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

March 26, 2010

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

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: SPENT FUEL POOL CRITICALITY (TAC NO. MD8251)

Dear Mr. Heacock:

The Commission has issued the enclosed Amendment No 248 to Renewed Facility Operating License No. NPF-49 for the Millstone Power Station, Unit No. 3 (MPS3), in response to your application dated July 13, 2007.

The amendment makes changes to the Technical Specifications (TS) for MPS3 spent fuel pool (SFP) storage requirements. By letter dated July 13, 2007 (Dominion Nuclear Connecticut, Inc. (DNC or the licensee), submitted a license amendment request for a stretch power uprate (SPU) of MPS3. Included in a supplement dated July 13, 2007, was a request to make changes to the TSs for MPS3 SFP storage. By letter dated March 5, 2008, DNC separated the MPS3 SFP storage requirements request from the MPS3 SPU request.

The July 13, 2007, request was supplemented by letters dated September 30, 2008, March 5, 2009, March 23, 2009, March 1, 2010, and March 5, 2010.

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,

Carleen J. Sanders, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures:
1. Amendment No. 248 to NPF-49
2. Safety Evaluation

cc w/encls: Distribution via Listserv

DOMINION NUCLEAR CONNECTICUT, INC., ET AL.

DOCKET NO. 50-423

MILLSTONE POWER STATION, UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 248
Renewed License No. NPF-49

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

   A. The application for amendment by the applicant dated July 13, 2007, as supplemented by letters dated July 13, 2007, March 5, 2008, September 30, 2008, March 5, 2009, March 23, 2009, March 1, 2010, and March 5, 2010, 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, as revised through Amendment No. 248, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the renewed license. DNC 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 30 days of issuance. Implementation of the amendment shall include revision of the Updated Final Safety Analysis Report, in accordance with 50.71(e), including but not limited to a detailed description of the actions required by TS 3.9.13 and when they apply, as well as, a detailed description of the steps to be taken and the required time frame for initiation of these steps, if a surveillance requirement associated with TS 3.9.13 is not met.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: March 26, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 248

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 marginal lines indicating the areas of change.

Remove	Insert
4	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 marginal lines indicating the areas of change.

Remove	Insert
3/4 9-16	3/4 9-16
3/4 9-20	3/4 9-20
3/4 9-21	3/4 9-21
5-6	5-6
----	3/4 9-22

(2) Technical Specifications

The Technical Specifications contained in Appendix A, revised through Amendment No248 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated into the license. DNC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) DNC shall not take any action that would cause Dominion Resources, Inc. (DRI) or its parent companies to void, cancel, or diminish DNC'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 Resources, 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 can not be amended in any material respect without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

REFUELING OPERATIONS

3/4.9.13 SPENT FUEL POOL - REACTIVITY

LIMITING CONDITION FOR OPERATION

3.9.13 The Reactivity Condition of the Spent Fuel Pool shall be such that k_{eff} is less than or equal to 0.95 at all times.

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

ACTION: With k_{eff} greater than 0.95:

a. Borate the Spent Fuel Pool until k_{eff} is less than or equal to 0.95, and

b. Initiate immediate action to move any fuel assembly which does not meet the requirements of Figures 3.9-1, 3.9-3, 3.9-4, or 3.9-5 to a location for which that fuel assembly is allowed.

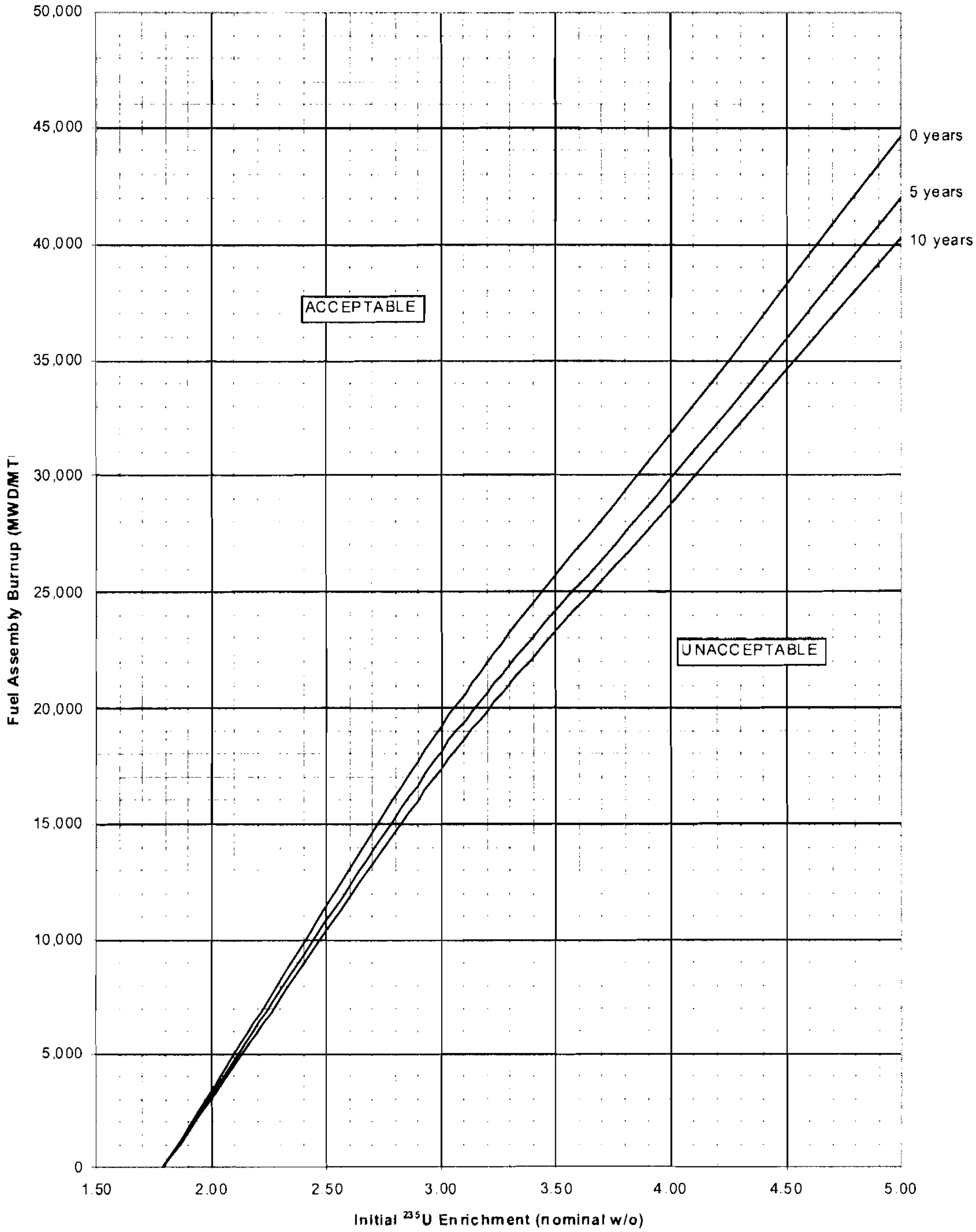
SURVEILLANCE REQUIREMENTS

4.9.13.1.1. Ensure that all fuel assemblies to be placed in Region 1 "4-OUT-OF-4" fuel storage are within the enrichment and burnup limits of Figure 3.9-1 by checking the fuel assembly's design and burn-up documentation.

4.9.13.1.2. Ensure that all fuel assemblies to be placed in Region 2 fuel storage are within the enrichment, decay time, and burnup limits of Figure 3.9-3 by checking the fuel assembly's design, decay time, and burn-up documentation.

4.9.13.1.3. Ensure that all fuel assemblies used exclusively in pre-uprate (3411 Mwt) conditions which are to be placed in Region 3 fuel storage are within the enrichment, decay time, and burnup limits of Figure 3.9-4 by checking the fuel assembly's design, decay time, and burn-up documentation. Ensure that all fuel assemblies used in post-uprate (3650 Mwt) conditions which are to be placed in Region 3 fuel storage are within the enrichment, decay time, and burn-up limits of Figure 3.9-5 by checking the fuel assembly's design, decay time, and burn-up documentation.

Figure 3.9-3 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 2 Storage Configuration



NOTE: For assemblies from Post-Uprate (3650 Mwt) Cores, the nominal fuel enrichment of blankets must be ≤ 2.6 w/o U-235, and nominal blanket length must be at least 6 inches on both ends of the fuel. Fuel batches A, B, C, and D may not be stored in Region 2.

Figure 3.9-4 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for Assemblies from Pre-Uprate (3411 Mwt) Cores

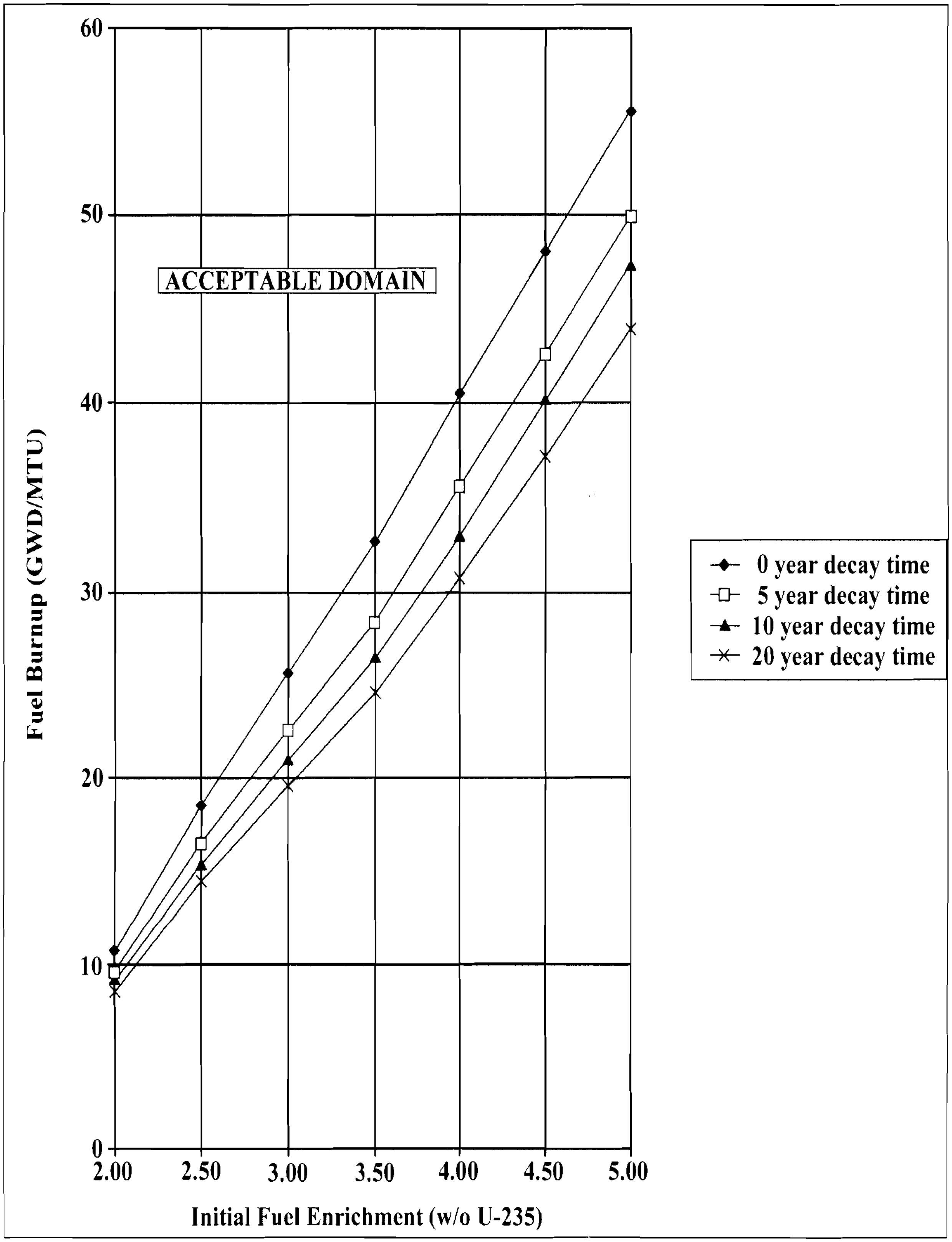
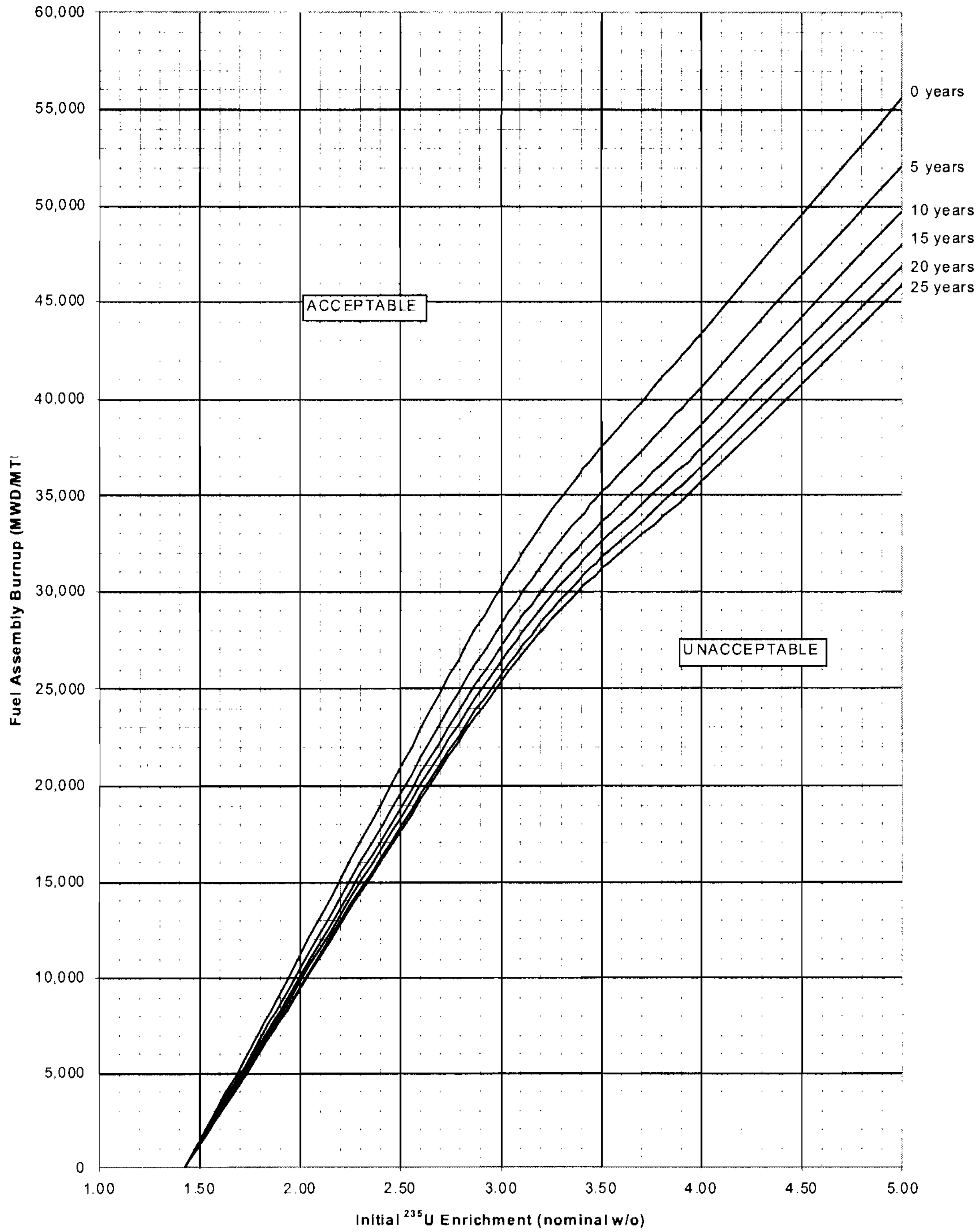


Figure 3.9-5 Minimum Fuel Assembly Burnup and Decay Time Versus Nominal Initial Enrichment for Region 3 Storage Configuration for Assemblies from Post-Uprate (3650 Mwt) Cores



NOTE: For assemblies from Post-Uprate (3650 Mwt) Cores, the nominal fuel enrichment of blankets must be ≤ 2.6 w/o U-235, and nominal blanket length must be at least 6 inches on both ends of the fuel.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 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 or equal to 0.95 when flooded with unborated water. The storage rack Regions are:

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, and can store fuel in 2 storage configurations:

(1) With credit for fuel burnup as shown in Figure 3.9-1, fuel may be stored in a "4-OUT-OF-4" storage configuration.

(2) With credit for every 4th location blocked and empty of fuel, fuel up to 5 weight percent nominal enrichment, regardless of fuel burnup, may be stored in a "3-OUT-OF-4" storage configuration. Fuel storage in this configuration is subject to the interface restrictions specified in Figure 3.9-2.

b. Region 2, a nominal 9.017 inch center to center distance, credits a fixed neutron absorber (BORAL) within the rack, and with credit for fuel burnup and fuel decay time as shown in Figure 3.9-3, fuel may be stored in all available Region 2 storage locations.

c. Region 3, a nominal 10.35 inch center to center distance, with credit for fuel burnup and fuel decay time as shown in Figure 3.9-4 for assemblies used exclusively in pre-uprate (3411 Mwt) cores or Figure 3.9-5 for assemblies used in post-uprate (3650 Mwt) cores, fuel may be stored in all available Region 3 storage locations. The Boraflex contained inside these storage racks is not credited.

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.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 248

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated July 13, 2007 (Reference 1), Dominion Nuclear Connecticut, Inc. (DNC or the licensee) submitted a license amendment request (LAR) for a stretch power uprate (SPU) of Millstone Power Station, Unit No. 3 (MPS3). Included in a supplement dated July 13, 2007 (Reference 2), was a request to make changes to the Technical Specifications for MPS3 spent fuel pool (SFP) storage. By letter dated March 5, 2008 (Reference 3), DNC separated the MPS3 SFP storage requirements request from the MPS3 SPU request.

The July 13, 2007 request was supplemented by letters dated September 30, 2008, (Reference 4), March 5, 2009, (Reference 5), March 23, 2009, (Reference 6), March 1, 2010, (Reference 7), and March 5, 2010 (Reference 8).

Changes in core operating parameters associated with an SPU result in Doppler broadening/spectral hardening of the neutron field which causes increased Plutonium-241 (^{241}Pu) production. Doppler broadening is the apparent increase in resonance absorption by a nuclei associated with an increased temperature of the nuclei. Spectral hardening is a reduction in moderation. The post-SPU discharged fuel is more reactive relative to pre-SPU discharged fuel at the same burnup. Therefore, the SFP storage requirements have to be adjusted to accommodate the more reactive fuel.

The SFP analysis was performed for the licensee by Westinghouse Electric Company, LLC (Westinghouse) in WCAP-16721-P, "Millstone Unit 3 Spent Fuel Pool Criticality Analysis" (Reference 2).

The supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination as published in the *Federal Register* (73 FR 2549). The SFP LAR no significant hazards consideration determination was noticed a second time, separate from the MPS3 SPU (74 FR 46241).

2.0 EVALATION

2.1 BACKGROUND

Currently the MPS3 SFP is divided into three Regions. The three Regions have physically different rack designs. Region 1 has two storage configurations. One is a '3-out-of-4' storage configuration with three fuel assemblies and one storage cell blocking device in a repeating pattern. The other is a '4-out-of-4' storage configuration controlled by a burnup and enrichment loading curve. Both Region 2 and Region 3 have a single storage configuration, controlled by respective burnup and enrichment loading curves.

The current MPS3 Technical Specification (TS) 3/4.9.13, "Spent Fuel Pool – Reactivity" and TS 3/4.9.14, "Spent Fuel Pool – Storage Pattern" provide requirements for controlling storage of spent nuclear fuel (SNF) and fresh nuclear fuel in the MPS3 SFP. TS 3/4.9.14 describes the '3-out-of-4' storage configuration in Region 1. There are no burnup requirements associated with this storage configuration; therefore, it allows for the storage of fresh nuclear fuel and depleted fuel. TS Figure 3.9-2 shows the interface requirements for the Region 1 '3-out-of-4' storage configuration with the Region 1 and Region 2 '4-out-of-4' storage configurations. TS 3/4.9.13 describes the enrichment and burnup limits for the '4-out-of-4' storage configurations in Regions 1, 2, and 3. The burnup and enrichment loading requirements associated with these storage configurations are found in TS Figures 3.9-1, 3.9-3, and 3.9-4, respectively. These TS are augmented by TS 5.6.1.1 in the design features section of the TS.

2.2 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50, Appendix A, Criterion 62 states that, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." The MPS3 Updated Final Safety Analysis commits MPS3 to meeting 10 CFR 50 Appendix A, Criterion 62.

The MPS3 licensing basis for the SFP is for k-effective (k_{eff}) to be less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level, if flooded with unborated water under nominal conditions. For abnormal/accident conditions, the MPS3 licensing basis allows credit for SFP soluble boron to maintain k_{eff} less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level. K_{eff} is the ratio of neutron production to absorption and leakage.

MPS3 TS 3.9.13 requires the reactivity condition of the SFP to be such that K_{eff} is less than or equal to 0.95 at all times. As stated above, K_{eff} shall be less than or equal to 0.95 without credit for soluble boron during non-accident conditions. The phrase "[w]hen K_{eff} is greater than 0.95:" which is adjacent to the word ACTION in TS 3.9.13, is a restatement of the limiting condition of operation. When K_{eff} is greater than 0.95, TS 3.9.13 ACTIONS "a" and "b" apply. TS 3.9.13 required ACTION "a" requires MPS3 to borate the spent fuel pool until k_{eff} is less than or equal to 0.95. TS 3.9.13 required ACTION "b" requires immediate action be taken to move any fuel assembly which does not meet the requirements of the enrichment and burnup limits for the given region, to a location for which the fuel assembly is allowed. The term "immediate action" in TS 3.9.13.b requires the specific action to begin without delay. The enrichment and burnup limits for each region of the SFP are provided as figures 3.9-1 and 3.9-3 through 3-9.5 in the TSs.

MPS3 TSs also require the surveillance requirements be met. Failure to meet the surveillance requirements, whether identified during performance of the surveillance or between performances of the surveillance, constitutes a failure to meet the TS limiting condition for operation. Thus, if any of the surveillances associated with TS 3.9.13 are identified as not being met at any time, TS 3.9.13 actions must be taken.

The initial submittal used an analytic acceptance criterion of k_{eff} <0.949 if flooded with unborated water, reserving 0.001 Δk_{eff} analytical margin to the licensing basis limit. The March 5, 2010, letter, revised the analytical acceptance criteria to 0.945 increasing the reserved analytical margin to 0.005 Δk_{eff}.

2.3 PROPOSED CHANGE

The Region 1 '3-out-of-4' storage configuration is not part of the requested change. The Region 1 '3-out-of-4' storage configuration is included in the analysis for consideration of interface requirements with the other storage configurations, but the requirements associated with the Region 1 '3-out-of-4' storage configuration are not being changed.

The Region 1 '4-out-of-4' storage configuration is a pattern of four fuel assemblies that have to meet the burnup and enrichment requirements of TS Figure 3.9-1. This storage configuration was reanalyzed in WCAP-16721-P. The analysis in WCAP-16721-P shows the current TS Figure 3.9-1 to be bounding, with the updated core operating parameters, input, and assumptions. Therefore, the licensee has not included a revised TS Figure 3.9-1 as part of this license amendment request.

The Region 2 '4-out-of-4' storage configuration is a pattern of four fuel assemblies that have to meet the burnup and enrichment requirements of TS Figure 3.9-3. This storage configuration was reanalyzed in WCAP-16721-P. The analysis in WCAP-16721-P shows the current TS Figure 3.9-3 to be non-bounding for the updated core operating parameters, input, and assumptions. The licensee has proposed a revised TS Figure 3.9-3 that has more stringent burnup and enrichment loading requirements, and also takes credit for the decay of ^{241}Pu and the build-up of Americium-241 (^{241}Am). The revised Figure 3.9-3 is open to both pre-SPU and post-SPU discharged fuel assemblies. However, there are restrictions captured in a note to the figure. The note includes axial blanket restrictions on post-SPU discharged fuel assemblies. Only the first four fuel batches at MPS3 were non-blanketed, and those are prohibited from being stored in Region 2.

The Region 3 '4-out-of-4' storage configuration is a pattern of four fuel assemblies that have to meet the burnup and enrichment requirements of TS Figure 3.9-4. This storage configuration was reanalyzed in WCAP-16721-P. The analysis in WCAP-16721-P shows the current TS Figure 3.9-4 to be non-bounding for the updated core operating parameters, input, and assumptions. The licensee has proposed a change to the title of Figure 3.9-4 to limit its applicability to pre-SPU discharged fuel assemblies. To accommodate post-SPU discharged fuel assemblies in Region 3, the licensee has proposed adding TS Figure 3.9-5.

In addition to the revised figures, the licensee has included changes to Surveillance Requirements 4.9.13.1.2 and 4.9.13.1.3, along with changes to the Design Features TS 5.6.1.1 to maintain consistency in the TSs.

2.4 TECHNICAL EVALUATION

2.4.1 Methodology

The NRC staff issued a publicly available memorandum on August 19, 1998, containing guidance for performing the review of SFP criticality analysis (Reference 9). This memorandum is known as the Kopp Letter. While the Kopp Letter does not specify a methodology for performing SFP criticality analysis, it does provide guidance. The guidance is relevant for boiling water reactors and pressurized water reactor (PWR), with borated and unborated water. Since its issuance, the Kopp Letter has been used for most PWR SFP criticality analysis, including the MPS3 analysis. The guidance in the Kopp Letter can be summarized as: determine the biases and uncertainties for the parameters affecting reactivity in the SFP and apply them in the conservative direction. The guidance allows for the statistical combination of uncertainties, provided the uncertainties are independent.

WCAP-16721-P describes the analysis methods used for the MPS3 SFP, including a description of the computer codes used to perform the criticality safety analysis.

2.4.2 Computer Code Validation

The analysis in WCAP-16721-P employs: (1) SCALE version 4.4 (SCALE 4.4), with the SCALE 4.4 versions of the 44- and 238-group Evaluated Nuclear Data File Version 5 (ENDF/B-V) neutron cross section libraries, and (2) the two-dimensional transport lattice code PHOENIX-P, with an Evaluated Nuclear Data File Version 6 (ENDF/B-VI) neutron cross section library. SCALE 4.4 is utilized for reactivity determinations of fuel assemblies in the MPS3 SFP. SCALE 4.4 is also used for in-SFP criticality simulations, specifically the Monte Carlo code KENO V.a that is part of SCALE 4.4. The PHOENIX-P code is used for simulation of in-reactor fuel assembly depletion.

Validation of SCALE 4.4 for purposes of fuel storage rack analyses is based on the analysis of 30 selected critical experiments from two experimental programs: 19 from the Babcock & Wilcox experiments in support of Close Proximity Storage of Power Reactor Fuel, and 11 from the Pacific Northwest Laboratory Program in support of the design of Fuel Shipping and Storage Configurations.

In addition to using the SCALE 4.4 code to perform the criticality analyses, the licensee employed the PHOENIX-P code to perform the fuel depletions used in the analysis. PHOENIX-P is a two-dimensional, multi-group transport theory lattice code. The multi-group cross sections are based on ENDF/B-VI. PHOENIX-P performs a two-dimensional 70-group nodal flux calculation which couples the individual sub-cell regions (pellet, cladding, and moderator) as well as surrounding rods via a collision probability technique. This 70-group solution is normalized by a coarse-energy-group S4 flux solution derived from a discrete ordinates calculation.

In responding to the NRC staff's questions, the licensee performed various scoping studies. In those scoping studies, the licensee used PARAGON in place of PHOENIX-P for in-reactor simulations to determine the isotopic concentrations and SCALE Version 5.1 for in-SFP simulations. PARAGON and SCALE 5.1 were used for the scoping studies because of their ease of use and runtime considerations. When the scoping studies indicated a revision to the burnup and enrichment loading curves was warranted, SCALE 4.4 was used for the in-SFP simulations to remain consistent with the validation that was performed for WCAP-16721-P. However, the isotopic concentrations used were those from PARAGON. PARAGON is a stand alone direct replacement for PHOENIX-P. For the purposes of this evaluation the depletion uncertainty is the same for either code.

2.4.3 Spent Fuel Pool

Description of the Spent Fuel Pool

Region 1 storage cells utilize a flux trap design with BORAL as a fixed neutron poison. This design's geometry consists of a stainless steel canister which controls the fuel assembly position within the array, BORAL fixed neutron poison panels located on the four outer walls of each canister, and finally a stainless steel wrapper that maintains the BORAL panel in place. The BORAL panels contain Boron-10 (B^{10}), at a nominal areal density of 0.0302 g/cm^2, and cover the entire length of the stored active fuel height. The Region 1 cells have a nominal north/south pitch (center-to-center distance between the storage cells) of 10.0 inches and a nominal east/west pitch of 10.455 inches.

Region 2 storage cells utilize a non-flux trap design with BORAL as a fixed neutron poison. This design's storage cells are formed by welding open stainless steel canisters together. Therefore, the Region 2 storage cells are a combination of individual canister storage cells and developed storage cells. The developed storage cells result from the welding process. As an example, the welding of four canisters produces a single developed storage cell at the center of the four canisters. BORAL fixed neutron poison panels are located on the four outer walls of each canister and are held in place by a stainless steel wrapper. Identical to the Region 1 panels, the Region 2 BORAL panels contain B^{10}, at a nominal areal density of 0.0302 g/cm^2 and cover the entire length of the stored active fuel height. The Region 2 cells have a nominal pitch of 9.017 inches.

Region 3 storage cells utilize a flux trap design with Boraflex as the fixed neutron poison. This design's geometry consists of a stainless steel canister, which controls the fuel assembly position within the array, Boraflex fixed neutron poison panels located on the four outer walls of each canister, and finally a stainless steel wrapper that maintains the Boraflex panel in place. The Boraflex fixed neutron poison is not credited here; the Boraflex is considered to be water for the purposes of this analysis. The Region 3 cells have a nominal pitch of 10.35 inches.

Spent Fuel Pool Mechanical Uncertainties

The material and configuration of the SFP racks contributes to the reactivity. The material contributes by providing a fixed neutron absorber and the configuration by controlling the fuel assembly spacing.

DNC's SFP analysis determined a separate uncertainty for the tolerances associated with each of the following parameters: cell pitch, cell wall thickness, cell internal dimension, BORAL loading, and the Boraflex wrapper thickness. An uncertainty was also determined for the assembly positioning within the cells. The uncertainty for each parameter was determined by comparing a nominal case, in which all parameters were defined at their nominal values, to cases where the parameters deviated by their manufacturing tolerance. When the comparison showed a positive reactivity increase greater than the KENO V.a case standard deviation, it was included as an uncertainty.

SFP Temperature Bias

NRC guidance provided in the Kopp Letter states the criticality analysis should be done at the temperature corresponding to the highest reactivity. When the SFP has a positive moderator temperature coefficient, the temperature corresponding to the highest reactivity would be the highest allowed operating temperature. Rather than determining the most reactive temperature and performing all of the analyses at that temperature, WCAP-16721-P performed the bulk of the analyses at a nominal temperature and then determined a temperature bias. In the original submittal, only temperatures above the nominal temperature used in the analysis were considered. In this submittal, a temperature bias was included for the non-poisoned Region 3 storage racks, not for the BORAL racks in Region 1 and Region 2. In response to NRC staff request for additional information (RAI) question No. 8 (Reference 4), the licensee determined that temperatures below the nominal introduced a temperature bias of approximately 0.00080 Δk_{eff} and 0.00075 Δk_{eff}. Since these are for separate regions they are not cumulative. Also in the response to RAI No. 8, the licensee indicated the additional temperature bias was compensated for by a radial leakage credit determined in the response to RAI No. 6 (Reference 4). Insufficient information has been provided to evaluate the radial leakage credit. However, in the response to RAI No. 30 (Reference 8), the licensee increased the reserved analytical margin to 0.005 Δk_{eff}. The 0.005 Δk_{eff} margin is more than enough to compensate for the 0.00080 Δk_{eff} and 0.00075 Δk_{eff} temperature bias introduced by temperatures lower than nominal. The NRC staff credits the reserved analytical margin rather than the radial leakage for the temperature bias.

2.4.4 Fuel Assembly

Selection of Bounding Fuel Assembly Design

MPS3 has been in operation since December 1975. During that time a variety of reload fuel regions containing different fuel assembly designs have been irradiated in the reactor. In the future, additional fuel assembly designs may be irradiated. Thus, the criticality safety analysis of the MPS3 SFP must take into account possible differences in the reactivity characteristics of different assembly types. For the purposes of this analysis, applicable fuel assembly types were surveyed to determine a reference fuel assembly design that would assure conservative results for the analysis.

The Westinghouse 17x17 standard fuel design (STD), Vantage 5-H fuel design (V5H), robust fuel assembly (RFA) and next generation fuel (NGF) assembly types are considered in the licensee's analysis. Simulations were performed for each storage configuration in this analysis to determine the fuel assembly combinations that produce the highest reactivity. The fuel assembly type that produced the highest reactivity was used in the analysis.

These simulations simplified the fuel assemblies by not modeling spacer grids, mixing grids, end fittings, or other structural components of the fuel assemblies. The NRC staff has determined that it is not always conservative to ignore structural components of the fuel assemblies. In Reference 4, the licensee provided the results of sensitivity studies that indicate, within the range of this analysis, ignoring the grid strips (spacer grids and mixing grids) is a reasonable assumption in all three Regions when soluble boron is not credited. However, the sensitivity studies indicate that for Region 3, there are some non-conservatism when soluble boron is credited. The non-conservatism starts at about 400 parts per million (ppm) of soluble boron. Since the limiting abnormal/accident condition for MPS3 is a misloading in Region 3, this non-conservatism could affect the amount of soluble boron necessary to overcome the limiting abnormal/accident condition. This is discussed in Section 2.4.6, "Soluble Boron Requirements."

Fuel Assembly Mechanical Tolerances

The licensee's SFP analysis determined a separate uncertainty for the tolerances associated with each of the following parameters: pellet diameter, cladding thickness, and fuel enrichment. The pellet diameter and cladding thickness uncertainties are determined at the maximum fresh fuel enrichment for each storage configuration. The fuel enrichment is determined separately for each enrichment step in the storage configuration analysis.

The analysis in WCAP-16721-P does not address the other fuel assembly mechanical tolerances. Although, it does state that pellet dishing and chamfering is not modeled, and a maximum Theoretical Density (TD) is used. Not addressing the other fuel assembly tolerances was supposed to be compensated for by not modeling pellet dishing and chamfering and using a maximum TD. However, in response to RAI No. 21 (Reference 6), the licensee explicitly modeled as-built pellet dishing and chamfering and used the TD for pre-SPU non-blanketed fuel types. The licensee addressed this by prohibiting the storage of fuel batches, A, B, C, and D in Region 2 via a footnote to TS Figure 3.9-3. These are currently the only fuel batches that contained non-blanketed fuel types at MPS3. In keeping with the existing licensing basis fuel batches, A, B, C, and D may still be stored in both Region 1 storage configurations and Region 3, provided they meet the requirements of TS 3.9-14 and TS Figures 3.9-1 and 3.9-4, as applicable.

2.4.5 Spent Fuel Characterization

For the SFP criticality analysis, the fuel must be characterized appropriately. Characterization of fresh fuel is based primarily on Uranium-235 (^{235}U) 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 carry through to the SNF. The standard practice has been to treat the uncertainties as unaffected by the depletion. The characterization of SNF is more complicated. Its characterization is based on the specifics of its initial conditions and its operational history in the reactor. This characterization has three main areas: a burnup uncertainty, the axial apportionment of the burnup, and the core operation that achieved that burnup.

Depletion Uncertainty

The current NRC guidance provided in the Kopp Letter states that for determining the depletion uncertainty, "[a] reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption." Rather than use the 5 percent reactivity decrement as the burnup uncertainty, WCAP-16721-P used an alternate method. Using the information provided by the licensee (Reference 4), the NRC staff was able to compare the burnup uncertainty method used in WCAP-16721-P with the method in the current NRC guidance. That comparison showed the resultant WCAP-16721-P burnup uncertainties to be comparable and slightly more conservative for the higher enrichments and associated higher burnup requirements. That comparison showed that the WCAP-16721-P burnup uncertainties are non-conservative for more moderate and lower enrichments and their associated burnup requirements. However, in WCAP-16721-P the burnup uncertainty is treated as a bias rather than an uncertainty. Since the burnup uncertainty is typically convoluted with other uncertainties, its overall impact is somewhat diluted; however, since WCAP-16721-P treats the burnup uncertainty as a bias, it is not convoluted. Therefore, even though the WCAP-16721-P burnup uncertainty is lower than would have been determined using the current NRC guidance, the net effect is that the total summation of biases and uncertainties is larger than they would have been if determined using the current NRC guidance. Therefore, the WCAP-16721-P method for determining and using the depletion uncertainty is conservative with respect to the current NRC guidance.

Burnup Profile and Apportionment

Another important aspect of fuel characterization is the selection of the 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 leakage. If a uniform axial burnup profile is assumed, then the burnup at the ends is over predicted. Analysis has shown that this results in an under prediction of k_{eff}, generally the under prediction becomes larger as burnup increases. This is known as the 'end effect.' Judicious selection of the axial burnup profile is necessary to ensure k_{eff} is not under predicted due to the end effect. NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis," (Reference 11) provides insight for selecting an appropriate axial burnup profile.

With respect to the burnup axial profile, WCAP-16721-P used the limiting axial burnup profile from the Department of Energy's topical report DOE/RW-0472, "Topical Report on Actinide-Only Burnup Credit for PWR Spent Fuel Packages," (Reference 12). However, DOE/RW-0472 does not identify a single 'limiting burnup profile,' instead it identifies a 'limiting burnup profile' for each of 12 specified burnup intervals. DOE/RW-0472 has not been reviewed and approved by the NRC staff. NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," (Reference 11) uses the same database as DOE/RW-0472, but in some cases NUREG/CR06801 determined a different 'limiting burnup profile' for each of the 12 specified burnup intervals. Both DOE/RW-0472 and NUREG/CR-6801 agree that using a profile from a given burnup interval is generally conservative for higher burnups but non-conservative for lower burnups. WCAP-16721-P uses a single profile for the entire range of burnups considered, including some that DOE/RW-0472 and NUREG/CR-6801 consider non-conservative. Additionally the axial nodalization of the distributed profile was less defined than that of DOE/RW-0472 or NUREG/CR-6801. WCAP-16721-P also used a uniform axial profile to compare to the distributed axial profile. The analysis in WCAP-16721-P assumed this would identify the limiting profile. A uniform axial burnup should be considered in the following manner: conservative for BU < 10 gigawatt day per metric ton uranium (GWD/MTU), non-conservative for BU > 20 GWD/MTU, indeterminate for BU between those values.

The distributed profile used in WCAP-16721-P is only indicated as conservative well above 20 GWD/MTU. Therefore, there was a range of burnups where two potentially non-conservative profiles were being compared to determine a limiting profile. Since there was not reasonable assurance that either profile was limiting, there was no reasonable assurance that the resultant was limiting. However, both DOE/RW-0472 and NUREG/CR-6801 focus on identifying limiting profiles for storage/transportation cask criticality analysis. The limiting profiles are selected to eliminate the possibility that a more limiting profile could be found. Given the broad nature of the limiting profile, it is possible that a site-specific evaluation would identify conservatism associated with those profiles or less restrictive profiles that could be used. In DNC's response to RAI No. 21 (Reference 6) the licensee performed a site-specific analysis using MPS3 distributed profiles and fuel designs. This analysis showed that the WCAP-16721-P treatment of the burnup profile was non-conservative in the intermediate range of burnups. The final treatment of the axial burnup profile is combined with the resolution of the core-exit temperature in the response to RAI No. 30 (Reference 8). In the response to RAI No. 30, the licensee combined the burnup dependent maximum core-exit temperature profile with the site-specific burnup profiles from the response to RAI No. 21 to determine a reactivity penalty with regard to the lower core-exit temperature and axial burnup profile used in WCAP-16721-P. Consistent with NUREG/CR-6801 the axial burnup profile used in WCAP-16721-P provides considerable margin at the highest burnups considered in this analysis and the margin is sufficient to offset the effect of the higher core-exit temperature for the remainder of the burnups.

BU History/Core Operating Parameters

NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR [light water reactor] Fuel," (Reference 10) provides some discussion on the treatment of depletion analysis parameters that determine how the burnup was achieved. While NUREG/CR-6665 is focused on criticality analysis in storage and transportation casks, the basic principals with respect to the depletion analysis apply generically to SFPs because the phenomena occurs in the reactor as the fuel is being used. In particular, the effects of the infinite lattice analysis are similar to those performed for SFP analyzes. The basic premise is to select parameters that maximize the Doppler broadening/spectral hardening of the neutron field resulting in maximum ^{241}Pu 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 effects are similar: Doppler broadening/spectral hardening of the neutron field resulting in the maximum ^{241}Pu production. NUREG/CR-6665 provides an estimate of the reactivity worth of these parameters. The largest effect is due to moderator temperature. NUREG/CR-6665 approximates the moderator temperature effect, in an infinite lattice of high burnup fuel, to be 90 pcm/°K. Thus a 10°F change in moderator temperature used in the depletion analysis would result in approximately a 0.005 Δk_{eff}. With the exception of boron concentration the WCAP-16721-P analysis used typical values rather than limiting values for the in-core depletion portion of the analysis.

For boron concentration NUREG/CR-6665 recommends using a conservative cycle average boron concentration. The WCAP-16721-P analysis used a constant conservative boron concentration throughout the depletion of the fuel assemblies of 1000 PPM. However, no information was provided with respect to past or predicted reactor cycle-average soluble boron concentrations at MPS3. Later analyses performed to support the response to RAI No. 21 (Reference 6) were slightly different. The response says for "[d]epletion calculations for each profile were performed at the same conditions described for the 4 zone model in the WCAP..." However, the unblanketed fuel response used "as-operated cycle soluble boron concentrations" thus eliminating any conservatism that might have been present. During the February 17, 2010, public meeting, the licensee identified a cycle which exceeded the soluble boron used in the analysis, even if it was averaged over two cycles. In response to RAI No. 26 (Reference 7), the licensee provided a table of reactor cycle-average soluble boron concentrations for the entire history of MPS3. That table identified that MPS3 Cycle 6 has exceeded the soluble boron used in the WCAP-16721-P depletion simulations. This non-conservatism would affect all of the fuel in MPS3 Cycle 6, fresh fuel for Cycle 6, plus re-inserts from Cycles 4 and 5. If the reactor cycle-average soluble boron concentrations are averaged over three cycles for these fuel assemblies, then the average reactor cycle-average soluble boron concentration is less than the soluble boron used in the WCAP-16721-P depletion simulations. But, if the average is taken over only two cycles, then the fresh fuel assemblies in Cycle 5 and Cycle 6 will have exceeded the soluble boron used in the WCAP-16721-P depletion simulations. Since it is credible that some of these fuel assemblies may have been discharged after only two cycles in the reactor, this non-conservatism needed to be addressed. To address this non-conservatism, the licensee stated that "[t]he core average moderator exit temperature for Cycle 6-7 fuel is more than 8 °F lower than the value assumed for uprated core conditions with minimum flow." If the average core-exit moderator temperature is lower, then the maximum core-exit moderator temperature would be lower as well. However, it is uncertain how much lower the Cycle 6 and 7 maximum core-exit moderator temperatures were as compared to the burnup dependent maximum core-exit moderator temperature determined in the response to RAI No. 30 (Reference 8). But, using NUREG/CR-6665 to evaluate the relative worth of the two effects it would not take much conservatism in the core-exit moderator temperature to offset the reactor soluble boron non-conservatism identified by the licensee. In addition, in the response to RAI No. 30, the licensee increased the reserved analytical margin to 0.005 Δk_{eff}. There may be margin in the maximum core-exit moderator temperature for Cycles 6 and 7 to accommodate the non-conservatism in the soluble boron concentration used in the depletion analysis, however there is margin present in the reserved analytical margin to account for this non-conservatism.

For fuel and moderator temperatures, NUREG/CR-6665 recommends using the maximum operating temperatures to maximize ^{241}Pu production. The focus is on the fuel moderator temperature since the fuel and moderator temperatures are linked and the moderator temperature has the larger effect. The WCAP-16721-P analysis used temperatures which are nominal according to the licensee's SPU (Reference 2). The MPS3 SPU LAR, Attachment 5 Table 2.8.3-1 lists the post-SPU nominal core inlet temperature as 556.4°F and the average temperature rise in the core as 71.6°F. This indicates that the nominal core exit temperature is 628°F. Therefore, it appears the temperature used in the analysis is a nominal value rather than a conservative value. In response to a staff question (Reference 4), the licensee indicated the SPU temperatures were based on the minimum allowed reactor coolant system (RCS) flow rate. Since the operating RCS flow rate will be higher than the minimum allowed, the operating temperatures will be lower. In the RAI response, the licensee indicates that they expect the SPU core average moderator temperature and vessel average moderator temperature will be approximately 2.5°F below the SPU's Table 2.8.3-1 values at nominal RCS flow. The RAI response also indicated the reactor vessel average temperature will be procedurally limited to the SPU's Table 2.8.3-1 value of 589.5°F. This provides some assurance that the temperatures in the SPU's Table 2.8.3-1 would not be exceeded. However the limiting moderator core exit temperature has not been identified. When the RCS flow is nominal, the average core exit temperature will be approximately 625.5°F. The maximum core exit temperature with nominal RCS flow has not been determined. Additionally, since full power operation is permitted at the minimum TS allowed RCS flow rate, nothing would prevent the licensee from operating at the minimum RCS flow rate. As the RCS flow rate decreases, the core moderator temperatures would increase, eventually to a point where 628°F would no longer be bounding. In response to staff questions, the licensee identified the maximum core exit temperature, which is higher than the 628°F used in WCAP-16721-P. The licensee addressed this by identifying that maximum core-exit temperature of the limiting fuel assembly will exceed the average core-exit temperature by a significant amount. In response to RAI No. 30 (Reference 8), the licensee determined a burnup dependent maximum core-exit temperature profile. The burnup dependency recognizes that a fuel assembly cannot physically be at the maximum core exit temperature for its entire life; it recognizes as it is depleted its relative power diminishes and so will its core exit temperature. The temperature profile was developed from the four most recent cycles and would include one cycle at SPU conditions. In response to RAI No. 30, the licensee combined the burnup dependent maximum core-exit temperature profile with the site-specific burnup profiles from the response to RAI No. 21 (Reference 6). The licensee did this to determine a reactivity penalty with regard to the lower core-exit temperature and axial burnup profile used in WCAP-16721-P. Consistent with NUREG/CR-6801, the axial burnup profile used in WCAP-16721-P provides considerable margin at the highest burnups considered in this analysis and is sufficient to offset the effect of the higher core-exit temperature.

NUREG/CR-6665 does not contain a recommendation for specific power and operating history. NUREG/CR-6665 estimated this effect to be on the order of 0.002 Δ k_{eff} using the operating histories it considered. Based on the difficulty of reproducing a bounding or even a representative power operating history, NUREG/CR-6665 merely recommends using a constant power level and retaining sufficient margin to cover the potential effect of a more limiting power history. The licensee used a constant core power for the depletion calculations. To address this item, the licensee added 0.002 Δ k_{eff} as a bias with the burnup penalty determination in the response to RAI No. 30 (Reference 8).

NUREG/CR-6665 does not have a specific recommendation for fuel assembly fixed burnable poisons such as Burnable Poison Rod Assemblies (BPRAs) and Axial Power Shaping Rod Assemblies (APSRs). NUREG/CR-6665 merely identifies fixed burnable poisons as another mechanism that hardens the neutron spectrum. Since a hardened neutron spectrum increases ^{241}Pu production, fixed burnable burnable poisons have the potential to increase the reactivity of the depleted fuel. NUREG/CR-6761, "Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit," (Reference 13), provides a more in-depth look at the effect of fixed burnable poisons on the final reactivity of a depleted fuel assembly. In response to a staff question (Reference 4), the licensee indicated that they do not utilize fixed burnable poisons such as BPRAs and APSRs. However, in response to another staff question (Reference 7), the licensee indicated that several cycles did utilize reactivity inserts. The licensee addressed this by evaluating the only three fuel batches, B, C, and D, at MPS3 that have ever had a reactivity control insert of any type. Batches B and C have a maximum enrichment of 3.7% without ^{235}U, while Batch D has a maximum enrichment of 3.8% without ^{235}U. These batches were evaluated on a Region-by-Region basis in the response to RAI No. 29 (Reference 7). The NRC staff finds the evaluation to be reasonable and acceptable. The use of any other reactivity control inserts of any type, other than those for fuel batches B, C, and D at MPS3 is not covered by the licensing basis.

NUREG/CR-6665 does not have a specific recommendation for integral burnable poisons such as Westinghouse's Integral Fuel Burnable Absorbers (IFBA). NUREG/CR-6665 merely identifies fixed burnable poisons as another mechanism that hardens the neutron spectrum. Since a hardened neutron spectrum increases ^{241}Pu production integral burnable poisons have the potential to increase the reactivity of the depleted fuel. NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit" (Reference 14), provides a more in-depth look at the effect of integral burnable poisons on the final reactivity of a depleted fuel assembly. NUREG/CR-6760 concluded it is non-conservative to ignore the presence of IFBA when performing the depletion portion of an SNF criticality analysis. WCAP-16721-P does not model integral burnable poisons. In response to a staff question (Reference 4), the licensee indicated that they do utilize the IFBA integral burnable poison. In that response the licensee claims that it is conservative to ignore the presence of IFBA when performing the depletion portion of an SNF criticality analysis. The licensee's submittal states that they performed two sets of analyses; one in which all residual IFBA were artificially removed after the depletion, and one in which all residual IFBA were retained after the depletion. The licensee's submittal indicates that when the residual IFBA are artificially removed, the effect of neutron spectral hardening is shown, but when the residual IFBA are left in the fuel assembly, the residual IFBA overcome the neutron spectral hardening with a conservative result. There is no indication in NUREG/CR-6760 that any residual IFBA were artificially removed in reaching its conclusions. The information presented in NUREG/CR-6760 indicates that any residual IFBA were left in the fuel assembly when determining the effect. In response to a staff question (Reference 8), the licensee reevaluated its modeling of IFBA and determined that it is non-conservative to ignore the presence of IFBA when performing the depletion portion of an SNF criticality analysis. The licensee addressed this by reevaluating its modeling of IFBA. The licensee had been modeling the IFBA as extending the full length of the fuel assembly. IFBA are not full length. Modeling the IFBA as full length results in excess residual IFBA that is sufficient to overcome the spectral hardening the IFBA cause. When modeled correctly, the licensee determined an IFBA penalty that is consistent with NUREG/CR-6760. The IFBA penalty was compiled with other penalties and adjustments in the response to RAI No. 30.

2.4.6 Determination of Soluble Boron Requirements

The MPS3 licensing basis does not credit soluble boron for nominal conditions. However, the MPS3 licensing basis does credit soluble boron for abnormal/accident conditions to ensure k_{eff} of the spent fuel storage racks, loaded with fuel of the maximum fuel assembly reactivity, will not exceed 0.95, at a 95 percent probability, 95 percent confidence level, during these conditions. The abnormal/accident analysis considered several scenarios, but determined that a miss-loading of a fresh fuel assembly into the Region 3 '4-out-of-4' storage configuration was the most limiting. This scenario requires 402 ppm of soluble boron to meet the revised analytical acceptance criterion of $k_{eff} < 0.945$, at a 95 percent probability, 95 percent confidence level, if flooded with borated water. MPS3 TS 3/4.1.2 requires 800 ppm of soluble boron to be present in the SFP whenever fuel assemblies are in the SFP.

Thus, there is some margin to the TS limit to address issues such as modeling the grid strips identified in Section 2.4.4 of this evaluation. The analysis provided in Reference 4 shows the potential non-conservatism to be small and would only require a few ppm of soluble boron for the most limiting abnormal/accident scenario described above. The biases and uncertainties were not determined with respect to the presence of soluble boron in the SFP. The changes in the analysis that were made to address the identified non-conservatisms in the unborated portion, would likely affect the determination of the soluble boron requirements. However, given the level of credit taken in the analysis these effects would likely be small and insufficient to challenge the TS limit of 800 ppm.

2.5 CONCLUSION

The MPS3 licensing basis for the SFP is for k_{eff} to be less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level, if flooded with unborated water under nominal conditions. For abnormal/accident conditions, the MPS3 licensing basis allows credit for soluble boron to maintain k_{eff} less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level.

The NRC staff's evaluation of the licensee submittal for the unborated criteria found several non-conservatisms. In the response to RAI No. 30 (Reference 8), the licensee showed: 1) a 0.002 Δk_{eff} bias to account for reactor operating power history; 2) an IFBA reactivity penalty to account for the spectral hardening effect the integral burnable absorbers have; 3) a combined reactivity penalty for the core-exit temperature and the axial burnup profile used in WCAP-16721-P; and 4) an increase in the reserved analytical margin from 0.001 Δk_{eff} to 0.005 Δk_{eff}. These were summed and converted to burnup penalties for the new Region 2 and Region 3 storage requirements. The burnup penalties were used to adjust the burnup and enrichment loading curves for post-SPU storage requirements in the revised Region 2 Figure 3.9-3 and the new Region 3 Figure 3.9-5.

The Region 1 portion of the analysis was evaluated, and since the '3-out-of-4' storage configuration has no burnup requirements associated with it and the '4-out-of-4' storage configuration is controlled by a burnup and enrichment loading curve that requires very little burnup, it was concluded that these portions of the analysis were not adversely affected. The current Region 3 Figure 3.9-4 is restricted to pre-SPU discharged fuel assemblies and is not part of this evaluation.

The above burnup penalties addressed most of the NRC staff's concerns with the initial analysis and subsequent RAI responses. To address the remainder, the licensee took the following actions:

- The current non-blanketed fuel is prevented from being stored in Region 2 by a footnote to the revised TS Figure 3.9-3.

- The current non-blanketed fuel is prevented from being stored in Region 3 by the new TS Figure 3.9-5 because this figure is only applicable to post-SPU discharged fuel assemblies. The current non-blanketed fuel can be stored in Region 3 as long as it meets the requirements of the TS Figure 3.9-4.

- Both the revised TS Figure 3.9-3 and the new TS Figure 3.9-5 have footnotes that delineate the axial blankets that may be credited in determining acceptability for storage. Fuel assemblies that do not meet those requirements must be stored as non-blanketed.

The licensee identified the two non-conservatisms listed below and the offsetting conservatisms. However, the NRC staff believes that applying some of the increased reserved analytical margin is the best way to address these non-conservatisms. There remains enough reserved analytical margin to support a reasonable assurance determination.

- A non-conservative temperature bias for Region 1 and Region 2 that was not included in the original analysis was identified.

- A non-conservatism associated with the amount of soluble boron used in the depletion analysis was identified.

The NRC staff's evaluation of the licensee submittal for the borated criteria found several potential issues, but used engineering judgment to determine that they are bound by the presence of excess soluble boron in the MPS3 SFP.

Therefore, the NRC staff finds, based on the above evaluation, that there is reasonable assurance that the SFP k_{eff} will be less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level, if flooded with unborated water under nominal conditions, and for abnormal/accident conditions the k_{eff} will be less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level with borated water.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 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 such findings (73 FR 2549 and 74 FR 46241). 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

5.0 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) 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.

6.0 REFERENCES

1. Incorporated letter 07-0450, Gerald T. Bischof, Vice President Nuclear Engineering, to USNRC document control desk, re: Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, License Amendments Request, Stretch Power Uprate, July 13, 2007. (ADAMS ML072000386)

2. Dominion Nuclear Connecticut, Incorporated letter 07-0450, Gerald T. Bischof, Vice President Nuclear Engineering, to USNRC document control desk, re: Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, License Amendments Request, Stretch Power Uprate-Supplemental Information, July 13, 2007. (ADAMS ML072000281)

3. Dominion Nuclear Connecticut, Incorporated letter 07-0450D, Gerald T. Bischof, Vice President Nuclear Engineering, to USNRC document control desk, re: Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, Supplemental Information Regarding Stretch Power Uprate License Amendments Request," March 5, 2008. (ADAMS ML080660108)

4. Dominion Nuclear Connecticut, Incorporated letter 08-0511A, Leslie N. Hartz, Vice President Nuclear Support Services, to USNRC document control desk, re: Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding A Spent Fuel Pool Storage License Amendment Request, September 30, 2008. (ADAMS ML082770113)

5. Dominion Nuclear Connecticut, Incorporated letter 09-084, J. Alan Price, Vice President Nuclear Engineering, to USNRC document control desk, re: Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding A Spent Fuel Pool Criticality License Amendment Request, March 5, 2009. (ADAMS ML090650513)

6. Dominion Nuclear Connecticut, Incorporated letter 09-084A, J. Alan Price, Vice President Nuclear Engineering, to USNRC document control desk, re: "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Regarding A Spent Fuel Pool Criticality License Amendment Request, March 23, 2009. (ADAMS ML090890189)

7. Dominion Nuclear Connecticut, Incorporated letter 10-072, Leslie N. Hartz, Vice President Nuclear Support Services, to USNRC document control desk, re: Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, Response to Request for Additional

Information Questions 26, 27, and 29 Regarding A Spent Fuel Pool Criticality License Amendment Request, March 1, 2010. (ADAMS ML100610104)

8. Dominion Nuclear Connecticut, Incorporated letter 10-072, Leslie N. Hartz, Vice President Nuclear Support Services, to USNRC document control desk, re: Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 3, Response to Request for Additional Information Questions 28 and 30 Regarding A Spent Fuel Pool Criticality License Amendment Request, March 5, 2010. (ADAMS ML100640343)

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 ML003728001)

10. NUREG/CR-6665, Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel (ADAMS ML003688150)

11. NUREG/CR-6801, Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis (ADAMS ML031110292)

12. DOE Topical Report DOE/RW-0472, "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages," dated September 1998, Revision 2 (ADAMS ML070780665 – non-publically available)

13. NUREG/CR-6761, "Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit" (ADAMS ML020770329)

14. NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit" (ADAMS ML020770436)

Principal Contributor: K. Wood

Date: March 26, 2010

March 26, 2010

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

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: SPENT FUEL POOL CRITICALITY (TAC NO. MD8251)

Dear Mr. Heacock:

The Commission has issued the enclosed Amendment No. 248 to Renewed Facility Operating License No. NPF-49 for the Millstone Power Station, Unit No. 3 (MPS3), in response to your application dated July 13, 2007.

The amendment makes changes to the Technical Specifications (TS) for MPS3 spent fuel pool (SFP) storage requirements. By letter dated July 13, 2007 (Dominion Nuclear Connecticut, Inc. (DNC or the licensee), submitted a license amendment request for a stretch power uprate (SPU) of MPS3. Included in a supplement dated July 13, 2007, was a request to make changes to the TSs for MPS3 SFP storage. By letter dated March 5, 2008, DNC separated the MPS3 SFP storage requirements request from the MPS3 SPU request.

The July 13, 2007, request was supplemented by letters dated September 30, 2008, March 5, 2009, March 23, 2009, March 1, 2010, and March 5, 2010.

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/
Carleen J. Sanders, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures:
1. Amendment No. 248 to NPF-49
2. Safety Evaluation

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ADAMS Accession NO.: ML100750024 *** via email** ^ - OGC Comment review 3/23/10

OFFICE	NRR/LPLI-2/PM	NRR/LPLI-2/LA	NRR/SRXB/BC	OGC	NRR/LPLI-2/BC
NAME	CSanders	ABaxter * ^	GCranston SMiranda for	DRoth NLO	HChernoff
DATE	3/25/2010	3/16/10	3/16/2010	3/17/2010	3/26/10

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