June 13, 2003

Mr. David L. Wilson
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 — ONE-TIME INSERVICE INSPECTION PROGRAM PLAN RELIEF REQUEST NO. 8 FOR LEAK TESTING THE “B” AND “G” MAIN STEAM SAFETY RELIEF VALVES (TAC NO. MB9538)

Dear Mr. Wilson:

The Nuclear Management Company’s, LLC (NMC’s), letter of June 12, 2003, submitted Relief Request No. 8 requesting relief from certain requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components.” The request applied to the fourth 10-year interval of the inservice inspection (ISI) examination plan for the Monticello Nuclear Generating Plant.

NMC is replacing the “B” and “G” main steam safety relief valve (SRV) assemblies at Monticello during its current planned maintenance outage to correct bypass leakage. In Relief Request No. 8, NMC requested a one-time relief from having to perform a system leakage test on the SRVs at a nominal operating pressure of about 1000 psig, as required by paragraph IWA-5211(a) of the ASME Code. Instead, NMC proposed performing the test and accompanying visual examination while operating at a minimum of 900 psig during the normal plant startup sequence. NMC requested relief pursuant to 10 CFR 50.55a(a)(3)(ii).

The NRC staff evaluated Relief Request No. 8 and concludes that NMC's proposed alternative of performing a one-time system leakage test and VT-2 examination of the SRV assemblies, while operating at a minimum of 900 psig during the normal plant start-up sequence, provides reasonable assurance of structural integrity of the SRV bolted connections. Furthermore, the NRC staff concludes that complying with the ASME Code requirement would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Therefore, the NRC staff authorizes NMC's proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii) for one-time use during normal startup from the current planned maintenance outage to replace the SRVs.

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

Therefore, the NRC staff authorizes NMC's proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii) for one-time use during normal startup from the current planned maintenance outage to replace the SRVs.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ONE-TIME INSERVICE INSPECTION PROGRAM PLAN

RELIEF REQUEST NO. 8 FOR LEAK TESTING THE “B” AND “G”

MAIN STEAM SAFETY RELIEF VALVES

NUCLEAR MANAGEMENT COMPANY, LLC

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

The Nuclear Management Company’s, LLC (NMC’s), letter of June 12, 2003, submitted Relief Request No. 8 requesting relief from certain requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components.” The request pertained to system leakage testing and VT-2 (visual) examination of the mechanical joints (bolted connections) on replacement “B” and “G” main steam safety relief valve (SRV) assemblies at the Monticello Nuclear Generating Plant. Relief Request No. 8 is for one-time use during normal startup activities from the current plant maintenance outage to replace SRVs at Monticello.

NMC is replacing the B and G main steam SRV assemblies at Monticello to correct bypass leakage. The 1995 edition of the ASME Code, Section XI, with the 1996 addenda, requires NMC to perform a system leakage test and VT-2 examination of the mechanical joints at a nominal operating pressure of about 1000 pounds per square inch gauge (psig) after replacing the SRVs.

In Relief Request No. 8, NMC requested a one-time relief from having to perform the system leakage test and VT-2 examination at nominal operating pressure (or when pressurized to nominal operating pressure and temperature) as required by paragraph IWA-5211(a) of the ASME Code. Instead, NMC proposed doing the test and accompanying VT-2 examination at a minimum of 900 psig during the normal plant startup following SRV repair/replacement activities. NMC requested relief pursuant to 10 CFR 50.55a(a)(3)(ii) stating that compliance with the required system leakage test under IWA-4540(c) would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Enclosure

2.0 REGULATORY EVALUATION

ISI of nuclear power plant components is performed in accordance with the ASME Code, Section XI, and applicable 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). The regulation at 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (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 ASME Code of record for Monticello for the fourth 10-year ISI interval is the 1995 edition through 1996 addenda.

3.0 TECHNICAL EVALUATION

3.1 Code Requirement

ASME Code, Section XI, paragraph IWA-5120(a) states that items subjected to repair/replacement activities shall be pressure-tested when required by paragraph IWA-4500. Paragraph IWA-4540(c) specifies that mechanical joints made in installation of pressure retaining items shall be pressure-tested in accordance with paragraph IWA-5211(a). Paragraph IWA-5211(a) stipulates conducting a system leakage test during operation at nominal operating pressure (or when pressurized to nominal operating pressure and temperature). In addition, paragraph IWB-5210(b) indicates that “system pressure tests and visual examinations shall be conducted in accordance with IWA-5000 and this Article. The contained fluid in the system shall serve as the pressurizing medium.”

3.2 Licensee’s Code Relief

NMC requested a one-time relief from having to perform a system pressure test and VT-2 examination at a nominal operating pressure of about 1000 psig (or when pressurized to nominal operating pressure and temperature) following repair/replacement activities performed on the B and G main steam SRVs during the current planned maintenance outage. Instead, NMC proposed doing the system leakage test and accompanying VT-2 examination (after a 10-minute hold time) with the subject valves uninsulated, while operating at a minimum of 900 psig during the normal plant startup from the current maintenance outage.

3.3 Identification of Components

Main steam safety relief valve assemblies, ASME Code Class 1, IWA-4540(c) and IWA-5211(a).

3.4 Licensee's Basis for Requesting Relief (as stated)

Nuclear Management Company, LLC (NMC) Monticello Nuclear Generating Plant (MNGP) recently completed a refueling outage on May 26, 2003. During the refueling outage, MNGP completed the system leakage test required by American Society of Mechanical Engineers (ASME) Section XI, Table IWB-2500-1, Category B-P, Item 15.10 and 10 CFR Part 50 Appendix G, Section IV.A.2.d. Following restart of the unit, the "B" and "G" main steam safety relief valve assemblies (SRVs) have indicated leakage, as determined by higher than normal temperatures in their respective discharge tailpipes.

MNGP has decided to conduct a planned unit shutdown and enter a maintenance outage to replace the affected SRV assemblies. The SRV assemblies are connected to the main steam piping with a bolted, mechanical joint. Replacing them for maintenance is considered a Repair-Replacement activity under the rules of ASME Section XI, 1995 Edition with the 1996 Addenda which is the current code of record for the 4th 10-Year ISI Interval. Following repair-replacement, a system leakage test is required by IWA-4540(c). The system leakage test at the nominal pressure associated with the reactor at 100% power would be approximately 1000 psig.

MNGP has identified three methods for performing the system leakage test on the mechanical joints associated with the repair-replacement activity that meet the requirements identified above. Several conditions associated with such testing represent an imposition on personnel safety, personnel radiation exposure, and challenges to the normal mode and manner of equipment operation.

Method No. 1 would perform the pressure test and VT-2 exam during normal startup procedures. During normal startup with normal power ascension, nominal operating pressure of 1000 psig is reached at a reactor power level of approximately 75%. If access to containment were permitted at this power level, personnel would be exposed to excessive radiation levels, including significant exposure to neutron radiation fields, which is contrary to current station ALARA [as low as reasonably achievable] practices.

Establishing the 1000 psig test condition at a more moderate power level (e.g. during plant startup at approximately 7% reactor power) and in the manner needed to address radiation concerns would require altering the normal operational mode of the steam pressure control system.

During the performance of plant startup procedures, the electric and mechanical pressure regulator (EPR and MPR) set points are established within their normal operational ranges (approximately 918 psig). Their primary function is to regulate the main steam system pressures as sensed near the inlet of the high-pressure turbine. Reactor pressure control at the nominal 1000 psig is achieved at higher reactor power levels as a function of the pressure control system and the induced differential pressure across the main steam isolation valves and main steam piping.

While it is technically feasible to manipulate these controls to establish the nominal system pressure of 1000 psig at lower power levels, doing so will affect core reactivity and could challenge plant safety systems, such as the reactor protection system (RPS). MNGP has not previously operated the EPR and MPR in this manner. Changing the set points outside of the normal range of operation for the purpose of performing this test at nominal operating pressure poses several operational challenges. The lack of experience and predictability of setting pressure regulators outside the normal range of operation could adversely impact personnel and reactor safety.

Method No. 2 implements the use of the reactor pressure boundary leakage test which meets the requirements of Table IWB-2500-1, Category B-P, Item 15.10: the reactor pressure vessel (RPV) is filled with coolant and the steam lines are flooded to provide a water-solid condition. Use of this method would result in multiple operational challenges.

During a maintenance outage, pressurization for the test would be provided by decay heat and the reactor recirculation pumps. To support the pressurization evolution, the normal decay heat removal system, residual heat removal (RHR) shutdown cooling, would be required to be removed from service and isolated from the vessel to be pressurized. This system is not designed to withstand pressures greater than 185 psig. Thus, the remaining system available for decay heat removal is the reactor water cleanup system (RWCU).

Application of ANSI /ANS-1994 decay heat code results in a significant level of decay heat load. The ratio of decay heat input versus the heat removal capacity provided by RWCU is approximately 4:1. Therefore, the decay heat generated by the reactor core will surpass the capacity of RWCU. The heat up rate of the vessel water will cause the temperatures to surpass 212°F prior to the initiation of the inspections.

Method No. 2 would present several operational challenges. The pressure increase would be obtained by balancing the flow into the vessel, which is provided by the control rod drive (CRD) system, with the flow out of the vessel provided by the RWCU system via the dump flow control valve and flow controller. This is the method used during refueling outages to complete the RPV system leakage test. A failure of a non-safety related component, such as the dump valve or flow controller, would cause the interruption of dump flow and would cause the RPV pressure to increase. The RPV pressure would increase until operator action would require the operating CRD pump to be tripped.

Due to the amount of decay heat being generated and the RWCU systems heat removal capacity, it is questionable whether the RPV would depressurize and may in fact continue to pressurize until further operator action would be required to depressurize the RPV. Operator actions may include one or more of the following: re-establishing RWCU dump flow; if the failure mechanism was no longer present, opening the main steam line drain valves, SRVs, or head vent line. Any of the last 3 of these actions would probably cause a rapid depressurization transient on the RPV.

Extensive valve manipulations, system lineups, and procedural controls are required in order to heat up and pressurize the primary system to establish the necessary test

pressure, during plant outage conditions, without the withdrawal of control rods. This test is expected to take approximately 1 day of outage time, and the additional valve lineups and system reconfigurations necessary to support this test impose an additional challenge to the affected systems. A normal plant startup then occurs, after completion and subsequent recover from the test procedure.

Method No. 3 would maintain the RPV at its normal level and use decay heat to produce sufficient steam pressure to conduct the test at nominal operating temperature.

At the projected time of shutdown for the maintenance outage, MNGP will have a runtime of approximately three weeks since startup from the Cycle 21 refueling outage. The maintenance of the SRV assemblies is projected to be completed within approximately 50 hours after plant shutdown. While the decay heat load is too high for the water-solid method discussed above, there is not sufficient decay heat available to perform the test within a reasonable time period to support completion of the maintenance outage. It would require a minimum of 25 hours to reach the pressure of 1000 psig needed to perform the test required by the Code based upon decay heat projections.

Each of the methods discussed above presents a hardship or unusual difficulty to NMC [without a compensating increase in the level of quality and safety].

3.5 Licensee's Proposed Alternative Examination (as stated)

The NMC proposes to perform a VT-2 examination on the mechanical joints of the SRV assemblies during the normal operational start-up sequence at a minimum of 900 psig following a 10 minute hold time (for uninsulated components) in lieu of the nominal operating pressure associated with 100% reactor power of approximately 1000 psig. In addition, if there is an unplanned shutdown with a drywell entry before the next refueling outage, another inspection of these bolted connections will be performed to look for any evidence of leakage.

Application of this alternative test maintains reasonable levels of personnel safety and reduces the opportunity for the introduction of undesirable operational challenges.

While NMC does not expect that leakage will occur, any leakage at the bolted connection [would] be related to the differential pressure across the connection. A 10% reduction in test pressure is not expected to result in the arrest of a leak that would occur at nominal operating pressure.

In the event that leakage would occur at the mechanical joints at higher pressures associated with 100% reactor power, leakage from these mechanical connections would be detected by the drywell monitoring systems, which include drywell pressure monitoring, the containment atmosphere monitoring system (CAM), and the drywell floor drain sumps. Leakage monitoring is required by Monticello Technical Specifications.

This alternative method for a system leakage test is particularly applicable for the MNGP maintenance outage, which is of limited scope, and where the only components on the

primary system that are being replaced are the main steam “B” and “G” safety relief valve assemblies attached via mechanical connections.

The NRC has authorized use of a similar alternative system leakage test method for the Cooper Nuclear Station in 1998 which permitted them to perform a system leakage test at a minimum of 900 psig following replacement of their SRV topworks, a mechanical joint, during a mid-cycle maintenance outage. The approval letter for the Cooper relief request was dated February 26, 1998.

3.6 NRC Staff Evaluation

The NRC staff evaluated NMC’s relief request and concludes that requiring NMC to comply with the required system leakage test at about 1000 psig would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety because of the following:

- NMC cannot isolate Monticello’s main steam SRVs from the reactor vessel. Thus, NMC would have to manipulate many valves, change system lineups, and establish procedural controls to heat up and pressurize the primary system in order to perform the system leakage test and VT-2 examination of the mechanical joints of the replaced “B” and “G” main steam SRVs without withdrawing control rods. NMC expects this would take about 1 day to do, and then it would have to reconfigure plant systems for normal plant startup. Also, NMC is concerned with controlling RPV pressure during the test due to the amount of decay heat being generated and the heat removal capacity of the RWCU system.

- A system leakage test cannot be performed at a nominal operating pressure of 1000 psig since this requires a reactor power level of about 75 percent. At this power level, plant personnel performing the VT-2 examinations inside containment would be exposed to excessive radiation levels which is contrary to ALARA practices. To address the radiation exposure concerns, NMC would have to alter the normal operational mode of the steam pressure control system to reach 1000 psig at a more moderate power level of about 7 percent reactor power. Doing this could affect core reactivity and could challenge plant safety systems. Additionally, NMC has not previously operated in this manner, and operating in an abnormal mode to perform the test at nominal operating pressure could adversely impact personnel and reactor safety.

- Maintaining the RPV at its normal level and using decay heat to produce steam pressure to conduct the test at nominal operating temperature is not feasible. There is not enough decay heat to perform the test within a reasonable time to support completing the maintenance outage on schedule. NMC estimates it would take a minimum of 25 hours to reach the 1000 psig ASME Code-required test pressure with available decay heat.

The NRC staff further concludes that performing the test and accompanying VT-2 examination at a minimum of 900 psig during the normal plant startup following SRV repair/replacement activities provides reasonable assurance of structural integrity. This is because this test pressure is adequate to cause the bolted connections to leak on the replacement SRVs during the test if NMC has not established a leak-tight connection. Should leakage occur later, Monticello has Technical Specifications requiring NMC to monitor leakage using the drywell pressure monitoring system, containment atmosphere monitoring system, and the drywell floor

drain sumps. In addition, NMC committed in its letter of June 12, 2003, to reinspect these bolted connections to look for any evidence of leakage if there is an unplanned shutdown with a drywell entry before the next refueling outage.

4.0 CONCLUSION

Based on the above, the NRC staff concludes that NMC's proposed alternative of performing a one-time system leakage test and VT-2 examination of the SRV assemblies with the subject valves uninsulated, while operating at a minimum of 900 psig during the normal plant startup sequence, provides reasonable assurance of structural integrity of the SRV bolted connections. Furthermore, the NRC staff concludes that complying with the ASME Code requirement would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the NRC staff authorizes NMC's proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii) for one-time use during normal startup from the current planned maintenance outage to replace the SRVs.

Principal Contributor: M. Padovan

Date: June 13, 2003