NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

June 12, 2003

L-MT-03-048
10 CFR 50.55a(a)

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET 50-263
LICENSE No. DPR-22

REQUEST FOR AUTHORIZATION OF INSERVICE INSPECTION PROGRAM
FOURTH 10-YEAR INTERVAL RELIEF REQUEST NO. 8

The Nuclear Management Company, LLC (NMC) hereby requests authorization by the NRC of the Monticello Nuclear Generating Plant (MNGP) Inservice Inspection (ISI) Fourth 10–Year Relief Request (RR) No. 8.

Following restart from MNGP refueling outage (RFO) 21, it was discovered that Main Steam Safety Relief Valves (SRVs) "B" and "G" were leaking. NMC has decided to conduct a planned unit shutdown for the purpose of replacing the leaking SRVs. The replacement of each SRV is a repair/replacement activity associated with a mechanical joint in accordance with American Society of Mechanical Engineers (ASME) Section XI.

Following the SRV replacement, the 1995 Edition of ASME Section XI with the 1996 Addenda requires a system leakage test and VT-2 examination be performed to verify the leak tightness of the mechanical joint (bolted connections). The test is required to be conducted at nominal operating pressure (approximately 1000 psig).

Attachment 1 contains a RR to perform the required VT-2 examination and the system leakage test at a minimum of 900 psig. The bases for this request are that personnel safety (ALARA) considerations and other plant operational constraints make such testing problematic and constitute and undue hardship. In addition, performance of a water-solid, full reactor pressure vessel leakage (10CFR50 Appendix G) test is not required since the plant successfully passed this test in the preceding RFO completed on May 26, 2003.

This RR makes the following commitment:

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.

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NMC requests NRC authorization to perform the proposed alternative test on a one-time basis in accordance with 10 CFR 50.55a(a)(3)(ii). NMC has determined that complying with the specified requirements would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Because the leaking SRVs manifested themselves immediately after the outage, NMC could not have anticipated the need for the proposed alternative test to the NRC early enough for the staff to process in the normal RR review timeframe. Therefore, due to these unforeseen and unexpected circumstances, NMC requests expedited authorization of the attached RR by June 16, 2003.

If you have any questions please contact John Fields (763-295-1663).

David L. Wilson
Site Vice President
Monticello Nuclear Power Plant

CC: Regional Administrator-III NRC
NRR Project Manager, NRC
Sr. NRC Resident Inspector, NRC
State of Minnesota Boiler Inspector
Hartford Insurance

Attachment 1 - ISI Relief Request Number: 8

Attachment 1

NUCLEAR MANAGEMENT COMPANY, LLC
MONTICELLO NUCLEAR GENERATING PLANT
DOCKET 50-263

JUNE 12, 2003

ISI RELIEF REQUEST NUMBER: 8

5 pages follow

Attachment 1

ISI RELIEF REQUEST NUMBER: 8

COMPONENT IDENTIFICATION

Code Class:	1
References:	IWA-4540(c) IWA-5211(a)
Examination Category:	Not Applicable
Item Number:	Not Applicable
Description:	System Leakage Pressure Test and accompanying VT-2 Examination at nominal operating pressure following Repair-Replacement activities involving Mechanical Joints.
Components:	Main Steam Safety Relief Valve Assemblies

CODE REQUIREMENTS

The 1995 Edition of American Society of Mechanical Engineers (ASME) Section XI with the 1996 Addenda, paragraph IWA-5120(a) states:

"Items subjected to repair/replacement activities shall be pressure tested when required by IWA-4500."

Paragraph IWA-4540(c) states:

"Mechanical joints made in installation of pressure retaining items shall be pressure tested in accordance with IWA-5211(a)."

Paragraph IWA-5211(a) states:

"A system leakage test conducted during operation at nominal operating pressure, or when pressurized to nominal operating pressure and temperature."

Paragraph IWB-5210(b) states:

"The 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."

BASIS FOR RELIEF

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 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.

PROPOSED ALTERNATIVE PROVISIONS

Pursuant to 10CFR50.55a(a)(3)(ii), 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.

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 to 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.

Attachment 1

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. This approval letter for the Cooper relief request was dated February 26, 1998.

CONCLUSION:

In summary, the proposed NMC alternative is to perform the system leakage test and VT-2 examination at 900 psig minimum after a 10 minute hold time in lieu of the pressure testing requirements of the 1995 Edition of ASME Section XI with the 1996 Addenda for mechanical joints following repair-replacement activities. 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.

Considering the hardship and unusual difficulty in performing the available methods for satisfying the code requirements and the ability to detect leakage in primary containment should it occur, this alternative will provide an acceptable verification of the leak integrity of the mechanical joint without putting the plant in a non-conservative operational condition and without unnecessary radiation exposure and safety challenges to personnel.

PERIOD FOR WHICH RELIEF IS REQUESTED

NMC requests NRC authorization to perform the proposed alternative test on a one-time basis for the system leakage tests following repair/replacement activities on the mechanical joints of SRVs "B" and "G" during the planned maintenance outage.