February 12, 2004

Mr. Joseph M. Solymossy
Site Vice President
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING PROPOSED AMENDMENT REQUEST FOR "SAFETY ANALYSES TRANSITION" (TAC NOS. MB8128 AND MB8129)

Dear Mr. Solymossy:

By application dated March 25, 2003, as supplemented by letters dated June 16, 2003, and January 14, 2004, the Nuclear Management Company, LLC (NMC), proposed to revise the Prairie Island Nuclear Generating Plant, Units 1 and 2, Technical Specifications to allow Westinghouse to perform many of the safety analyses that support operation of Prairie Island. These analyses include reactor core reload designs and are currently performed by NMC personnel. The Nuclear Regulatory Commission staff finds that the additional information identified in the enclosure is needed.

We emailed a request for additional information to Mr. R. Alexander (NMC) on February 5, 2004. We had a telephone discussion with L. Brown, et al. (Westinghouse), and your staff on February 2, 2004, to discuss the questions and to gain a mutual understanding. During a phone call on February 9, 2004, a mutually agreeable response date of February 25, 2004, was established.

Please contact me at (301) 415-4106 if future circumstances should require a change in the response date.

Sincerely,

/RA/

Anthony C. McMurtray, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosure: Request for Additional Information

cc w/encl: See next page

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Nuclear Management Company, LLC
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REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE REVIEW OF THE USE OF WESTINGHOUSE SAFETY ANALYSES
AND ASSOCIATED TECHNICAL SPECIFICATION CHANGES
PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2
DOCKET NOS. 50-282 AND 50-306

By application dated March 25, 2003, as supplemented by letters dated June 16, 2003, and January 14, 2004, the Nuclear Management Company, LLC (NMC), proposed to revise the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, Technical Specifications to allow Westinghouse to perform many of the safety analyses that support operation of Prairie Island. The Nuclear Regulatory Commission staff has the following questions related to the January 14, 2004, submittal:

The Table, Section, and page numbers refer to Attachment 5 of the submittal

1. Steam Generator Measurement Errors

   Table 5.1.2 (page 5-21) specifies the initial steam generator (SG) water level assumed in the non-loss-of-coolant accident (LOCA) analysis for Prairie Island. Table 5.1.5 (page 5-26) indicates that the low-low SG water level trip is credited in the analysis of the loss of normal feedwater (LONF) and loss of all AC power (LOAC) events to trip the reactor and to actuate auxiliary feedwater (AFW). Section 5.1.15.2 (page 5-274) indicates that the low SG wide range level signal is used in the anticipated transient without scram (ATWS) analysis to trip the reactor and the turbine, and actuate the AFW. However, insufficient information was provided regarding how the SG water level measurement errors are considered in the safety analysis.

   For the Westinghouse-manufactured SGs, Westinghouse issued a number of Nuclear Service Advisory Letters (NSALs) addressing the SG water level measurement errors. NSAL-02-3 and the revision to this NSAL deal with the uncertainty in the SG water level measurement caused by the placement of the mid-deck plate between the upper and lower pressure taps. This results in a delayed actuation of the SG water level low-low trip signal that trips the reactor and actuates the auxiliary feedwater. NSAL-02-4 deals with uncertainties in the SG water level measurement due to the void content of the two-phase mixture above the mid-deck plate that is not reflected in the SG water level setpoint calculation. This results in a premature actuation of the SG water level high-high trip signal that isolates the feedwater system. NSAL-02-5 deals with potential inaccuracies in the initial conditions assumed in safety analyses affected by SG water level. The safety analyses may not be bounding because the velocity head under some conditions may increase in the uncertainties in the SG water level control system. NSAL-03-09 indicates that Westinghouse has developed a program for the Westinghouse Owners Group that evaluates the effects on the SG water level control system uncertainties from various items. These items include the mid-deck plate, feedwater ring and feedwater ring supports, lower-deck plate supports, non-recoverable losses due to carryunder, decrease in subcooling due to carryunder, and transient conditions due to events such as the single-loop LONF, etc. Under the program, Westinghouse evaluated the design features of Westinghouse-designed SGs and other phenomena associated with Westinghouse SGs as they affect uncertainties with respect to the SG water level control system, and the SG water level low,

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low-low and high-high reactor trip functions. Westinghouse has documented the results of its program in WCAP-16115, “Steam Generator Level Uncertainties Program.” Westinghouse recommends that all licensees with plants using Westinghouse SGs review the WCAP-16115 results to determine the impact on plant-specific SG level uncertainty, in addition to the effects identified in NSALs 02-3, -4 and -5.

Please provide the following information:

(a.) Discuss how PINGP accounts for the applicable uncertainties documented in the Westinghouse NSALs and the guidance specified in WCAP-16115 in determining the initial SG water level and the SG water level (low and low-low) setpoints.

(b.) Provide information to demonstrate that the effects of water level uncertainties discussed in the NSALs and WCAP-16115 are appropriately considered in the analyses of the non-LOCA transients and ATWS presented in Attachment 5.

2. <u>Reactor Coolant Pump Coastdown Delay Times</u>

Section 5.1.1.2.2 (page 5-195) and 5.1.15.5 (page 5-292) indicate that in the analysis of the LOAC event, the reactor coolant pumps (RCPs) are assumed to lose power and begin coasting down two seconds after the reactor trip. Assumption 2 on pages 5-208 and 5-209 for the steam line break (SLB) analysis indicates that the RCPs are assumed to begin coasting down three seconds after the SLB initiation for the case without offsite power.

(a.) Please justify the adequacy of the RCP coastdown delay times assumed in the analysis of the LOAC and SLB events in predicting a minimum departure from nucleate boiling ratio (DNBR), the maximum pressurizer pressure and the minimum margin to overfill the pressurizer.

3. <u>Anticipated Transient Without Scram (ATWS) Analysis</u>

(a.) List the values of the moderator temperature coefficients (MTCs) assumed in each of the analyzed ATWS cases listed in Table 5.1.15-1 (page 5-273), and show that the values of MTC used in the ATWS analysis bound at least 95 percent of the fuel cycle time.

(b.) Figures 5.1.15.4-1 through 5.1.15.4-7 (pages 5-284 through 5-290) show the plant responses following an ATWS-LONF accident. Please provide a table listing the sequence of events for the ATWS-LONF accident and explain the causes of the changes in reactor coolant system temperatures (Figure 5.1.15.4-2 (page 5-285)), pressurizer and SG pressures (Figures 5.1.15.4-3 and -6 (pages 5-286 and -289)), and pressurizer and SG water volumes (Figure 5.1.15.4-4 and -7 (pages 5-287 and -290)).

cc:

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