April 27, 2005

Mr. Joseph M. Solymossy
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. 1-RR-4-2, 2-RR-4-2, 1-RR-4-3, 2-RR-4-3, 1-RR-4-4, AND 2-RR-4-4 FOR THE FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL (TAC NOS. MC3663 AND MC3664)

Dear Mr. Solymossy:

By a letter dated June 21, 2004, Nuclear Management Company, LLC (NMC or the licensee), submitted Relief Request (RR) Nos. 1-RR-4-2, 2-RR-4-2, 1-RR-4-3, 2-RR-4-3, 1-RR-4-4, and 2-RR-4-4 for the fourth 10-year Inservice Inspection Interval. The relief requests are described as below:

1-RR-4-2, 2-RR-4-2
Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested to perform VT-2 visual examinations in lieu of the required volumetric examinations.

1-RR-4-3, 2-RR-4-3
Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested to perform the VT-1, visual examination in lieu of the required VT-3, visual examination in conjunction with the American Society of Mechanical Engineers (ASME) Section XI Code Case N–566-1 [Note: Code Case N–566-1 has been incorporated in Regulatory Guide 1.147 *Inservice Inspection Code Case Acceptability ASME Section XI*, *Division 1*, Revision 13 for general use.]

1-RR-4-4, 2-RR-4-4
Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is requested from the provisions of Table IWB-2500-1, Category B-P, Item No. B15.50 for performing the VT-2 visual examination using reactor coolant as a pressurizing medium at a test pressure of 2235 psig.

A request for additional information was sent to the licensee on October 25, 2004, via e-mail (ADAMS Accession No. ML043210465). A followup telephone conference was held on November 24, 2004, between the U.S. Nuclear Regulatory Commission (NRC) staff and the NMC representatives. The licensee provided additional information in its letter dated January 28, 2005. The NRC staff evaluation for the above relief requests for the fourth 10-year ISI program which began December 21, 2004, and ends December 20, 2014, is as follows:

For 1-RR-4-2, 2-RR-4-2, the NRC staff determined that, compliance with the ASME Code volumetric examinations would result in hardship without a compensating increase in quality and safety and that the licensee's proposed alternative provides reasonable assurance of leakage and structural integrity of the regenerative heat exchangers and associated piping. Therefore, the licensee's proposed alternative contained in Request for Relief Nos. 1-RR-4-2 (Revision 0) and 2-RR-4-2 (Revision 0) is authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

For 1-RR-4-3, 2-RR-4-3, the NRC staff determined that the use of a VT-1 visual examination in lieu of the ASME Code-required VT-3 visual examination provides an acceptable level of quality and safety. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

For 1-RR-4-4, 2-RR-4-4, the NRC staff determined that the subject components would have to be redesigned in order for the licensee to perform the ASME Code-required tests which would place a significant burden on the licensee, and that the Code requirement is thus impractical. Additionally, the licensee's proposed alternative provides reasonable assurance of structural integrity of the subject components. Therefore, the licensee's request for relief is granted pursuant to 10 CFR 50.55a(a)(g)(6)(i). The NRC staff has determined that granting request for relief nos. 1-RR-4-4 (Revision 0) and 2-RR-4-4 (Revision 0) pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property, or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

A copy of our related safety evaluation is also enclosed.

Sincerely,

/RA/

L. Raghavan, Chief, 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

For 1-RR-4-2, 2-RR-4-2, the NRC staff determined that, compliance with the ASME Code volumetric examinations would result in hardship without a compensating increase in quality and safety and that the licensee's proposed alternative provides reasonable assurance of leakage and structural integrity of the regenerative heat exchangers and associated piping. Therefore, the licensee's proposed alternative contained in Request for Relief Nos. 1-RR-4-2 (Revision 0) and 2-RR-4-2 (Revision 0) is authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

For 1-RR-4-3, 2-RR-4-3, the NRC staff determined that the use of a VT-1 visual examination in lieu of the ASME Code-required VT-3 visual examination provides an acceptable level of quality and safety. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

For 1-RR-4-4, 2-RR-4-4, the NRC staff determined that the subject components would have to be redesigned in order for the licensee to perform the ASME Code-required tests which would place a significant burden on the licensee, and that the Code requirement is thus impractical. Additionally, the licensee's proposed alternative provides reasonable assurance of structural integrity of the subject components. Therefore, the licensee's request for relief is granted pursuant to 10 CFR 50.55a(a)(g)(6)(i). The NRC staff has determined that granting request for relief nos. 1-RR-4-4 (Revision 0) and 2-RR-4-4 (Revision 0) pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property, or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

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Prairie Island Nuclear Generating Plant, Units 1 and 2

cc:

Jonathan Rogoff, Esquire
Vice President, Counsel & Secretary
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

Manager, Regulatory Affairs
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

Manager - Environmental Protection Division
Minnesota Attorney General's Office
445 Minnesota St., Suite 900
St. Paul, MN 55101-2127

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
1719 Wakonade Drive East
Welch, MN 55089-9642

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

Administrator
Goodhue County Courthouse
Box 408
Red Wing, MN 55066-0408

Commissioner
Minnesota Department of Commerce
85 7th Place East, Suite 500
St. Paul, MN 55101-2198

Tribal Council
Prairie Island Indian Community
ATTN: Environmental Department
5636 Sturgeon Lake Road
Welch, MN 55089

Nuclear Asset Manager
Xcel Energy, Inc.
414 Nicollet Mall, R.S. 8
Minneapolis, MN 55401

John Paul Cowan
Executive Vice President & Chief Nuclear Officer
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

Craig G. Anderson
Senior Vice President, Group Operations
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION

REQUESTS FOR RELIEF NOS. 1-RR-4-2, 2-RR-4-2, 1-RR-4-3,

2-RR-4-3, 1-RR-4-4, AND 2-RR-4-4

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed and evaluated the information provided by Nuclear Management Company, LLC (the licensee) in its letter dated June 21, 2004, which proposed its Fourth 10-Year Interval Inservice Inspection Program Plan Requests for Relief Nos. 1-RR-4-2, 2-RR-4-2, 1-RR-4-3, 2-RR-4-3, 1-RR-4-4, and 2-RR-4-4 for Prairie Island Nuclear Generating Plant, Units 1 and 2. The licensee provided additional information in its letter dated January 28, 2005.

2.0 REGULATORY REQUIREMENTS

Inservice inspection (ISI) of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, 2, and 3 components is 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), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR Part 50, Section 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. 10CFR Part 50, Section 50.55a(g)(5)(iii) states, “if the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in § 50.4, information to support the determinations.”

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, "Rules for Inservice Inspection of Nuclear Power Plant Components," 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

ENCLOSURE

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 Part 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 fourth 10-year inservice inspection interval for Prairie Island Nuclear Generating Plant, Units 1 and 2, is the 1998 Edition through 2000 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI. The fourth 10-year ISI interval for Prairie Island Nuclear Generating Plant, Units 1 and 2 began on December 21, 2004.

3.0 TECHNICAL EVALUATION

Requests for Relief Nos. 1-RR-4-2 (Revision 0) and 2-RR-4-2 (Revision 0)

Component Identification

Code Class:	Class 1
Examination Category:	B-B and B-D
Item Number:	B2.60 and B3.160 (Inspection Program B)
Description:	Inspection of Regenerative Heat Exchanger Tubesheet-to-Head Welds and Nozzle Inner Radius Sections on the Primary Side of the subject exchanger.
Component Numbers:	See Table Below

Code Requirements
The 1998 Edition with 2000 Addenda of the ASME Code Section XI, Table IWB-2500-1, Category B-B, Item Number B2.60 requires a volumetric examination be performed on the primary side of the regenerative heat exchanger tubesheet-to-head weld. Examination Category B-D, Item Number B3.160 (Inspection Program B) requires a volumetric examination of the nozzle inside radius sections on the primary side of the subject exchanger.

Licensee's Proposed Alternative Examination (As Stated):
Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested to perform VT-2 visual examinations in lieu of the required volumetric examinations. As discussed below, this alternative will provide an acceptable level of quality and safety. The regenerative heat exchanger will receive a system leakage test prior to each unit startup after a refueling outage. During this system leakage test, the insulation will be removed from the affected areas and the components will receive a visual (VT-2) examination. The corresponding piping and component supports will continue to be inspected per the requirements of Section XI, as this relief does not affect them.

Licensee's Basis for Relief Request (As Stated):
The regenerative heat exchanger provides preheat for the normal charging water flowing into the reactor coolant system (RCS). Preheat is derived from normal letdown water coming from the RCS. The heat exchanger is actually three heat exchangers or sub-vessels of similar design and function. The Class 2, shell side of each heat exchanger has an outside shell diameter of 6.75 inches and is fabricated from SA312-304 austenitic stainless. The Class 1 primary head portion of the vessel is comprised of a hemispherical head fabricated from SA351-CF8 cast austenitic stainless and the forged tubesheet section of the vessel is fabricated from SA182-304 austenitic stainless and has integral penetrations adapted for connecting the 2 inch inlet and outlet process pipe by way of socket welding.

Ultrasonic examination (UT) of the head-to-tubesheet weld is limited to a single-sided examination from head side of the weld due to obstruction from the piping connection of the forged tubesheet. The UT examination is further hindered from the head side of the weld for obtaining meaningful ultrasonic data because of the hemispherical head material being cast austenitic stainless steel. UT sound beam attenuation and propagation properties in cast stainless steel are extremely difficult. It is recognized by the ASME Code Committee and the industry Performance Demonstration Initiative (PDI) that such examinations are difficult. ASME Section XI, Appendix VIII, Supplement 9 has been in "course of preparation" for several years, hence, there are no qualified examination procedures or personnel to conduct the required examinations.

UT of the nozzle inner radius sections of the forged tubesheet cannot be performed with reasonable assurance of obtaining satisfactory results because of geometric limitations of the nozzle section.

RCS leakage is closely monitored by the site in accordance with Prairie Island Technical Specifications (TSs). The TSs require that the RCS unidentified leakage rate be limited to one gallon per minute (gpm). In the event that the pressure boundary of the regenerative heat exchanger was breached with a throughwall leak, the leakage would be quickly detected.

In support of the proposed alternative, a similar Code Case is going through the ASME Code Committee. Over the course of two years, Westinghouse has conducted a survey of the regenerative heat exchanger service histories with respect to inspection costs, man-rem exposures, and safety being the major considerations in order to validate whether the current inspections required by the Code were warranted. No Westinghouse designed regenerative heat exchanger has exhibited leakage. Westinghouse also reviewed several relief requests and inspection reports to identify if there were any crack like indications found. None were reported. Westinghouse has also performed a series of weld joint fracture evaluations and finite element models. In all cases, the critical flaw depth exceeded the wall thickness. This leads to the deduction that there is no need to determine the flaw depth by volumetric examination.

The regenerative heat exchanger is a carefully designed and constructed component to ASME Code rules. The welds regions and nozzle inner radii of these components have not been designed for volumetric examination, and as such these examinations are time consuming and are dose intensive. These heat exchangers do not have a severe duty cycle, and service experience of the Westinghouse design has been good. Considering the low safety significance of these heat exchangers and large flaw tolerance, continuation of the volumetric and surface examinations results in an undo hardship without a commensurate increase in the level of quality and safety.

Previously, by letter dated November 15, 1995, Northern States Power (the licensee at the time) submitted Request for Relief No. 5 from the requirements of the ASME Code, Section XI, 1989 Edition, Table IWB-2500-1, Category B-B and B-D, regarding the inspection of the regenerative heat exchanger tubesheet-to-head welds and the nozzle inside radius sections. The NRC staff reviewed and evaluated the licensee's request for relief pursuant to 10 CFR 50.55a(g)(6)(i) [TAC No. M90187] and granted approval of a proposed alternative.

[Table 1 represents the list of components for which relief is requested]

<u>Table 1 Regenerative Heat Exchanger Tubesheet to Head Welds</u>

and Nozzle-to-Head Welds

Unit No.	ISI Summary No.	Code Item No.	Component Identification	Component Description
1	301077	B2.60	W-1	Tubesheet-to-Head Weld
1	301078	B2.60	W-2	Tubesheet-to-Head Weld
1	301079	B2.60	W-3	Tubesheet-to-Head Weld
1	303030	B3.160	N–1 IR	Nozzle Inner Radius
1	303031	B3.160	N–2 IR	Nozzle Inner Radius
1	303032	B3.160	N–3 IR	Nozzle Inner Radius
1	303033	B3.160	N–4 IR	Nozzle Inner Radius
1	303035	B3.160	N–5 IR	Nozzle Inner Radius
1	303037	B3.160	N–6 IR	Nozzle Inner Radius
2	501482	B2.60	W-1	Tubesheet-to-Head Weld
2	501536	B2.60	W-2	Tubesheet-to-Head Weld
2	501594	B2.60	W-3	Tubesheet-to-Head Weld
2	505024	B3.160	N–1 IR	Nozzle Inner Radius
2	505025	B3.160	N–2 IR	Nozzle Inner Radius
2	505026	B3.160	N–3 IR	Nozzle Inner Radius
2	505027	B3.160	N–4 IR	Nozzle Inner Radius
2	505028	B3.160	N–5 IR	Nozzle Inner Radius
2	505029	B3.160	N–6 IR	Nozzle Inner Radius

Response to a Request for Additional Information Dated January 28, 2005 (As Stated): Radiation levels near the regenerative heat exchangers are not solely dependent on these heat exchangers but on other equipment in the room, making it difficult to use shielding to significantly reduce dose. During the Unit 1 fall 2004 outage, one tubesbeet-to-head weld was ultrasonically examined. The system had been flushed prior to examination, to reduce personnel exposure, and the radiation levels remained high, at 320 mR/hour in the area and higher on contact.

Category B-B, Item Number B2.60

All six of the Examination Category B-B, Item Number B2.60 welds were volumetrically inspected during fabrication. No indications were identified during the course of these examinations.

No records of repairs to any of the subject welds were found.

Category B-D, Item Number B3.160

None of the Examination Category B-D, Item Number B3.160 inner radius sections have been ultrasonically inspected during fabrication, pre-service inspection, or inservice inspection.

No Fabrication/Pre-Service examinations have been found.

During the second interval, relief was requested from performing the required volumetric examination of the heat exchanger nozzle inside radius sections. The UT examination methods at that time did not adequately assess the condition of the specified examination volume. It was stated that an attempt would be made to perform the examinations if a proven outer surface, contact ultrasonic method was developed during the interval. The NRC approved this alternative by letter dated December 28, 1984.

During the third interval, relief was requested from performing the required volumetric examination of the heat exchanger nozzle inside radius sections. The nozzle area was visually inspected when the insulation was removed during the tubesheet-to-head weld examination. The NRC approved this alternative by letters dated November 15, 1995 and (TAC No. M90187) and February 22, 1996 (TAC No. M90186).

No records of repairs to any of the subject inner radius sections were found.

Abnormal Operating Procedure 1C4 AOPI, Reactor Coolant Leak

The Reactor Coolant Leak Abnormal Operating Procedure is in place to describe the methodology for determining the path of reactor coolant leakage and the necessary actions when symptoms associated with small reactor coolant leakage are detected.

The following symptoms will identify leakage from the regenerative heat exchangers and associated piping from the control room:

- Increased radiation levels
- Decreasing pressurizer level or a level deviation alarm
- Charging pump speed increase
- Decreasing volume control tank (VCT) level

The following automatic actions may occur:

- Charging pump speed increase in response to decreasing pressurizer level
- VCT automatic make-up
- Letdown isolation on low pressurizer level

If RCS inventory can not be maintained by available charging flow, then the reactor will be manually tripped and safety injection will be initiated.

Surveillance Procedure SP 1001AA, Daily RCS Leakage Test
In addition, the daily RCS leakage test is performed once every 24 hours to determine the leakage rate from the RCS. If the calculated leakage is greater than 0.1 gpm, then the identified leak rates are measured and subtracted to better quantify unidentified RCS leakage.

If the unidentified leakage rate is greater than 0.2 gpm or if the unidentified leakage rate is greater than 0.1 gpm and remains elevated for three consecutive days, then the RCS leakage investigation procedure to determine the unidentified leakage source(s) is performed.

Once 1C4 AOP1, reactor coolant leak, has been entered, an attempt will be made to determine the location of the leak. The following items would suggest a leak in containment:

- Increasing containment radiation levels
- Increasing containment temperature, pressure or humidity
- Sump level alarms
- Increasing fan coil condensate collection tank level

A containment entry would then be considered to determine the location of the leak.

If the leakage were identified to be from the regenerative heat exchanger or associated piping, they would be isolated.

If a containment entry was not made or the source of the leak could still not be identified, then an attempt to locate the leak would be made by sequentially isolating the service systems from the RCS. If the leakage were from the regenerative heat exchanger or associated piping, they would be isolated through this process.

A calculation has been performed to show that containment radiation monitor R-11 is capable of identifying a 1 gpm reactor coolant leak in containment. R-11 has been responsible for the identification of canopy seal leaks as small as 0.1 and 0.2 gpm in 1988 and 1994.

Staff Evaluation
The 1998 Edition with 2000 Addenda of the ASME Code, Section XI, Examination Category B-B Item B 2.60 and Examination Category B-D Item B3.160 require a volumetric examination to be performed on the regenerative heat exchanger tubesheet-to-head weld and nozzle inside radius section, respectively. Pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee proposed as an alternative to perform VT-2 examinations in lieu of the required volumetric examination. The licensee proposed that the VT-2 examinations of the regenerative heat exchanger will be performed during system leakage tests prior to each unit startup after a refueling outage. During this test, the licensee proposed to remove the insulation from the affected areas.

The shell side of each heat exchanger is fabricated from SA312-304 austenitic stainless steel, the Class 1 primary head portion of the vessel is a hemispherical head fabricated from SA351-CF8 cast austenitic stainless steel and the forged tubesheet section of the vessel is fabricated from SA182-304 austenitic stainless. Each heat exchanger has integral penetrations adapted for connecting the 2 inch inlet and outlet process pipe by way of socket welding. ASME Code-required volumetric examinations are limited to a single sided examination from head side of the weld, because the examination is obstructed by piping connected to the forged tubesheet.

Furthermore, UT examination of cast stainless steel is extremely difficult due to beam attenuation and propagation properties in cast stainless steel. All six of the subject tubesheet-to-head welds were volumetrically inspected during fabrication and no indications were identified during the course of these examinations. No records of repairs to any of the subject welds were found.

None of the inner radius sections have been ultrasonically inspected during fabrication, pre-service inspection, or ISI. The licensee's fabrication code did not require that the subject welds receive a volumetric examination. However, the subject welds were visually examined during the third interval. The inspection performed during the third 10-year interval was VT-1 examination in accordance with a request for relief approved by an NRC safety evaluation dated February 22, 1996. The licensee did not find any flaws in the last inspection and has not experienced any problems with the subject exchangers in approximately 30 years of operation at Prairie Island Nuclear Generating Plant, Units 1 and 2.

As noted above, the regenerative heat exchanger weld regions and nozzle inner radii of these components were not designed for volumetric examination, and as such volumetric examination of these regions are dose intensive. In the last inspection, even after flushing the regenerative heat exchanger, the dose rate was 320 mR/hr in the subject exchanger area and even higher on contact.

RCS leakage is closely monitored and, if unidentified leakage occurs in the subject regenerative heat exchanger, it would be detected quickly by the licensee. The licensee has developed the following symptoms that will allow leakage from the regenerative heat exchangers and associated piping to be identified from the control room: 1) increased radiation levels; 2) decreasing pressurizer level or a level deviation alarm; 3) charging pump speed increase; and 4) decreasing VCT level. In addition, the licensee identified the following automatic actions that may occur incase of a leakage: 1) charging pump speed increase in response to decreasing pressurizer level; 2) VCT automatic make-up; and 3) letdown isolation on low pressurizer level. The licensee also noted that if the RCS inventory can not be maintained by available charging flow, then the reactor will be manually tripped and safety injection will be initiated.

The licensee has Surveillance Procedure SP1001AA, Daily Reactor Coolant Svstern Leakage Test, in place that requires a RCS leakage test to be performed once every 24 hours to determine the leakage rate from the RCS. If the calculated leakage is greater than 0.1 gpm, then the identified leak rates are measured and subtracted to better quantify unidentified RCS leakage. If the unidentified leakage rate is greater than 0.2 gpm, or if the unidentified leakage rate is greater than 0.1 gpm and remains elevated for three consecutive days, then the licensee's RCS Leakage Investigation procedure to determine the unidentified leakage source(s) is performed.

In the event of detectable leakage, the subject components would be isolated. The licensee has several indicators that would suggest a leak in the containment. For example: 1) increasing containment radiation levels; 2) increasing containment temperature, pressure or humidity; 3) sump level alarms; and, 4) increasing fan coil condensate collection tank level. In addition, the licensee performed a calculation to show that containment radiation monitor R-11 is capable of identifying a 1 gpm reactor coolant leak in containment and should be able to detect leakage from the regenerative heat exchangers and associated piping.

Therefore, based on the above information, the NRC staff determined that compliance with the ASME Code volumetric examinations would result in hardship without a compensating increase in quality and safety and that the licensee's proposed alternative provides reasonable assurance of leakage and structural integrity of the regenerative heat exchangers and associated piping.

Request for Relief Nos. 1-RR-4-3 (Revision 0) and 2-RR-4-3 (Revision 0)

Component Identification

Code Class:	Class 1, 2, and 3
Category:	B-P
Item Number:	B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, B15.70
Description:	Pressure Retaining Bolting at Bolted Connections
Component Numbers:	All

Code Requirements

ASME Code, Section XI, 1998 Edition with 2000 Addenda, Subparagraph IWA-5250 (a)(2), requires a VT-3 examination of one bolt closest to the source of leakage.

Licensee's Proposed Alternative Examination (As Stated):

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested to perform a VT-1, visual examination in lieu of the required VT-3, visual examination in conjunction with ASME Section XI Code Case N–566-1 [*Corrective Action for Leakage Identified at Bolted Connections, Section XI*, *Division 1*]. As discussed below this will provide an acceptable level of quality and safety. [Note: Code Case N–566-1 has been incorporated in Regulatory Guide 1.147 *Inservice Inspection Code Case Acceptability ASME Section XI*, *Division 1*, Revision 13 for general use.]

Licensee's Basis for Relief Request (As Stated):

Prairie Island requests relief from the ASME Code, Section XI, 1998 Edition with 2000 Addenda, Subparagraph IWA-5250 (a)(2), regarding the actions to be taken when leakage occurs at a bolted connection on other than a gaseous system during the conduct of a system pressure test. Specifically, removal and examination of one bolt closest to the source of leakage would be by VT-1 visual examination in lieu of the Code required VT-3 visual examination. In addition PINGP [Prairie Island Nuclear Generating Plant] will use Code Case N–566-1 which establishes criteria for an engineering evaluation of the bolting condition as an alternative to removal and visual examination of all bolting.

The use of a VT-1 visual examination in lieu of the Code-required VT-3 visual examination will provide a comparable level of quality and safety. The ASME BSPV Code, Section XI, Table IWB-2500, Code Categories B-G-1 and B-G-2 require a VT-1 visual examination for Class 1 pressure retaining bolting. Guidance for performing VT-1 visual examinations of bolting are already incorporated within PINGP examination procedures. VT-1 visual examinations are considered more stringent than those associated with the VT-3 visual examination.

Staff Evaluation

If leakage is observed from a bolted connection ASME Code, Section XI, 1998 Edition with 2000 Addenda, Subparagraph IWA-5250 (a)(2), requires a VT-3 examination of one bolt closest to the source of leakage. The licensee proposed to perform a VT-1 as an alternative to the ASME Code required VT-3, visual examination in conjunction with the application of ASME Code Section XI Code Case N–566-1. Code Case N–566-1 establishes criteria for an engineering evaluation of the bolting condition as an alternative to removal and visual examination of all bolting.

When leakage occurs at a bolted connection on other than a gaseous system during the conduct of a system pressure test, the licensee will evaluate the bolted connection as required in Code Case N–566-1. If the engineering evaluation concludes that bolt is closest to the source of leakage, it should be removed and the bolt would be examined by a VT-1 examination in lieu of the ASME Code-required VT-3 examination. VT-1 examinations requirements are considered more stringent than those associated with the VT-3 visual examination. For example, VT-1 examinations are conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion or erosion. VT-3 visual examinations are conducted to determine the general mechanical and structural conditions of the component. Therefore, the NRC staff determined that the use of a VT-1 visual examination in lieu of the ASME Code-required VT-3 visual examination provides an acceptable level of quality and safety.

Request for Relief Nos. 1-RR-4-4 (Revision 0) and 2-RR-4-4 (Revision 0)

Component Identification

Code Class:	Class 1
Examination Category:	B-P
Item Number:	B15.50
Description:	NPS [Nominal Pipe Size] 1 inch Reactor Vessel flange leak-off connections from Reactor Vessel flange to 3/8" reducers.
Component Numbers:	Unit 1: Line Nos. 1-RC-9A and 9B Unit 2: Line Nos. 1-2RC-9A and 9B

Code Requirements

ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition 2000 Addenda, Table IWB-2500-1, Category B-P, Item No. B15.50 requires periodic VT-2 visual examination of Class 1 piping during the conduct of a system pressure test. This relief request involves Code requirements that mandate performance of a VT-2 visual examination during either the system pressure test or hydrostatic pressure test. Specifically, the requirement in Paragraph IWB-5221 (a) states a system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power, i.e., 2235 psig.

Licensee's Proposed Alternative Examination (As Stated):

Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is requested from the provisions of Table IWB-2500-1, Category B-P, Item No. B15.50 for performing the VT-2 visual examination using reactor coolant as a pressurizing medium at a test pressure of 2235 psig.

The required VT-2 visual examinations for the reactor vessel flange leak-off detection lines will be conducted during the regularly scheduled Class 1 system pressure test that is performed following each refueling outage. The reactor vessel flange leak-off lines will not be pressurized, during the VT-2 visual examinations, to RCS pressure (2235 psig) using reactor coolant as a pressuring medium. However, the examination will be conducted subsequent to pressurization of the reactor vessel flange leak-off lines with borated water during refueling operations. During refueling operations, the reactor vessel flange leak-off lines are pressurized due to the static head in the reactor cavity to approximately 10 psig. Since borated water leaves a crystalline residue, the proposed VT-2 visual examination provides reasonable assurance that through-wall leakage in the reactor vessel flange leak-off lines will be detected and corrected.

Licensee's Basis for Relief Request (As Stated):

The reactor pressure vessel head flange leakage detection lines are separated from the reactor pressure boundary by one passive membrane, a silver plated O-ring located on the vessel flange. A second O-ring is located on the opposite side of the tap in the vessel flange. This line is required during plant operation in order to indicate failure of the inner flange seal O-ring. Failure of the inner O-ring is the only condition under which these lines would be pressurized.

The configuration of this system precludes manual testing while the vessel head is removed because the odd configuration of the vessel tap, combined with the small size of the tap and the high test pressure requirement (2235 psig minimum), prevents the tap in the flange from being temporarily plugged. The hole opening in the flange is nominally only 0.815 inch in diameter and is smooth walled making a high pressure temporary seal very difficult. Failure of this seal could possibly cause ejection of the device used for plugging into the vessel.

A pneumatic test performed with the head installed is not practical due to the configuration of the top head. The top head of the vessel contains two grooves that hold the O-rings. The O-rings are held in place by a series of retainer clips. The retainer clips are contained in a recessed cavity in the top head. If a pressure test was performed from the leak-off line side with the head on, the inner O-ring would be pressurized in a direction opposite to what it would see in normal operation. This test pressure would result in a net inward force on the O-ring that would tend to push it into the recessed cavity that houses the retainer clips. The O-ring material is a thin silver plating and could very likely be damaged by this deformation into the recessed areas on the top head.

In addition to the problems associated with the O-ring design that make this testing impractical, it is also questionable whether a pneumatic test is appropriate for this line. Although the line will initially contain steam if the inner O-ring leaks, the system actually detects leakage rate by measuring the level of condensate in a collection chamber. This would make the system medium water at the level switch. Finally, the use of a pneumatic test performed at a minimum of 2335 psig would represent an unnecessary risk in safety for the inspectors and test engineers in the unlikely event of a test failure, due to the large amount of stored energy contained in air pressurized to 2235 psig. System leakage testing of these lines is impractical because the line will only be pressurized in the event of a failure of the inner O-ring. It is impractical to induce failure of the inner O-ring in order to perform a test.

Staff Evaluation

ASME Code Section XI, Table IWB- 2500-1, Category B-P, Item No. B15.50 requires periodic VT-2 visual examination of Class 1 piping during the conduct of a system pressure test at operating temperature and pressure.

The licensee proposed that the ASME Code required VT-2 visual examinations for the reactor vessel flange leak-off detection lines will be conducted during the regularly scheduled Class 1 system pressure test that is performed following each refueling outage. The reactor vessel flange leak-off lines will not be pressurized during the VT-2 visual examinations, to RCS pressure (2235 psig) using reactor coolant as a pressuring medium. The licensee noted that the examination will be conducted subsequent to pressurization of the reactor vessel flange leak-off lines with borated water during refueling operations. The licensee also noted that during refueling operations, the reactor vessel flange leak-off lines are pressurized due to the static head in the reactor cavity to approximately 10 psig.

The design of the reactor pressure vessel head flange makes the ASME Code- required leakage system tests impractical. The licensee’s sketch of the “Flange Leak Detection” system shows that the leakage detection lines are separated from the reactor pressure boundary by one passive membrane, a silver plated O-ring located on the vessel flange and a second O-ring is located on the opposite side of the tap in the vessel flange. This line is required during plant operation in order to indicate failure of the inner flange seal O-ring. The only condition under which the subject lines would be pressurized is the failure of the inner O-ring.

The licensee considered manual testing of the subject system while the vessel head is removed; however, the configuration of the vessel tap, combined with the small size of the tap and the high test pressure requirement (2235 psig minimum) prevents the tap in the flange from being temporarily plugged. It is very difficult to temporarily seal the tap because the hole opening in the flange is nominally only 0.815 inch in diameter and is smooth walled. Failure of this seal could possibly cause ejection of the device used for plugging into the vessel.

The licensee also considered a pneumatic test with the head installed; however, the licensee determined that it is not practical due to the configuration of the top head. There are two grooves in the top head of the vessel that hold the O-rings. The O-rings are held in place by a series of retainer clips. The retainer clips are contained in a recessed cavity in the top head. The inner O-ring would be pressurized in a direction opposite to what it would see in normal operation, if a pressure test was performed from the leak-off line side with the head on. As a result, a net inward force on the O-ring that would tend to push it into the recessed cavity that houses the retainer clips. The O-ring material has a thin silver plating which could very likely be damaged by this deformation into the recessed areas on the top head. Due to the large amount of stored energy contained in air pressurized to 2235 psig, the use of a pneumatic test would be a unnecessary risk for the inspectors and test engineers in the event of a test failure. Based on the licensee's description and the sketch provided of the reactor head seal, the NRC staff determined that the ASME Code requirements are impractical. The NRC staff determined that the subject components would have to be redesigned in order for the licensee to perform the ASME Code-required tests and to require the licensee to perform the tests would place a significant burden on the licensee.

As an alternative to the ASME Code required VT-2 examination during a system leakage test, the licensee proposed to perform the VT-2 examination subsequent to pressurization of the reactor vessel flange leak-off lines with borated water during refueling operations. The reactor vessel flange leak-off lines are pressurized during refueling operations to approximately 10 psig. In addition, the subject system contains borated water that leaves a crystalline residue which makes visual detection of leakage possible. Therefore, the licensee's proposed alternative provides reasonable assurance of structural integrity of the subject components.

4.0 CONCLUSIONS

The NRC staff has reviewed the licensee's submittial and concludes for Requests for Relief Nos. 1-RR-4-2 (Revision 0) and 2-RR-4-2 (Revision 0) that compliance with the ASME Code requirements would result in hardship without a compensating increase in quality and safety. Furthermore, the licensee's proposed alternative to perform a VT-2 visual examination and to monitor the RCS for leakage provides reasonable assurance of structural integrity of the subject regenerative heat exchanger components. Therefore, the licensee's proposed alternative contained in Request for Relief Nos. 1-RR-4-2 (Revision 0) and 2-RR-4-2 (Revision 0) is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year ISI program.

For Request for Relief Nos.1-RR-4-3 (Revision 0) and 2-RR-4-3 (Revision 0), the NRC staff concludes that the licensee's proposed alternative to perform a VT-1 visual examination in lieu of the ASME Code-required VT-3 visual examination of a bolt that is closest to the source of leakage, should it be removed after an engineering evaluation as noted in ASME Code Case N–566-1 is performed provides an acceptable level of quality and safety. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI program.

For Request for Relief Nos. 1-RR-4-4 (Revision 0) and 2-RR-4-4 (Revision 0), the NRC staff concludes that the ASME Code requirement to perform a VT-2 visual examination during the system leakage test is impractical due to the configuration of the 1 inch NPS Reactor Vessel flange leak-off connections from Reactor Vessel flange to 3/8" reducers. Furthermore, to require the licensee to perform the ASME Code required examinations would place a significant burden on the licensee because the subject components would have to be redesigned.

The NRC staff finds that the licensee's proposed alternative provides reasonable assurance of structural integrity of the subject components. Therefore, the licensee's request for relief is granted pursuant to 10 CFR 50.55a(a)(g)(6)(i) for the fourth 10-year ISI Interval. The NRC staff has determined that granting Request for Relief Nos. 1-RR-4-4 (Revision 0) and 2-RR-4-4 (Revision 0) pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property, or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Date: April 27, 2005