October 1, 2002

Mr. Mano Nazar
Site Vice President
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - EVALUATION OF RELIEF REQUEST NOS. 12 AND 13 FOR UNIT 1 AND NOS. 13 AND 14 FOR UNIT 2 ASSOCIATED WITH THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM (TAC NOS. MB5388, MB5389, MB5390, AND MB5391)

Dear Mr. Nazar:

By letter dated June 17, 2002, the Nuclear Management Company, LLC (the licensee) submitted four requests for relief for the third 10-year inservice inspection (ISI) interval at the Prairie Island Nuclear Generating Plant, Units 1 and 2. Specifically, the licensee submitted Relief Request Nos. 12 and 13 for Unit 1 and Nos. 13 and 14 for Unit 2.

In Relief Request No. 12 for Unit 1 and No. 13 for Unit 2, the licensee seeks relief from requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI (1989 edition, no addenda); Subsection IWA-5250(a)(2), regarding the removal of bolting and performance of a visual (VT-3) examination for degradation if leakage occurs at a bolted connection while conducting a system pressure test. The licensee's alternative would allow stopping leakage at the bolted connection, or if leakage is not stopped, the connection would be evaluated for joint integrity in accordance with specified criteria. For this alternative, the licensee proposes to selectively use Code Case N-566-1, "Corrective Action for Leakage Identified at Bolted Connections," for those instances where an engineering evaluation that addresses bolting susceptibility to corrosion and failure is more appropriate than removal and inspection of all fasteners.

In Relief Request No. 13 for Unit 1 and No. 14 for Unit 2, the licensee seeks relief from requirements of ASME Code, Section XI (1989 edition, no addenda), Subsection IWA-5242, regarding the removal of insulation from pressure-retaining bolted connections during the pressure test of systems borated for reactivity control. The licensee proposes an alternative based upon Code Case N-533, "Alternative Requirements for VT Visual Examination of Class 1 Insulated Pressure-Retaining Bolted Connections, Section XI, Division 1." The licensee's alternative would allow insulation removal and a VT-2 visual examination of the bolted connections to be done earlier in the refueling outage when the system is depressurized and at lower temperatures.

The U.S. Nuclear Regulatory Commission (NRC) staff's safety evaluation for these four relief requests is enclosed. As noted in the safety evaluation, the NRC staff concludes that the licensee's proposed alternatives provide an acceptable level of quality and safety to ensure the integrity of bolted connections. Therefore, the NRC staff authorizes the licensee's proposed alternatives described in Relief Request Nos. 12 and 13 for Unit 1 and Relief Requests Nos. 13 and 14 for Unit 2, pursuant to 10 CFR 50.55a(a)(3)(i), for the third 10-year ISI interval at Prairie Island, Units 1 and 2. The licensee's use of Code Cases N-566-1 and N-533 is authorized until such time as the code cases are published in a future version of Regulatory Guide (RG) 1.147, "Inservice Inspection Code Case Acceptability--ASME Section XI, Division 1," or in a succeeding document. At that time, if the licensee intends to continue implementing the alternatives based upon these code cases, the licensee must follow all applicable provisions of these code cases with the limitations or conditions specified in RG 1.147, if any, or in its succeeding document.

Sincerely,

/RA/

Tae Kim, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosure: Safety Evaluation

cc w/encl: See next page

The U.S. Nuclear Regulatory Commission (NRC) staff's safety evaluation for these four relief requests is enclosed. As noted in the safety evaluation, the NRC staff concludes that the licensee's proposed alternatives provide an acceptable level of quality and safety to ensure the integrity of bolted connections. Therefore, the NRC staff authorizes the licensee's proposed alternatives described in Relief Request Nos. 12 and 13 for Unit 1 and Relief Requests Nos. 13 and 14 for Unit 2, pursuant to 10 CFR 50.55a(a)(3)(i), for the third 10-year ISI interval at Prairie Island, Units 1 and 2. The licensee's use of Code Cases N-566-1 and N-533 is authorized until such time as the code cases are published in a future version of Regulatory Guide (RG) 1.147, "Inservice Inspection Code Case Acceptability--ASME Section XI, Division 1," or in a succeeding document. At that time, if the licensee intends to continue implementing the alternatives based upon these code cases, the licensee must follow all applicable provisions of these code cases with the limitations or conditions specified in RG 1.147, if any, or in its succeeding document.

Sincerely,

/RA/

Tae Kim, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosure: Safety Evaluation

cc w/encl: See next page

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Prairie Island Nuclear Generating Plant,
Units 1 and 2

cc:

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION

RELIEF REQUEST NOS. 12 AND 13 (FOR UNIT 1)

RELIEF REQUEST NOS. 13 AND 14(FOR UNIT 2)

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated June 17, 2002, the Nuclear Management Company, LLC (the licensee), submitted Relief Request Nos. 12 and 13 for Unit 1 and Nos. 13 and 14 for Unit 2, proposing alternatives to the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, for the Prairie Island Nuclear Generating Plant. In Relief Request No. 12 for Unit 1 and No. 13 for Unit 2, the licensee seeks relief from requirements of the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" (1989 edition, no addenda), Subsection IWA-5250(a)(2), regarding the removal of bolting and performance of a visual (VT-3) examination for degradation if leakage occurs at a bolted connection while conducting a system pressure test. As an alternative, the licensee proposes to selectively use Code Case N-566-1, "Corrective Action for Leakage Identified at Bolted Connections," for those instances where an engineering evaluation that addresses bolting susceptibility to corrosion and failure is more appropriate than removal and inspection of all fasteners. In Relief Request No. 13 for Unit 1 and No. 14 for Unit 2, the licensee seeks relief from requirements of ASME Code, Section XI (1989 edition, no addenda), Subsection IWA-5242, regarding the removal of insulation from pressure-retaining bolted connections during the pressure test of systems borated for reactivity control. The licensee proposes an alternative based upon Code Case N-533, "Alternative Requirements for VT Visual Examination of Class 1 Insulated Pressure-Retaining Bolted Connections, Section XI, Division 1."

The Nuclear Regulatory Commission (NRC) staff has reviewed and evaluated the licensee's requests for relief pursuant to 10 CFR 50.55a(a)(3)(i), as discussed below.

ENCLOSURE

2.0 REGULATORY EVALUATION

Inservice inspection (ISI) interval of the ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Code, and applicable addenda, as required by Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(g). 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 the licensee demonstrates (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 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 the Prairie Island Nuclear Generating Plant, Units 1 and 2, third 10-year ISI interval is the 1989 edition of the ASME Code, Section XI.

3.0 RELIEF REQUEST NO. 12 FOR UNIT 1 AND NO. 13 FOR UNIT 2

3.1 Components Affected

These relief requests affect all Class 1, 2, and 3 systems and components subject to Subsection IWA-5000 pressure testing.

3.2 Code Requirement

Section XI of the ASME Code, 1989 edition, no addenda, Subsection IWA 5250(a)(2) states:

> If leakage occurs at a bolted connection, the bolting shall be removed, VT-3 examined for corrosion, and evaluated in accordance with IWA-3100. The bolt selected shall be the one closest to the source of the leakage. When the removed bolt has evidence of degradation, all remaining bolting in the connection shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100.

3.3 Licensee's Proposed Alternative

The licensee proposes that:

> In lieu of the IWA-5250(a)(2) Corrective Measures, Prairie Island Nuclear Generating Plant proposes to use Nuclear Code Case N-566-1, "Corrective Action for Leakage Identified at Bolted Connections." Code Case N-566-1 would be selectively used for those instances where an engineering evaluation that addresses bolting susceptibility to corrosion and failure is more appropriate than removal and inspection of all fasteners.

3.4 Licensee's Basis for Relief

The licensee explains that:

> In certain situations, it is technically unnecessary to remove and inspect all the fasteners in a bolted connection that has evidence of leakage. The removal and subsequent inspection of all fasteners may not be necessary to maintain piping, component, and system integrity. For example, certain fasteners may not be exposed to leaking fluid due to the orientation of the leak, the leaking fluid may have little corrosive effect on the fasteners, or maintenance including fastener inspection or replacement may have just occurred on the bolted joint. Avoiding complete disassembly of radioactive components when a less intrusive method of assuring fastener integrity is available can also reduce personnel radiation exposure. In these types of circumstances, an evaluation can determine the susceptibility of fasteners to corrosion and failure due to leakage and can provide an acceptable level of quality and safety.
>
> ASME Code interpretation XI-1-92-01 states that new bolting or bolting that has received a visual inspection prior to installation and has not been in service does not have to be evaluated in accordance with this section. This is recognition by the Code that leakage in this situation is not subject to the requirements of IWA-5250(a)(2).

3.5 NRC Staff's Technical Evaluation

In accordance with the 1989 edition, no addenda, of the ASME Code, Section XI, when leakage occurs at bolted connections, one of the bolts shall be removed, VT-3 examined for corrosion, and evaluated in accordance with Subsection IWA-3100. In lieu of the Code-required removal of bolting to perform a VT-3 visual examination, the licensee has proposed to perform an evaluation of the bolted connection to determine the susceptibility of the bolting to corrosion and the potential for failure.

Most leakage at bolted connections is due to thermal expansion of the piping and bolting materials, with subsequent fluid seepage at the joint gasket. Therefore, proper torquing of the bolting, in most cases, should stop the leakage. After stopping a leak, the bolting and component material shall be evaluated to determine the susceptibility of the bolting to corrosion and failure considering all of the following factors:

(1) The number and service age of the bolts
(2) Bolt and component materials
(3) Corrosiveness of process fluid
(4) Leakage location and system function
(5) Leakage history at the connection or other system components
(6) Visual evidence of corrosion at the assembled connection

If the evaluation determines that the leaking condition has not degraded the fasteners, then no further action is necessary. If the initial evaluation indicates the need for a more in-depth evaluation, the bolt closest to the source of leakage will be removed, VT-3 examined, and evaluated in accordance with Subsection IWA-3100. In a situation where the leakage is not stopped, the licensee shall, along with the considerations of the above six factors, evaluate structural integrity of the bolting, consequences of continued operation, and the effect on the

system operability of continued leakage. Such an evaluation will be the basis for deferral of removing the bolt closest to the source of leakage. This alternative would allow the licensee to utilize a systematic approach and sound engineering judgement, provided that, as a minimum, all of the six evaluation factors listed above are considered. This proposed alternative engineering evaluation considers the factors necessary to identify degradation of the bolts in any leaking bolted connection. Accordingly, the use of this type of engineering evaluation is expected to result in: (1) the identification of such degradation, (2) corrective actions when appropriate, and (3) the avoidance of unnecessary joint disassembly when the bolts are fit for service. As a result, the licensee's alternative to the Code-required removal of bolting at a joint when leakage occurs will provide an acceptable level of quality and safety since the integrity of the joint will be maintained.

This alternative to the requirements of ASME Code, 1989 edition, no addenda, Subsection IWA-5250(a)(2), is consistent with Code Case N-566-1.

3.6 Conclusion

The NRC staff concludes that the licensee's proposed alternative to the requirements of Subsection IWA-5250(a)(2) is a technically sound engineering approach and will provide an acceptable level of quality and safety to ensure the integrity of bolted connections. Therefore, the NRC staff authorizes the licensee's proposed alternative, pursuant to 10 CFR 50.55a(a)(3)(i), for the third 10-year ISI interval for Prairie Island, Units 1 and 2. Use of Code Case N-566-1 is authorized until such time as the code case is published in a future version of Regulatory Guide (RG) 1.147, "Inservice Inspection Code Case Acceptability--ASME Section XI, Division 1," or in a succeeding document. At that time, if the licensee intends to continue implementing the alternative based upon this code case, it must follow all applicable provisions of Code Case N-566 with the limitations or conditions specified in RG 1.147, if any, or in its succeeding document.

4.0 RELIEF REQUEST NO. 13 FOR UNIT 1 AND NO. 14 FOR UNIT 2

4.1 Components Affected

These relief requests affect all Class 1 reactor coolant system (RCS) components.

4.2 Code Requirement

ASME Code, Section XI (1989 edition, no addenda), Subsection IWA-5242, requires that insulation be removed from pressure-retaining bolted connections for VT-2 visual examination on systems borated for the purpose of controlling reactivity. The VT-2 examination is also required to be performed at normal operating pressure and temperature.

4.3 Licensee's Proposed Alternative

The licensee proposes that:

In lieu of the IWA-5242 requirement to remove insulation from Class 1 pressure-retaining bolted connections to perform a VT-2 visual examination:

a) A system pressure test and VT-2 visual examination shall be performed each refueling outage for Class 1 connections without removal of insulation.

b) Each refueling outage, the insulation shall be removed from the bolted connections and a VT-2 visual examination shall be performed. The connections are not required to be pressurized. Any evidence of leakage shall be evaluated in accordance with IWA-5250.

These requirements are specified in Nuclear Code Case N-533 as approved by the Board of Nuclear Codes and Standards.

c) A four hour hold time on the system pressure test will be imposed.

4.4 Licensee's Justification for Alternative

The licensee explains that:

The above alternative is proposed pursuant to 10 CFR 50.55a(a)(3)(i) because it provides an acceptable level of quality and safety.

a) The proposed alternative will provide the same or greater level of leak detection/RCS integrity because:

- A four hour hold time will be imposed on the System Leakage Test thereby increasing the likelihood that any leaking connections will be identified during the VT-2 visual examination.

- After being pressurized for the whole fuel cycle, all connections will undergo an examination under environmental conditions that allow a more detailed inspection than would be possible if the connections were inspected hot. Leakage indications (boric acid residue) will be equally visible at reduced temperature and pressure.

b) The proposed alternative will reduce personnel exposure to hazards such as heat stress, radiation, and burn hazards. Conduct of examinations in hot areas would result in decreased examination efficiency due to shorter examiner stay times and would require the wearing of special protective equipment to protect against heat stress.

c) The proposed alternative will take advantage of established refueling outage activities to inspect for leaks at the beginning of the outage so that any corrective measures could be made a part of the outage scope. The proposed alternative reduces the number of RCS thermal cycles and the refueling outage duration.

4.5 Licensee's Basis for Relief

The licensee states that:

> Inspection of piping components at high temperatures are impractical because they pose significant safety hazards to personnel performing the examination, including heat stress, burns, and increased radiation exposure.
>
> Performance of the System Leakage Test per the code would require an additional reactor coolant system (RCS) thermal transient and an extension of the refueling outage. Prior to performing the VT-2 visual examination required by the System Leakage Test, the code requires the RCS pressure and temperature to be raised to 2235 psig and 547 °F. The code does allow the pressure and temperature to be lowered after attaining test conditions, however cooling down to perform the examination and the subsequent heat-up imposes an additional transient on the RCS.
>
> After the examination and before reactor startup the insulation has to be reinstalled. The additional cooldown and heat-up of the RCS and the replacement of insulation after the exam would result in an extension of the refueling outage.

4.6 NRC Staff's Technical Evaluation

The licensee's proposed alternative is consistent with Code Case N-533. Since the insulation will be removed during each refueling outage and the VT-2 visual examination will be performed during environmental conditions that allow for a more detailed inspection (the refueling outage), this alternative will provide the same or greater level of leak detection and RCS integrity while reducing personnel exposure to hazards such as heat stress, radiation, and burn hazards.

The additional imposition of a 4-hour hold time on the system leakage test following each refueling outage is appropriate. This hold time will increase the likelihood that leaking connections will be identifiable during subsequent VT-2 visual examination during the outage without imposing additional hazards to personnel.

The licensee's proposed alternative provides a thorough approach to ensuring the leak-tight integrity of systems borated for the purpose of controlling reactivity. First, the visual examination during the pressure test provides a means of detecting any significant leakage with the insulation in place. Second, by removing the insulation each refueling outage, the licensee will be able to detect minor leakage indicated by the presence of boric acid crystals or residue. This two-stage approach provides an acceptable level of quality and safety for bolted connections in systems borated for the purpose of reactivity control.

4.7 Conclusion

The NRC staff concludes that the licensee's proposed alternative to the requirements of Subsection IWA-5242 is a technically sound engineering approach and will provide an acceptable level of quality and safety to ensure the integrity of bolted connections in systems borated for the purpose of reactivity control. Therefore, the NRC staff authorizes the licensee's proposed alternative pursuant to 10 CFR 50.55a(a)(3)(i) for the remainder of the third 10-year ISI interval for Prairie Island, Units 1 and 2. Use of Code Case N-533 is authorized until such time as the code case is published in a future version of RG 1.147 or in a succeeding document. At that time, if the licensee intends to continue implementing the alternative based upon this code case, it must follow all applicable provisions of Code Case N-533 with the limitations or conditions specified in RG 1.147, if any, or in its succeeding document.

Principal Contributor: N. Sanfilippo

Date: October 1, 2002