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March 1, 2010

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No.: 10-072
NLOS/WDC R0
Docket No.: 50-423
License No.: NPF-49

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

Dominion Nuclear Connecticut, Inc. (DNC) submitted a stretch power uprate (SPU) license amendment request (LAR) for Millstone Power Station Unit 3 (MPS3) in letters dated July 13, 2007 (Serial Nos. 07-0450 and 07-0450A). The SPU license amendment request was supplemented in a letter dated December 13, 2007 (Serial No. 07-0450C). The SPU LAR included a revised spent fuel pool (SFP) criticality analysis with proposed changes in technical specification (TS) requirements. DNC separated the MPS3 SFP TS change request from the MPS3 SPU request via letter dated March 5, 2008 (Serial No. 07-0450D).

In a letter dated August 8, 2008, the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) regarding the SFP TS. DNC responded to RAI questions 1 through 19 in a letter dated September 30, 2008 (Serial No. 08-0511A). Additionally, in a letter dated February 2, 2009, the NRC requested additional information. DNC responded to RAI questions 20, 22, 23, and 25 in a letter dated March 5, 2009 (Serial No. 09-084) and to RAI questions 21 and 24 in a letter dated March 23, 2009 (Serial No. 09-084A). Subsequently, in a letter dated January 26, 2010, the NRC requested additional information.

Attachment 1 of this letter provides responses to RAI questions 26, 27 and 29. Responses to RAI questions 28 and 30 will follow in a separate letter.

The information provided by this letter does not affect the conclusions of the significant hazards consideration discussion in the December 13, 2007 DNC letter (Serial No. 07-0450C).

Should you have any questions in regard to this submittal, please contact Wanda Craft at 804-273-4687.

Sincerely,

Leslie N. Hartz
Vice President – Nuclear Support Services

Commitments made in this letter:
1. None.

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Support Services of Dominion Nuclear Connecticut, Inc. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 1st day of March, 2010.

My Commission Expires: 4/30/13.

Notary Public

GINGER L. ALLIGOOD
Notary Public
Commonwealth of Virginia
310847
My Commission Expires Apr 30, 2013

Attachment:

1. Attachment 1: Response to Request for Additional Information (RAI) Questions 26, 27 and 29 Regarding the Spent Fuel Pool Criticality License Amendment Request

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT 1

RESPONSE TO RAI QUESTIONS 26, 27 AND 29

REGARDING THE SPENT FUEL POOL CRITICALITY LICENSE AMENDMENT

REQUEST

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

RESPONSE TO RAI QUESTIONS 26 THROUGH 30 REGARDING THE SPENT FUEL POOL CRITICALITY LICENSE AMENDMENT REQUEST

Question 26

In response to RAI # 5 DNC states:

> NUREG/CR-6665, Reference 4, identifies both specific power and operating history effects as weakly correlated to increased reactivity for discharged fuel assemblies. The maximum impact noted in Reference 4 is approximately 0.00200 Δ Keff. This result is caused by reduced power operation near the end of assembly depletion. The depletion calculations supporting the analysis presented in WCAP-16721-P do not include part power operation. Instead, the soluble boron concentration is maintained at a constant value above the cycle average value for the entire depletion. The spectral hardening from the presence of boron, especially at the end of the cycle when the concentration is several hundred ppm above physical values, provides additional margin to account for this potential impact. The use of additional margin is the approach suggested in Reference 4 for accounting for the potential for operating history effects.

In order to determine if the conservatism gained from taking credit for using a higher than actual soluble boron concentration during the simulated depletion of the fuel in the reactor is enough to cover the 0.00200 Δ Keff, the NRC staff needs to know how much more than the actual boron concentration was used. Please provide additional information regarding how much soluble boron concentration was used during the simulated depletion of the fuel in the reactor compared to the actual.

Response:

With regard to specific power and operating history effects, NUREG/CR-6665, Reference 4, makes the following recommendation:

> "It appears that the optimum approach would be to assume a simple continuous-power operating history, and add in a margin to account for operating-history-induced effects."

Consistent with this recommendation, the depletions performed for WCAP-16721-P (including those for request for additional information (RAI) responses through RAI 30) used a constant power throughout the entire burnup range. The power modeled is consistent with uprated core conditions.

NUREG/CR-6665 cites a maximum power history effect of 0.00200 Δk_{eff}, but gives no guidance as to whether the effect should be considered a bias or uncertainty. The source of the recommendation of 0.00200 Δk_{eff} is ORNL/TM-12973 (Reference 3). According to Section 3.4 of ORNL/TM-12973:

> "However, the maximum nonconservatism is less than 0.2%. Thus for simplicity, it is probably best to assume a no-downtime exposure history for cases with both actinides and fission products present, and then include a 0.2% uncertainty in k_{eff}"

Also, it is important to note that the power history effect is not constant but varies with burnup and does not need to be applied to fresh fuel.

If the power history effect is treated as an uncertainty and is root-sum-squared with the other uncertainties, the maximum increase in total bias and uncertainty would be less than 0.00025 Δk_{eff}. For simplicity and conservatism, however, the power history effect will be treated as a constant bias of 0.00200 Δk_{eff} applied to burnt fuel in Regions 2 and 3. This bias will be converted into equivalent burnup using the relationship discussed in the response to RAI 30. For Region 1, the burnup limits are too low to be affected and no penalty for power history is needed because the power history effect is due to reduced power near the end of assembly depletion as noted in the RAI and increases with increasing burnup.

Regarding soluble boron, all simulated depletions were performed using 1000 ppm boron. Table 1 gives the cycle average boron concentration for Cycles 1-12 and the expected average for Cycle 13, which is currently operating at an uprated power of 3650 Mega Watts thermal (MWt).

Table 1 – Cycle Average Boron Concentrations

Cycle	Average Boron in parts per million (ppm)
1	624
2	785
3	737
4	929
5	912
6	1094
7	930
8	812
9	865
10	883
11	905
12	918
13	902

For fuel that will be depleted in uprated power cycles, recent cycles are indicative of current and projected fuel management and cycle average boron concentrations. The maximum cycle average boron concentration for the last 5 completed operating cycles is 918 ppm. Cycle 13 is currently operating at the uprated power (3650 MWt) and is designed to have a cycle average boron concentration of 902 ppm. For future cycles, cycle average boron is expected to remain 50 to 100 ppm below the assumed 1000 ppm.

Millstone Power Station Unit 3 (MPS3) fuel cycles, except for Cycle 6, have had cycle average boron concentrations below 930 ppm. Cycle 6 is an outlier because it was designed as a transition cycle to move from 18 month cycles to 24 month cycles. That transition was not pursued. In addition, Cycle 6 experienced a three year mid-cycle outage and was shut down prior to reaching the design end of cycle. The cycle average boron concentration for Cycle 6 was 1094 ppm. However, all fuel in Cycle 6 was depleted in at least one other cycle and all other cycles had lower cycle average boron concentrations. The highest two cycle average boron concentration for MPS3 cycles is 1008 ppm (lifetime weighted average of Cycles 6 and 7). NUREG/CR-6665 indicates that this amount of excess depletion boron (8 ppm) is worth less than 30 pcm for spent fuel pool (SFP) criticality.

There are several sources of margin to accommodate the 8 ppm of excess boron concentration for Cycle 6-7 fuel, including as-built fuel density and the pre-uprate as-operated moderator temperature. The 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. NUREG/CR-6665 indicates that this amount of moderator temperature difference (~4 °K) amounts to at least 140 pcm margin for depletion history.

The moderator temperature effect alone provides more than sufficient margin to offset the effect of the 8 ppm excess cycle average boron for the Cycle 6-7 fuel. Considering these factors, the assumption of 1000 ppm for simulated depletion of the fuel for the calculation of burnup credit requirements is sufficient for application in past MPS3 cycles and is conservative for application in projected future cycles.

Question 27

In RAI #21 the licensee performed new criticality analysis using site specific burnup profiles.

a. In RAI #21 the licensee determined penalties for burnups where the original analysis in WCAP-16721 was shown to be non-conservative. Those penalties were determined using a depletion code and criticality code different than that used in WCAP-16721. Since the development and utilization of the penalties constitutes more than just a comparison study, either provide a code validation or justify not performing a criticality code validation for the new code.

b. In Table 21-6 there is a Burnup Worth column that is used to determine the penalty. Explain the basis and origin for that column.

c. In RAI #21 the licensee indicates the depletion parameters used were the same as for WCAP-16721. Provide a list of the fuel assembly and depletion parameters used for each case in RAI #21.

d. As part of the response to RAI # 21 the licensee states, "Since the limiting Top-1/3 assembly comparison covers only existing No Blanket fuel from completed MPS3 cycles, credit has been taken for as-built fuel density, dish and chamfer fractions, Pre-Uprate (3411 Mwt) core power, and as operated cycle soluble boron concentration." In earlier RAI responses the licensee took credit for having modeled these parameters conservatively.

    i. Since the No Blanket fuel modeling no longer has those conservatisms explain how the earlier RAI responses are affected.

    ii. Since these parameters had been modeled conservatively in the original analysis there were no biases or uncertainties included for these parameters. Since the No Blanket fuel modeling no longer has those conservatisms explain how the biases and uncertainties are affected.

e. In its RAI #21 response, the licensee states the title to technical specification (TS) Figure 3.9-4 will be changed. That change was not provided as part of the revised TS markup. Please revise the marked-up TS page to indicate that change.

Response, Part 27a:

The analysis reported in WCAP-16721-P uses the PHOENIX-P lattice code to generate depleted isotopic number densities and SCALE Version 4.4 for three-dimensional Monte Carlo calculations to determine neutron multiplication factors (K_{eff}) in the SFP environment. SCALE Version 4.4 uses the BONAMI and NITAWL modules for cross section processing and the KENO V.a module for transport calculations. Reference 1 allows the use of NITAWL-KENO V.a and PHOENIX-P for SFP criticality safety calculations.

RAI 21 presents results from differential calculations performed using the PARAGON lattice code to generate depleted isotopic number densities and SCALE Version 5.1 for three-dimensional Monte Carlo K-eff calculations in the spent fuel environment.

PARAGON was approved for use in reactor physics calculations by the Nuclear Regulatory Commission (NRC) in Reference 2, which evaluated a large body of PARAGON benchmarking data. PARAGON is intended for use as a stand alone code or as a replacement for PHOENIX-P. Reference 2 states that PARAGON is "well qualified as a stand-alone code replacement for the PHOENIX-P lattice code, wherever the PHOENIX-P code is used in NRC-approved methodologies" and that "The PARAGON code can be used as a replacement for the PHOENIX-P lattice code". It is therefore concluded that PARAGON is an acceptable code for the generation of depleted isotopic number densities.

In order to address the concern about use of SCALE Version 5.1, penalty calculations that account for the effect of different axial nodalization and burnup profiles (RAI 21), the effect of Integral Fuel Boron Absorber (IFBA) (RAI 28), and the effect of higher assembly moderator exit temperature (RAI 30) will be performed using PARAGON and SCALE Version 4.4. The combined effect of higher assembly moderator temperature, axial nodalization, and conservative burnup profiles will be presented as part of the RAI 30 response. In this way, use of SCALE 5.1 has been limited to comparison studies and will not be used for any of the calculations used to determine TS burnup and enrichment requirements.

Response, Part 27b:

The burnup worth of a reactivity penalty was calculated by assuming a linear reactivity change over an interval of burnup near the point of potential non-conservatism. The difference in two SCALE Version 4.4 k_{eff} values was divided by the associated burnup difference to determine the reactivity worth associated with an increase in assembly average burnup. New reactivity and burnup penalties are being determined as part of the response to RAI 30, which will be submitted by separate correspondence. The determination of the burnup worth will be described in detail as part of that response.

Response, Part 27c:

Table 5 summarizes the fuel assembly and depletion parameters used in the original WCAP-16721-P analysis and in response to RAI 21 for fuel with natural and mid-enriched blankets. No blanket fuel was treated differently as discussed in response to RAI 21 but is no longer relevant as discussed in response to Part 27d.

Table 5 – Depletion and Fuel Assembly Parameters

Parameter	WCAP-16721-P	RAI 21, natural and mid-enriched blankets
Axial Burnup Distribution	Profile 5 from DOE Topical Report	MPS3 specific
Number of axial zones modeled	4	24
Tin	556.4 °F	556.4 °F
Tout	628 °F	628 °F
Soluble Boron present during depletion	constant 1000 ppm	constant 1000 ppm
Core Power	3650 MWt	3650 MWt
Theoretical Density of fuel	97.5%	97.5%
Fuel pellet shape	solid, right cylinder (i.e. no dishing or chamfering)	solid, right cylinder (i.e. no dishing or chamfering)
Design Basis Fuel Assembly	Westinghouse 17x17 STD	Westinghouse 17x17 STD
Fuel initial enrichments	3.0, 4.0, and 5.0 weight percent (wt%)	3.0, 4.0, and 5.0 wt%
Blankets modeled?	No	Yes

Response, Part 27d:

This question concerns the use of actual fuel parameters (fuel density, power level, boron concentration, etc.) for existing fuel instead of using bounding fuel parameters in Dominion Nuclear Connecticut's (DNC) response to RAI 21. RAI 21 was concerned about fuel reactivity effects due to axial burnup and isotopic conditions at the axial ends of the fuel assembly. The use of actual fuel parameters in DNC's response to RAI 21 was only for those existing fuel assemblies which have no axial blankets, and only for storage in Region 2. DNC did not credit actual fuel parameters in the RAI 21 response for fuel with natural blankets or mid-enriched blankets.

MPS3 fuel with no axial blankets is confined to fuel batches A, B, C and D, which were all operated at the pre-uprate power level of 3411 MWt. Fuel after batches A thru D have axial fuel blankets with an enrichment of 2.6 wt% (nominal) U235 or less. To eliminate the need to take credit for actual fuel parameters for these batches, DNC will

add a footnote to proposed TS Figure 3.9-3 for Region 2, which will preclude storage of fuel batches A, B, C and D in Region 2.

The basis for storage of no blanket fuel in the MP3 SFP is summarized as follows:

1) New no blanket fuel in Stretch Power Uprate (SPU) cycles
   a. In the response to RAI 21, DNC proposed adding a footnote to revised TS Figure 3.9-3 (all Region 2 fuel) and Figure 3.9-5 (Region 3 uprate fuel) requiring a maximum nominal blanket enrichment of 2.6 wt% U235 and a minimum nominal blanket length of 6 inches for fuel operated at uprate conditions. No blanket fuel depleted at SPU conditions will be precluded from storage in Regions 2 and 3 by this footnote. This effectively precludes use of new no blanket fuel.

   b. The Region 1 maximum burnup requirement is 8000 megawatt day per metric ton uranium (MWd/MTU) at 5 wt% U235. At or below this low burnup value, axial reactivity effects are not significant and a flat axial power shape is bounding, so the type of blanket is not important. Since a flat axial burnup profile was one of the considered shapes in WCAP-16721, and the flat axial shape is bounding at these burnup values, axial reactivity effects have been appropriately considered, at both the pre-uprate and uprated power levels for storage in Region 1 of "no blanket" fuel.

2) Existing no blanket fuel from pre-SPU cycles

   a. Region 1 storage of existing no blanket fuel is acceptable for the reasons given for new no blanket fuel. In addition, the maximum fuel enrichment of batch A-D fuel is 3.8 w/o U235. Per TS Figure 3.9-1 (Region 1), this 3.8 wt% U235 enrichment requires less than 1000 MWd/MTU of fuel burnup for storage in Region 1 of the spent fuel pool in a 4-out-of-4 configuration, which is too low a burnup for axial effects to be significant. In addition, the minimum burnup on any of the batch A-D fuel assemblies is more than 17,000 MWd/MTU (including 4% reduction for measured burnup uncertainty). Further, DNC has not proposed any changes to TS Figure 3.9-1, which is the current basis for storage of this fuel. DNC has submitted, as part of WCAP-16721, an analysis for Region 1 that shows the required fuel enrichments and fuel burnup limits for fuel at the SPU power level are bounded by TS Figure 3.9-1.

   b. DNC will add a footnote to proposed TS Figure 3.9-3 for Region 2, which will preclude storage of fuel batches A, B, C and D from being stored in Region 2.

c. For Region 3 of the spent fuel pool, TS Figure 3.9-4 provides the required enrichment and fuel burnup to allow storage of fuel in Region 3 for pre-uprate fuel. The current design basis, including TS Figure 3.9-4, is not altered or affected by the WCAP-16721 analysis provided in this Technical Specification change, and the only change to TS Figure 3.9-4 (as proposed in the original TS change for this amendment) is to change the title of the figure to reflect it is valid only for fuel assemblies from pre-uprate (3411 Mwt) cores. MPS3 no blanket fuel currently in the SFP from pre-uprate (3411 MWt) cores continues to be able to be stored in Region 3 per this TS figure. This figure was previously approved by the NRC, and was retained in the Technical Specifications to allow fuel from pre-uprate (3411 Mwt) cores to be stored in Region 3, provided the fuel meets the requirements of the TS figure.

Response, Part 27e:

This question concerns whether there are additional changes to TS Figure 3.9-4 that have not yet been provided. TS Figure 3.9-4 was proposed to be changed in the original submittal (Dominion Letter Serial Number 07-0450) by adding to the title of the figure to reflect it is valid only for fuel assemblies from pre-uprate (3411 MWt) cores. The proposed TS Figure 3.9-4 was not changed by any subsequent RAI response. The DNC response to RAI question 21 was referring to the title change previously proposed for TS Figure 3.9-4 in the original submittal.

Question 29

In DNC's response to RAI #5 DNC states their fuel management does not use fixed burnable absorbers.

a. Has MPS3 ever used any other flux suppression devices such as hafnium inserts to reduce the neutron dose to reactor vessel welds?
b. If so, how do they affect the reactivity of the discharged fuel assemblies?

Response:

No flux suppression devices have been used at MPS3. MPS3 fuel designs do not currently use fixed neutron absorbers, nor are any planned. From Cycle 3 thru the present (Cycle 13), only Integral Fuel Burnable Absorber (IFBA) has been used as a neutron absorber.

MPS3 did use some fixed neutron absorbers in Cycles 1 and 2 at the pre-uprate power level of 3411 MWt. The fixed absorbers were Pyrex glass rods, which were used in selected fuel assemblies in fuel batches B, C and D in Cycles 1 and 2. An explanation is provided next as to why these fuel assemblies in batches B, C and D which contained Pyrex rods in Cycles 1 and 2 may be stored in the MPS3 SFP under the proposed TS amendment.

- Region 1 Fuel Storage: Storage of fuel in Region 1 is restricted to the fuel enrichment and fuel burnup limits of TS Figure 3.9-1. DNC has not proposed any changes to TS Figure 3.9-1. DNC has submitted, as part of WCAP-16721, an analysis for Region 1 that shows the required fuel enrichments and fuel burnup limits for fuel at the SPU power level are bounded by TS Figure 3.9-1. Since TS Figure 3.9-1 limits bound SPU operation, no changes were proposed for TS Figure 3.9-1. TS Figure 3.9-1 does not require any fuel burnup for storage of fuel in the 4-out-of-4 configuration for enrichments below 3.7 wt% U235. Batch B and C fuel and most of the batch D fuel are below an enrichment of 3.7 wt% U-235 and therefore require no fuel burnup to be stored in Region 1 in a 4-out-of-4 configuration. Therefore, there is no need to consider the fuel history effects of fixed neutron absorber on fuel reactivity for this fuel with a required fuel burnup of 0 MWd/MTU. A small number of batch D fuel assemblies have a maximum fuel enrichment of 3.8 wt% U235, which per TS Figure 3.9-1 requires less than 1000 MWd/MTU of fuel burnup to allow storage in Region 1 in a 4-out-of-4 configuration. All of the batch D fuel assemblies with a fuel enrichment greater than 3.7 wt% U235 have at least a fuel burnup of 38000 MWd/MTU, including a 4% reduction for burnup uncertainity. The fuel history effects of a fixed neutron absorber on fuel reactivity at 1000 MWd/MTU of fuel burnup is very small, and negligible when considering the 38000 MWd/MTU actual minimum fuel burnup. In summary, the storage of batch B, C and D fuel assemblies

that contained fixed neutron absorbers in Cycles 1 and 2 are acceptable for 4-out-of-4 storage in Region 1 per TS Figure 3.9-1.

- Region 2 Fuel Storage: As described in the response to RAI question 27(d), DNC has modified proposed TS Figure 3.9-3 to preclude the storage of fuel batches A, B, C and D in Region 2 of the SFP. The proposed TS Figure 3.9-3 covers both pre-uprate fuel and post-uprate (SPU) fuel. Since batch B, C and D (which contained Pyrex rods) are not allowed to be stored in Region 2, this eliminates any question of fixed neutron absorber history effects for fuel batches B, C and D on assembly reactivity for Region 2 fuel storage.

- Region 3 Fuel Storage: TS Figure 3.9-4 specifies the enrichment and fuel burnup limits for storage of fuel in Region 3 for pre-uprate fuel. Proposed TS Figure 3.9-5 specifies the enrichment and fuel burnup limits for storage of fuel in Region 3 for SPU uprate fuel. DNC has proposed changing the title of TS Figure 3.9-4 to make it clear it applies only to pre-uprate fuel. The analysis that supports application of TS Figure 3.9-4 for pre-uprate fuel was previously approved by the NRC. Fuel batches B, C and D which contained fixed neutron absorbers in Cycles 1 and 2 are acceptable for fuel storage in Region 3 per TS Figure 3.9-4, consistent with the existing NRC approved analysis for Region 3.

The entire set of proposed TS changes, including proposed TS Figures 3.9-3 and 3.9-5 which are being modified as a result of the presently submitted responses to RAI 26-RAI 30, will be included as part of the separate responses to be submitted with RAI 28 and RAI 30.

References:

1. Laurence Kopp (USNRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998.

2. Letter from H. N. Berkow (NRC) to J.A. Gresham (Westinghouse), "Final Safety Evaluation for the Westinghouse Topical Report WCAP-16045-P, Revision 0, 'Qualification of the Two-Dimensional Transport Code PARAGON' (TAC No. MB8040)," March 18, 2004, ADAMS Accession Number: ML040780402.

3. M. D. DeHart, "Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages", ORNL/TM-12973, May 1996

4. C. V. Parks et.al., "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," NUREG/CR-6665, February, 2000.