May 18, 2000

Mr. Michael F. Hammer
Site General Manager
Monticello Nuclear Generating Plant
Northern States Power Company
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - COMPLETION OF LICENSING ACTION FOR GENERIC LETTER (GL) 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS" (TAC NO. M96835)

Dear Mr. Hammer:

The NRC staff issued GL 96-06 on September 30, 1996, to all holders of operating licenses for nuclear power reactors, except for those licenses that have been amended to possession-only status. GL 96-06 requested information from licensees related to two concerns: (1) water hammer and two-phase flow in the cooling water systems that serve the containment air coolers and (2) thermally-induced overpressurization of isolated water-filled piping sections in containment. On November 13, 1997, the staff issued Supplement 1 to GL 96-06 informing licensees about ongoing efforts and new developments associated with GL 96-06 and providing additional guidance for completing corrective actions. You responded in letters dated October 25, 1996, January 28 and October 30, 1997, February 2, May 6 and August 5, 1998, August 12, 1999, and April 12, 2000. The results of the NRC staff's review of your responses to GL 96-06 follow.

Water Hammer and Two-Phase Flow

You provided your assessment of the water hammer and two-phase flow issues in a letter dated January 28, 1997, and additional information was submitted in a letter dated August 5, 1998. Based on the staff's review of the information provided, it is the staff's understanding that: (a) the drywell coolers are not required for accident mitigation, and (b) procedures will be revised to require reactor building closed cooling water system flow to the drywell coolers to be isolated during the event scenarios of interest if there is a possibility that water hammer could occur when containment integrity is required. The staff considers the GL 96-06 issues of water hammer and two-phase flow at Monticello are closed.

Thermally Induced Overpressurization

In your response of January 28, 1997, you identified 13 penetrations as potentially vulnerable to a water solid volume that may be subjected to an increase in pressure due to heating of the trapped fluid. You determined that one of the lines (main steam drain line) is not susceptible to thermally induced overpressurization since it is kept hot during normal operation and will cool when the steam is isolated. You also stated that stresses in two of the pipe lines are within the design-basis American Society of Mechanical Engineers (ASME) Code-allowable limits. You determined that three of the penetrations are operable based on the criteria of Appendix F to Section III of the ASME Code and the other three are operable based on administrative controls requiring the line between isolation valves be drained. For your long term corrective action, you revised the operating procedure to ensure that these lines are isolated during operations and the pipe line between the isolation valves is drained. The remaining penetrations were determined to be acceptable based on design-basis analysis.

In response to the staff's request for additional information of September 16, 1997, you submitted a letter dated October 30, 1997, that provided evaluation summaries of all affected penetrations; committed to install pressure relief devices on four penetrations during the March 1988 refueling outage; and committed to increase the margin between postulated pressure and the design pressure for three penetrations. Your letter of February 2, 1998, revised your commitment and stated that you would install pressure relieving devices prior to startup from the 1998 refueling outage, with the exception of the residual heat removal (RHR) shutdown cooling line and the instrument air/alternate $N_2$ system, which will be resolved during the 1999 refueling outage. In your letter of May 6, 1998, you confirmed that you had installed pressure relief devices on three penetrations during the 1998 refueling outage and the issue for the remaining four penetrations would be resolved during the 1999 refueling outage.

Based on your further review, you once again revised your commitment to resolve the issue for the four penetrations. Your letter of August 12, 1999, stated that the RHR shutdown cooling inboard isolation valve would be replaced with a new double disc gate valve that will relieve overpressure through a hole in the vessel side valve disc. The system isolation between the shutdown cooling inboard isolation valve and the reactor vessel consists of a single manual valve. In order to lower overall modification risk, you committed to install this modification during the May 2001 refueling outage when offloading of the full core is planned. The RHR shutdown cooling line in its current configuration meets the ASME Section III, Appendix F operability criteria. For the remaining three penetrations (instrument air/alternate $N_2$ system), you committed to install higher pressure-rated devices or upgrade certification of existing devices prior to startup from the January 2000 refueling outage. Your letter dated April 12, 2000, stated that the instrument air/alternate $N_2$ system penetration modifications had been completed during the January 2000 refueling outage. In addition, you informed the staff that you had determined that a more appropriate resolution to the RHR line is to install a bypass line with a check valve to allow flow from between the shutdown cooling isolation valves around the inboard isolation valve and back to the primary system. You committed to install this modification during the 2001 refueling outage. The staff finds your evaluation and corrective action reasonable and acceptable and concludes that the GL 96-06 issue of thermally induced pressurization of piping runs penetrating the containment at Monticello is closed.

Finally, the staff concludes that all requested information has been provided; therefore, we consider GL 96-06 to be closed for your facility.

Sincerely,

***/RA/***

Carl F. Lyon, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-263

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Finally, the staff concludes that all requested information has been provided; therefore, we consider GL 96-06 to be closed for your facility.

Sincerely,

***/RA/***

Carl F. Lyon, Project Manager, Section 1
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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-263

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