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

March 26, 2010

Mr. Mark A. Schimmel
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power – Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 – REQUESTS FOR ADDITIONAL INFORMATION RE: LICENSE AMENDMENT REQUEST TO ADOPT THE ALTERNATIVE SOURCE TERM METHODOLOGY (TAC NOS. ME2609 AND ME2610)

Dear Mr. Schimmel:

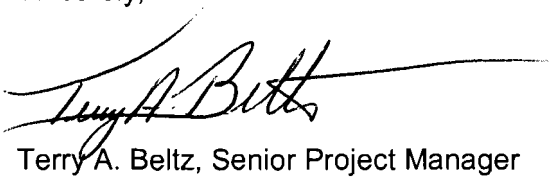
By letter to the U.S. Nuclear Regulatory Commission (NRC) dated October 27, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093160583), Northern States Power Company, a Minnesota corporation (NSPM), submitted a license amendment request for the Prairie Island Nuclear Generating Plant, Units 1 and 2. The proposed amendment would adopt the Alternative Source Term (AST) methodology, in addition to technical specification changes supported by the AST design basis accident radiological consequences analysis. The proposed amendment would also incorporate Technical Specification Task Force (TSTF)-490, "Deletion of E-Bar Definition and Revision to RCS [reactor coolant system] Specific Activity Tech Spec," Revision 0.

To complete their technical review, the NRC staff provided draft requests for additional information (RAIs) to NSPM (ADAMS Accession Nos. ML100620213, ML100680414, and ML100750041). The draft RAIs were discussed with your staff to resolve outstanding concerns or provide additional clarity regarding the requested information.

The finalized RAIs are being issued as an enclosure to this letter. As agreed upon with Ms. Amy Hazelhoff, please respond to the enclosed RAIs no later than April 30, 2010.

If you have any questions, please contact me at (301) 415-3049.

Sincerely,



Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosure:
As stated

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REQUESTS FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST (LAR) REGARDING

ALTERNATIVE SOURCE TERM (AST)

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

TAC NOS. ME2609 AND ME2610

ELECTRICAL ENGINEERING BRANCH (EEEB)

EEEB RAI 1

Provide the changes in the emergency diesel generator (EDG) loading due to this LAR.

Provide the EDG loads for Units 1 and 2 pre- and post-AST, and confirm that adequate margin exists post-AST. Confirm that EDG testing envelopes the loading requirements due to this LAR.

EEEB RAI 2

Confirm that no credit has been taken for non-safety related system(s) due to this LAR.

If yes, then a) describe the impact on the EDG loading due to these systems, b) describe how electrical and physical separation, and single failure criteria, have been met, and c) describe how the operators will be notified (e.g., control room annunciators) in the event that these systems become inoperable.

EEEB RAI 3

Confirm that there are no changes to the environmental qualification program due to this LAR.

MECHANICAL AND CIVIL ENGINEERING BRANCH (EMCB)

EMCB RAI 1

Page 9 of the enclosure in Reference 1 indicates that credit will be taken for isolation dampers, which make up a portion of the Auxiliary Building Normal Ventilation System, as part of the proposed AST implementation. Various justifications are provided for crediting these non-safety related dampers as part of the proposed implementation. However, no justification is provided regarding the structural and seismic ruggedness of this equipment. Appendix A to 10 CFR Part 100 requires that structures, systems, and components necessary to assure the capability of the plant to mitigate the consequences of accidents, which could result in exposures comparable to the guideline exposures provided in 10 CFR Part 100, be designed to remain functional during and after a safe shutdown earthquake.

Please discuss the methodologies used to demonstrate the seismic ruggedness and/or seismic qualification of the aforementioned dampers. Additionally, please provide the references which provide the regulatory acceptance bases of these methodologies.

REACTOR SYSTEMS BRANCH (SRXB)

The following requests for additional information are associated with the NRC staff review of the steam generator tube rupture (SGTR) overfill analysis in support of the AST LAR application, as discussed in the Enclosure to Reference 1, Pages 106 and 107.

Please provide the following information for the NRC staff to continue its review:

SRXB 1

Discuss the methods used for the SGTR overfill analysis. If the methods were previously approved by NRC, list the NRC safety evaluation reports approving the methods. If the methods were not reviewed and approved by NRC, address acceptability of the methods.

The information should include a description and justification of reactor coolant system (RCS) models with safety injection simulation, models for determination of the primary-to-secondary break and steam relief flow rates, and steam generator (SG) water level model accounting for the effects of bubble formation during depressurization on the SG water level for a SGTR event.

SRXB 2

Provide a list the nominal values with measurement uncertainties and the corresponding values used in the SGTR overfill analysis for the following applicable plant parameters:

- Initial RCS pressure
- Initial SG water inventory
- Safety injection actuation pressure setpoint
- Safety injection flow versus RCS pressure
- Safety injection system pump delay time
- SG relief valve pressure setpoint
- Auxiliary feedwater actuation setpoint and delay time
- Auxiliary feedwater flow rate per SG
- Auxiliary feedwater temperature
- Time of loss of offsite power
- Delay times for reactor trip and turbine trip
- Decay heat model and initial value in percentage of the rated power level

Discuss the effects of an increase or decrease in the value for each of the above plant parameters on the SG water level calculations during a SGTR and address the adequacy of the values used in the SGTR overfill analysis in minimizing the margin to SG overfill.

SRXB 3

List operator action times for the following applicable operator actions as determined by the plant simulator in accordance with the Emergency Operating Procedure E-3 for a SGTR:

- Identify and isolate the rupture SG
- Initiate RCS cooldown
- Initiate RCS depressurization
- Terminate safety injection flow
- Establish charging flow
- Establish RCS letdown
- Reopen pressurizer power-operated relief valve

Discuss the operator actions credited in the SGTR overfill analysis and provide a sequence of events for the SGTR including the above operator action times, and calculated times for the RCS cooldown, RCS depressurization and equalization of RCS and ruptured SG pressure. The information should show that the operator actions and their associated times assumed in the analysis were identical with that determined by the plant simulator.

SRXB 4

List the single failure events considered in the SGTR overfill analysis and identify the worst single failure used in the analysis that resulted in a minimum margin to the SG overfill.

SRXB 5

Provide the results of the SGTR overfill analysis for the following applicable plant parameters:

- Pressurizer pressure versus time
- Secondary pressures and SG water volumes versus time for both intact and rupture SGs
- Total primary to secondary break flow and total integrate primary to secondary break flow versus time
- SG relief flow and integrated SG relief flow versus time for both intact and rupture SGs

The results indicate the following: the calculated RCS break and SG relief flowrates are consistent with the primary and secondary pressures; there is no unexplainable thermal-hydraulic phenomenon; the RCS pressure and the rupture SG secondary pressure are equal; and, the SG does not overfill with water.

References:

1. Letter from Mark Schimmel, Xcel Energy, to the U.S. Nuclear Regulatory Commission Document Control Desk, Re: Prairie Island Nuclear Generating Plant, Units 1 and 2, Dockets 50-282 and 50-306, License Nos. DPR-42 and DPR-60, "License Amendment Request to Adopt Alternative Source Term Methodology," dated October 27, 2009 (ADAMS Accession No. ML093160583).

If you have any questions, please contact me at (301) 415-3049.

Sincerely,

/RA/

Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosure:
As stated

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ADAMS Accession No.: ML100820298 * concurrence via memo ** concurrence via e-mail

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