December 20, 2007

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 NO. 3 - REQUEST FOR ADDITIONAL INFORMATION REGARDING STRETCH POWER UPRATE AMENDMENT REQUEST (TAC NO. MD6070)

Dear Mr. Christian:

By letter dated July 13, 2007, as supplemented on July 13, September 12, and November 19, 2007, Dominion Nuclear Connecticut, Inc. submitted a stretch power uprate license amendment request for Millstone Power Station, Unit No. 3. The proposed license amendment would allow an increase in the maximum authorized core power level from 3,411 megawatts thermal (MWt) to 3,650 MWt, and would make changes to the Technical Specifications, as necessary, to support operation at the stretch power level.

In order to complete its review of the reports, the U.S. Nuclear Regulatory Commission staff has determined that additional information is needed, and requires a response to each of the enclosed questions. The questions were sent by e-mail on November 16, 2007, and were discussed via teleconference on November 20, 27, 29, December 4 and 6, 2007, with your staff to ensure that the questions were understandable, the regulatory basis was clear and to determine if the information was previously docketed. Mr. Ron Thomas of your staff agreed to respond within 30 days of the date of this letter.

Please note that if you do not respond to this letter within the prescribed response times 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, I can be reached at (301) 415-3100.

Sincerely,

/ra/

John G. Lamb, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosure: As stated

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

MILLSTONE POWER STATION, UNIT NO. 3

STRETCH POWER UPRATE LICENSE AMENDMENT REQUEST

TAC NO. MD6070

DOCKET NO. 50-423

By letter dated July 13, 2007, as supplemented on July 13, September 12, and November 19, 2007, Dominion Nuclear Connecticut, Inc. (DNC or licensee) submitted a stretch power uprate (SPU) license amendment request for Millstone Power Station, Unit No. 3 (MPS3). The proposed license amendment would allow an increase in the maximum authorized core power level from 3,411 megawatts thermal (MWt) to 3,650 MWt, and would make changes to the Technical Specifications (TS), as necessary, to support operation at the stretch power level.

The U.S. Nuclear Regulatory Commission (NRC) staff has been reviewing the submittal and has determined that additional information is needed to complete its review.

Reactor Systems Branch

SRXB-07-0088 (2.8.5.3.1-4)

Loss of Forced Reactor Coolant Flow - The results of this accident are discussed in terms of no violations to the departure from nucleate boiling ratio limit. Provide results showing conformance with other relevant acceptance criteria confirmed, or explain why these acceptance criteria are acceptably not analyzed.

SRXB-07-0089 (2.8.5.4.5-10)

Chemical And Volume Control Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant - Page 2.8-298 lists three acceptance criteria. Confirm that the third, regarding fuel temperature, is analogous to the second acceptance criterion listed in Final Safety Analysis Report (FSAR) Section 15.4.6. The staff requests this confirmation because specific values are not listed in the SPU licensing report (LR).

SRXB-07-0090 (2.8.5.5-4)

Please provide information regarding the qualification of the power-operated relief valves (PORVs) as safety-related equipment (i.e., the PORVs' automatic control system and their ability to relieve water).

ENCLOSURE

Containment and Ventilation Branch

SCVB-07-0091

Millstone Power Station Unit 3 (MPS3) FSAR lists the following initial containment conditions selected for the subcompartment analyses:

Pressurizer Subcompartment: Temperature 100° F; Air partial pressure 8.9 pounds per square inch absolute (psia); relative humidity 10 percent.

Steam Generator and Upper Reactor Cavity Subcompartments: Temperature 120° F; air partial pressure 9.0 psia; relative humidity 50 percent.

Section 2.6.2 of Attachment 5 of the licensing amendment request (LAR) does not list the initial conditions used for subcompartment analysis. Please provide the initial conditions used for SPU subcompartment analysis. In case any of these conditions deviate from the current licensing bases, provide appropriate justification for the deviation.

SCVB-07-0092

Section 2.6.2.2.2, fifth paragraph of Attachment 5 of the LAR states that the pre-SPU analysis for short term mass and energy (M&E) release within the steam generator cubicle is conservatively based on a frictionless Moody critical flow model and bounds the SPU condition. Please justify why the pre-SPU M&E release bounds the M&E release for the SPU condition.

SCVB-07-0093

Pressurizer spray line break loss-of-coolant accident (LOCA) M&E release data are not provided in LAR Attachment 5, Section 2.6.3. Section 2.6.3.1.2.2.3, seventh paragraph states the SPU changes "could increase the spray line mass and energy releases by as much as 3.4 percent." Please verify that the calculated SPU M&E release data for the pressurizer spray line break is within 3.4 percent from current M&E release data provided in FSAR Table 6.2-31 using the same assumptions and initial conditions, and is within the margin.

SCVB-07-0094

Pressurizer surge line break LOCA M&E release data are not provided in LAR Attachment 5, Section 2.6.3. Section 2.6.3.1.2.2.3, eighth paragraph states the SPU changes "could increase the surge line mass and energy release by as much as 15.75 percent on mass released and 11.27 percent on energy released." Please verify that the calculated SPU M&E release data for the pressurizer surge line break is within 15.75 percent on mass released and within 11.27 percent on energy released from the M&E release data provided in FSAR Table 6.2-32 using the same assumptions and initial conditions.

SCVB-07-0095

FSAR Section 6.2.1.1.3.4 provides a description of the heat sink model and Tables 6.2-1 and 6.2-2 provide thermal and physical properties of the heat sinks in the current licensing basis analysis. It is stated in LAR Attachment 5, Section 2.6.1.2.2.2 that the passive heat sinks are unchanged by SPU. Please verify that the proposed SPU containment analyses use the same description, thermal and physical properties of the heat sinks as in the above FSAR section and tables. Provide appropriate justification for any variations from the current licensing bases. Please explain how the changes in the sump strainer were accounted for in the passive heat sinks.

SCVB-07-0096

LAR Attachment 5, Section 2.6.1.2.2.3, under heading “Depressurization Analysis,” states that the initial containment conditions that yield slowest containment depressurization are the maximum pressure, temperature, and relative humidity. This is not consistent with the results presented in Table 2.6.1.2.2-6. The table shows that the slowest depressurization during the first hour gives a pressure of 30.2 psia for the initial condition minimum relative humidity of zero percent, and the slowest depressurization during first five hours gives a pressure of 22.6 psia for the initial condition of minimum temperature and minimum relative humidity. Please clarify or remove the inconsistency.

SCVB-07-0097

LAR Attachment 5, Section 2.6.1.2.2.3, under heading “Peak Temperature Analysis,” mentions the calculated containment temperature profile, but does not provide results for the limiting case which gives the slowest cooldown rate. Please provide the accident case which gives the slowest cooldown rate and specify for which initial conditions it occurs.

SCVB-07-0098

LAR Attachment 5, Section 2.6.2.3, third paragraph, states that for the pressurizer surge line break, the steam generator compartment differential pressure increased by approximately 5 percent from the current licensing basis and is bounded by the current analysis results for the steam generator compartment. What is the margin in the current licensing basis and what will be the new margin with the increased differential pressure of 5 percent in the steam generator compartment?

SCVB-07-0099

LAR Attachment 5, Section 2.6.3.2.2.3, provides the sixteen cases analyzed for M&E release for the SPU main steam line break (MSLB) which are different from the current licensing bases MSLB break cases in FSAR Table 6.2-22. Please provide appropriate justification for the differences.

SCVB-07-0100

LAR Attachment 5, Table 2.6.5-1 provides the maximum sump water temperature for the current analysis as 260° F and the SPU value as 225° F. What are the reasons for the difference? For the current value determined using LOCTIC model, and the SPU value determined using GOTHIC model, please verify that same assumptions and input parameters were used in the two analyses. For different assumptions and/or input parameters provide appropriate justification and indicate if any conservatism used in the current analysis was reduced in the SPU analysis.

SCVB-07-0101

LAR Attachment 5, Table 2.6.6-1 provides the parameters for emergency core cooling system (ECCS) containment backpressure analysis. Please provide justification for changing the minimum initial containment pressure to 8.9 psia and temperature to 80° F from their corresponding FSAR values of 10.4 psia and 90° F.

SCVB-07-0102

LAR Attachment 5, Section 2.6.2.3 states that “The design of the pressurizer cubicle has been evaluated for this pressure increase and determined to be acceptable. Analysis of the pressurizer cubicle walls has demonstrated that the current design pressure for the limiting wall element remains bounding with no net decrease in the design basis margin (i.e. the margin between the current design pressure utilized in the structural analysis and the allowable design pressure associated with the limiting wall element).”

Please explain why there is no net decrease in the design margin between the current design pressure utilized in the structural analysis and the allowable design pressure associated with the limiting wall element?

SCVB-07-0103

LAR Attachment 5, Section 2.6.6.2 states, “Table 2.6.6-2 provides the structural heat sink data used in the ECCS containment backpressure boundary condition analysis. The structural heat sink data has been updated to reflect re-validation of the data and implemented design changes, including the sump strainer.” Please explain what is meant by “to reflect re-validation of the data and implemented design changes, including the sump strainer.”

SCVB-07-0104

NRC Generic Letter (GL) 96-06, issue number 3 states: “Thermally induced overpressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident-mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage. Corrective actions may be needed to satisfy system operability requirements.”

Please verify that this issue was reconsidered for the SPU conditions, and confirm that the piping systems that penetrate the containment which are susceptible to thermal expansion of the fluid and overpressurization, will remain within their design limits.

Balance-of-Plant Branch

SBPB-07-0105

In Attachment 5, Section 2.5.4.3, Reactor Auxiliary Cooling Water System, the licensee states that the following design change was required to the plant component cooling water (CCP) system in order to support SPU conditions: "A design change to increase the design temperature of the CCP system between the residual heat removal system heat exchangers and the CCP heat exchangers from 150° F to 160° F, and increase the CCP system operating temperature during cooldown modes of operation, will be performed during the SPU implementation." Since SPU conditions will necessitate a change in design conditions, then the resulting change can not be bounded by previous system design conditions. Provide a description of the design change, the impact on system components, and any physical modifications required to support the change.

Instrumentation and Controls Branch

EICB-07-0106

The license amendment request (LAR), dated July 13, 2007, proposed following Technical Specifications (TS) changes associated with instrument setpoints:

1. TS Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, Functional Unit 18c, Power Range Neutron Flux, P-8, the nominal trip setpoint is increased from 37.5% to 50.0% and Allowable Value is increased from ≤ 38.1% to ≤50.6% of RTP.

2. TS Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints, Functional Unit 11, Cold Leg Injection Permissive, P-19, Nominal Trip Setpoint is specified as 1900 psia, and Allowable Value as ≥ 1897.6 psia.

To support NRC assessment of the acceptability of the LAR with regard to setpoint changes, please provide the following for each setpoint to be added or modified:

A. Setpoint Calculation Methodology: Provide documentation (including sample calculations) of the methodology used for establishing the limiting setpoint (or NSP) and the limiting acceptable values for the As-Found and As-Left setpoints as measured in periodic surveillance testing as described below. Indicate the related Analytical Limits and other limiting design values (and the sources of these values) for each setpoint.

B. Instrument Functionality: Describe the measures to be taken to ensure that the associated instrument channel is capable of performing its specified safety functions in accordance with applicable design requirements and associated analyses. Include in your discussion information on the controls you employ to ensure that the as left trip setting after completion of periodic surveillance is consistent with your setpoint methodology. Also, discuss the plant corrective action processes (including plant procedures) for restoring channels to operable status when channels are determined to be "inoperable" or "operable but degraded." If the controls are located in a document other than the TS (e.g., plant test procedure), describe how it is ensured that the controls will be implemented.

Accident Dose Branch

AADB-07-0107

Please provide additional information describing, for each design basis accident affected by the proposed stretch power uprate (SPU), all the basic parameters used in the dose consequence analyses. For each parameter, please indicate the current licensing basis (CLB) value, the revised value where applicable, as well as the basis for any changes to the CLB. The staff notes that much of the requested information has been provided in Table 2.9.4 of the license amendment request (LAR). The staff requests that the information in Table 2.9.4 be expanded to include all of the basic parameters whether or not the individual parameter is being changed for the SPU amendment. The staff also requests that the information be presented in separate tables for each affected accident, as was done for the alternative source term LAR.

Millstone Power Station, Unit No. 3

cc:

Lillilan M. Cuoco, Esquire
Senior Counsel
Dominion Resources Services, Inc.
Building 475, 5th Floor
Rope Ferry Road
Waterford, CT 06385

Edward L. Wilds, Jr., Ph.D.
Director, Division of Radiation
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

First Selectmen
Town of Waterford
15 Rope Ferry Road
Waterford, CT 06385

Mr. J. W. "Bill" Sheehan
Co-Chair NEAC
19 Laurel Crest Drive
Waterford, CT 06385

Mr. Evan W. Woollacott
Co-Chair
Nuclear Energy Advisory Council
128 Terry's Plain Road
Simsbury, CT 06070

Senior Resident Inspector
Millstone Power Station
c/o U.S. Nuclear Regulatory Commission
P. O. Box 513
Niantic, CT 06357

Ms. Nancy Burton
147 Cross Highway
Redding Ridge, CT 00870

Mr. Joseph Roy,
Director of Operations
Massachusetts Municipal Wholesale
Electric Company
Moody Street
P.O. Box 426
Ludlow, MA 01056

Mr. J. Alan Price
Site Vice President
Dominion Nuclear Connecticut, Inc.
Building 475, 5th Floor
Rope Ferry Road
Waterford, CT 06385

Mr. Chris Funderburk
Director, Nuclear Licensing and
Operations Support
Dominion Resources Services, Inc.
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

Mr. David W. Dodson
Licensing Supervisor
Dominion Nuclear Connecticut, Inc.
Building 475, 5th Floor
Rope Ferry Road
Waterford, CT 06385