August 8, 2008

Mr. David A. Christian
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION - UNIT 3 - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE SPENT FUEL POOL CRITICIALTY AMENDMENT REQUEST (TAC NO. MD8251)

Dear Mr. Christian:

By letter dated July 13, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072000386), Dominion Nuclear Connecticut, Inc. (DNC) submitted a license amendment request (LAR) for a stretch power uprate (SPU) of Millstone Power Station, Unit 3 (MPS3). Included via a supplement dated July 13, 2007 (ADAMS Accession No. ML072000281), was a request to amend the MPS3 spent fuel pool (SFP) storage requirements. By letter dated March 5, 2008 (ADAMS Accession No. ML080660108), DNC separated the MPS3 SFP storage requirements request from the MPS3 SPU request. The U.S. Nuclear Regulatory Commission staff has reviewed the information DNC provided on the SFP storage requirements and determined that additional information is required in order to complete the evaluation as set forth in the enclosure.

The draft questions were sent to Mr. Geoffrey Wertz, of your staff, to ensure that the questions were understandable, the regulatory basis for the questions was clear, and to determine if the information was previously docketed. During a phone call with Mr. Wertz on July 17, 2008, it was agreed that you would provide a response within 45 days of the date of this letter. Please note that this response time may impact DNC's requested review date of November 14, 2008. This impact has been discussed with Mr. Wertz and Mr. William Bartron. Also note that if you do not respond to this letter by the agreed-upon date or provide an acceptable alternate date in writing, we may reject your application for amendment under the provisions of Title 10 of the *Code of Federal Regulations*, Section 2.108. If you have any questions, please contact me at (301) 415-1603.

Sincerely,

/ra/ (John G. Lamb for)

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

Docket No. 50-423

Enclosure:
Request for Additional Information

cc w/ enclosure: See next page

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REQUEST FOR ADDITIONAL INFORMATION

SPENT FUEL POOL CRITICALLITY

MILLSTONE POWER STATION UNIT 3

DOCKET NUMBER 50-423

Dominion Nuclear Connecticut, Inc. (DNC) submitted a license amendment request (LAR) for a stretch power uprate (SPU) of Millstone Power Station, Unit 3 (MPS3) dated July 13, 2007 (Reference 1). Included in supplemental information to DNC's LAR was a request to amend the MPS3 spent fuel pool (SFP) storage requirements (Reference 2). The Technical Specification (TS) changes associated with the SFP are found in Attachment 3 of Reference 1. The technical justification for the SFP LAR is located in Attachment 2 of Reference 2. The request to amend the MPS3 SFP storage requirements was separated from the SPU LAR in Reference 3, dated March 5, 2008.

The MPS3 SFP is divided into three regions. The storage racks in Regions I and II contain a permanently installed absorber: BORAL. The storage racks in Region III contain a permanently installed absorber: Boraflex. Boraflex is subject to known degradation mechanisms in SFP environments. Due to the degradation at MPS3, Boraflex is not credited for maintaining SFP sub-criticality requirements.

There are no proposed TS changes for Region I. Region I currently has two possible storage configurations: a 4-out-of-4 storage configuration in which the fuel assemblies must meet the burnup/enrichment requirements in TS Figure 3.9-1, and a repeating 3-out-of-4 storage configuration in which one cell in the 2x2 array must be empty, as shown in TS Figure 3.9-2. There is no limitation being imposed on Region I with respect to pre-SPU or post-SPU fuel.

Currently, Region II has one storage configuration: a 4-out-of-4 storage configuration in which the fuel assemblies must meet the burnup/enrichment requirements in TS Figure 3.9-3. The LAR proposes to change TS Figure 3.9-3 to incorporate Decay Time into the burnup/enrichment requirements. The Decay Time includes the decay of Plutonium – 241 (^{241}Pu) and the corresponding buildup of Americium – 241 (^{241}Am). There is no limitation being imposed on Region I with respect to pre-SPU or post-SPU fuel.

Currently, Region III has one storage configuration: a 4-out-of-4 storage configuration in which the fuel assemblies must meet the burnup/enrichment requirements in TS Figure 3.9-4. TS Figure 3.9-4 currently includes Decay Time in the burnup/enrichment requirements. The LAR proposes to change TS Figure 3.9-4 to limit it to pre-SPU fuel assemblies. Figure 3.9-5 is being proposed as an addition to include post-SPU fuel assemblies in Region III. Figure 3.9-5 will include Decay Time in the burnup/enrichment requirements.

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

Enclosure

As required in 10 CFR 50.68(b)(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."

As required in 10 CFR 50.68(b)(4), "If no credit for soluble boron is taken, the k-effective [k_{eff}] 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."

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

The current MPS3 TS 5.6.1.1 states, "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 current MPS3 TS 3.9.13 limiting condition of operation (LCO) states, "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." However, with k_{eff} greater than 0.95 Action a for MPS3 TS 3.9.13 states, "Borate the Spent Fuel Pool until k_{eff} is less than or equal to 0.95, and. . ." The previous MPS3 SFP license amendment, (Reference 4) demonstrated that k_{eff} was less than or equal to 0.95 with unborated water under nominal conditions and that k_{eff} was less than or equal to 0.95 with borated water under abnormal/accident conditions. Therefore, the MPS3 licensing basis is to maintain the SFP k_{eff} 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 to maintain the SFP k_{eff} less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level with borated water under abnormal/accident conditions.

The current LAR intends to maintain the MPS3 SFP k_{eff} 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 to maintain the SFP k_{eff} less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level with borated water under abnormal/accident conditions.

The Nuclear Regulatory Commission (NRC) staff requests responses to the following questions in order to continue its review of the MPS3 SFP LAR.

1) WCAP-16721 Section 2.2, Axial Burnup Distribution Modeling, indicates benchmark analyzes where performed to justify the axial nodalization used in the criticality analysis. Provide the description and results of those benchmarks analyzes.

2) WCAP-16721 Section 2.2, Axial Burnup Distribution Modeling, states, "Input to this analysis is based on a limiting axial burnup profile data provided in the Department of Energy [DOE] Topical Report, as documented in Reference 12. The burnup profile in the DOE topical report is based on a database of 3169 axial burnup profiles for pressurized water reactor [PWR] fuel assemblies compiled by Yankee Atomic. This profile is derived from the burnups calculated by utilities or vendors based on core-follow calculations and in-core measurement data." However, the DOE Topical Report, (Reference 8 herein) does not

have a limiting axial burnup profile. Rather, burnup is divided into 12 groups with each interval having a limiting axial burnup profile. For ease of use, those 12 groups are compressed into three intervals. The axial burnup profile indicated in Figure 2-1 is indicated by the DOE Topical Report and NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis," (Reference 6) as being conservative for burnups greater than or equal to 30 giga watt day per metric ton uranium (GWD/MTU), but non-conservative below 30 GWD/MTU. WCAP-16721 has used this burnup profile for burnups of 5, 15, and 25 GWD/MTU. Provide a justification for the use of this axial burnup profile below 30 GWD/MTU.

3) WCAP-16721 Section 2.2, Axial Burnup Distribution Modeling, states, "A key aspect of the burnup credit methodology employed in this analysis is the inclusion of an axial burnup profile correlated with feed enrichment and discharge burnup of the depleted fuel assemblies. This effect can be important in the analysis of the fuel assembly characteristics when the majority of spent fuel assemblies stored in the MPS3 spent fuel pool have a discharge burnup well beyond the limit for which the assumption of a uniform axial burnup shape is conservative. Therefore, it is necessary to consider both uniform and axially distributed burnup profiles, and the more conservative representation will be utilized to determine fuel assembly storage requirements." Subsequent statements indicate that only a uniform axial burnup profile was used for the "Region I 4-out-of-4" storage configuration whereas a uniform axial burnup profile was used in the Region II and Region III storage configurations to compare with the results of the axially distributed profile mentioned above. These statements indicate WCAP-16721 has used a uniform axial burnup profile for burnups of 5, 15, 25, 35, 45, and 55 GWD/MTU. A uniform axial burnup profile is generally accepted as conservative at low burnup. There is a transition point at which the uniform axial profile becomes non-conservative. As burnup increases beyond this transition point, the uniform axial burnup profile becomes ever more non-conservative. However, exactly where a uniform axial burnup profile transitions from conservative to non-conservative is not generically known. There is no evidence in the LAR that this transition point was established for MPS3 fuel. Based on literature familiar to the NRC staff, a uniform axial burnup should be considered in the following manner: conservative for burnup (BU) < 10 GWD/MTU, non-conservative for BU > 20 GWD/MTU, indeterminate for BU between those values. Provide a justification for the use of the uniform axial burnup profile above 10 GWD/MTU.

4) WCAP-16721 Section 2.2, Axial Burnup Distribution Modeling, states, "Table 3-5 lists the fuel and moderator temperatures employed in the depletion calculations for the assembly-average burnup model and each node of the . . . axial burnup model. These values are based on conservative temperature profiles for Millstone Unit 3 at uprated conditions. The use of uprated conditions for depletion calculations - with increased power, moderator temperatures and fuel temperatures - lead to increased reactivity determinations at any given burnup relative to fuel irradiated in the core prior to the uprate." Table 3-5 indicates the core exit temperature used in the analysis is approximately 628° Fahrenheit (F). MPS3 SPU LAR, (Reference 1), 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 makes the nominal core exit temperature 628°F. Therefore, the temperature used in the analysis is a nominal value rather than a conservative value. MPS3 SPU LAR, (Reference 1), Attachment 5, Table 2.8.3-1 lists the pre-SPU nominal core exit temperature would be approximately 623.5°F. So while the post-SPU nominal core exit temperature exceeds the

pre-SPU core exit temperature, it is not clear whether or not it bounds the pre-SPU maximum core exit temperature. Provide a justification for the use of the nominal core moderator and fuel temperatures in the depletion calculations.

5) NRC staff guidance is to use the most reactive fuel (Reference 7). NUREG/CR-6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR [light water reactor] Fuel," (Reference 8) 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, since the phenomena occur in the reactor as the fuel is being used. The results have some translation to SFP criticality analyses, especially when the discussion includes the effect in an infinite lattice analysis, similar to that 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 effect is similar: Doppler broadening/spectral hardening of the neutron field resulting in maximum ^{241}Pu production. NUREG/CR-6665 provides an estimate of the reactivity worth of these parameters. Other than moderator/fuel temperature and soluble boron, none of the other core operating parameters are discussed in WCAP-16721. Provide a discussion of these other core operating parameters and their impact on the MPS3 SFP criticality analysis.

6) WCAP-16721 Section 2.4, Methodology Assumptions, lists some of the assumptions used in the criticality analysis. One assumption states, "The design basis limit for k_{eff} is conservatively reduced from 0.95 to 0.949 for this analysis." However, maintaining k_{eff} less than or equal to 0.95 at all times is not a design basis limit, it is a regulatory limit. Therefore, the analysis is only reserving 0.001 delta (Δ) k_{eff} analytical margin to the MPS3 licensing basis limit. Any identified non-conservatism or potential non-conservatism that can not be offset by an explicitly known conservatism will erode the reserved analytical margin. A significant erosion of the reserved analytical margin will preclude a reasonable assurance determination by the NRC staff. Please provide the following information regarding the assumptions listed in Section 2.4.

   a) Nominal and tolerances for fuel stack density. Identify the conservatism associated with using a fuel stack density of 10.686 gm/cm^3. Identify how this conservatism carries through to depleted fuel.

   b) Nominal and tolerances for pellet dishing and chamfering. Identify the conservatism associated with modeling pellets as full right circular cylinders. Identify how this conservatism carries through to depleted fuel.

   c) Justification for not modeling fuel assembly grids.

   d) The reactivity worth of not modeling uranium -234 (^{234}U) uranium-236 (^{236}U) in fresh fuel. Identify how this conservatism carries through to depleted fuel.

   Alternatively, the licensee may quantify conservatisms elsewhere in the analysis.

7) WCAP-16721 Section 3.5, Fuel Assembly Design Parameters, states, "The design parameters of the Westinghouse 17x17 STD [standard], V5H [vantage 5H], RFA [robust fuel assembly] and NGF [next generation fuel] fuel assembly types are summarized in Table 3-4 and Figure 3-3. Note that for the purposes of this analysis, the RFA and NGF dimensions are identical and will be treated as one fuel type. Simulations were performed for each storage configuration in this analysis to determine the fuel assembly combinations that produce the highest reactivity."

   a) WCAP-16498-P, "17x17 Next Generation Fuel (17x17 NGF) Reference Core Report," (Reference 9) has not yet been reviewed and approved by the NRC staff. Therefore NGF should not be considered as part of the NRC staff's review of this LAR. What are the licensee's plans to address the final approved version of WCAP-16498-P affects on this analysis?

   b) Provide a description and results of the simulations used to determine the fuel assembly combinations that produce the highest reactivity.

8) WCAP-16721 Section 4.2, Bias and Uncertainty Calculations, states, "Applicable biases factored into this evaluation are: 1) the methodology bias deduced from the validation analyses of pertinent critical experiments; and 2) any reactivity bias, relative to the reference analysis conditions, associated with operation of the spent fuel pool over a temperature range of 32°F to 160°F." However, WCAP-16721 Section 2.1.1, The SCALE Code, states, "The SCALE version 4.4 version of the 238-group ENDF/B-V neutron cross section library is also utilized in this analysis. However, this library is only utilized for off-nominal temperature simulations (greater than 68 °F). The 238-group library is a general purpose library that is applicable at all temperatures. The 44-group library was collapsed using a representative spectrum from a 17x 17 PWR assembly at 68°F, so any deviations from these conditions should be considered as potentially moving outside the basis of applicability for this specialized library. In addition, these calculations are only considered in a relative sense, to establish the reactivity changes due to temperature deviations. Since there is no need to quantify the absolute magnitude of the reactivity at these conditions, a comprehensive validation analysis is not performed for the 238-group neutron cross section library."

   a) Provide a description of how the effect of temperatures below 68°F was evaluated.

   b) Provide a description and results of the simulations used to determine the presence or absence of a temperature bias.

9) WCAP-16721 Section 4.2, Bias and Uncertainty Calculations, states, "For item c., the fuel rod manufacturing tolerance for the reference design fuel assembly consists of the following components; an increase in pellet diameter […], a decrease in fuel cladding thickness […] and an increase in fuel enrichment of […]." However, neither the Region I 3-out-of-4 nor the Region I 4-out-of-4 storage configurations include the pellet diameter as an uncertainty. Provide a quantitative justification for why the pellet diameter was not included in those uncertainty determinations.

10) WCAP-16721 Section 4.2, Bias and Uncertainty Calculations, states, "For item d., the following component tolerances are varied to their outer bounds: the stainless steel canister inner dimension and thickness, the storage cell center-to-center spacing, the BORAL poison loading and the Boraflex channel wrapper thickness."

a) Provide a quantitative justification for why uncertainties for the BORAL panel width, thickness and wrapper material thickness are not included the Region I and Region II storage configurations.

b) Provide a quantitative justification for why uncertainties for the "…stainless steel canister inner dimension and thickness, the storage cell center-to-center spacing…," are not included for Region II.

c) Provide a quantitative justification for why an uncertainty for the "……stainless steel canister inner dimension …," is not included for Region III.

11) WCAP-16721 Section 4.2, Bias and Uncertainty Calculations, states, "In the case of the tolerance due to positioning of the fuel assembly in the storage cells (item e.), all nominal calculations are carried out with fuel assemblies centered in the storage cells. Simulations are run to investigate the effect of off-center position of the fuel assemblies for each of the fuel assembly storage configurations. These cases positioned the assemblies as close as possible in four adjacent storage cells and at intermediate positions in between. For the Region I and Region II racks containing BORAL, the centered or intermediate positions may yield the highest reactivity due to the proximity of the BORAL fixed neutron poison panels and any increased reactivity at intermediate positions is accounted for as an uncertainty. The highest reactivity condition for the Region III racks is determined to occur when the assemblies are as close as allowable within the storage cells." However, the details of those simulations are never discussed and only Region III has an uncertainty attributed to Off-Center Assembly Positioning.

a) Provide a description and results of the simulations used to determine the fuel assembly off-center combinations that produce the highest reactivity for each storage configuration.

12) WCAP-16721 Section 4.2, Bias and Uncertainty Calculations, states, "For item f., a depletion determination uncertainty […] is included in the final summation of biases and uncertainties. Since the depletion determination uncertainty is dependent on the magnitude of the burnup credited in the analysis, it is determined iteratively at each initial enrichment considered in a storage configuration." This method of determining the burnup uncertainty is a deviation from NRC staff guidance contained in Reference 7. The NRC staff estimates that the method used to determine the burnup uncertainty may result in a smaller uncertainty than the method contained in the NRC staff guidance. However, as this analysis actually treats the burnup uncertainty as a bias, the alternate method may be acceptable. To make that determination, the NRC staff requests the following information:

a) The basis for the WCAP-16721-P method for determining the burnup uncertainty.

b) The zero (0) burnup reactivity for each initial enrichment considered in a storage configuration.

13) It is unclear from WCAP-16721-P Table 4-5, Table 4-6, and Table 4-7, whether or not the Enrichment Uncertainty contains the KENO V.a case uncertainties for those storage configurations as it does for the "Region I 3-out-of-4" storage configuration as indicated by Table 4-1. Confirm that the KENO V.a case uncertainties are included in all Enrichment Uncertainties.

14) The Depletion Uncertainty for 3.0 without Initial Enrichment in WCAP-16721-P Table 4-6, "Region II" Storage Configuration Total Biases and Uncertainties Results, appears to be

incorrect. Confirm the actual depletion uncertainty for 3.0 weight percent initial enrichment for Region II.

15) The Depletion Uncertainty for 3.0 without Initial Enrichment in WCAP-16721-P Table 4-7, "Region III" Storage Configuration Total Biases and Uncertainties Results, appears to be incorrect. Confirm the actual depletion uncertainty for 3.0 weight percent initial enrichment for Region III.

16) The k_{eff} values in WCAP-16721-P Table 4-8, Table 4-9, and Table 4-10 storage configurations all have KENO V.a case uncertainties associated with them. However, those KENO V.a case uncertainties are not used in determining the Burnup Uncertainty or the burnup necessary to meet the target k_{eff} values. This is non-conservative. Therefore, since the burnup necessary to meet the target k_{eff} values is used directly in establishing the burnup/enrichment relationships in Section 5, this non-conservatism acts as a bias. Given the small amount of reserved analytical margin, the licensee should identify offsetting conservatism in the analysis.

17) WCAP-16721 Section 4.5.2, Soluble Boron Required to Mitigate Postulated Accident Effects, does not consider the misloading of a 5.0 weight percent ^{235}U fresh fuel assembly into location required to be empty in the Region I 3-out-of-4 storage configuration. Provide a justification for not including this accident scenario; include a description of the blocking device used and the controls which govern it.

18) Describe the process used to determine that fuel assemblies have attained proper burnup for storage in the burnup dependent racks.

19) Describe the process used to control movement of items within the SFP.

REFERENCES

1. Dominion Nuclear Connecticut, Inc., 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 Amendment Request, Stretch Power Uprate," July 13, 2007. (ADAMS ML072000386)

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

3. Dominion Nuclear Connecticut, Inc., 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, License Amendment Request, Stretch Power Uprate – Supplemental Information," March 5, 2008. (ADAMS ML080660108)

4. U.S. NRC letter to Northeast Nuclear Energy Company, "Millstone Nuclear Power Station, UNIT NO. 3 – Notice of Issuance of Amendment to Facility Operating License and Final Determination of No Significant hazards Consideration (TAC NO. MA5137)," dated November 28, 2000. (ADAMS ML003771974)

5. DOE/RW-0472, "Topical Report on Actinide-Only Burnup Credit for PWR Spent Fuel Packages," Revision 2, September 1998.

6. NUREG/CR-6801, “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis”

7. 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)

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

9. WCAP-16498-P, “17x17 Next Generation Fuel (17x17 NGF) Reference Core Report,” March 2008. (ADAMS ML081010602)

Millstone Power Station, Unit No. 3

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