March 25, 2004

Mr. Thomas J. Palmisano
Site Vice President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT — THIRD 10-YEAR INTERVAL INSERVICE INSPECTION (ISI) REQUEST FOR RELIEF RR-17 (TAC NO. MC0593)

Dear Mr. Palmisano:

The Nuclear Management Company, LLC's (NMC's) letter of August 27, 2003, requested a one-time relief from the requirements of the 1986 Edition of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code) Section XI at Monticello Nuclear Generating Plant (MNGP). NMC's relief request involved a repair/replacement activity on the topworks of main steam safety relief valve (SRV) "G."

The Nuclear Regulatory Commission (NRC) staff evaluated NMC's request and concludes that compliance with the ASME Code-required system pressure test following the repair/replacement activity of the main steam SRV "G" topworks would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The NRC staff further finds that the proposed alternative provides reasonable assurance of the structural integrity of the component. Therefore, the NRC staff authorizes NMC's proposed alternative on a one-time basis pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year ISI interval at MNGP. Enclosed is our safety evaluation.

Sincerely,

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

THIRD 10-YEAR INTERVAL INSERVICE TESTING PROGRAM

RELIEF REQUEST NO. 17

NUCLEAR MANAGEMENT COMPANY, LLC

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

The Nuclear Management Company, LLC's (NMC's) letter of December 6, 2002 (Relief Request No. 7), requested Nuclear Regulatory Commission (NRC) approval of an alternative to allow NMC to use the 2001 edition of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, "Rules for Inservice Inspection [ISI] of Nuclear Power Plant Components," for repair and replacement activities for the fourth 10-year interval of the Monticello Nuclear Generating Plant (MNGP) ISI Program.

NMC's letter of January 23, 2003, said that MNGP's third 10-year ISI interval would expire on May 31, 2003, and that its fourth ISI 10-year interval would begin on May 1, 2003, thus creating overlapping intervals. However, NMC did not indicate which interval would apply to its ISI Repair/Replacement Program (RRP).

During startup testing on May 24, 2003, following a refueling outage, NMC identified that the temperature of the discharge tailpipe on main steam safety relief valve (SRV) "G" was higher than normal. This indicated that the SRV was leaking. NMC decided to replace the SRV "G" topworks assembly in order to correct the leakage. On May 25, 2003, NMC submitted a letter to the NRC requesting approval of Relief Request No. 7 insofar as it applied to the bolted connections of SRV "G," in lieu of the then current MNGP RRP. The NRC verbally granted a one-time relief for SRV "G" on the same day, based on the understanding that MNGP was in the fourth ISI 10-year interval. During the course of a conference call between the NRC and NMC held on May 29, 2003, it became apparent that there was a mis-communication between the NRC and NMC regarding which ISI interval the relief applied to. Relief Request No. 7 was written for the fourth 10 -year interval, but the repair/replacement activity was performed under the third 10-year interval RRP. Based on discussions with NMC personnel on May 25 and May 29, 2003, the NRC staff asked NMC to provide a relief request to clearly document that the relief requested on May 25, 2003, applied to the third 10-year ISI interval for repair/replacement of a bolted connection. Accordingly, NMC's letter of August 27, 2003, asked for a one-time authorization to perform the proposed alternative test in accordance with 10 CFR 50.55a(a)(3)(ii). This safety evaluation assesses that request. The NRC staff reviewed NMC's

ENCLOSURE

relief request of December 6, 2002, and authorized NMC's proposed alternative in its letter of October 3, as corrected on December 31, 2003.

2.0 REGULATORY EVALUATION

The ISI of ASME Code Class 1, Class 2, and Class 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states in part that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable ASME Code of record for the third 10-year ISI for MNGP is the 1986 Edition of the ASME Code, Section XI. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

3.0 TECHNICAL EVALUATION

3.1 Component for Which Relief Is Requested

ASME Section XI, Class 1, Table IWB-2500-1 Examination Category B-P, Main Steam SRV "G."

3.2 ASME Code Requirements

The code requirements for the applicable item is given in subarticles IWA-5214(e) and IWB-5221, of ASME Section XI, 1986 edition.

IWA-5214(e) states that "if only disassembly and reassembly of mechanical joints of a component are involved, a system pressure test of IWA-5211(a), (b), or (c) shall be acceptable in lieu of the system hydrostatic test."

IWB-5221 states that "... the system leakage test shall be conducted at a test pressure not less than the nominal operating pressure associated with 100 percent rated operating power."

3.3 <u>NMC's Proposed Alternative</u>

Following the repair of the Main Steam SRV "G," a VT-2 visual inspection for leakage was performed during start-up at 150 psi and 900 psi.

3.4 <u>NMC's Basis for Relief</u> (as stated):

> The Class 1 System Leakage Test required by Table IWB-2500-1, Category B-P had already been completed for the outage. The indicated leakage of the SRV was discovered subsequent to the Class 1 System Leakage Test and therefore, to meet the requirements as specified in IWA-5214(e), another Class 1 System Leakage Test would be required for this one mechanical connection along with a VT-2 examination. This test and examination would have required the reactor pressure vessel to be filled with coolant and the steamlines flooded to provide a water-solid condition.
>
> Extensive valve manipulations, system lineups, and procedural controls would have been required in order to heat up and pressurize the primary system to establish the necessary test pressure during plant outage conditions. The additional valve lineups and system reconfigurations necessary to support this test would have imposed an additional challenge to the affected systems. A normal plant startup would then occur, after completion and subsequent recovery from the test procedure.
>
> The required heatup and cooldown during the performance of the pressure test would have added a thermal cycle(s) to various components within the scope of the thermal fatigue-monitoring program. Furthermore, this evolution would have placed the primary system in a condition where it was more susceptible to Low Temperature Over Pressure events.
>
> Pursuant to 10 CFR 50.55a(a)(3)(ii), compliance with the specified requirements of the Code noted above and 10 CFR 50.55a(g) would have resulted in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.5 <u>NRC Staff's Evaluation</u>

The ASME Code requires that, after repair/replacement activities of the main steam SRV "G" topworks, a system pressure test be conducted at a pressure not less than the nominal operating pressure associated with 100 percent rated reactor power. At MNGP, this corresponds to 1000 psig. However, performing the test as required by the ASME Code would have resulted in hardship. The repair/replacement activity associated with main steam SRV "G" occurred after the Class 1 system leakage test had been completed for the outage. Performing the required pressure test would have required filling the reactor pressure vessel with water and flooding the steam lines to provide a water-solid condition. It would also have required heating-up and pressurizing the primary system to establish the necessary test pressure during plant outage conditions. The additional activities would add a thermal cycle to various components within the scope of the thermal fatigue-monitoring program.

NMC's proposed alternative included a VT-2 visual inspection for leakage during start-up when the system was being pressurized. The valve assembly was first inspected when pressurized to 150 psig and then again at 900 psig. This test reasonably demonstrated the structural integrity of the bolted connection. The NRC staff has determined that for this situation, a visual examination of the component at 900 psig satisfied the ASME Code requirement, which is to detect leakage and to assure structural integrity after the reassembly of the bolted connection. If the reassembled valve was going to leak at the nominal operating pressure, it would also likely leak at 900 psig, although at a lower rate.

According to NMC, the drywell monitoring systems would detect leakage that would occur in the valve at higher pressures associated with nominal reactor power. These systems include drywell pressure monitoring, the containment atmosphere monitoring system, and the drywell floor drain sumps. The NRC staff agrees that monitoring such leakage provides additional assurance of the integrity of the component.

Based on the above evaluation, the NRC staff finds that performing the alternative examination provides reasonable assurance of the leakage and structural integrity of the valve, and that compliance with the ASME Code-specified requirements would result in hardship without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

The NRC staff concludes that complying with the ASME Code-required system pressure test following the repair/replacement activity of the main steam SRV "G" topworks would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The NRC staff further concludes that the proposed alternative provides reasonable assurance of the structural integrity of the component. Therefore, NMC's request is authorized on a one-time basis pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year ISI interval at MNGP.

All other ASME Code Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: B. Fu

Date: March 25, 2004