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Dominion®

November 19, 2007

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2378

Serial No.: 07-0751
NLOS/MAE: R0
Docket No.: 50-423
License No.: NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST

Dominion Nuclear Connecticut, Inc. (DNC) submitted a stretch power uprate license amendment request for Millstone Power Station Unit 3 (MPS3) in letters dated July 13, 2007 (Serial Nos. 07-0450 and 07-0450A) and supplemented the submittal by letter dated September 12, 2007 (07-0450B). The NRC staff forwarded a request for additional information (RAI) in an October 29, 2007 letter. The response to this RAI is provided in the attachment to this letter.

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

Should you have any questions in regard to this submittal, please contact Ms. Margaret Earle at 804-273-2768.

Very truly yours,

Gerald T. Bischof
Vice President Nuclear Engineering

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Gerald T. Bischof, who is Vice President Nuclear Engineering of Dominion Nuclear Connecticut, Inc. He has affirmed before me that he 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 his knowledge and belief.

Acknowledged before me this 19th day of November, 2007.

My Commission Expires: August 31, 2008.

Margaret B. Bennett
Notary Public

MARGARET B. BENNETT
Notary Public
Commonwealth of Virginia
My Commission Expires Aug 31, 2008

Commitments made in this letter: None

Attachment

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT

LICENSE AMENDMENT REQUEST

STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

Stretch Power Uprate License Amendment Request

Response to Request for Additional Information

Dominion Nuclear Connecticut, Inc. (DNC) submitted a stretch power uprate (SPU) license amendment request for Millstone Power Station Unit 3 (MPS3) in letters dated July 13, 2007 (Serial Nos. 07-0450 and 07-0450A) and supplemented the submittal by letter dated September 12, 2007 (Serial No. 07-0450B). The NRC staff forwarded a request for additional information (RAI) in an October 29, 2007 letter. The response to the RAI is provided below.

NRC Question CVIB-07-001

Title 10 of the Code Federal Code of Federal Regulations (10 CFR) Part 50, Appendix G, provides fracture toughness requirements for ferritic materials (low alloy steel or carbon steel) materials in the reactor coolant pressure boundary (RCPB) components. 10 CFR Part 50, Appendix G also identifies that RCPB materials must satisfy the criteria in Appendix G of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests.

Consistent with the requirements specified in 10 CFR Part 50, Appendix G, the licensee performed fracture toughness analyses of the ferritic reactor vessel (RV) materials.

The staff requests that the licensee confirm that these analyses bound the fracture toughness requirements of the ferritic RCPB components other than the RV components at MPS, Unit 3

DNC Response

For the ferritic RCPB components other than the reactor vessel (i.e., pressurizer and steam generators), the fracture toughness requirements were addressed in the original ASME Code analyses. The results of these Code required 10 CFR 50 Appendix G analyses are documented in each component's Code stress report. The changes associated with SPU have no impact on the 10 CFR 50 Appendix G analyses documented in the pressurizer and steam generator stress reports. Therefore, the fracture toughness requirements associated with 10CFR50 Appendix G for the ferritic RCPB components other than the reactor vessel will continue to be satisfied after implementation of SPU.

NRC Question CVIB-07-002

Section 2.1.2.2-1 delineates the difference between 1/4T neutron fluence (E > 1 MeV) estimated in the license extension application and the value proposed in the current application for SPU, i.e., 1.97×10^{19} n/cm^2 and 1.63×10^{19} n/cm^2, respectively. Similarly, Section 2.1.3.2.3 indicates a similar difference between the peak inside surface vessel fluence to 54 effective full power years (EFPYs) of operation. Apparently, both calculations were carried out using methodologies adhering to the guidance in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

Please describe the physical reasons that justify the lower fluence value for the 54 EFPYs under SPU conditions.

DNC Response

The change in estimated neutron fluence (both surface and 1/4T) is due to new information received from analysis of surveillance capsule W removed in October 2005 at 13.8 EFPYs, and from recent fluence analyses performed in 2006 to support the SPU. The neutron fluence estimates cited in the license renewal application were based on projections from surveillance capsule X removed at 8.0 EFPYs.

The recent fluence calculations performed for capsule W and SPU were based on three-dimensional synthesized neutron flux distributions, where forward transport calculations were performed in (r,θ), (r,z), and (r) geometry. The fluence calculations performed for capsule X were based on two-dimensional (r,θ) analyses using forward transport and adjoint calculations. As noted in Section 1.3.4 of Regulatory Guide 1.190, the use of a synthesized three-dimensional approach, with both radial and axial transport calculations, is generally more accurate. Additional details of the neutron fluence analyses are provided below.

The reactor vessel inner surface neutron fluence (E > 1 MeV) of 1.97×10^{19} n/cm^2 at 32 EFPY, as cited in the license renewal application, is discussed in WCAP-15405, Revision 0 "Analysis of Capsule X from Northeast Nuclear Energy Company Millstone Unit 3 Reactor Vessel Radiation Surveillance Program," May 2000. WCAP-15405 was submitted to the NRC on May 17, 2000, "Millstone Nuclear Power Station Unit No. 3 Submittal of Second Reactor Vessel Surveillance Capsule Report" (ADAMS Accession No. ML003717395). This WCAP documents the analysis and evaluation of surveillance capsule X removed at the end of the sixth cycle after 8.0 EFPY. The results of the evaluation of capsule X were used to project the estimated fluence from cycle 6 to the end of life at 32 EFPY (40-year operation), and at 54 EFPY for 60-year operation. Very conservative assumptions were made for estimating the end of life fluence. Projections for future operation were based on the assumption that neutron flux values from Cycles 4 through 6 would continue to be applicable throughout plant life. From Table 6-2 of WCAP-15405, the average neutron flux for Cycles 4 – 6 is 1.93×10^{10} n/cm^2-sec at the limiting azimuthal location. This equilibrium flux value was used in the fluence projections. From Table 6-13 of WCAP-15405 the highest calculated inner surface fluence at 32 EFPY was 1.97×10^{19} n/cm^2. At 54 EFPY, the calculated fluence was 3.31×10^{19} n/cm^2.

In October 2005, at the end of cycle 10, surveillance capsule W was removed for evaluation and analysis. The results of the evaluations and analysis of this surveillance capsule are summarized in WCAP-16629-NP, "Analysis of Capsule W from the Dominion Nuclear Operating Company Millstone Unit 3 Reactor Vessel Radiation Surveillance Nuclear Program," Revision 0, September 2006. WCAP-16629-NP was submitted to the NRC on October 2, 2006, "Dominion Nuclear Connecticut, Inc. Millstone Power Station Unit 3 Submittal of Third Reactor Surveillance Capsule Report" (ADAMS Accession No. ML062850221). In addition to the evaluation and analysis of the surveillance capsule, this WCAP also contains projections of the maximum fluence from Cycle 10 to end of life. From Table 6-2 of WCAP-16629-NP, the maximum fluence was estimated to be 1.75×10^{19} n/cm^2 at 32 EFPY, and 2.96×10^{19} n/cm^2 at 54 EFPY. These projections were carried out based on an assumed core power uprate from 3411 MWt to 3650 MWt at the onset of Cycle 11. The projections were based on the assumption that the relative spatial core power distributions and associated plant operating characteristics from Cycle 10 were representative of future plant operation. For Cycles 11 and beyond, the equilibrium flux values are based on Cycle 10 plus a multiplier of 1.07 to account for the assumed power uprate. It is worth noting that the equilibrium flux values shown in Table 6-2 of WCAP-16629-NP are less than those reported in WCAP-15405. This can be attributed to the continued used of low leakage core designs at Millstone Unit 3. Additionally, added refinements in the fluence calculation methodology were used in WCAP-16629-NP as compared to the methodology used in WCAP-15405. The fluence analysis per WCAP-16629-NP employed a P_5 legendre expansion and an S_{16} angular quadrature discretization, whereas the fluence analyses per WCAP-15405 employed a P_3 legendre expansion and an S_8 angular quadrature discretization.

For the Stretch Power Uprate (SPU) analysis, these neutron fluence projections have been updated to reflect the current uprate schedule. Preliminary designs of transition and equilibrium fuel cycles for the SPU program have been completed. These designs describe a pre-uprate Cycle 12 operating at 3411 MWt, two transition cycles (Cycles 13 and 14) operating at 3650 MWt (SPU power level) and an equilibrium fuel cycle (Cycle 15) also operating at 3650 MWt. Using these designs the projected maximum inner surface fluence at 32 EFPY is 1.63×10^{19} n/cm^2, and at 54 EFPY is 2.70×10^{19} n/cm^2.

The neutron transport methodology used in the Capsule W and SPU calculations follows the guidance and meets the requirements of Regulatory Guide 1.190 "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The Capsule X fluence analyses per WCAP-15405 were compliant with the requirements of Draft Regulatory Guide DG-1053, which was the precursor to Regulatory Guide 1.190. Additionally, the methods used to determine the pressure vessel neutron exposure make use of the NRC approved methodology described in WCAP-14040-NP-A "Methodology Used to Develop Cold Overpressurization Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1996.

NRC Question CVIB-07-003

In 2001, one Babcock and Wilcox (B&W) licensee experienced failures of two control rod drive mechanism 17-4 precipitation hardened (PH) lead screw male couplings. The failures were attributed to thermal embrittlement of the 17-4 PH martensitic stainless steel materials.

The staff issued NRC Information Notice (IN) 2007-02, "Failure of Control Rod Drive Mechanism Lead Screw Male Coupling at a Babcock and Wilcox Designed Facility." In IN 2007-02, the staff reiterated the importance of implementing frequent visual and surface examinations for identifying thermal embrittlement in 17-4 PH martensitic stainless steel reactor vessel internals (RVI) components.

The staff requests that the licensee provide the following information with respect to monitoring the aging degradation of the 17-4 PH martensitic stainless steel materials used in RVI components at MPS3:

(1) Identify 17-4 PH martensitic stainless steel RVI components at MPS3, if any.

(2) Identify the method of inspection that was performed thus far on these components and provide information regarding any aging degradation that was identified thus far in these components.

DNC Response

(1) There are no 17-4 PH materials used in the reactor vessel internals.

(2) This question is not applicable since there are no 17-4 PH materials used in the reactor vessel internals.

NRC Question SBPB-07-004

In Attachment 5, on page 2.5-114, under the section "Turbine Driven Feedwater Pump Turbine Control Valves," the licensee stated in the application that "[the] engineering evaluation to confirm whether or not more steam flow is required for turbine driven feedwater pump turbines for SPU conditions is in progress."

The NRC staff cannot initiate a review of this section until the results of the evaluation and a description of any impacts or modifications are submitted.

Please provide the results of the evaluation and a description of any impacts or modifications.

DNC Response

The GE engineering evaluations of the need for the feedwater turbine control valve modifications have been completed. The evaluations concluded that no hardware modifications to those valves are necessary in support of the SPU.

NRC Question SBPB -07-005

In Attachment 5, on page 2.5-22, under the section, "Turbine Results," DNC performed high-pressure turbine rotor phased array testing of the tangential-entry dovetail regions of the wheel rim, which revealed indications on the turbine first stage wheel. DNC is currently evaluating either installing a new high-pressure rotor, or disassembling first stage buckets, conducting inspections, removing indications and possible repairs. DNC needs to identify their proposed resolution and affects on the proposed SPU before the NRC staff can complete its review.

DNC Response

During the next refueling outage (October 2008) DNC will be removing buckets from both of the first stage wheels of the Millstone HP Rotor. The wheels will be machined to a new bucket attachment dovetail configuration. This machining will remove the subject wheel indications. New longer shank buckets will be installed on the machined wheel. The new dovetail configuration and buckets are being designed by the turbine OEM for the SPU conditions. This repair will reestablish the normal 10-year HP Rotor inspection interval.