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

January 8, 2016

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

SUBJECT: MILLSTONE POWER STATION, UNIT 3 – REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO ADOPT DOMINION CORE DESIGN AND SAFETY ANALYSIS METHODS (CAC NO. MF6251)

Dear Mr. Heacock:

By application dated May 8, 2015 (Agencywide Documents Access and Management System Accession No. ML15134A244), Dominion Nuclear Connecticut, Inc. (the licensee), submitted a license amendment request for changes to the Technical Specifications enabling the use of Dominion nuclear safety and reload core design methods for Millstone Power Station, Unit 3.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the information provided by the licensee and has determined that a response to the enclosed request for additional information (RAI) is needed in order to complete the review. As discussed with your staff during a recent phone call, the NRC staff requests that you support an audit during the week of February 22, 2016, and respond to the enclosed RAI no later than February 29, 2016.

If you have any questions regarding this matter, please contact me at 301-415-1030.

Sincerely,

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

Docket No. 50-423

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST

TO ADOPT DOMINION CORE DESIGN AND SAFETY ANALYSIS METHODS

DOMINION NUCLEAR CONNECTICUT, INC.,

MILLSTONE POWER STATION, UNIT NO. 3

CAC NO. MF6251

By application dated May 8, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15134A244), Dominion Nuclear Connecticut, Inc. (DNC), the licensee for Millstone Power Station, Unit No. 3 (MPS3) , submitted a license amendment request (LAR) for changes to the Technical Specifications (TSs) enabling use of Dominion nuclear safety and reload core design methods for MPS3. Based on its review of this LAR, the U.S. Nuclear Regulatory Commission staff has determined the following additional information is necessary to continue its technical review.

RAI-1 (SNPB): Parameter Uncertainties

Section 3.2 and Table 3.2-1 of Attachment 6 to the LAR list parameters and their uncertainties. Describe how these uncertainties were used to quantify the total uncertainty for each of these parameters.

RAI-2 (SNPB): Calculations of Critical Heat Flux Correlation Uncertainty Factor

(a) Provide detailed calculations to obtain the critical heat flux (CHF) correlation uncertainty factor using the 95% upper confidence limit on the VIPRE-D/WRB-2M and ABB-NV code correlation pair measured-to-predicted (M/P) CHF ratio and standard deviation.

(b) Equation 3.2 of Section 3.3 of Attachment 6 is used to calculate the upper confidence limit and is a correction factor that gives one-sided 95% upper confidence limit on the estimated standard deviation of a given population. Demonstrate, by an example, how this equation is used in the statistical analysis for CHF correlations.

RAI-3 (SNPB): Selection of 5-Percent Code Uncertainty

Code uncertainty in the statistical design limit (SDL) analysis is said to account for any differences between the licensee's VIPRE-D and other thermal hydraulic codes in which the WRB-2M and ABB-NV data were correlated, and any effect due to the modelling of a full core with a correlation based on bundle test data. Provide justification for the selection of 5% code uncertainty as a conservative value under all types of thermal margin calculations; including full core, half-symmetry, quarter-symmetry and 1/8 symmetry.

Enclosure

RAI-4 (SNPB): Explanation of the Randomization Process

Section 3.1 of Attachment 6 states that each minimum departure from nucleate boiling ratio (MDNBR) is randomized by a code/correlation uncertainty factor as described in Reference 1 using the upper 95% confidence limit on the VIPRE-D/WRB-2M and VIPRE-D/ABB-NV code/correlation pair measured-to-predicted CHF ratio standard deviation. Explain the randomization process to obtain the randomized values that appear in Tables 3.6-3 and 3.6-4.

RAI-5 (SNPB): Impact of NRC Information Notice (IN)-2014-01, on the DNBR SDL Determination

Section 3.5 of Attachment 6 quantifies the code uncertainty that is used in the calculation of SDL of DNBR. However, in Section 3.6.1, which discusses the impact of IN-2014-01, the code uncertainty, Fc, is neglected. Explain why the code uncertainty is not used for the calculation shown in Section 3.6.1.

RAI-6 (SNPB): Different Values for the Linear Regression Coefficient

Table 3.8-1 and Table 3.8-2 of Attachment 6 list the linear regression coefficient, R^2, for the verification of the nominal statepoints for the MPS3 17x17 RFA-2 fuel with VIPRE-D/WRB-2M and VIPRE-D/ABB-NV correlations, respectively. There is considerable difference in R^2 for the parameters in the two sets for the correlations. Explain the basis for the reliability of the selected nominal statepoints for the two correlations in the context of this significant difference.

RAI-7 (SNPB): Nuclear Safety Advisory Letter (NSAL)-09-5, Revision 1

NSAL-09-5 Revision 1, "Relaxed Axial Offset Control FQ Technical Specifications Actions," identified a technical concern which is applicable to MPS3 SR 4.2.2.1.2.e. which may not provide assurance that the non-equilibrium $F_Q(Z)$ limiting conditions for operation (LCO) limit will not be met between the performance of the required 31 Effective Full Power Day (EFPD) core power distribution surveillances. Westinghouse has determined in NSAL-09-5 that if F_Q^W (Z) is not within the LCO limit following a surveillance performed at ≥ 75% RTP, the following actions should be administratively implemented with NRC Administrative Letter 98-10, in addition to the current Required Actions contained in the plant specific F_Q Technical Specifications:

1. Reduce the maximum allowable power by 3% for each 1% $F_Q^W(Z)$ exceeds the limit within 4 hours.

2. Reduce the power range neutron flux – high trip setpoints ≥ 1% for each 1% that the maximum allowable power level is reduced within 72 hours.

3. Reduce the overpower ΔT trip setpoints by ≥ 1% for each 1% that the maximum allowable power level is reduced within 72 hours.

4. Perform SR 3.2.1.1 and SR 3.2.1.2 prior to increasing THERMAL POWER above the limit of action 1. Note that this action must be completed whenever the $F_Q^W(Z)$ limit is not met following a surveillance performed at ≥ 75% RTP.

The licensee has listed the actions in LAR Attachment 2 that shall be taken with Specification 4.2.2.1.2.c not being satisfied as INSERT "A". Insert "A", item a, states: "Within 15 minutes, control the AFD [axial flux difference] to within the new reduced AFD limits specified in the COLR [core operating limits report] that restores $F_Q(Z)$ to within its limits." Also, item b states: "Reduce the THERMAL POWER by the amount specified in the COLR that restores $F_Q(Z)$ to within its limits within 4 hours,"

(1) What is the reason for specifying the reduction in AFD limits and the reduction in the amount of thermal power specified in the COLR rather than in the TSs?

(2) Specify the amounts for the AFD limit reduction and THERMAL POWER reduction that are proposed. Specify the reason for the amounts of reduction if they deviate from the NSAL-09-5 Revision 1 actions listed above.

RAI-8 (SRXB): Acceptability of WCAP-14441

On page 10 of Attachment 4, the LAR indicated that the Westinghouse BORDER code and the constituent equations discussed in WCAP-14441 are used to verify the cycle-specific boration requirements in the reload safety analysis. Justify the use of the WCAP-14441 methods for the stated purpose in the reload safety analysis.

RAI-9 (SRXB): Reanalysis for the RETRAN Benchmarking Cases

The licensee indicated in an e-mail dated June 30, 2015 (ADAMS Accession No. ML15349A808) that it identified a discrepancy between the MPS3 RETRAN Base Model Pressurizer Shell Heat Conductor and the Dominion RETRAN topical report (TR). The MPS3 RETRAN base input deck models the pressurizer shell as a heat conductor, which differs from TR, VEP-FRD-41, which states that "Dominion continues to model the non-equilibrium wall as an adiabatic surface." Each of the five benchmarking cases supporting the LAR were reanalyzed with needed correction.

Discuss the results of the reanalysis for the five benchmarking cases, and provide the modified graphs showing the changes and text for the affected cases, including the affected loss of normal feedwater benchmark case.

RAI-10 (SRXB): Loss of Load (LOL) Benchmark Analysis – RCS Pressurization Rate

On page 9 of Attachment 5 to the LAR, the licensee indicated that "the Dominion case trips slightly earlier than the FSAR [final safety analysis report] data because of the higher RCS [reactor coolant system] pressurization rate", and Table 4.1-2 showed that for the LOL [loss of load/turbine trip] analysis, the calculated peak RCS pressure for the Dominion case is 2,717.19 psia [pounds-per-square-inch, absolute], which is 12.22 psi lower than the peak pressure of 2,729.41 psia for the FSAR case.

Discuss the differences of the models, input parameters or assumptions used in the LOL analyses for the Dominion case and the FSAR case that will result in:

(1) a higher RCS pressurization rate before the reactor trip for the Dominion case and;

(2) a lower peak RCS pressure for the Dominion case against the FSAR case discussed above.

RAI-11 (SRXB): Locked Rotor (LR) Benchmark Analysis - Over-Pressurization and DNB

On page 14, the licensee stated for the LR analysis that, based on the data comparison between the Dominion case and FSAR case, "both the initial under-prediction of the heat flux response, followed by an over-prediction during the rod insertion is indicative of the fuel rod heat transfer being modeled differently in the vendor methods than in the Dominion model." It further stated that "the over- prediction of both nuclear power and heat flux will lead to conservative results at the limiting point in the transient for both RCS over-pressurization and DNB during rod insertion."

Discuss the differences of the fuel rod heat transfer models used in the LR analyses for the Dominion case and the FSAR case, and justify a higher peak RCS pressure of 2,680.75 psia and a higher peak cladding temperature (PCT) of 1,760.0 ^{0}F predicted for the Dominion case, compared to lower corresponding values of 2,616.65 psia and 1,718.3 ^{0}F (shown in Table 4.2-2 and Table 4.2-4, respectively) for the FSAR case.

RAI-12 (SRXB): Loss of Normal Feedwater Benchmark Analysis – Pressurizer Water Volume Response

The pressurizer water volume response shown in Figure 4.3-6 indicated that the Dominion analysis predicts the same trends as the FSAR data, but calculates lower values in the period from 63 to 900 seconds, followed with a strong in-surge during the second heat-up period in the transient. The calculated maximum water volume of 1588.96 ft^3 is lower than the FSAR case of 1730.85 ft^3. The licensee indicated the deviations of the pressurizer water volume response are attributed to differences in the main steam safety valves (MSSV) modeling, as well as differences in the pressurizer spray models.

Discuss the differences in the MSSV model and pressurizer spray model used in the Dominion analysis and the FSAR case and justify the deviations discussed above for the pressurizer water volume response.

RAI-13 (SRXB): Main Steam Line Break RETRAN Benchmark Analysis - Split Vessel Nodal Scheme

The licensee proposed to use the reactor vessel nodal scheme in Figure 2-2 in the analysis of the main steam line break (SLB), which is an asymmetric response transient with lower temperature in the core next to the ruptured steam generator (SG) and higher temperature in other side of the core. Figure 2-2 represented the nodal scheme change by adding volume to create a second parallel flow path through the active core from the lower plenum to the upper plenum. The licensee indicated that the nodal diagram can represent RCS loop temperature asymmetries.

Discuss the model of the mixing flow between the cold-and-hot sides of the core and address the adequacy of the flow mixing model for the SLB analysis, when use the nodal scheme in Figure 2-2.

RAI-14 (SRXB): Control Rod Bank Withdrawal at Power (RWAP) Benchmark Analysis - Higher Core Power Rate of Increase

The core power response in Figure 4.5-1 shows that for the RWAP 1 pcm/sec case, its rate of increase for the Dominion model is greater than the FSAR data. The faster power increase rate leads to the Dominion modeling tripping on high neutron flux at about 73 seconds, and the lower power increase rate for the FASR case results in a reactor trip on an OTΔT [Overtemperature delta T] signal at about 93 seconds.

Discuss the differences of the nodding, input parameters, models or assumptions used in analyses of the RWAP 1 pcm/sec case for the Dominion case and the FSAR case, and justify the greater increase of the core power rate observed in the analysis of the Dominion case.

RAI-15 (SRXB): RWAP Analysis - DNBR Calculations

The results of MPS3 FSAR Chapter 15 non-LOCA analyses indicated the RWAP event is the most limiting event in terms of the margin to the safety limit DNBR in the category of the anticipated operating occurrences (AOOs). Since the licensee also proposed to use the RETRAN and Dominion VIPRE-D method to perform DNBR calculations for assessing the fuel integrity during AOOs and accidents, the RETRAN benchmarking analysis for both RWAP 1 pcm/sec and 100 pcm/sec cases should be performed to include the results of the DNBR calculation by using the Dominion VIPRE-D method. The requested information includes a comparison of Dominion VIPRE-D analyses to the MPS3 FSAR analysis of record (AOR) showing that the calculated DNBRs for both cases are compatible with the AOR and the allowable range of the use of the NRC-approved DNBR correlation in VIPRE-D for the Dominion method is not exceeded.

RAI-16 (SRXB): Feedwater Line Break Analysis

MPS3 FSAR (2012 Version), Section 15.2.8 discussed the feedwater line break (FLB) analysis for both cases with and without offsite power available. The FSAR AOR presented transient results including nuclear power, core heat flux, total reactivity, pressurizer pressure, total RCS flow, feedwater break flow, loop temperature, intact loop temperature, and SG pressure. FSAR Figures 15.2-13 and 15.2-19 indicated that a post-trip return-to-power will occur for the case with offsite power available, and core will remain subcritical throughout the transient time of several seconds for the case without offsite power available. Also, page 15-2-16 indicated that the FLB is the most limiting event in the decrease in secondary removal category. The analysis of the FLB will use a broad scope of the models in RETRAN, including feedwater break flow model, RC pumps coastdown model, SG heat transfer model, and reactivity feedback model.

Perform the RETRAN benchmarking analysis for the FLB event for both cases with and without offsite power available. The information to be provided should show that: the values of the plant parameters and assumptions used in the Dominion FLB analysis are consistent with that used in the FSAR AOR; the results are compatible with the AOR; and there is no unexplainable thermal-hydraulic phenomena throughout the transients.

RAI-17 (SRXB): Steam Generator Tube Rupture (SGTR) Analysis

MPS3 FSAR 15.6-3 discussed the SGTR analysis for two cases: (1) the SG overfill margin analysis that is used to validate the assumption of no water leaked from the affected SG to atmosphere; and (2) the mass release analysis that is used as input to a computer code for calculating the dose releases. This analysis involved simulation of the mitigating strategies directing operators to identify and isolate the ruptured SG, cooldown the RCS to establish subcooling margin, depressurize to restore RCS inventory, and terminate safety injection to stop primary-to-secondary leakage.

Perform the RETRAN benchmarking analysis for the SGTR event for both SG overfill and mass releases cases. The information to be provided should show that RETRAN is capable of simulating the operator actions specified in FSAR and discussed above and the results of the SG overfill and mass releases analyses are compatible with to the AOR.

RAI-18 (SRXB): TS 6.9.1.6.b Reference List - Addition of NRC-Approved Dominion Methodologies

On page 14 of Attachment 1 to the LAR, the licensee indicated that it proposed to apply the Dominion's analysis methods to MPS3 in the reload design and safety analysis for licensing applications. The methods are documented in the TRs as follows:

- TR-1: VEP-FRD-42-A, "Reload Nuclear Design Methodology."
- TR-2: VEP-NE-1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."
- TR-3: VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."
- TR-4: DOM-NAF-2-P-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code."
- TR-5: DOM-NAF-1-P-A, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations."
- TR-6: VEP-FRD-41-P-A, "VEPCO Reactor System Transient Analyses Using the RETRAN Computer Code."

The licensee proposed to add TR-1 through TR-4 to TS 6.9.1.6.b and leave TR-5 and TR-6 out of the TSs. Each of the applicable referenced TRs in the TSs will be included in the COLR reference list with the complete identification (i.e., report number, title, revision, date, and any supplements) for a specific reload cycle. The licensee clarified on page 16 that in accordance with guidance of Generic Letter (GL) 88-16, "a methodology in the COLR reference list is intended to satisfy these criteria: (1) it is used to determine core operating limits, and (2) it has been previously approved by the NRC."

Discuss the purposes of the methods documented in TR-5 and TR-6 for use in the reload design and safety analysis and justify the adequacy of not including TR-5 and TR-6 in TS 6.9.1.6.b (and thus being able to not reference TR-5 and TR-6 in the COLR). As part of the discussion, address compliance with the above discussed two criteria established in meeting the GL 88-16 guidance for the case without inclusion of TR-5 and TR-6 in TS 6.9.1.6.b.

January 8, 2016

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

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