March 21, 2003

Mr. Mano Nazar
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 - FINAL CLOSEOUT OF RESPONSES TO GENERIC LETTER 96-06 (TAC NOS. M96854 AND M96855)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission (NRC) staff issued Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," on September 30, 1996. In GL 96-06, the NRC staff requested licensees to determine for postulated accident conditions, if (1) containment air cooler cooling water systems are susceptible to either waterhammer or two-phase flow conditions, and (2) piping systems that penetrate containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

The Nuclear Management Company, LLC (NMC), the licensee, previously the Northern States Power Company, provided its assessment for the Prairie Island Nuclear Generating Plant, Units 1 and 2, in a letter dated January 28, 1997, and additional information was provided in letters dated September 15, 1997, May 15, 1998, January 8, 1999, September 30, 2002, and February 4, 2003.

This letter addresses the final closeout of GL 96-06 and is divided into two parts. The first part is the issue of waterhammer and two-phase flow. The second part is the thermal overpressurization issue.

In reference to the waterhammer and two-phase flow issue, it is the NRC staff's understanding that you adhered to the analytical methodology that was established by the Electric Power Research Institute (EPRI) for evaluating the GL 96-06 waterhammer issue. This EPRI methodology is documented in EPRI Technical Reports 1003098 and 1006456 (previously known as EPRI Report TR-113594), and found acceptable by the NRC in an evaluation dated April 3, 2002. Your September 30, 2002, response provided the information required by Section 3.3 of the NRC evaluation, and we understand that the Prairie Island Nuclear Generating Plant, Units 1 and 2, are not vulnerable to the waterhammer and two phase flow concerns discussed in GL 96-06. With respect to the two phase flow issue, it is the NRC staff's understanding that a number of modifications (some of which were initiated prior to issuance of GL 96-06) have been made to the Cooling Water system focused primarily on reducing or eliminating potential flow diversion paths, thereby enhancing system response during an accident. The NRC staff is satisfied with your response and consider the waterhammer and two-phase flow elements of GL 96-06 to be closed.

In reference to the thermal overpressurization issue, you summarized your review of fluid systems that are susceptible to overpressurization due to thermal expansion of internal fluid in the submittal dated January 28, 1997. You concluded that all except certain pipe segments associated with the reactor coolant pump seal water (RCPSW) return line, the sample lines, and the safety injection (SI) test line were dispositioned based on high operating temperature of the internal fluid, or the presence of certain pressure relief devices such as thermal relief valves, diaphragm valves, expansion bellows, check valves, and containment venting. You performed detailed evaluation of these affected pipe segments associated with the RCPSW return line, the sample lines, and the SI test line, and determined that their calculated internal pressures did not exceed the design basis maximum allowable pressure for these piping. However, in your submittal dated January 8, 1999, you indicated that a more detailed analysis was performed for the SI test line, and the newly calculated internal pressure exceeded the maximum allowable values. As a result, you proposed to implement corrective actions to alleviate the potential overpressure problem for the SI test line by the end of each unit's next refueling outage, which occurred in April 1999, for Unit 1, and May 2000, for Unit 2. The NRC staff finds the corrective action acceptable.

Installation of a check valve, a relief valve, a diaphragm valve, an expansion bellow or applicable pressure relief devices, is an acceptable means for relieving pressure of a solid water volume. The NRC staff also finds that verifying the calculated internal pressures to be within the Code allowable pressure is an acceptable means to demonstrate the operability of affected piping segments. Therefore, the NRC staff concludes that your corrective actions, proposed plant modifications, and verifying the calculated internal pressures to be within the Code allowable provide an acceptable resolution for the issue of thermally-induced pressurization of piping runs penetrating the containment, and the closed piping segments inside the containment. The NRC staff is satisfied with your response and consider the thermally-induced pressurization element of GL 96-06 to be closed.

Please be advised that the NRC staff has not performed a detailed review of your waterhammer analysis, thermally-induced pressurization analysis, or the associated modifications that were made. They are subject to future NRC audit or inspection activity if any.

This completes the NRC staff's efforts regarding GL 96-06 for Prairie Island Nuclear Generating Plant, Units 1 and 2, under TAC Nos. M96854 and M96855.

If you have any questions regarding this matter, please contact me at (301) 415-1446.

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 and 50-306

cc: See next page

In reference to the thermal overpressurization issue, you summarized your review of fluid systems that are susceptible to overpressurization due to thermal expansion of internal fluid in the submittal dated January 28, 1997. You concluded that all except certain pipe segments associated with the reactor coolant pump seal water (RCPSW) return line, the sample lines, and the safety injection (SI) test line were dispositioned based on high operating temperature of the internal fluid, or the presence of certain pressure relief devices such as thermal relief valves, diaphragm valves, expansion bellows, check valves, and containment venting. You performed detailed evaluation of these affected pipe segments associated with the RCPSW return line, the sample lines, and the SI test line, and determined that their calculated internal pressures did not exceed the design basis maximum allowable pressure for these piping. However, in your submittal dated January 8, 1999, you indicated that a more detailed analysis was performed for the SI test line, and the newly calculated internal pressure exceeded the maximum allowable values. As a result, you proposed to implement corrective actions to alleviate the potential overpressure problem for the SI test line by the end of each unit's next refueling outage, which occurred in April 1999, for Unit 1, and May 2000, for Unit 2. The NRC staff finds the corrective action acceptable.

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Please be advised that the NRC staff has not performed a detailed review of your waterhammer analysis, thermally-induced pressurization analysis, or the associated modifications that were made. They are subject to future NRC audit or inspection activity if any.

This completes the NRC staff's efforts regarding GL 96-06 for Prairie Island Nuclear Generating Plant, Units 1 and 2, under TAC Nos. M96854 and M96855.

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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
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