April 23, 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, UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE APPLICATION OF LEAK-BEFORE-BREAK TECHNOLOGY TO REPLACEMENT STEAM GENERATOR DESIGN (TAC NO. MB7509)

Dear Mr. Solymossy:

By letter dated January 29, 2003, the Nuclear Management Company, LLC (NMC), submitted a letter regarding the application of leak-before-break (LBB) technology to the replacement steam generator design for the Prairie Island Generating Plant, Unit 1. The Nuclear Regulatory Commission (NRC) staff finds that the additional information identified in the enclosure is needed.

A draft of the request for additional information was e-mailed to Mr. T. Higgins (NMC) on March 7, 2003. A phone call was held between K. Albright (NMC), R. Pearson (NMC), O. Nelson (NMC), M. Baker (Framatone), C. McCauley (Framatone), J. Wu (NRC), and myself on March 28, 2003, to discuss the questions and to gain a mutual understanding. Also, the phone call established a mutually agreeable response date of 45 days from the date of this letter.

Please contact me at (301) 415-1446 if future circumstances should require a change in this response date.

Sincerely,

/RA/

John G. Lamb, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-282

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE APPLICATION OF LEAK-BEFORE-BREAK

TECHNOLOGY TO STEAM GENERATOR REPLACEMENT DESIGN

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

DOCKET NO. 50-282

By letter dated January 29, 2003, the Nuclear Management Company, LLC (NMC), submitted a letter regarding the application of leak-before-break (LBB) technology to the replacement steam generator design for the Prairie Island Generating Plant, Unit 1. The NRC staff has the following questions:

1. Does "reanalysis" mean: (a) reactor coolant loop (RCL) reanalysis and requalification? OR (b) leak-before-break (LBB) re-demonstration by reevaluation or validation of the original evaluation?

2. Does NMC plan to submit results of 1(b) above for the NRC staff's review and approval whether new LBB loads are enveloped by the original LBB loads, or LBB is reevaluated and re-demonstrated?

3. Does NMC plan to make a statement in response to (2) above that there are no new breaks due to cutting and welding of the steam generator replacement project?

4. The submittal stated that the RCL will be designed for the postulated breaks in the largest branch lines (surge, residual heat removal, safety injection system/accumulator injection). Are postulated breaks in the main feedwater and main steam lines going to be reanalyzed? Will they be reanalyzed for the same break locations and conditions?

5. Are there any support changes associated with the steam generator replacement project?

6. Relating to the reactor coolant system (RCS) reanalysis, question 1(a) above, for thermal and seismic conditions, are there any stiffness changes which may impact the natural frequencies of the RCS and structural responses at controlling locations for LBB demonstration? Please explain how these impacts are addressed in question 1(b) above.

7. How does NMC plan to document the RCL reanalyses associated with the replacement steam generator? Will the reanalyses be governed by the same design specifications as the original steam generator?

ENCLOSURE

cc:

Jonathan Rogoff, Esquire
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