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

Dominion

APR 11 2001

Docket No. 50-336
B18381

RE: 10 CFR 50.90

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

Millstone Nuclear Power Station, Unit No. 2
Changes to Technical Specifications
Updating List of Documents Describing the Analytical Methods Specified in
Technical Specification 6.9.1.8b (TSCR 2-4-01)

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Millstone Unit No. 2 Technical Specifications. The proposed changes will update the list of documents, describing the analytical methods used to determine the core operating limits, specified in Technical Specification 6.9.1.8b. The reason for these changes is to incorporate the most recent, Nuclear Regulatory Commission (NRC) approved, methodology documents in Millstone Unit No. 2 Technical Specifications. These changes will update the documents describing the analytical methods used in the current Small Break Loss of Coolant Accident analysis (SBLOCA), setpoint methodology, and Non-LOCA methodology. In addition, the revision number and the date of documents listed in Technical Specification 6.9.1.8b will be deleted. The method of referencing topical reports in Technical Specification by citing the report number and title only has been approved by the NRC in Standard Technical Specification Change Traveler TSTF-363.[1]

Attachment 1 provides a discussion of the proposed changes and the Safety Summary. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides the marked-up version of the appropriate pages of the current Technical Specifications. Attachment 4 provides the retyped pages of the Technical Specifications.

A001

(1) TSTF-363, "Revise Topical Report References in ITS 5.6.5, COLR," Approved on February 22, 2000.

Environmental Considerations

DNC has reviewed the proposed License Amendment Request against the criteria of 10 CFR 51.22 for environmental considerations. The proposed changes will update the list of documents, describing the analytical methods used to determine the core operating limits, specified in Technical Specification 6.9.1.8b. These changes will not increase the type and amounts of effluents that may be released offsite. In addition, this amendment request will not increase individual or cumulative occupational radiation exposures. Therefore, DNC has determined the proposed changes will not have a significant effect on the quality of the human environment.

Conclusions

The proposed changes do not involve a significant impact on public health and safety (see the Safety Summary provided in Attachment 1) and do not involve a Significant Hazards Consideration pursuant to the provisions of 10 CFR 50.92 (see the Significant Hazards Consideration provided in Attachment 2).

Plant Operations Review Committee and Nuclear Safety Assessment Board

The Plant Operations Review Committee and Nuclear Safety Assessment Board have reviewed and concurred with the determinations.

Schedule

We request issuance of this amendment prior to restart from refueling outage 14, which is currently scheduled in early February of 2002, with the amendment to be implemented within 30 days of issuance.

State Notification

In accordance with 10 CFR 50.91(b), a copy of this License Amendment Request is being provided to the State of Connecticut.

There are no regulatory commitments contained in 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.

Raymond P. Necci
Vice President - Nuclear Technical Services

Subscribed and sworn to before me

this 11th day of April, 2001

Notary Public

Date Commission Expires:

SANDRA J. ANTON
NOTARY PUBLIC
COMMISSION EXPIRES
MAY 31, 2005

Attachments (4)

cc: H. J. Miller, Region I Administrator
D. S. Collins, NRC Project Manager, Millstone Unit No. 2
S. R. Jones, Senior Resident Inspector, Millstone Unit No. 2

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 Nuclear Power Station, Unit No. 2

Change to Technical Specifications
Updating List of Documents Describing the Analytical Methods Specified in Technical Specification 6.9.1.8b (TSCR 2-4-01)
Discussion of Changes and Safety Summary

Change to Technical Specifications
Updating List of Documents Describing the Analytical Methods Specified in Technical Specification 6.9.1.8b (TSCR 2-4-01)
Discussion of Changes and Safety Summary

Introduction

Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Millstone Unit No. 2 Technical Specifications. The proposed changes will update the list of documents, describing the analytical methods used to determine the core operating limits, specified in Technical Specification 6.9.1.8b.

Description of Proposed Change

The proposed changes will update the documents describing the Nuclear Steam Supply System (NSSS) vendor's methodology given in Technical Specification 6.9.1.8b. The following are the proposed changes:

1. The method of referencing topical reports in Technical Specification by citing the report number and title only has been approved by the Nuclear Regulatory Commission (NRC) in Standard Technical Specification Change Traveler TSTF-363.[(1)] DNC proposes to adopt TSTF-363 for Millstone Unit No. 2. The revision number and the date are deleted from the documents contained in sections 6.9.1.8b.1 through 6.9.1.8b.15 in accordance with TSTF-363. This method of referencing topical reports would allow DNC to use current topical reports to support limits in the Core Operating Limit Report (COLR) without having to submit an amendment to the facility operating license every time the topical report is revised. The COLR, which is part of the Millstone Unit No. 2 Technical Requirements Manual (TRM), would provide specific information identifying the particular approved topical reports used to determine the core limits for the particular cycle in the COLR report. This would eliminate unnecessary expenditure of NRC and DNC resources, and would ease the burden of Technical Specification submittal and approval needed to license reload fuel.

2. The documents contained in section 6.9.1.8b.6 and 6.9.1.8b.7, describing Small Break Loss of Coolant Accident analysis (SBLOCA) methodology, will be replaced with the document listed in Insert A. This change is required to identify the most recent methodology description for SBLOCA analysis.[(2)] The document contained in

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(1) TSTF-363, "Revise Topical Report References in ITS 5.6.5, COLR," Approved on February 22, 2000.

(2) EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model S-RELAP5 Based," Framatome ANP, March 2001.

section 6.9.1.8b.7 is deleted and documents 6.9.1.8b.8 through 6.9.1.8b.15 are renumbered.

3. The document contained in section 6.9.1.8b.15 (renumbered 6.9.1.8b.14), describing setpoint methodology, will be replaced with the document listed in Insert B. This change is required to identify the most recent setpoint methodology for Combustion Engineering type reactors.(3)

4. A new document is added as 6.9.1.8b.15. This document describes Standard Review Plan (SRP) Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors. The NRC is still in the process of reviewing this document (TAC MA7192). In the event that the NRC review is not completed prior to issuing of this amendment, this document will be removed from the list and the NRC will be provided with updated section 6.9.1.8.b pages.

The documents listed in Inserts A and B have already been approved by the NRC.

<u>Safety Summary</u>

The proposed changes will update the documents describing the NSSS vendor's methodology given in Technical Specification 6.9.1.8b. Replacing the documents contained in section 6.9.1.8b.6 and 6.9.1.8b.7 with the document listed in Insert A and deleting the documents contained in section 6.9.1.8b.7 is required to identify the most recent methodology description for Small Break Loss of Coolant Accident analysis (SBLOCA).

The use of the revised methodology constitutes an improvement over the previous methodology and is in accordance with the NRC's Safety Evaluation Report (SER). Therefore, these proposed changes will have no adverse effect on plant safety.

Replacing the document contained in section 6.9.1.8b.15 (renumbered 6.9.1.8b.14) with the document listed in Insert B is required to identify the most recent setpoint methodology for Combustion Engineering type reactors. The document listed in Insert B has already been approved by the NRC. The use of the revised methodology constitutes an improvement over the previous methodology, provides a conservative setpoint determination, and is in accordance with the NRC's SER. Therefore, these proposed changes will have no adverse effect on plant safety.

The topical report associated with the document listed in 6.9.1.8b.15 is being reviewed by the NRC. Document 6.9.1.8b.15 will be added upon NRC approval of the topical report. Adding a new document as 6.9.1.8b.15 is required to identify Standard Review Plan (SRP) Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors. The

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(3) EMF-1961(P)(A) Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, July 2000.

use of this methodology constitutes an improvement over the previous methodology and is in accordance with the NRC's SER. Therefore, this proposed change will have no adverse effect on plant safety.

Deleting the revision number and the date from the documents contained in sections 6.9.1.8b.1 through 6.9.1.8b.15 has no impact on the actual analytical methods used to determine the core operating limits, nor does it have impact on the calculations performed for current or future reloads. This change is administrative in nature. Therefore, this proposed change will have no adverse effect on plant safety.

Attachment 2

Millstone Nuclear Power Station, Unit No. 2
Change to Technical Specifications
Updating List of Documents Describing the Analytical Methods Specified in
Technical Specification 6.9.1.8b (TSCR 2-4-01)
Significant Hazards Consideration

Proposed Revision to Technical Specifications Updating List of Documents Describing the Analytical Methods Specified in Technical Specification 6.9.1.8b (TSCR 2-4-01) Significant Hazards Consideration

Significant Hazards Consideration

In accordance with 10 CFR 50.92, Dominion Nuclear Connecticut, Inc. (DNC) has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change in document 6 and the deletion of document 7 of Technical Specification 6.9.1.8b are made to identify the most recent, Nuclear Regulatory Commission (NRC) approved, model used in Small Break Loss of Coolant Accident (SBLOCA) applications. This methodology meets the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K. This change has no impact on plant equipment operation. Since the change only affects the SBLOCA analysis, it cannot affect the likelihood or consequences of accidents. Therefore, this change will not increase the probability or consequences of an accident previously evaluated.

   The proposed change in document 15 (renumbered 14) of Technical Specification 6.9.1.8b is made to identify the most recent, NRC approved, setpoint methodology for Combustion Engineering type reactors. This change has no impact on plant equipment operation. Since the change only affects the setpoint analysis and is an approved methodology, it cannot affect the likelihood or consequences of accidents. Therefore, this change will not increase the probability or consequences of an accident previously evaluated.

   The proposed change to add a new document as 6.9.1.8b.15 is required to identify the most recent Non-LOCA methodology to be used in the Millstone Unit No. 2 Non-LOCA analysis. The use of this methodology will demonstrate that the acceptance criteria for Non-LOCA events are met. This change has no impact on plant equipment operation. Since the change only affects the Non-LOCA analysis and is an approved methodology, it cannot affect the likelihood or consequences of accidents. Therefore, this change will not increase the probability or consequences of an accident previously evaluated.

   Deleting the revision number and the date from the documents contained in sections 6.9.1.8b.1 through 6.9.1.8b.15 has no impact on the actual analytical methods used to determine the core operating limits, nor does it have impact on the calculations performed for current or future reloads. This change is administrative in nature. This change has no impact on plant equipment operation nor does it affect the likelihood or consequences of accidents. Therefore, this change will not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. These changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

   The proposed changes have no impact on plant equipment operation. The use of the revised methodologies in accordance with NRC's Safety Evaluation Report will meet the acceptance criteria for accident analysis. Therefore, the proposed changes will not result in a reduction in a margin of safety.

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Attachment 3

Millstone Nuclear Power Station, Unit No. 2

Change to Technical Specifications
Updating List of Documents Describing the Analytical Methods Specified in Technical Specification 6.9.1.8b (TSCR 2-4-01)
Marked Up Pages

MONTHLY OPERATING REPORT (Con't)

Administrator, Region I, and one copy to the NRC Resident Inspector, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.8 a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle.

3/4.1.1.1	SHUTDOWN MARGIN - T_{avg} > 200°F
3/4.1.1.2	SHUTDOWN MARGIN - $T_{avg} \leq$ 200°F
3/4.1.1.4	Moderator Temperature Coefficient
3/4.1.3.6	Regulating CEA Insertion Limits
3/4.2.1	Linear Heat Rate
3/4.2.3	Total Integrated Radial Peaking Factor - F_r^T
3/4.2.6	DNB Margin

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1) EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 - Methodology Description, Volume 2 - Benchmarking Results," Siemens Power Corporation~~, January 1997~~.

2) ANF-84-73 ~~Revision 5~~ Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels~~, July 1990~~.

3) XN-NF-82-21(P)(A) ~~Revision 1~~, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company~~, September 1983~~.

4) EMF-84-093(P)(A) ~~Revision 1~~, "Steamline Break Methodology for PWRs," Siemens Power Corporation~~, February 1999~~.

5) XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company~~, October 1983~~.

Replace with Insert A

6) ~~XN-NF-82-49(P)(A) Revision 1, "EXXON Nuclear Company Evaluation Model EXEM PWR Small Break Model," Advanced Nuclear Fuels Corporation, April 1989.~~

~~7) XN-NF-82-49(P)(A) Revision 1 Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model," Siemens Power Corporation, December 1994.~~

7 ~~8~~) EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation~~, June 1999~~.

Page 6-18a, Insert A

EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model S-RELAP5 Based," Framatome ANP.

CORE OPERATING LIMITS REPORT (CONT.)

8 ~~9~~) XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized water reactors," Exxon Nuclear Company~~, October 1983~~.

9 ~~10~~) XN-NF-621(P)(A) ~~Revision 1~~, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company~~, September 1983~~.

10 ~~11~~) XN-NF-82-06(P)(A) ~~Revision 1~~ and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company~~, October 1986~~.

11 ~~12~~) ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation~~, December 1991~~.

12 ~~13~~) XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company~~, November 1988~~.

13 ~~14~~) ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation~~, May 1992~~. Replace with insert B

14 ~~15~~) ~~XN-NF-507(P)(A) Supplements 1 and 2, "ENC Setpoint Methodology for C.E. Reactors: Statistical Setpoint Methodology," Exxon Nuclear Company, September 1986~~.

Insert C

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Deleted

Page 6-19, Insert B

EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation.

Page 6-19, Insert C

15) EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP.

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Attachment 4

Millstone Nuclear Power Station, Unit No. 2

Change to Technical Specifications
Updating List of Documents Describing the Analytical Methods Specified in Technical Specification 6.9.1.8b (TSCR 2-4-01)
Retyped Pages

MONTHLY OPERATING REPORT (Con't)

Administrator, Region I, and one copy to the NRC Resident Inspector, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.8 a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle.

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3/4.1.3.6	Regulating CEA Insertion Limits
3/4.2.1	Linear Heat Rate
3/4.2.3	Total Integrated Radial Peaking Factor - F_r^T
3/4.2.6	DNB Margin

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1) EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 - Methodology Description, Volume 2 - Benchmarking Results," Siemens Power Corporation.

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4) EMF-84-093(P)(A), "Steamline Break Methodology for PWRs," Siemens Power Corporation.

5) XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company.

6) EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model S-RELAP5 Based," Framatome ANP.

7) EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation.

CORE OPERATING LIMITS REPORT (CONT.)

8) XN-NF-78-44(NP)(A), "A Generic Analysis of the Control rod Ejection Transient for Pressurized water reactors," Exxon Nuclear Company.

9) XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company.

10) XN-NF-82-06(P)(A) and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company.

11) ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation.

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13) ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation.

14) EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation.

15) EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP.

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Deleted