Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

Docket No. 50-423
B18990

RE: 10 CFR 50.90

SEP 18 2003

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

Millstone Power Station, Unit No. 3
Response to Request For Additional Information
License Basis Document Change Request (LBDCR) 3-15-02
Relocation of Some Technical Specification Parameters to The Core Operating Limits Report and Updating The Description of Analytical Methods Used to Determine Core Operating Limits

By a letter dated April 7, 2003,[1] Dominion Nuclear Connecticut, Inc. (DNC) proposed to amend Operating License NPF-49 by incorporating changes into the Millstone Unit No. 3 Technical Specifications. The proposed changes relocate some Technical Specification parameters to the Core Operating Limits Report (COLR) and update the description of analytical methods used to determine core operating limits.

By a facsimile dated September 4, 2003,[2] a Request For Additional Information (RAI) was received from the Nuclear Regulatory Commission staff, which contains six questions related to the aforementioned license amendment request.

Attachment 1 provides the DNC response to the September 4, 2003, RAI. The additional information provided in this letter will not affect the conclusions of the Safety Summary and Significant Hazards Consideration discussion in the DNC April 7, 2003, letter.

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(1) J. A. Price letter to the U.S. NRC, "Millstone Power Station, Unit No. 3, License Basis Document Change Request (LBDCR) 3-15-02, Relocation of Some Technical Specification Parameters to The Core Operating Limits Report and Updating The Description of Analytical Methods Used to Determine Core Operating Limits," dated April 7, 2003.

(2) V. Nerses (NRC) facsimile, "Millstone Power Station, Unit No. 3, Facsimile Transmission, Draft Request For Additional Information (RAI) to be Discussed in an Upcoming Conference Call (TAC No. MB8387)," dated September 4, 2003.

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There are no regulatory commitments contained within this letter.

If you should have any questions regarding this submittal, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price
Site Vice President - Millstone

Sworn to and subscribed before me

this 18 day of September, 2003

Diane M. Phillipo
Notary Public

My Commission expires

DIANE M. PHILLIPO
NOTARY PUBLIC
MY COMMISSION EXPIRES 12/31/2005

Attachment (1)

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Unit No. 3
Millstone Senior Resident Inspector

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

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Attachment 1

Millstone Power Station, Unit No. 3

License Basis Document Change Request (LBDCR) 3-15-02
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Response to Request For Additional Information

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Response to Request For Additional Information**

Question 1:

In Generic Letter 88-16 entitled, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," the NRC established a generic process that eliminated the need to modify the technical specifications to update cycle-specific parameters. A central element of the process included the addition of a COLR to include cycle-specific parameters and NRC-approved methodologies. A licensee may add approved methodologies to the COLR list using the standard technical specification modification process. When the staff reviews licensee requests to relocate technical specifications to the COLR, it uses two important conditions to determine applicability. These two conditions are the following: 1) is the parameter cycle specific? and 2) is there an NRC approved methodology, which calculates the parameter in an acceptably conservative manner?

a. The staff requests that the licensee provide a detailed discussion of how refueling boron concentration meets the two conditions.

Response:

The limit of 2600 ppm for the refueling boron concentration stated in the Limiting Condition for Operation and the Action statement of Technical Specification (TS) 3/4.9.1.1 is relocated to the Core Operating Limits Report (COLR). The limit for the refueling boron concentration satisfies the two conditions delineated in Generic Letter 88-16 for the relocation of cycle specific physics parameters to the COLR, namely, the limit is cycle specific, and is calculated in an acceptably conservative manner using a Nuclear Regulatory Commission (NRC) approved methodology.

The limit for the refueling boron concentration is cycle specific:

The current TS 3/4.9.1.1 specifies a K_{eff} shutdown margin limit of 0.95 and a boron concentration limit of 2600 ppm. The reload cycle analysis for each cycle includes the calculation of the minimum boron concentration corresponding to a K_{eff} shutdown margin limit of 0.95. This value is a function of burn-up of the fuel, reactivity parameters and other cycle design conditions that will vary from cycle-to-cycle. The current boron concentration limit of 2600 ppm represents a conservative bound to the cycle dependent parameter.

<u>The limit for the refueling boron concentration is calculated using an NRC approved methodology:</u>

The method used for calculating this limit is described in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology." WCAP-9272-P-A is listed in Millstone Unit No. 3 Technical Specifications Section 6.9.1.6.b, which describes the analytical methods used to determine the core operating limits. This reference has been reviewed and approved by the NRC and fulfills the requirements for inclusion in Section 6.9.1.6.b. Section 6.9.1.6.b states, "The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC." Therefore, the limit for the refueling boron concentration is calculated using an NRC approved methodology.

<u>Question 2:</u>

During refueling, the reactor cavity, the transfer canal and the spent fuel pool form a single connected mass. Please describe the steps taken to ensure that there will not be a dilution event in either the reactor cavity or the spent fuel pool. How will you ensure that the concentrations are the same in the reactor cavity and the spent fuel pool?

<u>Response:</u>

<u>The steps taken to ensure that there will not be a dilution event in either the reactor cavity or the spent fuel pool are as follows:</u>

TS 3/4.9.1.1 states, "Additionally, the CVCS valves of Specification 4.1.1.2.2 shall be closed and secured in position." The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the Reactor Cooling System (RCS) including the reactor core, refueling cavity, and the spent fuel pool (SFP) (when the SFP is aligned to the RCS). This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

Additionally, Chemistry Surveillance Procedure SP 3863 provides instructions for sampling and analyzing the filled portions of the RCS and refueling cavity (including the SFP when the SFP is aligned to the RCS) for boron to ensure compliance with the requirements of TS 3/4.9.1.1. This procedure is performed at least once per 72 hours in MODE 6.

<u>How the reactor cavity and the spent fuel pool ppm concentrations are ensured to be the same:</u>

There is no specific Technical Specifications requirement for the concentrations in the reactor cavity and the SFP to be the same. However, the Millstone Unit

No. 3 Technical Specifications (3/4.9.1.1) requires the boron concentration of all filled portions of the RCS and the refueling cavity (which includes the SFP when the SFP is physically aligned to the RCS) to be sufficient to ensure either a $K_{eff} \leq 0.95$, or a boron concentration of greater than or equal to the cycle specific limit for the refueling boron concentration (currently 2600 ppm), whichever is more restrictive. So as long as the minimum boron concentration requirement of the Technical Specifications is met, the individual boron concentrations of the RCS, refueling cavity, and SFP (when aligned to the RCS) can be allowed to vary.

Operating procedure OP 3210A, "Refueling Preparation," ensures that 1) the RCS is borated to the requirements of TS 3/4.9.1.1 and 2) that the spent fuel pool boron concentration is greater than the RCS prior to the contained volumes being connected for refueling operations. Applicable procedures will be modified to ensure that the more restrictive boron concentration requirement between TS 3/4.9.1.1 and TS 3/4.9.1.2 is met prior to connecting the RCS and SFP.

When the reactor cavity and spent fuel pool are physically connected, Chemistry Surveillance Procedure SP 3863, "Reactor Coolant and Reactor Vessel Refueling Cavity Analysis for Boron," implements the boron monitoring requirements of TS 3/4.9.1.1. This monitoring is performed at the same frequency for the RCS and SFP. The cycle specific limit for the refueling boron concentration (currently 2600 ppm) will continue to be retained in the surveillance even after the implementation of the proposed TS 3/4.9.1.1 change.

Question 3:

WCAP-8301 is requested to be added to the TS list of COLR documents. Methodologies added to the COLR must be reviewed and approved. WCAP-8301 does not have the "-A" designation. Is this a reviewed and approved methodology? If so, please provide reference to the staff safety evaluation and provide justification for its use at Millstone 3. Include the date and code revision that you plan to use.

Response:

WCAP-8301[(1)] describes the LOCTA code, which was developed to resolve the heat conduction problem for a nuclear fuel rod accounting for the thermal-mechanical behaviors for the fuel pellet and cladding and the variation in the heat transfer coefficient on the outer surface of the cladding as a function of the coolant properties. The LOCTA code was used in conjunction with WFLASH (evaluation model prior to NOTRUMP) to determine the Small Break Loss of Coolant Accident (SBLOCA) transient response and Peak Cladding Temperature (PCT).

(1) WCAP-8301, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," (Proprietary), June 1974.

WCAP-8471-P-A[(2)] outlines the basis for the August 1974 version of the Westinghouse ECCS evaluation models. SER for WCAP-8471-P-A states that:

"The NRC staff has completed its review of the Westinghouse ECCS evaluation model, which is comprised of References 1 through 20. We closely followed the development of the Westinghouse ECCS evaluation model and utilized the referenced reports to determine the compliance of the Westinghouse evaluation model with Appendix K to 10 CFR Part 50... We conclude:

...

2. *That References 1 through 20, which constitute the consolidated description of the Westinghouse ECCS evaluation model, may be incorporated by reference in licensing applications as an accepted ECCS evaluation model. This acceptance applies to the Westinghouse ECCS model and does not constitute acceptance of the individual reports for any purpose other than for ECCS analyses".*

WCAP-8301 is Reference 14 in WCAP-8471-P-A. LOCTA-IV was used in the LOCA analysis for the initial licensing of Millstone Unit No. 3. NUREG-1031[(3)] documents the NRC review and approval of the LOCA analysis. In NUREG-1031, the NRC concluded that the analysis was acceptable and was performed with an evaluation model that had been previously reviewed and approved. WCAP-8301 is included in the Bibliography of NUREG-1031.

The revision and date of the code are those bounded by the approved topical report (WCAP-8301).

Question 4:

WCAP-10054-P-A (Addendum 2) was requested to be added to the TS list of COLR documents. This report has been reviewed and approved by the NRC. However, no specific evaluation was included to document its applicability to Millstone 3. Please provide justification for its use at Millstone 3. If it has been used previously, please provide reference to the staff approval for use at your plant. The safety evaluation includes the condition that "The generic submittal includes validation that the limiting break location has not shifted away from the cold legs to the hot or pump suction legs." Please include the date and the code revision that you plan to use.

(2) WCAP-8471-P-A, "The Westinghouse ECCS Evaluation Model: Supplementary Information," April 1975.

(3) NUREG-1031, "Safety Evaluation Report Related to The Operation of Millstone Nuclear Power Station, Unit No. 3," July 1984.

Response:

WCAP-10054-P-A[4] describes the NOTRUMP code used for small break LOCA analysis and is currently included in the Technical Specification list of COLR documents (item 8 of TS 6.9.1.6.b). The Millstone Unit No. 3 small break LOCA analysis utilizing the NOTRUMP and LOCTA-IV methodology was described in the Plant Safety Evaluation submitted to the NRC for Cycle 4.[5] In the Plant Safety Evaluation, it is confirmed that the limiting break location is the cold leg. The Millstone Unit No. 3 small break LOCA analysis using NOTRUMP was reviewed and approved by the NRC as documented in a letter dated March 11, 1991.[6]

In this request, we are adding Addendum 2 Revision 1 of WCAP-10054-P-A[7] to the Technical Specification list of COLR documents. As described in WCAP-10054-P-A Addendum 2 Revision 1, the addendum addresses the addition of the COSI model to the NOTRUMP code. WCAP-10054-P-A Addendum 2 Revision 1 provided sensitivity studies using the COSI model and SI injected into the broken loop on a sample 4-loop plant with a thermal power of 3250 MWt and an ECCS design typical of newer Westinghouse designs using 12 foot cores. These conditions are representative of Millstone Unit No. 3, and therefore the use of the COSI model and SI injected into the broken loop is directly applicable to Millstone Unit No. 3. A SBLOCA reanalysis was recently completed for Millstone Unit No. 3. The reanalysis used the latest version of the NOTRUMP-EM (WCAP-10054-P-A, WCAP-10079-P-A,[8] and WCAP-10054-P-A Addendum 2 Revision 1), which included the COSI model and SI injected into the broken loop (WCAP-10054-P-A Addendum 2 Revision 1). As such, WCAP-10054-P-A Addendum 2 Revision 1 should be referenced in Section 6.9.1.6.b of the Technical Specifications.

The revision and date of the code are those bounded by the approved topical reports (WCAP-10054-P-A, WCAP-10079-P-A, and WCAP-10054-P-A Addendum 2 Revision 1).

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(4) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

(5) Letter B13627 from E. J. Mroczka to U. S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3 Proposed Change to Technical Specifications Cycle 4 Reload Submittal," dated November 1, 1990.

(6) U. S. Nuclear Regulatory Commission letter to E. J. Mroczka, "Issuance of Amendment No. 60 to Facility Operating License No. NPF-49 (TAC NO. 77924)," dated March 11, 1991.

(7) WCAP-10054-P-A Addendum 2 Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

(8) WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

Question 5:

WCAP-8745-P-A was requested to be added to the TS list of COLR documents. This report has been reviewed and approved by the NRC, however no specific evaluation was included to document its applicability to Millstone 3. Does your plant operate under constant axial offset control?

Response:

Currently, Millstone Unit No. 3 Technical Specifications allow use of relaxed axial offset control (RAOC) or base load operation for the control of axial power distributions. The constant axial offset control (CAOC) procedure was utilized in Cycles 1 through 3. Issuance of Amendment No. 60 to the Millstone Unit No. 3 Operating License[(6)] supported improved Departure from Nucleate Boiling (DNB) analysis methods, use of VANTAGE 5H fuel and RAOC.

The NRC Safety Evaluation for Amendment No. 60 approved changes in the DNB analysis methodology (use of WRB-1 and WRB-2 correlations), fuel design changes (use of VANTAGE 5H fuel) and plant operating procedures (RAOC). Each of these changes is supported by an NRC approved topical report.[(9),(10),(11)] As a result of these changes, Over Temperature Delta T (OTΔT) and Over Power Delta T (OPΔT) trip setpoints were revised and reflected in the safety analysis.

Implementation of new DNB correlations, new fuel design and a new plant operating procedure at Millstone Unit No. 3 were accounted for in the application of WCAP-8745-P-A that generated the revised reactor trip setpoints. The scope and applicability discussion in the NRC Safety Evaluation of WCAP-8745 acknowledges the topical reports in WCAP-8762-P-A,[(9)] WCAP-10444-P-A[(10)] and WCAP-10216-P-A-R1A,[(11)] and concludes that the effect on core power distributions must be evaluated. This evaluation was performed in support of Amendment No. 60, which was reviewed and approved by the NRC. Therefore, WCAP-8745-P-A remains applicable to Millstone Unit No. 3 with the current operating procedures allowed by the Technical Specifications.

Question 6:

Page 1 of Attachment 1 of the April 7, 2003, submittal states that the proposed changes are consistent with the changes included in Technical Specifications Task Force (TSTF) Traveler TSTF-339, Revision 2. Specifically the proposed changes

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(9) F. E. Motley, K. W. Hill, F. F. Cadek, and J. Shefchek, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762-P-A, July 1984.

(10) Davidson, S. L. ed. et al., "VANTAGE 5 Fuel Assembly Reference Core Report," WCAP-10444-P-A, September 1985.

(11) R. W. Miller, et al., "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," WCAP-10216-P-A-R1A, Rev. 1, February 1994.

relocate several Technical Specification parameters to the COLR consistent with WCAP-14483. Identify and justify any deviation from TSTF-339, Revision 2 for its use at Millstone 3.

Response:

TSTF-339, Revision 2, provides the option to relocate a number of parameters from the Technical Specifications to the Core Operating Limits Report. The Technical Specification changes provided in TSTF-339 are based upon the Technical Specification format and content given in NUREG-1431 for the Improved Standard Technical Specifications. Since Millstone Unit No. 3 has not converted to NUREG-1431 some of the changes given in the TSTF-339 are more complicated to implement in the current Millstone Unit No. 3 Technical Specifications. Thus, each of the changes in the TSTF was evaluated to determine the scope of this submittal. The results of this evaluation are summarized below.

Parameter	Evaluation
1. Safety Limit Parameters	Not proposed for relocation. The TSTF splits the current single safety limit curve into two separate parameters. It has been determined that the current Millstone Technical Specification is adequate and that the added complexity of two parameters is unnecessary.
2. Overtemperature ΔT Coefficients	Proposed for relocation to the COLR.
3. Overpower ΔT Coefficients	Proposed for relocation to the COLR.
4. Pressurizer pressure	Proposed for relocation to the COLR.
5. RCS average temperature	Proposed for relocation to the COLR.
6. RCS total flow rate	Not proposed for relocation. Unlike NUREG-1431, RCS total flow rate is not included in the Millstone Unit No. 3 TS 3/4.2.5 or in Table 3.2-1.