June 9, 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 — FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN RELIEF REQUEST NO. 5 (TAC NO. MB6956)

Dear Mr. Wilson:

The Nuclear Management Company's, LLC (NMC's), letter of December 6, 2002, as supplemented April 28, 2003, submitted several requests for 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 requests applied to the fourth 10-year interval of the inservice inspection (ISI) examination plan for the Monticello Nuclear Generating Plant. This letter and the enclosed safety evaluation address Relief Request No. 5, only.

In Relief Request No. 5, NMC requested relief from having to 1) remove a bolt from a leaking control rod drive bolted connection and 2) perform the ASME Code-required VT-3 visual examination at the leaking connection without first performing an evaluation of this leakage. NMC requested relief pursuant to 10 CFR 50.55a(a)(3)(ii).

The NRC staff evaluated Relief Request No. 5 and concludes that NMC's proposed alternative to removing a bolt at a leaking CRD bolted connection without first performing an evaluation of this leakage provides reasonable assurance of operational readiness of the subject bolted connection. 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 the fourth 10-year ISI interval.

In addition, the NRC staff concludes that NMC's proposal to perform a VT-1 visual examination on all CRD bolting that will be reused in lieu of a VT-3 visual examination provides an acceptable level of quality and safety. Therefore, the NRC staff authorizes NMC's proposed alternative pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Enclosed is our safety evaluation.

Sincerely,

/RA/

L. Mark Padovan, Project Manager, 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

In addition, the NRC staff concludes that NMC's proposal to perform a VT-1 visual examination on all CRD bolting that will be reused in lieu of a VT-3 visual examination provides an acceptable level of quality and safety. Therefore, the NRC staff authorizes NMC's proposed alternative pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

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L. Mark Padovan, Project Manager, Section 1
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Office of Nuclear Reactor Regulation

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Enclosure: Safety Evaluation

cc w/encl: See next page

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cc:

Jonathan Rogoff, Esquire
General Counsel
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
2807 W. County Road 75
Monticello, MN 55362

Manager, Regulatory Affairs
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

Robert Nelson, President
Minnesota Environmental Control
Citizens Association (MECCA)
1051 South McKnight Road
St. Paul, MN 55119

Commissioner
Minnesota Pollution Control Agency
520 Lafayette Road
St. Paul, MN 55155-4194

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Commissioner
Minnesota Department of Health
717 Delaware Street, S. E.
Minneapolis, MN 55440

Douglas M. Gruber, Auditor/Treasurer
Wright County Government Center
10 NW Second Street
Buffalo, MN 55313

Commissioner
Minnesota Department of Commerce
121 Seventh Place East
Suite 200
St. Paul, MN 55101-2145

Adonis A. Neblett
Assistant Attorney General
Office of the Attorney General
445 Minnesota Street
Suite 900
St. Paul, MN 55101-2127

Mr. Roy A. Anderson
Executive Vice President
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

John Paul Cowan
Chief Nuclear Officer
27780 Blue Star Memorial Highway
Covert, MI 49083

Jeffrey S. Forbes
Senior Vice President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West Country Road 75
Monticello, MN 55362-9637

Nuclear Asset Manager
Xcel Energy, Inc.
550 15th St., Suite 1000
Denver, CO 80202

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN

RELIEF REQUEST NO. 5

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, as supplemented April 28, 2003, submitted several requests for 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 requests applied to the fourth 10-year interval of the inservice inspection (ISI) examination plan for the Monticello Nuclear Generating Plant. This safety evaluation addresses Relief Request No. 5, only.

In Relief Request No. 5, NMC requested relief from having to 1) remove a bolt from a leaking control rod drive (CRD) bolted connection and 2) perform the ASME Code-required VT-3 visual examination at the leaking connection without first performing an evaluation of this leakage. NMC requested relief pursuant to 10 CFR 50.55a(a)(3)(ii).

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 Code of record for the Monticello Nuclear Generating Plant, fourth 10-year ISI interval is the 1995 edition through 1996 addenda of the ASME Code.

3.0 TECHNICAL EVALUATION

3.1 Code Requirement

ASME Code, Section XI, Paragraph IWA-5250(a)(2), requires that if leakage occurs at a bolted connection on other than a gaseous system, one of the bolts shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100.

3.2 Licensee's Code Relief

NMC requested relief from having to 1) remove a bolt from a leaking CRD bolted connection and 2) perform the ASME Code-required VT-3 visual examination at the leaking CRD connection without first performing an evaluation of this leakage under General Electric Company (GE) Guidelines. NMC's submittals of July 29 and December 20, 1993, for the third 10-year ISI interval for Relief Request No. 7, contained this GE Operation and Maintenance Guideline information.

3.3 Identification of Components

Bolted CRD Housing Joint, Table IWB-2500-1, Category B-P, Item No. B15.10

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

> 10 CFR Part 50, Section 50.55a(a)(3), which states, (in part):
>
> "Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when ...
>
> > (ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."
>
> The CRD (Control Rod Drive) housings are flanged connections beneath the reactor vessel that are used to secure the 121 CRD mechanisms in position below the vessel. Each of the 121 CRD to CRD housing bolted joints utilizes eight bolts, washers, and nuts to hold the CRD mechanism in position. The joint also utilizes three hollow metal O-rings to provide a watertight seal capable of withstanding full reactor pressure at normal operating temperatures.
>
> The CRD housing joints are VT-2 examined as part of the periodic Reactor Pressure Vessel Leakage and Hydrostatic pressure tests. These tests are conducted with the vessel temperature much less than the design operating temperature. For a typical test, the vessel temperature would be <212°F, as compared to a normal operating temperature of about 540°F. It is not unusual for these bolted joints to leak slightly

during periodic reactor vessel pressure tests conducted at test temperatures below normal operating temperature. [In its letter dated April 28, 2003, the licensee stated that the Monticello pressure-temperature (P-T) curves referenced in Monticello Technical Specifications (TS) 3.6/4.6.B (with the associated Reference Temperature Nil Ductility Transition Shift) are used to develop the temperature during the system leakage test to prevent brittle fracture. This results in a temperature of <212°F.]

Compliance with Code Requirement IWA-5250(a)(2) represents a hardship (burden) in the case of the CRD housing bolted joints because:

1. Examining the bolting would involve the accumulation of considerable personnel radiation exposure, since the work must be performed in a relatively high dose rate area inside the drywell, immediately below the reactor vessel. Typical shutdown dose rates in the vicinity of the bolting flanges would be on the order of 50 to 100 mr/hr.

2. Since the reactor pressure vessel test is critical path item, the additional time needed to depressurize the vessel, remove the bolting, perform the exam, and then re-pressurize the vessel to retest the joint would delay plant startup from an outage by an equivalent amount of time. The cost of such delays is significant, since it is estimated that the cost of extending the duration of an outage is $379,000 per day (including replacement power costs)(this is estimated cost submitted in 1993 (see TAC No. M82545 referenced in "Status" section)

Compliance with Code requirement IWA-5250(a)(2) would not result in a compensating increase in quality or safety because:

1. CRD Housing joint leakage during (relatively) low temperature testing is not unexpected due to the design of the bolted joint. This joint is unusual in that it has hollow metal o-rings that require the CRD housing bolts to be tightened within a specific torque range in order to function properly at normal operating temperature. Thus, the bolts cannot simply be tightened to stop leakage as might be done for a conventional gasketed joint. As noted previously, GE developed guidance to evaluate any CRD housing leakage to determine if the leakage will persist at normal operating temperature/pressure and should therefore be corrected.

2. Leakage that is found to be acceptable per the guidance is not considered adverse to quality or safety and need not be corrected prior to startup. This type of analysis is consistent with Section XI.

3. Code paragraph IWB-3142 allows analysis of the leakage for acceptability. Performance of the VT-3 bolting examination does not represent a corrective action for the joint leakage and will not reduce the likelihood of joint leakage upon retest. Therefore, the VT-3 bolting examination does not contribute to increased quality or safety.

4. The bolts in the CRD housing connection are periodically examined when the joint is disassembled, per Table IWB-2500-1, Item B7.80 (1995 Edition with no Addenda per 10 CFR 50.55A Paragraph (b)(2)(xxi)(B)) and Procedure 9309, "Changeout Selected CRD's - Maintenance" and Commitment No. M92076A. Four of the eight bolts on each housing joint were replaced with new bolts in 1991 under Work Control Record (WCR) 91-01909. It was also reported in General Electric SIL [Service Information Letter] 483 that only three uniformly distributed housing bolts are required to support the CRD mechanism. These factors provide a high degree of confidence in the long term safety and integrity of the CRD housing joints.

3.5 <u>Licensee's Proposed Alternative Examination</u> (as stated)

Pursuant to 10 CFR 50.55a(a)(3)(ii), the following alternative is proposed. Any leakage found at a CRD housing bolted joint during a periodic pressure test performed at a temperature much less than operating temperature will be evaluated to determine whether it will stop leaking at operating temperature. If this evaluation shows the leak will stop as temperature increases to normal operating temperature, no further action will be taken. The acceptance criteria is based on guidance provided by General Electric and is included in the VT-2 tests for the joint (Note: This criteria was submitted for NRC review during the Request for Relief process previously approved on October 18, 1994, therefore it is not included in this submittal). If the leakage is determined to be unacceptable according to the General Electric guidelines and the joint is disassembled to correct the leak, any CRD bolting that is reused will be examined by the VT-1 examination method (10 CFR 50.55a(b)(2)(xxi)(B) dated September 26, 2002).

Upon approval of this relief request, MNGP commits to revise the applicable pressure test procedure to perform a VT-1 exam in lieu of a VT-3 exam specified by IWA-5250(a)(2) on all CRD bolting that will be reused when the GE acceptance criteria has been exceeded and disassembly is required to correct the leak.

3.6 <u>NRC Staff Evaluation</u>

The ASME Code, Section XI, requires that the source of leakage detected during the conduct of a system pressure test be located and evaluated for corrective measures. If leakage occurs at a bolted connection, a bolt nearest the leakage is to be removed and VT-3 visually examined. NMC is requesting relief from the Code requirement to remove the bolt at a leaking CRD housing connection if an evaluation indicates that the leak would eventually stop when normal operating temperature is reached. In addition, NMC proposed that if the leakage is determined to be unacceptable according to the GE guidelines and if the joint is disassembled to correct the leak, any CRD bolting that is reused will be VT-1 visually examined.

Removing a bolt from a leaking CRD bolted connection constitutes a hardship or unusual difficulty because maintenance personnel would accumulate considerable radiation exposure when doing this. Plant personnel must perform the maintenance in a relatively high dose rate area inside the drywell, immediately below the reactor vessel. NMC indicated that typical shutdown dose rates in the vicinity of the bolted flanges would be about 50 to 100 mr/hr.

The CRD housings, designed by GE, have an inherent characteristic where the bolted joints leak slightly during periodic, reactor pressure vessel tests conducted at test temperatures below normal operating temperature. NMC's letter of April 28, 2003, said that GE has provided guidance for CRD flange leakage evaluations. The guidance states that drip-type leaks of 30 drops-per-minute, or less, which show a constant or decreasing leak rate over an 8-hour period at reactor pressures greater than or equal to 1000 psig, do not require any corrective maintenance action. GE determined that corrective maintenance is not required because a decreasing leak rate will eventually seal without being internally pressurized, provided the flange bolts remain properly torqued. NMC monitors the CRD bolting during the hydrostatic pressure test performed during each startup. Furthermore, when the plant is operating at rated temperature and pressure, the drywell drain sump monitoring system provides indication of leakage. The TSs provide action levels for unidentified leakage in the drywell. Unidentified leakage is typically less than 5 percent of the TS limit of 5 gpm during a normal operating cycle. Additionally, there is also a TS limit of a 2 gpm increase in unidentified leakage in a 24-hour period.

The NRC staff determined that because a boiling-water reactor does not contain borated water, the corrosion resulting from boric acid is not a concern. Also, the use of NMC's proposed alternative for corrective measures for leaking CRD bolted connections provides a reasonable assurance of leak tightness of the subject connection. NMC's April 28, 2003, letter discussed past operating experience related to the these bolted connections that supports its approach. In addition, NMC has currently installed 200 cap screws of the new GE design to meet GE's recommendation in SIL-483. GE redesigned the cap screws with new American Society for Testing and Materials SA-540 Grade B23, Class 4 material which has higher mechanical properties and better controlled chemistry than that of the old design. In addition, NMC's letter of April 28, 2003, stated that GE determined that the degradation mechanism for the original designed cap screws is attributed to stress-corrosion cracking in a crevice region of the cap screw with possible aggravation by fabrication irregularities. Magnesium sulfide inclusions and surface pitting may have contributed in some cases. It was also probable that water leakage into the bolted connection contributed to stress corrosion. The new style cap screws are intended to be less susceptible to these degradation mechanisms. Furthermore, NMC intends to use the new cap screw design or an improved cap screw design for future replacements.

Based on the above, the NRC staff concludes that NMC's proposed alternative to removing a bolt at a leaking CRD bolted connection without first performing an evaluation of this leakage under GE Guidelines provides reasonable assurance of operational readiness of the subject bolted connection. 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.

NMC also proposed to perform an ASME Code, Section XI, VT-1 visual examination in lieu of a VT-3 visual examination specified by IWA-5250(a)(2) on all CRD bolting that will be reused when the GE acceptance criteria has been exceeded and disassembly is required to correct the leak. The VT-1 visual examination requirements are more stringent than that of the requirements for a VT-3 visual examination. Therefore, NMC's proposed alternative to perform a VT-1 visual examination in lieu of a VT-3 visual examination provides an acceptable level of quality and safety.

4.0 CONCLUSION

Based on the above, the NRC staff concludes that NMC's proposed alternative to removing a bolt at a leaking CRD bolted connection without first performing an evaluation of this leakage provides reasonable assurance of operational readiness of the subject bolted connection. 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 the fourth 10-year ISI interval.

In addition, based on the above, the NRC staff concludes that NMC's proposal to perform a VT-1 visual examination on all CRD bolting that will be reused in lieu of a VT-3 visual examination provides an acceptable level of quality and safety. Therefore, the NRC staff authorizes NMC's proposed alternative pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: T. McLellan

Date: June 9, 2003