April 24, 2003

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. 10 AND 11 FOR THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM (TAC NOS. MB7820 AND MB7821)

Dear Mr. Solymossy:

By letter dated February 14, 2003, the Nuclear Management Company, LLC (NMC), submitted Relief Request Nos. 10 and 11 (RR-10 and RR-11) for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. In RR-10 and RR-11, NMC proposed an alternative to use the root mean square (RMS) value of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies the depth sizing criterion of ASME Code, Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c).

The enclosure provides the U. S. Nuclear Regulatory Commission (NRC) staff's safety evaluation for RR-10 and RR-11. As noted in the safety evaluation, the NRC staff concludes that NMC's proposed alternatives provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff authorizes NMC's proposed alternatives described in RR-10 and RR-11 for the third 10-year inservice inspection intervals.

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

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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 NO. 10 (UNIT 1) AND RELIEF REQUEST NO. 11 (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 February 14, 2003, the Nuclear Management Company, LLC (the licensee), submitted Relief Request Nos. 10 and 11 (RR-10 and RR-11) for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. In RR-10 and RR-11, the licensee proposed an alternative to use the root mean square (RMS) value of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies the depth sizing criterion of ASME Code, Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c). The licensee proposed this alternative for the third 10-year inservice inspection (ISI) Intervals at Prairie Island, Units 1 and 2.

ISI of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) Class 1, Class 2, and Class 3 components shall be performed in accordance with the ASME Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” and applicable editions and 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, in part, that alternatives to the requirements of paragraph (g) may be used, when authorized by the U. S. Nuclear Regulatory Commission (NRC), if the licensee demonstrates that: (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 preservice 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) on the date 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

## 2.0 CODE REQUIREMENTS FOR WHICH RELIEF IS REQUESTED

The licensee proposed an alternative to ASME Code, Section XI, Appendix VIII, Supplement 4, Subparagraph 3.2(c), which states that the UT performance demonstration results must be plotted on a two-dimensional plot with the measured depth plotted along the ordinate axis and the true depth plotted along the abscissa axis. For qualification, the plot must satisfy the following statistical parameters: (1) slope of the linear regression line is not less than 0.7, (2) the mean deviation of flaw depth is less than 0.25 inches, and (3) correlation coefficient is not less than 0.70.

## 3.0 LICENSEE'S PROPOSED ALTERNATIVE TO THE CODE

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee proposed an alternative to use the RMS value of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies the depth sizing criterion of ASME Code, Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c).

## 4.0 EVALUATION

The licensee proposed an alternative to ASME Code, Section XI, Appendix VIII, Supplement 4, Subparagraph 3.2(c) which imposes three statistical parameters for depth sizing. The first parameter, 3.2(c)(1), pertains to the slope of a linear regression line. The linear regression line is the difference between actual versus true value plotted along a through-wall thickness. For Supplement 4 performance demonstrations, a linear regression line of the data is not applicable because the performance demonstrations are performed on test specimens with flaws located in the inner 15-percent through-wall. The differences between actual versus true value produce a tight grouping of results, which resemble a shot gun pattern. The slope of a regression line from such data is extremely sensitive to small variations, thus making the parameter of Subparagraph 3.2(c)(1) a poor and inappropriate acceptance criterion. The second parameter, 3.2(c)(2), pertains to the mean deviation of flaw depth. The value used in the code is too lax with respect to evaluating flaw depths within the inner 15 percent of wall thickness. Therefore, the licensee proposed to use the more appropriate criterion of 0.15 inch RMS of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies Subparagraph 3.2(a), as the acceptance criterion. The third parameter, 3.2(c)(3), pertains to a correlation coefficient. The value of the correlation coefficient in Subparagraph 3.2(c)(3) is inappropriate for this application since it is based on the linear regression from Subparagraph 3.2(c)(1).

The United States nuclear utilities created the Performance Demonstration Initiative (PDI) to implement performance demonstration requirements contained in Appendix VIII of Section XI of the ASME Code. To this end, the PDI has developed a performance demonstration program for qualifying UT equipment, procedures, and personnel. The PDI was aware of the inappropriateness of Subparagraph 3.2(c) early in the development of its program. The PDI brought the issue before the appropriate ASME committee, which formalized eliminating the use of Supplement 4, Subparagraph 3.2(c) in Code Case N-622. The NRC staff representatives participated in the discussions and development process of the code case. Based on the above, the NRC staff believes that the use of Subparagraph 3.2(c) requirements in this context is inappropriate and that the proposed alternative to use the RMS value of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies the criterion of ASME Code, Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c), will provide an acceptable level of quality and safety[1].

5.0 CONCLUSION

Based on the discussion above, the NRC staff has concluded that the alternative proposed in RR-10 and RR-11 for the third 10-year interval will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff authorizes the proposed alternative for the third 10-year interval. All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: R. Hernandez

Date: April 24, 2003

[1] The information which would have been required for Appendix VIII, Supplement 4, Subparagraph 3.2(c)(1) is still required and valid for the sizing qualification of Appendix VIII, Supplement 6.