



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
May 28, 2019

Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Dominion Nuclear
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 – ISSUANCE OF AMENDMENT
NO. 273 REGARDING TECHNICAL SPECIFICATION CHANGES FOR SPENT
FUEL STORAGE AND NEW FUEL STORAGE (EPID L-2018-LLA-0126)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 273 to Renewed Facility Operating License No. NPF-49 for Millstone Power Station, Unit No. 3 (Millstone 3), in response to your application dated May 3, 2018, as supplemented by letters dated November 29, 2018; March 27, 2019; and May 7, 2019.

The amendment revises the Technical Specifications (TSs) to reflect the results and constraints of a new criticality safety analysis for fuel assembly storage in the Millstone 3 fuel storage racks. Specifically, the amendment implements the following items associated with fuel assembly storage at Millstone 3: (1) increases the TS minimum spent fuel pool soluble boron concentration, (2) revises allowed storage patterns and initial enrichment/burnup/decay time for fuel assemblies in the spent fuel pool to meet k_{eff} requirements under normal and accident conditions; (3) permits the storage of any fuel assembly with certain enrichment that contains a rod cluster control assembly in Region 2 without restriction, and (4) implements a revised criticality analysis for the new fuel storage racks using the updated methods for the spent fuel pool criticality analysis for consistency.

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,

A handwritten signature in black ink, appearing to read "Richard V. Guzman" followed by a small "for" written in a cursive style.

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

Docket No. 50-423

Enclosures:

1. Amendment No. 273 to NPF-49
2. Safety Evaluation

cc: Listserv



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

DOMINION ENERGY NUCLEAR CONNECTICUT, INC., ET AL

DOCKET NO. 50-423

MILLSTONE POWER STATION, UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 273
Renewed License No. NPF-49

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Dominion Energy Nuclear Connecticut, Inc. (the licensee) dated May 3, 2018, as supplemented on November 29, 2018, March 27, 2019, and May 7, 2019, 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.

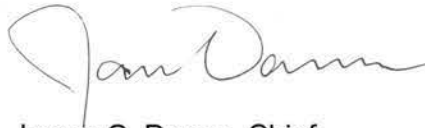
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-49 is hereby amended to read as follows:

- (2) Technical Specifications

The Technical Specifications contained in Appendix A, revised through Amendment No. 273 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated into the license. DENC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: May 28, 2019

Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: May 28, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 273

MILLSTONE POWER STATION, UNIT NO. 3

RENEWED FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

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 a marginal line indicating the area of change.

Remove
4

Insert
4

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 a marginal line indicating the areas of change.

Remove
1-7
3/4 9-1a
3/4 9-16

3/4 9-17
3/4 9-18
3/4 9-19
3/4 9-20
3/4 9-21
3/4 9-22
5-6
5-6a

Insert
1-7
3/4 9-1a
3/4 9-16
3/4 9-16a
3/4 9-17
3/4 9-18
3/4 9-19
3/4 9-20
3/4 9-21
3/4 9-22
5-6
5-6a

(1) Technical Specifications

The Technical Specifications contained in Appendix A, revised through Amendment No. 273 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated into the license. DENC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

- (3) DENC shall not take any action that would cause Dominion Energy, Inc. or its parent companies to void, cancel, or diminish DENC's Commitment to have sufficient funds available to fund an extended plant shutdown as represented in the application for approval of the transfer of the licenses for MPS Unit No. 3.
- (4) Immediately after the transfer of interests in MPS Unit No. 3 to DNC*, the amount in the decommissioning trust fund for MPS Unit No. 3 must, with respect to the interest in MPS Unit No. 3, that DNC* would then hold, be at a level no less than the formula amount under 10 CFR 50.75.
- (5) The decommissioning trust agreement for MPS Unit No. 3 at the time the transfer of the unit to DNC* is effected and thereafter is subject to the following:
- (a) The decommissioning trust agreement must be in a form acceptable to the NRC.
 - (b) With respect to the decommissioning trust fund, investments in the securities or other obligations of Dominion Energy, Inc. or its affiliates or subsidiaries, successors, or assigns are prohibited. Except for investments tied to market indexes or other non-nuclear-sector mutual funds, investments in any entity owning one or more nuclear power plants are prohibited.
 - (c) The decommissioning trust agreement for MPS Unit No. 3 must provide that no disbursements or payments from the trust, other than for ordinary administrative expenses, shall be made by the trustee until the trustee has first given the Director of the Office of Nuclear Reactor Regulation 30 days prior written notice of payment. The decommissioning trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the NRC.
 - (d) The decommissioning trust agreement must provide that the agreement cannot be amended in any material respect without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

* On May 12, 2017, the name "Dominion Nuclear Connecticut, Inc." changed to "Dominion Energy Nuclear Connecticut, Inc."

DEFINITIONS

VENTING

1.39 VENTING shall be 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.

1.40 Deleted

1.41 Deleted

CORE OPERATING LIMITS REPORT (COLR)

1.42 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Unit Operation within these operating limits is addressed in individual specifications.

1.43 Deleted

1.44 Deleted

REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

- 3.9.1.2 The soluble boron concentration of the Spent Fuel Pool shall be greater than or equal to 2600 ppm.

APPLICABILITY:

Whenever fuel assemblies are in the spent fuel pool.

ACTION:

- a. With the boron concentration less than 2600 ppm, initiate action to bring the boron concentration in the fuel pool to at least 2600 ppm within 72 hours, and
- b. With the boron concentration less than 2600 ppm, suspend the movement of all fuel assemblies within the spent fuel pool and loads over the spent fuel racks.

SURVEILLANCE REQUIREMENTS

- 4.9.1.2 Verify that the boron concentration in the fuel pool is greater than or equal to 2600 ppm at the frequency specified in the Surveillance Frequency Control Program.

REFUELING OPERATIONS

3/4.9.13 SPENT FUEL POOL - STORAGE

LIMITING CONDITION FOR OPERATION

3.9.13 The spent fuel storage requirements necessary to maintain K_{eff} within limits shall be met.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTION:

- a. For a fuel assembly stored in Region 1A - initiate immediate action to move any assembly which does not meet Surveillance Requirement 4.9.13.1.1 to Region 1B.
- b. For a fuel assembly stored in Region 2 that does not contain a Rod Cluster Control Assembly - initiate immediate action to move any assembly which does not meet the requirements of Figure 3.9-2 to a location for which that fuel assembly is allowed.
- c. For a fuel assembly stored in Region 3 - initiate immediate action to move any assembly which does not meet the requirements of Figure 3.9-3 to a location for which that fuel assembly is allowed.

SURVEILLANCE REQUIREMENTS

----- NOTE -----
The Region 1 Fuel Storage Loading Schematic (Figure 3.9-1) designates each storage location as either Region 1A or Region 1B.

Regarding fuel assemblies that contain a Rod Cluster Control Assembly for storage in Region 2 - if the enrichment and burnup of a given assembly is not in the "Acceptable" domain of Figure 3.9-2 (e.g., the assembly requires a Rod Cluster Control Assembly to be stored in Region 2), then the assembly must be located in an acceptable Region 1 storage location before its Rod Cluster Control Assembly can be inserted or removed.

Initial enrichment is the maximum initial planar volume averaged as-built U-235 enrichment in the assembly. If the assembly has axial blankets the lower enriched fuel is not credited in determining the enrichment. Also, fuel burnup is the volume averaged burnup of the assembly as determined using the measured reaction rates.

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

- 4.9.13.1.1. Ensure that all fuel assemblies to be placed into a Region 1A storage location, with an initial enrichment greater than 4.75 w/o U-235, have achieved a fuel burnup greater than or equal to 2.0 GWD/MTU or contain a minimum of twelve (12) Integral Fuel Burnable Absorber (IFBA) Rods by checking the fuel assembly's location, design, and burnup documentation. Fuel assemblies with an initial enrichment less than or equal to 4.75 w/o U-235 may be stored in Region 1A without restriction.
- 4.9.13.1.2. Ensure that all fuel assemblies to be placed in Region 1B are stored consistent with the Fuel Storage Loading Schematic specified in Figure 3.9-1 by checking the fuel assembly's storage location. All fuel assemblies with an initial enrichment less than or equal to 5.0 w/o U-235 may be stored in Region 1B without restriction.
- 4.9.13.1.3. Ensure that all fuel assemblies to be stored in Region 2 - that do not contain a Rod Cluster Control Assembly - are within the enrichment and burnup limits of Figure 3.9-2 by checking the fuel assembly's design and burnup documentation. A fuel assembly that contains a Rod Cluster Control Assembly may be stored in Region 2 without restriction.
- 4.9.13.1.4. Ensure that all fuel assemblies to be stored in Region 3 are within the enrichment, burnup, and decay time limits of Figure 3.9-3 by checking the fuel assembly's design, burnup, and decay time documentation.

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Figure 3.9-1 Region 1 Fuel Storage Loading Schematic

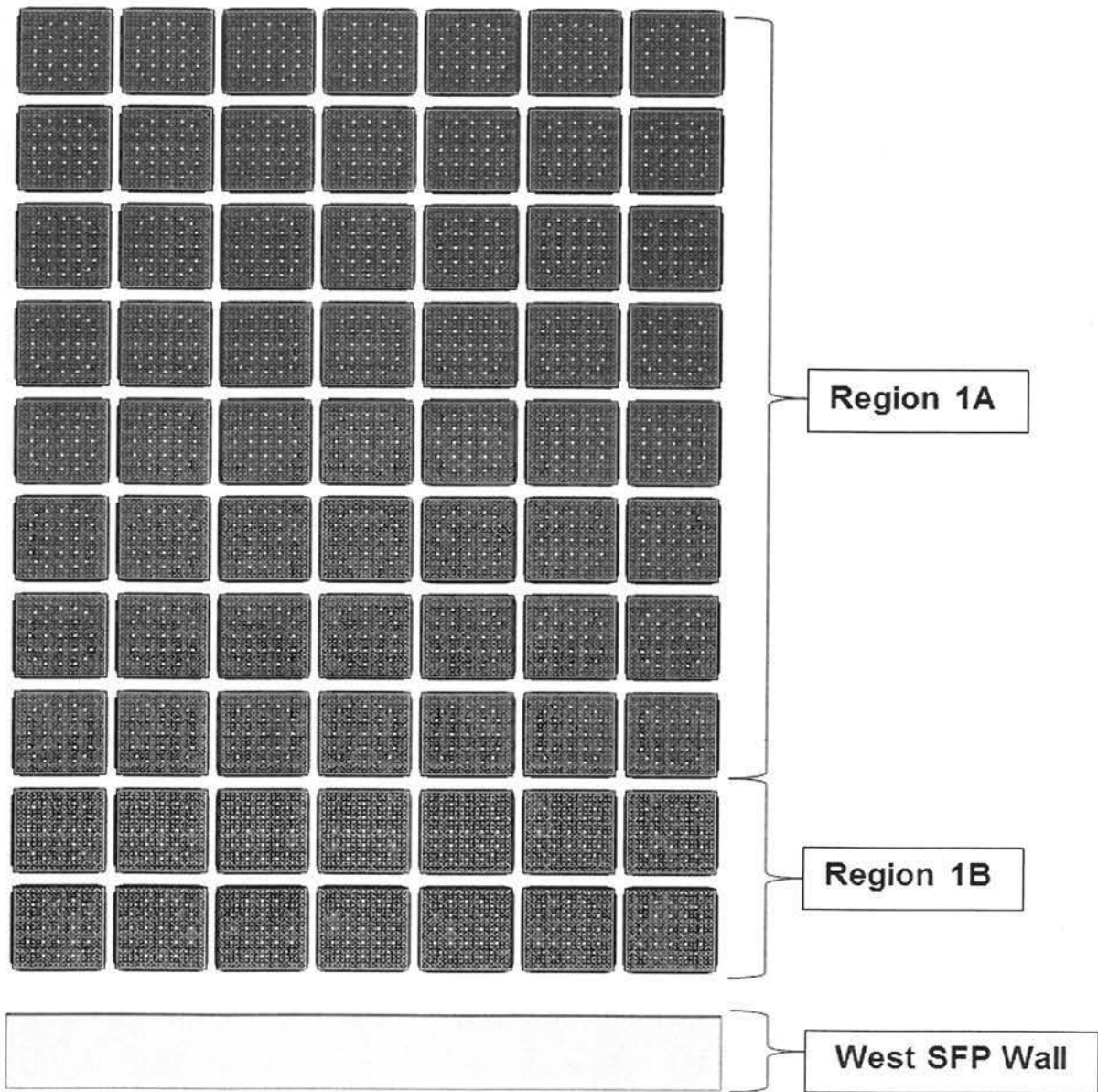


Figure 3.9-2 Minimum Fuel Assembly Burnup versus Initial Enrichment for Region 2
 Storage Configuration
 (Fuel Assemblies without Rod Cluster Control Assemblies)

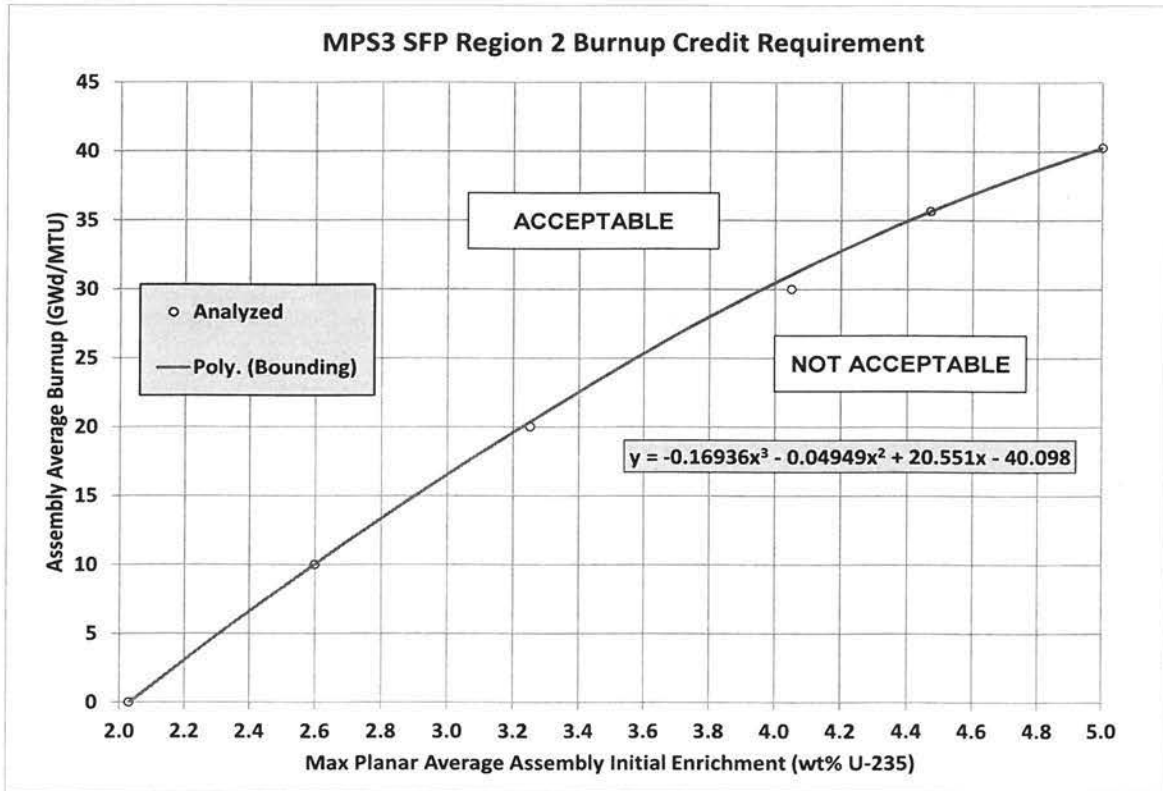
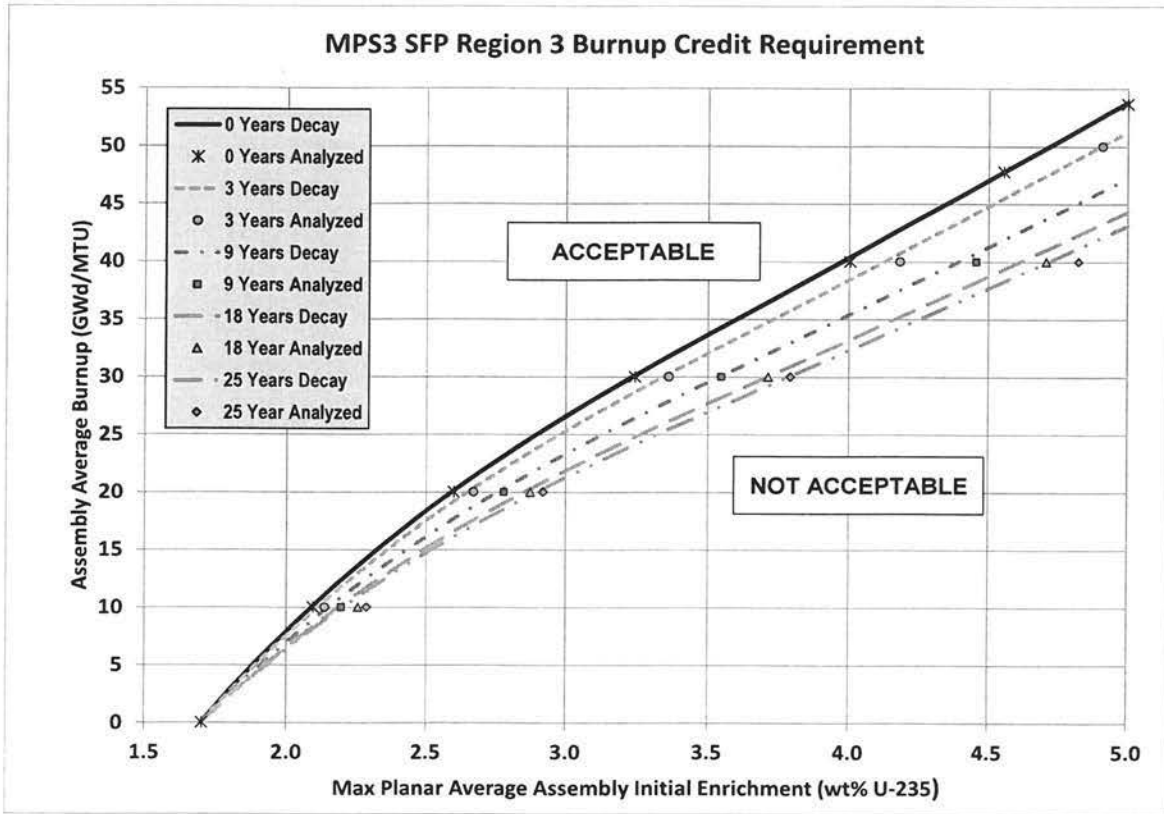


Figure 3.9-3 Minimum Fuel Assembly Burnup and Decay Time versus Initial Enrichment for Region 3 Storage Configuration



The burnup curve equations have the following polynomial format (bounding):

$$BU[GWD/MTU] = \alpha_4 * wt\%^4 + \alpha_3 * wt\%^3 + \alpha_2 * wt\%^2 + \alpha_1 * wt\%^1 + \alpha_0$$

Burnup Credit Curve Polynomial Coefficients

Region	Decay Time (Years)	α_4	α_3	α_2	α_1	α_0
3	No Credit	-0.2459	4.208	-26.80	88.70	-92.00
3	3 Years	-0.2338	4.001	-25.48	84.34	-87.47
3	9 Years	-0.2153	3.684	-23.46	77.66	-80.54
3	18 Years	-0.2020	3.458	-22.02	72.88	-75.59
3	25 Years	-0.1964	3.361	-21.40	70.84	-73.47

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5.6 FUEL STORAGE

CRITICALITY

- 5.6.1 The New Fuel Storage Racks, a nominal 22.125 inch center to center distance, credit a fixed neutron absorber (BORAL) within the rack and are designed and shall be maintained with:
- a. K_{eff} less than or equal to 0.95 with the storage racks fully loaded with the highest reactivity fuel and flooded with potential moderators,
 - b. K_{eff} less than or equal to 0.98 with the storage racks fully loaded with the highest reactivity fuel and optimum moderation of the racks.

The spent fuel storage racks are made up of 3 Regions which are designed and shall be maintained to ensure a K_{eff} less than 1.0 when flooded with unborated water, and K_{eff} less than or equal to 0.95 with 600 ppm soluble boron in the spent fuel pool water. The storage rack regions are as follows:

- a. Region 1, a nominal 10.0 inch (North/South) and a nominal 10.455 inch (East/West) center to center distance, credits a fixed neutron absorber (BORAL) within the rack. Each Region 1 fuel storage rack contains two storage sub-regions - Region 1A and Region 1B:
 - (1) Region 1A - Fuel assemblies meeting one of the following three criteria may be stored in Region 1A storage locations:
 - i. initial enrichment less than or equal to 4.75 w/o U-235, or
 - ii. initial enrichment less than or equal to 5.0 w/o U-235 with a fuel burnup greater than or equal to 2.0 GWD/MTU, or
 - iii. initial enrichment less than or equal to 5.0 w/o U-235 that contain a minimum of 12 Integral Fuel Burnable Absorber (IFBA) rods.
 - (2) Region 1B - Fuel assemblies with an initial enrichment less than or equal to 5.0 w/o U-235 shall be stored per the Fuel Storage Loading Schematic shown in Figure 3.9-1 (the two rows against the spent fuel pool west wall are designated Region 1B).
- b. Region 2, a nominal 9.017 inch center to center distance, credits a fixed neutron absorber (BORAL) within the rack and either fuel burnup as shown in Figure 3.9-2 or takes credit for containing a Rod Cluster Control Assembly.
- c. Region 3, a nominal 10.35 inch center to center distance, credits fuel burnup and decay time as shown in Figure 3.9-3. These racks contain Boraflex which is not credited.

DESIGN FEATURES

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 45 feet.

CAPACITY

5.6.3 The spent fuel storage pool contains 350 Region 1 storage locations, 673 Region 2 storage locations and 756 Region 3 storage locations, for a total of 1779 fuel storage locations. An additional Region 2 rack with 81 storage locations may be placed in the spent fuel pool, if needed. With this additional rack installed, the Region 2 storage capacity is 754 storage locations. The total storage capacity of the spent fuel pool is limited to no more than 1860 fuel assemblies.

5.7 DELETED



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 273

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-49

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated May 3, 2018, as supplemented by letters dated November 29, 2018; March 27, 2019; and May 7, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18128A049, ML18340A028, ML19092A332, and ML19135A067, respectively), Dominion Energy Nuclear Connecticut, Inc. (the licensee) submitted a license amendment request (LAR) to modify the Technical Specifications (TSs) at Millstone Power Station, Unit No. 3 (Millstone 3 or MPS3).

The amendment would revise the TSs to reflect the results and constraints of a new criticality safety analysis for fuel assembly storage in the Millstone 3 fuel storage racks. Specifically, the amendment would implement the following items associated with fuel assembly storage at Millstone 3: (1) increase the TS minimum spent fuel pool (SFP) soluble boron concentration, (2) revise allowed storage patterns and initial enrichment/burnup/decay time for fuel assemblies in the SFP to meet k_{eff} requirements under normal and accident conditions, (3) permit the storage of any fuel assembly with certain enrichment that contains a rod cluster control assembly in Region 2 without restriction, and (4) implement a revised criticality analysis for the new fuel storage racks using the updated methods for the SFP criticality analysis for consistency.

The supplemental letters dated November 29, 2018; March 27, 2019; and May 7, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission's (NRC or the Commission) original proposed no significant hazards consideration determination as published in the *Federal Register* on August 7, 2018 (83 FR 38735).

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance documents that the NRC staff used in the review of the LAR are listed below.

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC). GDC 61, "Fuel storage and handling and radioactivity Control," requires in part, that, "These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety." GDC 62, "Prevention of criticality in fuel storage and handling," 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), and accordingly, must comply with the following 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 regulations in 10 CFR 50.36, "Technical specifications," contain the requirements for the content of TSs. The regulations in 10 CFR 50.36(b) require TSs to be derived from the analyses and evaluation included in the safety analysis report and amendments thereto. As required by 10 CFR 50.36(c)(4), the TSs will include design 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.

The NRC staff also reviewed the proposed LAR against the guidance of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," and Section 9.1.2, "New and Spent Fuel Storage." These sections of NUREG-0800 provide guidance regarding the specific acceptance criteria and review procedures to ensure that the proposed changes satisfy the requirements in 10 CFR 50.68 and GDC 62. Additionally, Nuclear Energy Institute (NEI) Topical Report 16-03-A, "Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools" (NEI 16-03-A), is endorsed by the NRC staff as guidance to the industry for developing adequate monitoring programs for fixed neutron absorbers in SFPs.

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 Millstone 3 SFP. The request also sought to revise the TSs for the new fuel storage area (NFSA). This evaluation presents the results of the NRC staff's review of the nuclear criticality safety (NCS) analysis, which was provided as Attachment 5 to the LAR and updated through the supplements. The criticality safety basis is thus composed of both the original criticality analysis and the licensee's supplements in response to the NRC's requests for additional information (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 NFSA, the analysis shows that NFSA rack maximum k_{eff} will be no greater than 0.95 when the NFSA is flooded with unborated water at a 95 percent probability, 95 percent confidence level, and will be no greater than 0.98 if the NFSA is flooded with low density water (i.e., at optimum moderation conditions) at a 95 percent probability, 95 percent confidence level.

There are three storage rack designs in the Millstone 3 SFP that are grouped into three areas designated as Regions 1, 2, and 3. The Regions 1 and 2 storage racks are of the typical SFP two-region design for a pressurized-water reactor (PWR). The basic component is stainless steel boxes with a neutron-absorbing material (NAM) attached to the outside of each face. In this case, the NAM is Boral. For Region 1, the boxes are attached to a structural grid that provides some space between each box. The resultant array is attached to a base plate, forming a storage module. Region 1 has two sheets of NAM and a water gap between each storage cell. Region 1 is primarily for the storage of fresh or lightly burned fuel. Region 2 has similar boxes attached at the corners, forming what is called an "egg crate," with formed cells being created between the manufactured cells. Region 2 has one sheet of NAM and no water gap between the storage cells. Region 2 is primarily intended for storage of fuel that had completed its service life. Region 3 is similar to Region 1, except the NAM is Boraflex, and 6x6 arrays of cells are bordered by a stainless steel wall.

New fuel assemblies may be stored in what are normally dry conditions in the Millstone 3 NFSA. The licensee submitted a new NCS analysis for the NFSA but did not propose any revisions to the NFSA TSs.

3.2 Proposed Changes

3.2.1 NCS Analyses and Fuel Storage Requirements

The proposed SFP TSs significantly revise the organization and storage requirements for the SFP. The current Millstone 3 TSs divide the SFP into Regions 1, 2, and 3. The proposed TSs would retain the three-region designation, but with new storage requirements for each region. Specifically, as provided in Attachment 2 to the LAR, the amendment would revise the following TSs and TS figures:

- TS 1.40
- TS 1.41
- TS 3/4.9.1.2
- TS 3/4.9.13
- TS 3/4.9.14
- TS Figure 3.9-1
- TS Figure 3.9-2
- TS Figure 3.9-3
- TS Figure 3.9-4
- TS Figure 3.9-5
- TS 5.6.1.1

There are several proposed TS changes that either impact NCS analyses or implement changes in fuel storage requirements. The TSs for the NFSA are not being changed.

3.3 Method of Review

This safety evaluation (SE) involves a review of the licensee's NCS analyses for the Millstone 3 NFSA and the SFP that were provided as Attachment 5 to the LAR and updated through the supplements. The review was performed consistent with Sections 9.1.1 and 9.1.2 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 (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" (ADAMS Accession No. ML003728001) (hereafter the "Kopp memo"). 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 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 thereafter, including this LAR analysis.

The NRC staff also used interim staff guidance document entitled, "Final Division of Safety Systems Interim Staff Guidance, DSS-ISG-2010-01, Revision 0, 'Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools,'" dated September 2010 (ADAMS Accession No. ML110620086); notice of availability published in the *Federal Register* on October 13, 2011 (76 FR 63676), 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 it applies 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 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 Millstone 3 SFP are described in the criticality analysis that was provided as Attachment 5 to the LAR and updated through the supplements. The methodology is specific to this analysis and is not appropriate for other applications.

3.4.1.1 Computational Methods

The Millstone 3 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 Millstone 3 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. Millstone 3 used the T5-DEPL depletion sequence from SCALE 6.0 to perform its depletion step. The T5-DEPL depletion sequence from SCALE 6.0 does not have an NRC-approved method governing its use, nor does it have the feedback mechanism of being used by licensees for licensed core design work. 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

The licensee used the SCALE 6.0 T5-DEPL sequence for burned fuel compositions. Guidance in the Kopp memo recommends that, "In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption." This guidance is based on engineering judgment and

on the use of NRC-approved reactor depletion codes and operational experience of NRC licensed reactors.

In a previous application from the licensee, the NRC questioned the applicability of the Kopp depletion uncertainty methodology to the SCALE 6.0 T5-DEPL sequence.¹ The licensee provided sufficient information for the NRC staff to accept the use of the Kopp depletion uncertainty methodology to its analysis. In its LAR, the licensee cited prior approvals as precedent and justified its applicability to this request.

The NRC staff finds that the licensee's analysis provides reasonable assurance that previous guidance regarding the depletion validation is applicable to this Millstone 3 LAR. Consistent with the guidance provided in the Kopp memo, the licensee's analysis has incorporated an uncertainty equal to 5 percent 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 5 to the LAR and updated through the supplements. The validation set includes critical configurations from the International Handbook of Evaluated Criticality Safety Benchmark Experiments and French Haut Taux de Combustion critical experiments from NUREG/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data" (ADAMS Accession No. ML082880452). The suite of experiments was similar to those used in NRC-approved license amendments for other facilities. The validation was performed in a manner consistent with NUREG-6698, "Guide for Validation of Nuclear Criticality Safety Computational Methodology" (ADAMS Accession No. ML050250061) and included an evaluation for a temperature bias. Therefore, this validation of SCALE 6.0 CSAS5 is 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 performed a sensitivity analysis to determine the most reactive temperature for each region. The sensitivity analysis included SFP with water temperatures of 32 degrees Fahrenheit (°F), 68 °F (base case), 110 °F, and 150 °F. The water densities used were adjusted consistent with the water temperatures being modeled. The most reactive temperature for each region was used in the base calculations for each region. 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

The Millstone 3 SFP contains three storage rack designs (Regions 1, 2, and 3). The rack design and the modeling of the racks are described in Attachment 5 to the LAR (Section 4 of the criticality safety evaluation report) and are summarized below.

¹ NRC-issued License Amendment No. 327 for Millstone, Unit No. 2, dated June 23, 2016 (ADAMS Accession No. ML16003A008)

- Region 1 racks have a flux trap design and contain BORAL® neutron absorber panels. Region 1 racks are subdivided into Regions 1A and 1B. Region 1B occupies two rows of Region 1 rack storage cells adjacent to the west SFP wall. Region 1B credits neutron leakage at the interface with the Region 1 wall.
- Region 2 racks contain BORAL® neutron absorber panels but do not use a flux trap design. A fuel assembly must meet the requirements of the Region 2 burnup curve to be stored in this region. However, any fuel assembly with initial enrichments \leq [less than or equal to] 5.0 weight/percent U-235 that contains a rod cluster control assembly may also be stored in Region 2 without restriction.
- Region 3 racks have a flux trap design and contain Boraflex neutron absorber material. Boraflex is not credited in the criticality safety analysis and is modeled as SFP water. A fuel assembly must meet the requirements of the Region 3 burnup curve, which credits decay time to be stored in Region 3.

3.4.2.3 SFP Storage Rack Models Manufacturing Tolerances and Uncertainties

The analysis, as documented in Attachment 5 to the LAR, 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 to determine the effect in each region. 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 Fuel Assembly

3.4.3.1 Bounding Fuel Assembly Design

MPS3 uses a 17x17 lattice fuel with a center instrument tube and 24 guide tubes. Four fuel designs have been used but all the designs are similar for the criticality analysis. The initial fuel design, which is designated "Standard," used all inconel grids. The current fuel design is the Westinghouse RFA-2 design. The current and legacy fuel designs are listed in Table 3.1 of Attachment 5 to the LAR. For its analysis, the licensee has used a composite fuel design intended to represent current and legacy fuel designs. The composite fuel design is listed in Table 13.2. Use of a composite fuel design is an accepted practice to simplify the analysis. The NRC staff has reviewed the licensee's use of a composite fuel design and considers it acceptable.

3.4.3.2 Fuel Assembly Manufacturing Tolerances and Uncertainties

The analysis, as documented in Attachment 5 to the LAR, 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 to determine the effect in each region. 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

The licensee used the Kopp depletion uncertainty methodology, which the NRC determined was acceptable for this application, as discussed in Section 3.4.1.1.1.

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, the burnup at the ends is over-predicted. Analysis discussed in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis" (ADAMS Accession No. ML031110292), 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 Millstone 3 SFP criticality analysis used the bounding axial burnup profiles from NUREG/CR-6801. Since the distributed burnup profiles from NUREG/CR-6801 are not always bounding at lower burnup levels, the licensee considered uniform burnups for those scenarios. The NRC staff considers the licensee's approach to axial burnup profiles to be appropriate. While a significant fraction of the Millstone 3 fuel has axial blankets, the analysis treated all fuel as non-blanketed, effectively adding more highly enriched fuel to the end of each assembly. This is an acceptable approach to conservatively address the blanketed fuel.

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 Millstone 3 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 U.S. Department of Energy document, DOE/RW-0496, "Horizontal Burnup Gradient Datafile for PWR Assemblies," with information on the sensitivity of k_{eff} to radial burnup tilt provided in Section 6.1.2 of NUREG/CR-6800, "Assessment of Reactivity Margins and Loading Curves for PWR Burnup-Credit Cask Designs" (ADAMS Accession No. ML031110280). 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 $0.002 \Delta k$. 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. 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" (ADAMS Accession No. ML003688150), 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 Millstone 3 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 (i.e., 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 percent mille per degree Kelvin ($^{\circ}\text{K}$). Thus, a 10°F change in moderator temperature used in the depletion analysis would result in $0.005 \Delta k$. 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 Millstone 3 analysis used bounding values for current and expected operation. The NRC staff finds the moderator and fuel temperatures used by the licensee are acceptable for this application.

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 1,050 parts per million (ppm) for all cycles. This value bounds all but one earlier cycle and is at least 100 ppm higher than the current cycle averages. The licensee justified using the 1,050 ppm for the non-bounded cycle by considering the cycle average soluble boron of the previous and subsequent cycles to generate a two-cycle average. The NRC staff considers the use of cycle average and two-cycle averages to be different methodologies. The use of multi-cycle averages or fuel assembly lifetime soluble boron average could be non-conservative, especially if the higher soluble boron cycle occurred just prior to the fuel being placed in the SFP. This possibility has not been sufficiently vetted for the practice to be considered generally acceptable. However, Figure 8.2 of Attachment 5 to the licensee's May 3, 2018, application shows the previous and post two cycles of the outlier to be sufficiently less than 1,050, and therefore, there is sufficient margin to offset the use of 1,050 ppm for the outlier cycle.

Use of a cycle average soluble boron may be non-conservative for mid-cycle or early termination of the cycle. In its March 27, 2019, response to the NRC's RAI on the topic, the licensee provided options to use a fuel assembly lifetime soluble boron average or actual depletion parameters.

During a clarification phone call, the NRC staff informed the licensee that the information presented in its response was methodology changes that were not supported by the submittal. In its letter dated May 7, 2019, the licensee revised the RAI response to remove those options. The licensee retained an acceptable approach to offsetting the non-conservatism of a mid-cycle shutdown or early termination of the cycle. As a result, the NRC staff finds the boron concentration used in the licensee's analysis is acceptable.

3.4.3.3.5 Integral and Fixed Burnable Absorbers

In the past, MPS3 used two types of fixed burnable absorbers. The first two cycles used the fixed burnable absorber: Pyrex Burnable Poison (BP). All subsequent cycles have only used the integral burnable absorber: Integral Fuel Burnable Absorber (IFBA). Although the licensee has not used the Wet Annular Burnable Absorber (WABA) fixed integral burnable absorber their use was included in the analysis to support future use. The licensee states the Pyrex BP is not bounded by the WABA, but used the WABA in the analysis and dispositioned the legacy use of the Pyrex BP. Based on the NRC staff's review, the licensee has appropriately treated the use of its fixed integral burnable absorbers.

The licensee stated that its maximum projected WABA loading will bound its maximum IFBA loading, and therefore, used WABA in the analysis. The WABA and IFBA can be used simultaneously, as the analysis did include a portion for the combined use of WABA and the IFBA. The analysis did not consider the combined use of maximum WABA and maximum IFBA, but rather considered combinations of the two that would remain bounded by the use of WABA alone. Therefore, the analysis sets a limit on the combined use of WABA and IFBA.

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 licensee identified its control element assemblies insertion history. It established an average control rod insertion of 6.25 inches. Figure 8.5 of Attachment 5 to the LAR indicates that several fuel assemblies exceeded that average amount. The information provided in Figure 8.5 indicates 6.25 inches bounds current operation. Therefore, the staff finds this modeling of the effects of control element assemblies usage is acceptable.

3.4.3.3.7 Credited Nuclides

The licensee described the isotopes used in the Millstone 3 analysis. The licensee provided an acceptable treatment of volatile and gaseous fission products.

3.4.4 Non-Standard Fuel Configurations/Reconstituted Fuel

A fuel rod storage canister in the Millstone 3 SFP allows for storage up to 52 fuel rods. The licensee analyzed the fuel rod storage canister and determined it can be stored in any storage cell that could have a fuel assembly.

Millstone 3 has one non-standard fuel assembly with two rods missing on the periphery. The licensee's analysis determined that those empty lattice locations did increase the fuel assembly's reactivity. While Millstone 3 currently only has one fuel assembly with empty lattice locations, it could have more in the future. In its March 27, 2019, response to the NRC's RAI on the topic, the licensee's reply included options to use actual depletion parameters. During a clarification phone call, the NRC staff informed the licensee those were changes to the

methodology that were not supported by the submittal. In its letter dated May 7, 2019, the licensee chose to revise its RAI response to remove those options. The licensee retained an acceptable approach to evaluating future empty lattice locations. As a result, the NRC staff finds the licensee's analysis acceptable.

The licensee provided a description of its fuel reconstitution process. As part of this process, the fuel assembly to be reconstituted will have empty cells on all four faces (i.e., will have no face-adjacent fuel assemblies). The licensee demonstrated that configuration was sufficiently neutronically isolated to not impact the reactivity of the SFP. However, it was not clear from the licensee's initial submittal that the requirement for no face-adjacent fuel assemblies applied to all regions. In its letter dated May 7, 2019, the licensee affirmed that the requirement applied to Region 1, as well as Regions 2 and 3.

3.4.5 Determination of Soluble Boron Requirements

Section 50.68 of 10 CFR requires that the k_{eff} of the Millstone 3 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 is changing its SFP TS governing SFP soluble boron to require a minimum of 2,600 ppm.

The licensee's analysis for the normal static condition indicates the regulatory requirement is met with 600 ppm of soluble boron in the SFP.

The licensee's analysis considered the following abnormal/accident conditions: loss of SFP cooling, inadvertent placement of a fuel assembly outside the SFP storage racks, misloading of multiple fuel assemblies in the SFP storage racks, a dropped fuel assembly, fuel handling error, and a seismic event. The licensee determined the multiple misloading scenario is limiting. The licensee performed a conservative multiple misloading analysis that showed compliance with the regulatory requirement with 2,600 ppm of soluble boron in the SFP.

3.5 Millstone 3 NFSA NCS Analysis

Although the licensee is not revising its NFSA TSs, the NRC staff reviewed the analysis to set a new analysis of record for the licensee. This section documents the review of the NFSA NCS analysis. The licensee's NFSA contains the NAM Boral, which is atypical for a new fuel storage system.

3.5.1 NFSA NCS Analysis Method

Compliance with 10 CFR 50.68(b)(2) and 10 CFR 50.68(b)(3) is demonstrated in Section 7 of Attachment 5 to the LAR and updated through the supplements.

For the NFSA the licensee used the same SCALE 6.0 CSAS5 KENO V.a-based criticality analysis sequence and the SCALE ENDF/B-VII 238 neutron energy group library it used to analyze the SFP (details in Section 3.4.1.1.2 of this SE).

The licensee performed sensitivity analyses for temperature, NFSA wall material, NFSA wall thickness, and foam flooding of the rest of the building outside the NFSA. Since the foam flooding of the rest of the building indicated an increase in the optimum moderation scenario,

the licensee should have considered full density flooding outside the NFSA. However, considering the margin in the analysis, it would be unlikely to change the conclusion.

3.5.2 NFSA Fuel Storage Racks

The steel structures that hold the individual storage cells in place 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. The Boral panels have cutouts for the steel structure that hold the individual storage cells in place, and those cutouts were considered. Fuel was modeled at minimum spacing and as sitting on the floor of the NFSA. Since worst-case spacing and location were modeled, uncertainty analysis for the fuel storage rack parameters is not needed.

3.5.3 Fuel Assemblies

For the NFSA, the licensee used the same composite fuel assembly it used to the SFP (details in Section 3.4.3.1 of this SE).

3.5.4 New Fuel Storage Racks and Fuel Assembly Manufacturing Tolerances and Uncertainties

The licensee used the same method for the new fuel storage racks that it used for the SFP (details in Sections 3.4.2.3 and 3.4.3.2 of this SE).

3.6 Boral Surveillance Program

3.6.1 Background

At Millstone 3, Boral is credited as the NAM in the SFP fuel racks. This LAR proposes changes to the SFP and new fuel storage racks criticality analyses. The proposed changes would increase the TS minimum required soluble boron concentration in the SFP, as well as revise spent fuel storage patterns.

Boral is a NAM that was previously approved for use in the Millstone 3 SFP by NRC-issued License Amendment No. 189 to Millstone 3, dated November 28, 2000 (ADAMS Accession No. ML003744387). Boral is a cermet made from aluminum and boron carbide, clad in 1,100 alloy aluminum. Additionally, the chemical compatibility of Boral with the materials utilized in the SFP racks, as well as the SFP environment, was evaluated in the letter dated November 28, 2000.

3.6.2 Boral Surveillance Program Licensee Description

The licensee currently uses a Boral coupon monitoring program. The purpose of this program is to detect potential degradation of the Boral material, ensure that the proper monitoring and trending occur, and ensure that the appropriate corrective actions are implemented if degradation is detected.

In its LAR, the licensee references its response to Generic Letter (GL) 2016-01, "Monitoring of Neutron-Absorbing Materials in Spent Fuel Pools" (ADAMS Accession No. ML16312A064), as well as its response to RAIs related to GL 2016-01 (ADAMS Accession No. ML17338A057), dated November 22, 2017, to provide a description of its Boral surveillance program.

The GL response and associated RAI responses stated that the coupons are located in a spent fuel cell that is surrounded with spent fuel assemblies. The coupons were surrounded by freshly discharged fuel for the first five cycles in order to expose the coupons to a higher temperature and gamma dose than the rest of the inservice material. These coupons were pre-characterized in order to compare the pre- and post-duty characteristics. Additionally, the coupons were manufactured at the same time as the inservice material, using material from all three lots as the Boral in the inserts. The licensee also stated that it conducts testing, including B-10 areal density measurements (by chemical analysis), visual examinations, dimensional measures, and weight and specific gravity measurements, on the coupons.

In addition, the licensee stated that the Boral coupon program acceptance criteria are that the coupons show no signs of degradation, and the B-10 areal density is bounded by the SFP criticality analysis. The results from the coupon tests are compared to the SFP criticality analysis, and percent loss in B-10 areal density is used to monitor degradation.

The licensee also stated that the Boral coupon program testing frequency will be incorporated into the Updated Final Safety Analysis Report, and will follow the guidance in NEI 16-03-A. The licensee stated this change will be incorporated following the end of Cycle 21 for Millstone 3.

3.6.3 Boral Surveillance Program NRC Staff Evaluation

The NRC staff has reviewed the contents of the LAR related to the properties of Boral NAM, compatibility of the Boral NAM in the SFP environment, and the Boral coupon monitoring program. The staff's review was conducted using guidance from SRP Sections 9.1.1 and 9.1.2. In addition, the staff considered the references to the GL 16-01 response provided by the licensee. The staff used this information to determine if the Boral NAM and the associated Boral surveillance program provide reasonable assurance that the appropriate parts of 10 CFR 50.68, GDC 61, and GDC 62 are met.

Boral has been previously approved for use by the staff at commercial power reactors, as well as at Millstone 3. Due to the chemical composition of Boral and operating experience from previously approved and installed Boral in other SFPs, the staff has reasonable assurance that the Boral will remain compatible with the SFP environment in the SFP.

The NRC staff also reviewed the details of the coupon monitoring program as provided in the licensee's response to GL 16-01.

The staff finds the coupon testing program acceptable because the coupons are in a location that will accelerate the radiation and temperature dose relative to the rest of the inservice material. In addition, the coupons are pre-characterized so that any test data may be monitored and trended against the as-installed data. The licensee conducts measurements of the B-10 areal density (via chemical analysis), visual examinations, dimensional measures, and weight and specific gravity measurements, on an appropriate interval given the NAM used. Measuring the B-10 areal density of the NAM is essential to verifying the assumption of B-10 areal density used in the SFP criticality analysis. The B-10 areal density measurements, in conjunction with the other measurements described, provide the staff reasonable assurance that the coupon monitoring program will be able to detect signs of potential degradation in the NAM.

The NRC staff finds the approach to evaluating the coupon monitoring program results acceptable because the licensee will evaluate the results of the coupon tests and compare them to the stated acceptance criteria. The acceptance criteria are acceptable as they ensure the assumption for B-10 AD value assumed in the SFP criticality analysis is maintained,

and trending the percent loss of areal density will help to project future degradation that may impact the B-10 AD assumed in the SFP criticality analysis. Appropriate acceptance criteria, and the trending of potential degradation, provide the staff reasonable assurance that the results of the coupon monitoring program will be able to detect signs of potential degradation in the NAM.

In addition, the licensee proposed to update final safety analysis report (FSAR) Section 9.1.2.3 to include coupon test frequencies in accordance with NEI 16-03-A, which states the test frequency will be at least once per 10 years. The NRC staff reviewed the licensee's proposal to include the coupon test frequency in the FSAR and found it acceptable because the frequency is in accordance with a staff approved topical report, and because it will become part of the current licensing basis for the plant and provide reasonable assurance that the coupons will be tested at an appropriate frequency while in service.

3.7 Conclusions

In its analysis, the licensee has made appropriate simplifications and used less than maximum bounding values that are considered acceptable by the NRC staff. The NRC staff has completed its review of the Millstone 3 SFP and NFSA NCS analyses, which are documented in the licensee's LAR and supplements. Based on the above, the NRC staff concludes that there is reasonable assurance that the Millstone 3 SFP and NFSA fuel storage racks meet the applicable regulatory requirements in 10 CFR 50.68.

Additionally, the NRC staff reviewed information from the licensee's Boral coupon program, and determined that the program as described in the licensee's LAR and referenced response to GL 16-01, will provide reasonable assurance that the licensee will be able to detect degradation of the neutron absorbing material before its ability to perform its intended safety function is impacted. Additionally, the staff has reviewed the information provided regarding material compatibility with the SFP environment and has reasonable assurance that Boral will be chemically compatible with the SFP environment. On this basis, the staff concludes that the contents of the program and the use of Boral in the SFP meet the applicable requirements of 10 CFR 50.68 and GDC 61 and 62, and are, therefore, acceptable.

The NRC staff determined that the proposed TSs will continue to be based on the analyses and evaluations included in the safety analysis report and amendments thereto in accordance with 10 CFR 50.36(b). The NRC staff also determined that the proposed TSs will continue to include design features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety, in accordance with 10 CFR 50.36(c)(4). Therefore, the NRC staff has determined that the proposed TSs are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment on May 6, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that

may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on this finding (August 7, 2018; 83 FR 38735). 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.

Principal Contributors: Kent Wood
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Date: May 28, 2019

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 – ISSUANCE OF AMENDMENT NO. 273 REGARDING TECHNICAL SPECIFICATION CHANGES FOR SPENT FUEL STORAGE AND NEW FUEL STORAGE (EPID L-2018-LLA-0126) DATED MAY 28, 2019

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