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

October 17, 2008

Mr. Michael D. Wadley
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 - 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, as supplemented by letters dated August 4 and August 26, 2008, Nuclear Management Company, LLC, submitted a request for Technical Specifications changes to allow the use of Westinghouse 0.422-inch outside diameter (OD) 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 October 6, 2008, 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, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-282 and 50-306

Enclosure:
Request for Additional Information

cc w/encl: Distribution via ListServ

REQUEST FOR ADDITIONAL INFORMATION (RAI)

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

In reviewing the Nuclear Management Company, LLC (NMC), submittal dated June 26, 2008, as supplemented by letters dated August 4 and August 26, 2008, which requested technical specification 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 the Prairie Island Nuclear Generating Plant, Units 1 and 2, the U.S. Nuclear Regulatory Commission (NRC) staff has determined that the following information is needed in order to complete its review:

The license amendment request included a revisitation of the accident and transient analyses, some of which were re-analyzed, and the loss-of-coolant accident (LOCA) analyses, which were affected by the requested fuel transition.

1. Page 1-7, Table 1-1 of the PINGP 422V+ Reload Licensing Report states, "The power uncertainty was reduced to account for installation of a more accurate flow measurement system used in the power measurement. The [Revised Thermal Design Procedure (RTDP)] analyses completed within this report were thus completed at a bounding high power level to confirm acceptable operation at any power level, including measurement uncertainties of 0.5 percent or more, up to 1,683 [megawatts thermal (MWt)]." Please justify the uncertainty reduction:

    a. Explain what flow measurement system was installed.

    b. Provide reference to applicable supporting documentation, such as topical reports describing the flow measurement system.

    c. Briefly describe the flow measurement system installation and calibration process.

2. The information contained in Table 1-1, discussed in RAI 1 above, appears slightly inconsistent with the information contained in Table 4-1, on Page 4-8, of the licensing report, which states, "A power level of 1,677 MWt has been used for all RTDP thermal-hydraulic design analyses. For analyses explicitly modeling parameter uncertainties, a power level of 1,683 MWt was used." Please provide additional information about the analytic incorporation of power uncertainty to bring these two statements into clearer alignment.

3. Page 4-4 of the licensing report states "There is a maximum 9.0-percent transition core [departure from nucleate boiling ratio (DNBR)] penalty for the 400V+ fuel which will be offset by a 6.0-percent [FdH] reduction in burned 400V+ fuel based on a conservative 1.5-percent DNBR: 1-percent [FdH] sensitivity." This treatment of DNBR margin trade-off is presented as axiomatic; however, the NRC staff is unfamiliar with this sensitivity. Please provide a basis for this statement.

    a. Reference an appropriate licensing topical report where this sensitivity is described.

b. If the sensitivity is not discussed in a licensing topical report, please provide a phenomenological discussion of the peaking behavior of previously irradiated fuel and explain how the changes in peaking behavior result in increased DNBR margin.

c. Presumably, the steady-state peaking effects of previously irradiated fuel that result in a DNBR margin increase are propagated through DNBR transient analyses. Is the margin increase observed above based on a steady-state or transient power shape?

4. Table 4-3 on Page 4-10 of the licensing report lists a 15 psi increase in RTDP pressure uncertainty; however, the NRC staff was unable to locate a discussion of this increase in the licensing report. Please explain.

5. Section 5.1 of the licensing report discusses the Rod Withdrawal Accident from a Subcritical Condition. An isothermal temperature coefficient of +5 pcm/°F is assumed, and the accident initiates at 547°F. As a part of the justification for these assumptions, the licensing report states, "...after the initial neutron flux peak, the isothermal temperature coefficient can affect the succeeding rate of power increase." Regarding the selection of parameters affecting heat transfer, the analysis is designed so that it "yields a larger peak heat flux."

   a. Confirm whether the assumed isothermal temperature coefficient is bounding of that at lower assumed temperatures.

   b. Explain how the effect of selecting input conditions to maximize heat flux results in a conservative hot rod fuel temperature, or how other input assumptions correct or compensate for the maximized heat flux.

6. The introduction of larger fuel assemblies will reduce the volume of water in the core. Thus, a chemical and volume control system malfunction could dilute the core more rapidly. Confirm that the assumed reactivity insertion rate associated with a boron dilution is bounding for the 422V+ core, which will have a reduced volume.

7. The reactor coolant pump locked rotor/shaft break analysis presents hot spot cladding inner temperature as a function of time. This result is attained based on standard Westinghouse analytic assumptions used to maximize fuel energy delivery to the cladding.

   a. Why is fuel centerline temperature not an acceptance criterion for this postulated accident scenario?

   b. How is the fuel centerline temperature affected by the assumptions discussed above?

   c. What is the predicted peak fuel centerline temperature for this accident?

8. Page 5-72 of the licensing report discusses the triviality of differences arising from postulating a LOCA in a transition core as opposed to the analyzed equilibrium 422V+ core. It is stated, "Even for larger [small-break loss-of-coolant accidents (SBLOCAs)], the thermal-hydraulic response is quasi-one dimensional..." This statement is offered to

assess the significance of the potential for flow redistribution between the 400V+ and the 422V+ fuel assemblies. Presumably, the quasi-one dimensionality of transition cores has been assessed. Please provide a summary of this assessment to help substantiate the quasi-one dimensionality claim.

9. In consideration of the Westinghouse position on loop seal plugging, the licensing report discusses the effects that gaps between the core barrel upper plenum nozzles and the vessel may have on the effectiveness of vapor relief. Please provide an assessment of this statement in terms of the figures of merit discussed in the 1997 report, NSD-NRC-97-5092, "Core Uncovery Due to Loop Seal Re-Plugging During Post-LOCA Recovery." Particularly, consider some of the stable or unstable core uncovery envelopes, and evaluate the effects of a change in $K/A^2$ would have on the core uncovery envelopes.

October 17, 2008

Mr. Michael D. Wadley
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 - 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, as supplemented by letters dated August 4 and August 26, 2008, Nuclear Management Company, LLC, submitted a request for Technical Specifications changes to allow the use of Westinghouse 0.422-inch outside diameter (OD) 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 October 6, 2008, 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,

/RA/

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-282 and 50-306

Enclosure:
Request for Additional Information
cc w/encl: Distribution via ListServ

DISTRIBUTION:
PUBLIC
RidsNRRPMTWengert Resource
RidsAcrsAcnw_MailCTR Resource
RidsRgn3MailCenter Resource
LPL3-1 R/F
RidsNrrLATHarris Resource
RidsOgcRp Resource
RidsNrrDorlDpr Resource
RidsNrrDorlLpl3-1 Resource
RidsNrrDssSrxb Resource
B. Parks, NRR

ADAMS Accession Number: ML082810645

| OFFICE | LPL3-1/PM | LPL3-1/LA | NRR/DSS/SRXB/BC | LPL3-1/BC |
|---|---|---|---|---|
| NAME | TWengert | THarris | GCranston | LJames |
| DATE | 10/14/08 | 10/9/08 | 10/14/08 | 10/17/08 |

OFFICIAL RECORD COPY