblies in each box. The current TS included burnup limits on fuel in these

CFSBs. Since no further consolidation operations are planned, the burnup credit TS limits are no longer needed and have been removed from the proposed TS. Instead, the analysis shows that these specific CFSBs reduce rack K_{eff} significantly.

Criticality analysis for all current NSFCs was provided as Attachment 4 to Reference 4. A list of NSFCs is provided in Table 4.0-1 of Attachment 4. The analysis for each configuration demonstrated that, under normal conditions, storage of each NSFC either had no impact or reduced storage rack K_{eff}. Accident analysis, provided in Section 4.9 of Attachment 4, demonstrated that the K_{eff} does not increase K_{eff} including bias and uncertainty above 0.95 when the pool is at the minimum soluble boron allowed by the proposed TS (2100 ppm).

The licensee has provided a description of its fuel reconstitution process. As part of this process, the fuel assembly being reconstituted will be neutronically decoupled from the other fuel assemblies in the SFP. Only one fuel rod will be removed at a time. The original fuel rod will be returned to its original location, or it will be replaced with a stainless steel rod or a fuel rod with naturally occurring U-235. With one exception, fuel assemblies with an empty lattice location have not been stored with other fuel assemblies. Based on this description, the NRC finds the licensee's fuel reconstitution process and storage acceptable.

The staff has identified implementation activities that the licensee has committed to (Reference 6) that are associated with non-standard fuel configurations/reconstituted fuel (see Section 3.9).

3.4.5 Determination of Soluble Boron Requirements

Section 50.68 of 10 CFR requires that the K_{eff} of the MPS2 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. This requirement applies to all normal and abnormal/accident conditions.

The licensee's initial analysis for the normal static condition, was based on a model of the entire SFP. Monte Carlo models that large can be complex. In response to an RAI, the licensee revised its analysis to demonstrate that the K_{eff} for each Region was equal to or less than 0.95 at a 95-percent probability, 95-percent confidence level for normal static conditions with 550 ppm of soluble boron. This is less than the 600 ppm currently in MPS2 TS (5.6.1.e), therefore, the staff finds the analysis results showing 50 ppm of soluble boron margin is acceptable. The licensee's initial analysis considered normal non-static conditions such as fuel handling and inspection. The licensee's analysis used engineering judgment to conclude that its normal non-static conditions are bounded by the normal static conditions due to the neutronic decoupling that occurs during both conditions. This conclusion was not altered by the RAI responses. The NRC staff agrees with the licensee's conclusion and, therefore, finds the licensee's approach acceptable.

The licensee's initial analysis also addressed abnormal/accident conditions based on a model of the entire SFP. The licensee did not initially consider the accident where multiple fuel assemblies are misloaded, however, in response to an RAI the licensee revised their analysis to include multiple misload scenarios on a smaller model.

The licensee limited its consideration of multiple misloading scenarios to fresh fuel misloadings in Regions 1 and 2, while Regions 3 and 4 were only analyzed for misloadings of underburned fuel. The rationale for accepting the difference in fuel burnup modeling assumptions for misloaded fuel between certain regions is based on the following: (1) restriction on storage of fresh fuel in Regions 3 and 4, (2) there are visual cues for fuel handlers that would allow them to recognize errors in move sheets, and (3) the restrictions and visual cues will be included in fuel handling procedures and training program. The NRC staff considers the licensee's multiple misload modeling assumptions reasonable given the provided rationale. The staff has identified implementation activities that the licensee has committed to (Reference 6) that are associated with its fuel handling procedures and training program (see Section 3.9).

For the Region 1 and Region 2 multiple misloading analysis, the licensee modeled an 8x10 array of fresh 4.85 wt % U-235 fuel assemblies without burnable poisons. The analyzed accident has been reasonably defined as a misloading of 80 fresh unpoisoned fuel assemblies. The Region 1 and 2 racks are the same design with different loading requirements. Therefore, the multiple misloading would have the same effect in both regions. The licensee determined 2100 ppm of soluble boron was sufficient to maintain K_{eff} less than or equal to 0.95 at a 95-percent probability, 95-percent confidence level.

For the Region 3 and Region 4 multiple misloading analysis, the licensee modeled a 9x10 array of 4.2 wt % U-235 fuel assemblies with 10 GWD/MTU of burnup. This represents approximately 31 GWD/MTU less burnup than the requirement for 4.2 wt % U-235 fuel in Region 3 and approximately 16 GWD/MTU less burnup than the requirement for 4.2 wt % U-235 fuel in Region 4; note that the model also assumed all required empty storage cells were filled. The analyzed accident has been reasonably defined as a misloading of 90 under burned fuel assemblies. The Region 3 and 4 racks are the same design with different loading requirements. Therefore, the multiple misloading would have the same effect in both Regions. The licensee determined 2100 ppm of soluble boron was sufficient to maintain K_{eff} less than or equal to 0.95 at a 95-percent probability, 95-percent confidence level.

The licensee's RAI response also included a revised misloading of a single fresh fuel assembly in Region 3 and 4. The licensee determined 2000 ppm of soluble boron was sufficient to maintain K_{eff} less than or equal to 0.95 at a 95-percent probability, 95-percent confidence level with a single fresh fuel assembly misloaded in Region 3 or 4.

The staff finds the licensee's determination of the soluble boron requirements under normal and abnormal/accident conditions is consistent with current guidance, and sufficiently conservative to provide reasonable assurance that the regulatory requirement is met.

The 2100 ppm of soluble boron required to mitigate the multiple misloading accident is greater than the current MPS2 TS requirement. The proposed change to TS to increase the required soluble boron to 2100 ppm is supported by the licensee's analysis. Therefore, the staff finds the proposed TS change acceptable.

The SFP boron dilution event is based on the ability of the licensee to detect and terminate the event before reaching the soluble boron required to maintain K_{eff} less than 0.95 under normal conditions (i.e., 600 ppm). Following this licensee amendment that soluble boron requirement

will remain unchanged at 600 ppm. The licensee is proposing to increase the TS required soluble boron to 2100 ppm to accommodate the multiple misloading accident. This is an increase from the current 1720 ppm requirement. The licensee's current boron dilution analysis is based on the event starting at 1720 ppm and being terminated before reaching 600 ppm. The staff finds that since no new boron dilution sources are introduced by this license amendment and the new initial starting soluble boron concentration will be higher, the licensee's current boron dilution analysis remains bounding.

3.4.6 Margin Analysis and Comparison with Remaining Uncertainties

This section provides evaluation of additional conservatism in the licensee's analysis and evaluation of items that may have been treated non-conservatively.

3.4.6.1 Potential Non-Conservatisms

From the review of the RAI responses provided in Reference 4, this section describes the potential non-conservatisms associated with the SFP criticality analysis.

- The licensee performed a validation study to quantify any potential temperature dependent bias in K_{eff} calculations. The staff determined that the bias calculated from the data was increased by one standard deviation. To be consistent with acceptable uncertainty analysis practice, the bias should have been determined from the linear regression slope after increasing the slope by two standard deviations. Over the full temperature range of the analysis, this small underestimate of the bias would be worth no more than 0.00051 Δk.

- The K_{eff} validation suite did not include any non-HTC MOX experiments. The licensee evaluated additional non-HTC MOX experiments, but then concluded that, since non-HTC MOX experiments made the bias less conservative, that the non-HTC MOX experiments could be excluded from the bias and uncertainty calculation. The staff determined that a figure provided by the licensee showing the non-HTC MOX experiment results shows that including the non-HTC MOX experiments, which appear to have a larger population variance, might significantly increase the 95/95 bias uncertainty. It is unlikely that including the non-HTC MOX experiments would increase the uncertainty by 50%, but this value is assumed for purposes of quantifying the potential non-conservatism. This would increase the overall K_{eff} validation 95/95 bias uncertainty by 0.0015 Δk. Some of this would be offset by the reduction in the bias term and by the quadratic combination with other uncertainties. However, for analysis of potential non-conservatisms, a value of 0.0015 Δk is used.

- Uncertainties or biases associated with potential radial burnup gradients were not included. Instead, as discussed in Section 3.4.3.3.3 the NRC staff estimated the effect could be worth up to 0.002 Δk.

- Sensitivity cases were performed to justify the assumption that it is conservative to model the Boraflex as water. Confirmatory calculations were performed comparing Boraflex modeled as water to Boraflex modeled with the B_4C replaced with water and the

rest of the Boraflex material in place. These calculations showed that the Boraflex water-only model yielded a K_{eff} value that was 0.0005 Δk lower than the Boraflex with B_4C removed model.

The total potential non-conservatisms are determined using a standard summation approach and result in 0.0045 Δk. This potential non-conservatism is offset by the analysis conservatisms described in the next section.

3.4.6.2 Analysis Conservatisms

The analysis includes several aspects that add margin to the analysis. These include an administrative margin of 0.005 Δk that the licensee proposes for NRC purposes and an additional margin of 0.0041 Δk that the licensee proposes. The additional margin represents margin to the regulatory limit after the biases and total 95/95 uncertainty are applied and is based on the smallest margin observed for Region 2 with Region 2A 4.85 wt % fuel burned to 16 GWd/MTU and Region 2B 4.5 wt % fuel burned to 30 GWd/MTU.

The staff finds that there is adequate margin in the licensee's analysis to demonstrate compliance with regulatory limits.

3.4.6.3 Conclusion on Analysis of Margins

Considering both the potential non-conservatisms identified in Section 3.4.6.1 and the conservatisms identified in Section 3.4.6.2 of this SE, the staff concludes that the available margin offsets the potential non-conservatisms and is, therefore, acceptable.

3.5 The MPS2 NFSR NCS Analysis

Section 5 of the NCS analysis covers new fuel storage in the NFSR. This section documents the review of the NFSR NCS analysis.

3.5.1 NFSR NCS Analysis Method

Compliance with 10 CFR 50.68(b)(2) and 10 CFR 50.68(b)(3) is demonstrated in the criticality analysis dated November 2012, which was provided as Attachment 4 to the December 17, 2012, letter and updated through the subsequent supplements, modeling a simplified representation of fresh fuel stored in the NFSR. The licensee's analysis demonstrates that the K_{eff} value, including bias and uncertainties, for both full density water and optimum moderation conditions is more than 0.01 Δk below the applicable limits from 10 CFR 50.68.

The SCALE 6.0 CSAS5 KENO V.a-based criticality analysis sequence and the SCALE ENDF/B-VII 238 neutron energy group library were used in the licensee's analysis to calculate the K_{eff} value for fresh fuel in the NFSR fuel storage racks. The CSAS5 sequence has a long history of use for this type of analysis and, therefore, the staff finds it acceptable.

The CSAS5 sequence and the ENDF/B-VII 238 group library validation study is presented in Appendix A of the criticality analysis (Attachment 4 to Reference 4) and the RAI 20 response

states the bias and 95/95 bias uncertainty from the validation are 0.007 Δk ± 0.0060. This bias and bias uncertainty are similar to values reported for other analyses and, therefore, the staff finds it acceptable.

Other than LEU-COMP-THERM-046, the critical experiments used in the validation study were near room temperature or 293 °K. In response to RAI 42, the licensee performed an additional validation study to estimate the bias associated with elevated temperature K_{eff} calculations. This study modeled 17 critical configurations from the IHECSBE LEU-COMP-THERM-046 evaluation that had temperatures ranging from 287.22 °K to 358.31 °K. The validation study quantified the change in the bias as a function of temperature to be -1.14e-5 +/-0.56e-5 Δk/°K, with the uncertainty reported at a 95-percent confidence value. This results in a bias sensitivity of -1.70e-5 Δk per degree-K at the upper 95% confidence interval limit. Thus, the licensee-calculated temperature bias is zero at 293 °K and -0.0014 Δk at 373 °K. Since the sign of the bias sensitivity is negative, this indicates the possibility of a slight K_{eff} under prediction at elevated temperatures; therefore, the licensee added 0.0014 Δk to the K_{eff} results at 373 °K.

The NRC staff performed a confirmatory calculation of the slope of the bias using an inverse variance (including both Monte Carlo and critical experiment uncertainties) weighted linear least square fit which yielded a value of -1.14e-5 +/- 0.61e-5 Δk/°K, with the uncertainty being one standard deviation. Thus, the bias sensitivity was calculated to be -2.36e-5 Δk/°K at the upper 95% confidence interval limit and the bias at 373 °K was estimated to be as high as -0.0019 Δk. The difference between the licensee's calculation and the NRC staff's confirmatory calculation indicates that the temperature bias may be underestimated by as much as 0.0005 Δk. Since there is adequate margin to cover this possible non-conservatism, the NRC staff considers the slight non-conservatism acceptable.

The NRC staff reviewed the computational method and supporting validation described above and based on the results finds them acceptable.

3.5.2 NFSR Fuel Storage Racks

The steel structures that comprise the NFSR fuel storage racks were conservatively not modeled. Without the steel, the rack model simplifies down to constraints on the spacing and location of the fuel assemblies. All rack structures were modeled as water at the calculation-specific water density. Fuel was modeled at minimum spacing and as sitting on the floor of the NFSR. Since worst case spacing and location were modeled, uncertainty analysis for the fuel storage rack parameters is not needed.

In response to RAI 17, a conservative concrete composition model was used in the revised analysis. The staff finds the NFSR rack and concrete wall and floor models are conservative and, therefore, acceptable.

3.5.3 Fuel Assemblies

The revised criticality analysis performed for this LAR included a sensitivity study at both full water density and at the water density where the optimum moderation peak occurred (i.e., 4% of

full density) for all fuel types ever used at the MPS2. The most reactive design was used for the NFSR criticality design.

The fuel assembly design is conservative in that natural uranium blankets and fuel rods containing Gd_2O_3 are modeled with unpoisoned 5.0 wt % U-235. Since the bounding maximum fuel enrichment was used in the criticality analysis, the impact of using the planar average enrichment in the NFSR TS is bounded. Other than the active lengths of the fuel rods, other fuel assembly components were replaced with water. The staff finds this approach is conservative for NFSR analysis.

Based on the NRC staff's review of the above, the fuel assembly model used in the NFSR NCS analysis is conservative and acceptable.

3.5.4 Analysis of Margins

The licensee determined that the maximum K_{eff} value, including biases and uncertainties, for the full density flooding case was 0.9365. The applicable limit is 0.95. The margin to this limit is 0.0135 Δk. The licensee determined that the maximum K_{eff} value, including biases and uncertainties, for the optimum moderation case was 0.9397. The applicable limit is 0.98. Thus, the margin to this limit is 0.0403 Δk and the NRC staff finds this acceptable.

One small potential non-conservatism was identified in the review. It is possible that the temperature bias at 373 °K should have been 0.0005 Δk higher. However, the staff finds that the margins to the limits are large enough to cover this potential non-conservatism.

3.6 Elimination of Boraflex Credit

Boraflex is used in the spent fuel storage racks for absorption of neutrons. Degradation of the Boraflex in the racks could result in an increase in the reactivity of the spent fuel configuration. The licensee justified their request to stop crediting Boraflex in Regions 1 and 2 in the SFP by performing criticality and accident analyses. The licensee will credit soluble boron and fuel placement methodology for SFP criticality control. These analyses were done in accordance with 10 CFR 50.68(b)(4) "Criticality accident requirements" acceptance criteria.

The staff evaluated whether or not the changes proposed by the licensee are sufficient to maintain sub criticality without crediting Boraflex. The proposed methodology for fuel placement in the SFP racks in combination with the soluble boron was found to be an acceptable means for the licensee to maintain sub criticality. The licensee's criticality analysis report as supplemented contained the acceptance criteria, assumptions made, design and input data, the methodology, and the results which the staff found to be in agreement with 10 CFR 50.68 requirements for the safe operation of the SFP. The report also included an analysis of possible risks and accidents that are associated with the proposed methods to control criticality in the SFP. The proposed TS states that the SFP boron concentration shall be at least 600 ppm, that the licensee relies on the use of the developed burn-up versus enrichment curves when following the fuel placement methodology, and that the licensee follows the fuel loading rules proposed in the LAR.

The staff has reviewed the impact of the licensee's request to remove Boraflex credit at MPS2 on the GL 96-04 commitments to monitor Boraflex degradation for MPS2. The proposed criticality analysis methodology does not take credit for Boraflex; therefore, the staff finds the proposed changes are acceptable and concludes that Boraflex credit including the GL 96-04 commitments to monitor Boraflex degradation for MPS2 will no longer be necessary upon the staff's approval of the criticality analysis methodology.

3.7 Borated Stainless Steel Rodlets

Three rodlets are already inserted into the guide tubes of each fuel assembly stored in Region 3 of the pool and whose enrichment-burnup characteristics conform to the requirements given in Figure 3.9-1C of the modified TS. The rodlets are made of borated stainless steel containing 2 weight percent of boron.

The poison rodlets for MPS2 consist of solid cylinders. The diameter of rodlets is smaller by 0.08 inches and it is 1.75 inches longer than the individual fingers in a CEA. Since nominal internal diameter of the guide tube is 1.035 inches over most of its fuel length, a clearance of 82 mils exists between the rodlet and the guide tube walls with the exception of the dashpot region where the clearance is 48 mils. This clearance is larger than the corresponding clearance for the fingers of the CEA and any crud accumulated in the gap should not interfere with removal or insertion of the rodlets.

Carpenter Technology Corporation manufactured the type 304 B7, Grade A, borated SS rodlets (2 wt % boron). They were manufactured in accordance with the requirements of standards ASTM A 887-89 and ASTM A 484-91. Borated stainless steel is a two phase alloy, composed of a complex boride phase in an austenitic chromium-nickel-iron matrix. In general, the physical properties of the material resemble those of 304 austenitic stainless steel. However, the yield strength, ultimate tensile strength and hardness increase with increasing levels of boron and ductility and impact strength are decreased. These properties also vary with the exposure to neutron fluency, but no significant changes occur for neutron fluences below 10^{17} neutron/square centimeter (n/cm^2). This value is much higher than $2x10^{12}$ n/cm^2 which is the maximum anticipated neutron fluency reached during the lifetime of the SFP in MPS2 (which includes the original 40 year service life plus an additional 20 years for life extension and another 20 years for extended storage in the SFP). Since poison rodlets do not carry any loads when inserted in the guide tubes, mechanical properties of the material are not of primary importance.

Although intergranular corrosion resistance of borated stainless steel exposed to acidic conditions decreases with increased boron content, long term tests with borated stainless steel have indicated that no measurable corrosion effects take place in the SFP environment. It is not expected, therefore, that any significant corrosion degradation of poison rodlets will occur during their service life. However, in order to have assurance that at all times, there is enough poison material for reactivity control, the licensee stated in its RAI response dated May 28, 2013 (Reference 3) that it has confirmed that their program for surveillance and inspection is consistent with that described in the letter dated July 16, 1993, from Northeast Nuclear Energy Company to the NRC (Reference 20). This surveillance program, at approximately 5-year intervals, will have 1-percent of the rodlets visually inspected for any material degradation.

Currently, 27 rodlets, which have been in continuous use at MPS2 since installation in 1994, are designated as surveillance rodlets. These rodlets are re-inspected during subsequent periods to permit trending.

Based on the above evaluation, the staff concludes that the poison rodlets made of borated stainless steel, with the material characteristics described in the licensee's submittals, will resist material degradation in the SFP environment. Verification and trending of their conditions by periodic inspections provides reasonable assurance that the integrity of the neutron absorbing material required for reactivity control will not diminish from that assumed in the analysis. Therefore, the staff concludes that the continued use of poison rodlets made of borated stainless steel is acceptable.

3.8 Conclusions

The NRC staff has completed its review of the MPS2 SFP and NFSR NCS analyses, which are documented in the licensees December 17, 2012, application and updated through the subsequent supplements, and concludes that there is a reasonable assurance that the MPS2 SFP and NFSR fuel storage racks meet the applicable regulatory requirements in 10 CFR 50.68.

In addition, the staff concludes that conservatisms and potential non-conservatisms have been adequately addressed in the analysis via licensee margins and confirms compliance with NRC regulatory limits.

The analysis and methodology used in the licensee's application is unique to the MPS2 SFP. Additionally, the NRCs evaluation included consideration of offsetting effects and is therefore specific to and applicable to MPS2 only. There are several aspects about the analysis and methodology that will make future changes complex. Therefore, the licensee should ensure full and appropriate consideration of these offsetting effects when evaluating potential changes to the facility and methods of evaluation under the auspices of 10 CFR 50.59 that may impact these criticality analyses.

Additionally, the staff has reviewed the impact of the licensee's request to remove Boraflex credit at MPS2 on the GL 96-04 commitments to monitor Boraflex degradation for MPS2. The proposed criticality analysis methodology does not take credit for Boraflex; therefore, the staff finds the proposed changes are acceptable and concludes that the GL 96-04 commitments will no longer be necessary upon the approval of the criticality analysis methodology proposed for MPS2.

Based on its review of the licensee's program for surveillance and inspection of the borated stainless steel rodlets, the staff also determined that the borated stainless steel rodlets will be adequately monitored for use in the SFP. The staff has reasonable assurance that the program for surveillance and inspection, which includes visual inspection, will be able to detect potential degradation of the borated stainless steel before it could impair its neutron absorption capability. Therefore, the staff concludes that the use of borated stainless steel as a neutron absorber rodlet in the spent fuel racks is acceptable with the use of the program for surveillance and inspection.

3.9 Implementation Activities

As stated in its letter dated June 1, 2016 (Reference 6), the licensee will ensure the following activities are performed prior to implementation of Amendment 327:

1) The licensee will ensure that its fuel handling procedures include the stipulation that the reactivity control devices must be installed prior to placement of the fuel into the Region 3 storage locations (Section 3.2.1).

2) The licensee will ensure that its core reload procedures include the stipulation that the burnup history, core operating parameters, burnable absorber use, and CEA usage remains bounded by this criticality safety analysis (Section 3.4.3.3.4, 3.4.3.3.5, and 3.4.3.3.6).

3) The licensee will ensure that its fuel reconstitution procedures include the stipulations (1) the fuel assembly being reconstituted will be neutronically decoupled from the other fuel assemblies in the SFP, (2) only one fuel rod will be removed at a time, (3) the original fuel rod will be returned to its original location, or it will be replaced with a stainless steel rod or a fuel rod with naturally occurring U-235, or if the entire rod cannot be removed or the removed rod cannot be replaced, then an analysis similar to that performed in Attachment 4 of the licensee's supplemental letter dated July 21, 2015 (Reference 4) for fuel assembly P-26 may be performed (Section 3.4.4).

4) The licensee will ensure that its fuel handling procedures and training program are updated to include restrictions on storing fresh fuel in Regions 3 and 4, visual cues to allow error recognition in move sheets, and training to these procedures are completed (Section 3.4.5).

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified on December 7, 2015, of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in 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 published in the Federal Register (FR) on June 11, 2013 (78 FR 35060) that the amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or

environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) 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 J. A. Price, Dominion Nuclear Connecticut, Inc. to Document Control Desk, U. S. Nuclear Regulatory Commission on "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 2, License Amendment Request Regarding Proposed Technical Specifications Changes for Spent Fuel Storage," Serial No. 12-678, dated December 17, 2012 (ADAMS Accession Nos. ML12362A391 and ML12362A392).

2. Letter from E. S. Grecheck, Dominion Nuclear Connecticut, Inc., to Document Control Desk, U. S. NRC, on "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 2, Supplement to License Amendment Request Regarding Proposed Technical Specifications Changes for Spent Fuel Storage," Serial No. 12-678A, dated February 25, 2013 (response to Feb. 11, 2013 NRC ltr) (ADAMS Accession No. ML13064A380).

3. Letter from E. S. Grecheck, Dominion Nuclear Connecticut, Inc., to Document Control Desk, U. S. NRC, on "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 2, Response to Request for Additional Information Regarding Proposed Technical Specifications Changes for Spent Fuel Storage," Serial No. 13-283, dated May 28, 2013 (response to Apr 26 RAI to NRC) (ADAMS Accession No. ML13155A140).

4. Letter from M. D. Sartain, Dominion Nuclear Connecticut, Inc. to Document Control Desk, U. S. Nuclear Regulatory Commission on "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 2, Response to Second Request for Additional Information Regarding Proposed Technical Specification Change for Spent Fuel Storage," Serial No. 14-625, dated July 21, 2015 (ADAMS Accession No. ML15209A729).

5. Letter from M. D. Sartain, Dominion Nuclear Connecticut, Inc., to Document Control Desk, U. S. NRC, on "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 2, Elimination of the Boraflex Monitoring Program," December 18, 2015 (ADAMS Accession No. ML15357A028).

6. Letter from M. D. Sartain, Dominion Nuclear Connecticut, Inc. to Document Control Desk, U.S. Nuclear Regulatory Commission on "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 2, Regulatory Commitments for Proposed Technical Specification

Change for Spent Fuel Storage," Serial No. 16-224, dated June 1, 2016 (ADAMS Accession No. ML16159A220).

7. E-mail from R. Guzman to W. Craft, "NRC Staff Request for Additional Information - Proposed TS Changes for MPS2 Spent Fuel Storage (TAC No. MF0435)," July 20, 2015 (ADAMS Accession No. ML15201A767) (in reference to June 11, 2014 draft RAI, ADAMS Accession No. ML15197A191).

8. Letter from T. L. Harpster to T.C Feigenbaum, Connecticut Yankee Atomic Power Company, Northeast Nuclear Energy Company, to USNRC, "Haddam Neck Plant, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, Response to Generic Letter 96-04, Boraflex Degradation in Spent Fuel Storage Racks," October 24, 1996.

9. NRC Memorandum from L. Kopp to T. Collins, Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998 (ADAMS Accession No. ML003728001).

10. "Final Division of Safety Systems Interim Staff Guidance, DSS-ISG-2010-01, Rev. 0, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," U. S. Nuclear Regulatory Commission, Division of Safety Systems (ADAMS Accession No. ML110620086); Notice of availability published in the Federal Register on October 13, 2011 (76 FR 64676).

11. "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," ORNL/TM-2005/39, Version 6, Vols. I–III, Oak Ridge National Laboratory, Oak Ridge, Tennessee, January 2009. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-750.

12. G. Radulescu, I. C. Gauld, G. Ilas, and J. C. Wagner, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses-Isotopic Composition Predictions," NUREG/CR-7108 (ORNL/TM-2011/509), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, April 2012 (ADAMS Accession No. ML12116A124).

13. D.E. Mueller, K.R. Elam, and P.B. Fox, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," NUREG/CR-6979, ORNL/TM-2007/083, U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, September 2008 (ADAMS Accession No. ML082880452).

14. "International Handbook of Evaluated Criticality Safety Benchmark Experiments," NEA/NSC/DOC(95)03/I-IX, Organization for Economic Co-operation and Development, Nuclear Energy Agency, 2010.

15. J.C. Wagner, M.D. DeHart, and C.V. Parks, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis," NUREG/CR-6801 (ORNL/TM-2001/273), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 2003 (ADAMS Accession No. ML031110292).

16. C. Kang, et al., "Horizontal Burnup Gradient Datafile for PWR Assemblies," DOE/RW-0496, U.S. Department of Energy, Office of Civilian Radioactive Waste Management, [http://web.ornl.gov/sci/nsed/rnsd/BUCPublications._files/HorizontalBurnupGradient Database_DOE-RW-0496.pdf], May 1997.

17. J.C. Wagner and C.E. Sanders, "Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs," NUREG/CR-6800 (ORNL/TM-2002/6), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 2003 (ADAMS Accession No. ML031110280).

18. C.V. Parks, M.D. DeHart, and J.C. Wagner, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," NUREG/CR-6665 (ORNL/TM-1999/303), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, February 2000 (ADAMS Accession No. ML003688150).

19. C. E. Sanders and J. C. Wagner, "Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit," NUREG/CR-6760 (ORNL/TM-2000/321), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 2002 (ADAMS Accession No. ML020770436).

20. Letter from J. F. Opeka, Northeast Nuclear Energy Company to the Nuclear Regulatory Commission, "Millstone Nuclear Power Station Unit 2, Response to Request for Additional Information," July 16, 1993.

Principal Contributors: K. Wood
E. Wong

Date: June 23, 2016

June 23, 2016

Mr. David A. Heacock
President and Chief Nuclear Officer
Dominion Nuclear
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION CHANGES FOR SPENT FUEL STORAGE (TAC NO. MF0435)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 327 to Renewed Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2 (MPS2), in response to your application dated December 17, 2012, as supplemented on February 25, 2013, May 28, 2013, July 21, 2015, December 18, 2015, and June 1, 2016.

The amendment revises the MPS2 Technical Specifications (TSs) to reflect the results and constraints of a new criticality safety analysis for fuel assembly storage in the MPS2 fuel storage racks. Specifically, the amendment revises TS 1.39 "Storage Pattern," TS 3.9.18, "Spent Fuel Pool - Storage," TS 3.9.19, "Spent Fuel Pool - Storage Patterns," TS 5.3.1 "Fuel Assemblies," TS 5.6.1, "Criticality," and TS 5.6.3, "Capacity." The amendment will implement the following items associated with fuel storage at MPS2: (1) allow removal of Boraflex credit; (2) eliminate reactivity credit for Boraflex panels in current regions A and B of the spent fuel pool; (3) revise allowed storage patterns for fuel assemblies in the spent fuel pool to meet effective neutron multiplication factor (K_{eff}) requirements under normal and accident conditions; (4) revise alphanumeric designation of spent fuel regions from Regions A, B, and C to Regions 1, 2, 3, and 4 to clearly distinguish from existing designations; (5) allow use of control element assemblies as well as borated stainless steel poison rodlets in Region 3; and (6) eliminate requirement to use spent fuel rack-cell blocking devices.

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

Sincerely,
/RA/
Richard V. Guzman, Sr. Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 327 to DPR-65
2. Safety Evaluation

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K. Wood, NRR

Accession No.: ML16003A008 *Safety evaluation input provided by memos. No substantial changes made.

OFFICE	DORL/LPL1-1/PM	DORL/LPL1-1/LA	DSS/SRXB/BC	DE/ESGB/BC
NAME	RGuzman	KGoldstein	CJackson*	GKulesa*
DATE	5/31/16	1/11/16	10/15/15	08/27/13
OFFICE	DSS/STSB/BC	OGC	DORL/LPL1-1/BC	DORL/LPL1-1/PM
NAME	RElliott	CKanatas	TTate	RGuzman
DATE	1/21/16	6/20/16	6/23/16	6/23/16

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