November 26, 2003

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 June 16, 2003, 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 (NRC) staff finds that the additional information identified in the enclosure is needed.

We emailed two separate requests for additional information to Mr. D. Vincent (NMC) on August 6 and August 11, 2003. We had telephone discussions with R. Creighton (Westinghouse), L. Brown, et al. (Westinghouse), and your staff on August 11, 2003, to discuss the questions and to gain a mutual understanding. During a phone call on November 25, 2003, a mutually agreeable response date of January 23, 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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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 June 16, 2003, 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 NRC staff has the following questions related to these submittals:

1. Regarding the proposed use of Westinghouse safety analyses to support safe operation of the Prairie Island Nuclear Generating Plant (PINGP), provide a list of the titles of the topical reports that document the methodologies (including empirical correlations such as the critical heat flux correlation with its safety limit) and computer codes used in the conversion to Westinghouse safety analyses for supporting the March 25, 2003, application. For each listed report, (1) discuss the purposes of the methodologies and computer codes documented in the report; (2) list the associated NRC acceptance letter and any safety evaluation approving the use of the report previously issued to the PINGP; (3) address the compliance with each of the limitations imposed on use of the report; and (4) identify changes to the NRC-approved report and address acceptability of the changes.

2. Regarding the PINGP-specific analyses, for each event analyzed with the Westinghouse computer codes, provide the results of the transient analysis and demonstrate that the results meet the applicable acceptance criteria. For each transient analyzed, (1) describe the nodalization scheme and demonstrate its adequacy; (2) list the models and empirical correlations specified in the computer codes as “options” that are selected for the transient analysis, and demonstrate that the selected models and correlations are adequate; (3) discuss how the computer codes are used related to iteration between the codes; (4) discuss the plant initial conditions and values of plant parameters (specific to fuel design, nuclear physics and thermal-hydraulics characteristics) assumed in the transient analysis, and also justify that the values used are adequate and result in a worst case.

3. Add to Technical Specification 5.6.5.b the titles of the topical reports that document applicable methodologies and computer codes used in the conversion to the Westinghouse safety analyses.

4. On Page 4 of Exhibit A of the March 25, 2003, application, the $f(\Delta I)$ function is proposed to be removed from the OPΔT equation in Table 3.3.1-1 (page 8). This proposed change appears to be inconsistent with NUREG-1431, Revision 2, “Standard Technical Specifications - Westinghouse Plants.” Table 3.3.1-1 (page 6 of 6) of the NUREG indicates that the $f(\Delta I)$ function in the OPΔT equation is a constant and can be specified in the core operating limits report (COLR). Also, page A-20 of WCAP-10216-A, Revision 1A, “Relaxation of Constant Axial Offset Control,” indicates that current 17x17 plants with the constant axial offset control (CAOC) operate based on an analysis without OPΔT $f(\Delta I)$ function since OTΔT $f(\Delta I)$ function is more restrictive. If the need for an OPΔT $f(\Delta I)$ function is indicated by the relaxed axial offset control (RAOC) analysis, the OTΔT $f(\Delta I)$ function should be changed such that it should be more restrictive. Provide information to

demonstrate that it is adequate to remove the f(ΔI) function from the OPΔT equation in the conversion process from the CAOC to the ROAC.

Additionally, Westinghouse has verified, via “plant-specific analyses” for several 14x14 and 15x15 plants, that an OPΔT f(ΔI) function can be set to zero. Provide information to confirm that the “plant-specific analyses” are applicable to PINGP with the proposed RAOC operation.

5. On Page 5 of Exhibit A of the March 25, 2003, application, the allowable value of the pressurizer pressure-low reactor trip is proposed to be changed from 1760 psig to 1845 psig. Provide the rationale for the change. Specify the actual plant pressurizer pressure-low reactor trip setting and provide analyses which verify that with the increased allowable value, from 1760 psig to 1845 psig, the actual plant setting, with consideration of the instrumentation uncertainties, remains unchanged. Also, identify the analytical value used in the non-loss-of-coolant-accident analysis with the Westinghouse methodologies and computer codes and provide the analysis which shows that this analytical value is lower than the proposed increased allowable value of 1845 psig.

6. On Page 5 of Exhibit A of the March 25, 2003, application, it is indicated that the sensitivity factors, operating parameters, and VIPRE model used in the safety analysis do not differ significantly from those used in WCAP-11397. Provide the results of the analysis that determines the design limit departure from nucleate (DNBR) using the revised thermal design procedure. The information should include a table similar to Table 3-1 of WCAP-11397, including values of the DNBR design limit, operating parameters, measurement uncertainties, and sensitivity factors.

cc:

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