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

February 11, 2009

Mr. Michael D. Wadley
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO LICENSE AMENDMENT REQUEST FOR TECHNICAL SPECIFICATIONS CHANGES TO ALLOW USE OF WESTINGHOUSE 0.422-INCH OD 14X14 VANTAGE+ FUEL (TAC NOS. MD9142 AND MD9143)

Dear Mr. Wadley:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated June 26, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081821037), as supplemented by letters dated August 4 (ADAMS Accession No. ML082210260), August 26 (ADAMS Accession No. ML082400518), November 14, 2008 (ADAMS Accession No. ML083190820), and January 30, 2009 (ADAMS Accession No. ML090300684), Nuclear Management Company, LLC, a predecessor license holder to the Northern States Power Company, a Minnesota corporation, submitted a request for Technical Specification changes to allow the use of Westinghouse 0.422-inch outside diameter 14X14 VANTAGE+ fuel for Prairie Island Nuclear Generating Plant, Units 1 and 2.

The NRC staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is addressed in the enclosure to this letter. During a discussion with your staff on February 6, 2009, it was agreed that you would provide a response within 30 days of the date of this letter.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-4037.

Sincerely,

Thomas J. Wengert

Thomas J. Wengert, 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:
Request for Additional Information

cc w/encl: Distribution via ListServ

<u>REQUEST FOR ADDITIONAL INFORMATION</u>

<u>PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2</u>

<u>DOCKET NOS. 50-282 AND 50-306</u>

In reviewing the Nuclear Management Company, LLC*, a predecessor license holder to the Northern States Power Company, a Minnesota corporation (the licensee), submittal dated June 26, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081821037), as supplemented by letters dated August 4 (ADAMS Accession No. ML082210260), August 26 (ADAMS Accession No. ML082400518), November 14, 2008 (ADAMS Accession No. ML083190820), and January 30, 2009 (ADAMS Accession No. ML090300684), which requested technical specification (TS) changes related to a change in fuel type from Westinghouse 0.400-inch outside diameter (OD) Vantage+ (400V+) fuel to Westinghouse 0.422-inch OD Vantage+ (422V+) fuel for Prairie Island Nuclear Generating Plant, Units 1 and 2 (PINGP), the U.S. Nuclear Regulatory Commission (NRC) staff has determined that the following information is needed in order to complete its review:

<u>Containment and Ventilation Systems (SCVB) Requests for Additional Information (RAIs)</u>

1. In Enclosure 1, Attachment 4, Section 5.3.3 of the June 26, 2008, license amendment request (LAR), it is stated that [PINGP] has applied [leak-before-break (LBB)] to the main [reactor coolant system (RCS)] piping. Please verify that ruptures of piping that are not subject to LBB do not result in subcompartment pressurizations that exceed pressure or structural design limits.

2. As described in Enclosure 1, Attachment 4, Section 5.4.1.2.3 of the LAR, liquid entrainment is included in the break flow for the main steamline break inside containment analysis. In the LAR, the licensee states that the entrainment characteristics for large steamline breaks are not sensitive to the steam generator design and that the NRC staff agreed to this position for a [Point Beach Nuclear Plant (Point Beach)] LAR.

   However, the NRC staff notes that this agreement was based upon different Westinghouse steam generator designs than those at PINGP.

   (a) Please explain why the same conclusion holds for the ANP 56/19 steam generator.

   (b) For the referenced Point Beach analyses an additional 0.1 was added to the quality. Was this done for PINGP steamline break mass and energy release analyses?

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* On September 22, 2008, Nuclear Management Company, LLC (NMC), transferred its operating authority to Northern States Power Company, a Minnesota corporation (NSPM). By letter dated September 3, 2008, NSPM stated that it would assume responsibility for actions and commitments submitted by NMC.

Mechanical and Civil Engineering Branch (EMCB) RAIs

3. EMCB RAI-1:

   (a) Section 2.4, Cladding Stress and Strain, first bullet, page 2-6 of Reference 1 mentions that the stress limit is based on the American Society of Mechanical Engineers (ASME) code. Provide the specific section, subsection, and edition of the ASME code utilized.

   (b) Section 2.4, Cladding Fatigue, page 2-8 of Reference 1 states that the cumulative usage factor (CUF) is less than the design limit. Please provide the numerical value of the computed CUF for cladding of 422V+ fuel cladding as well as for the current 400V+ fuel in operation at PINGP.

   (c) Section 2.4, End Plug Weld Integrity, page 2-9 of Reference 1 states that the fuel system will not be damaged due to excessive end plug weld tensile pressure differential loads. Please provide the computed values of the end plug weld tensile pressure differential loads for the 422V+ and 400V+ fuel rod end plug welds along with the acceptable or allowable limits.

4. EMCB RAI-2: Section 2.5.2, Grid Load Analysis, page 2-10 of Reference 1:

   (a) Provide numerical values of the computed maximum grid impact force (combined from safe-shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) analyses) and the allowable grid strength for the homogeneous and mixed cores.

   (b) Provide a summary of fuel assembly (fuel rods and thimble tubes) stresses for operating-basis earthquake, and combined SSE seismic and LOCA loadings along with the corresponding allowable stresses for the 422V+ assembly design.

5. EMCB RAI-3: Section 6.1.4, pages 6-2 to 6-5 of Reference 1 addresses mechanical system evaluations for LOCA and seismic loads using a mathematical model consisting of 3 sub-models of reactor vessel, internals such as core barrel, and support plates and fuel based on the ANSYS finite element code.

   (a) Provide a summary of the results of maximum combined LOCA and SSE loads and the corresponding allowable loads for 14X14 type guide tubes.

   (b) Provide a summary of the flow induced vibration levels and the acceptance limits during normal operation at the fuel upgrade analyzed reactor coolant system conditions.

   (c) Provide a summary of the results (computed stresses, allowable stress limits, and fatigue usage factors) of structural evaluations of the reactor vessel internal components for the 422V+ assembly design.

Reference 1: Enclosure 1, Attachment 4 (Prairie Island Units 1 and 2 422V+ Reload Transition Licensing Report) of NMC's application dated June 26, 2008, License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14X14 VANTAGE+Fuel.

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/RA/

Thomas J. Wengert, Senior Project Manager
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NAME	TWengert	THarris	KManoly*	RDennig	LJames
DATE	02/10/ 09	02/10/09	01/26/09	02/10/09	02/11/09

*via memo dated 1/26/09 **OFFICIAL RECORD COPY**