January 5, 2006

Mr. Thomas J. Palmisano
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 - ISSUANCE OF RELIEF REQUEST FOR THE RISK-INFORMED INSERVICE INSPECTION PROGRAM (TAC NOS. MC5644 AND MC5645)

Dear Mr. Palmisano:

By letter dated December 29, 2004, as supplemented by letter dated August 30, 2005, Nuclear Management Company, LLC (NMC) requested a relief to implement a risk-informed inservice inspection (RI-ISI) program as an alternative to the American Society of Mechanical Engineers (ASME) Code, Section XI, 1998 Edition with Addenda through 2000 for Class 1, Code Category B-J and B-F and Class 2, Code Category C-F-1 and C-F-2 piping welds at Prairie Island Nuclear Generating Plant, Units 1 and 2 during the fourth 10-year inservice inspection interval.

The proposed RI-ISI program was developed in accordance with the methodology contained in the U.S. Nuclear Regulatory Commission approved Electric Power Research Institute (EPRI) topical report EPRI TR-112657 Revision B-A with limited application of Code Case N–578 as contained in the scope of the above EPRI topical report.

The results of our review indicate that your proposed RI-ISI program is an acceptable alternative to the requirements of the ASME Code, Section XI for inservice inspection of Class 1 and 2 piping, Examination Categories B-F, B-J, C-F-1, and C-F-2 welds. Therefore, your request for relief is authorized for the fourth 10-year inspection interval (December 21, 2004, to December 20, 2014) at both units of Prairie Island pursuant to Title 10 of the *Code of Federal Regulation* Section 50.55a(a)(3)(i) on the basis that the alternative provides an acceptable level of quality and safety.

Sincerely,

/RA by Tim Kobetz for/

L. Raghavan, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures: Safety Evaluation

cc w/encls: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RISK-INFORMED INSERVICE INSPECTION PROGRAM

NUCLEAR MANAGEMENT COMPANY, LLC

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated December 29, 2004, as supplemented by letter dated August 30, 2005, Nuclear Management Company, LLC (NMC) requested a relief to implement a risk-informed inservice inspection (RI-ISI) program as an alternative to the American Society of Mechanical Engineers (ASME) Code, Section XI, 1998 Edition with Addenda through 2000 for Class 1, Code Category B-J and B-F and Class 2, Code Category C-F-1 and C-F-2 piping welds at Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2 during the fourth 10-year inservice inspection interval. The licensee's program references the ASME Section XI, Code Case N–578 and the Electric Power Research Institute (EPRI) topical report (TR) EPRI TR-112657, Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," dated December 1999, which was previously reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC). The NRC staff has previously accepted certain segments of Code Case N – 578 in EPRI TR-112657 inspection methodology but not the code case in its entirety. Therefore, the licensee's use of Code Case N–578 will be limited to the extent contained in the scope of the above EPRI topical report.

The licensee proposed the RI-ISI program as an alternative to the requirements in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i).

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements set forth in Section XI of the Code to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations of 10 CFR 50.55a(g) also state that ISI of the ASME Code, Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Code and applicable addenda, except where specific written relief has been granted by the NRC. The objective of the ISI program as described in Section XI of the ASME Code and applicable addenda is to identify conditions (i.e., flaw indications) that are precursors to leaks and ruptures in the pressure boundary of these components that may impact plant safety. The regulations also require that, during the first 10-year ISI interval and during subsequent intervals, the licensee's ISI program complies with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference into 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. Prairie Island began its fourth 10-year interval on December 21, 2004, using the 1998 Edition of Section XI of the ASME Code with addenda through 2000.

According to 10 CFR 50.55a(a)(3)(i), the NRC Director of the Office of Nuclear Reactor Regulation may authorize alternatives to the requirements of 10 CFR 50.55a(g), if an applicant demonstrates that the proposed alternatives would provide an acceptable level of quality and safety.

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," defines the safety principles for an acceptable RI-ISI program as follows:

(1) The proposed change meets current regulations unless it is explicitly related to a requested exemption.

(2) The proposed change is consistent with the defense-in-depth philosophy.

(3) The proposed change maintains sufficient safety margins.

(4) When proposed changes result in an increase in risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

(5) The impact of the proposed change should be monitored using performance measurement strategies.

In addition, RG 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping," describes methods acceptable to the NRC staff for integrating insights from probabilistic risk assessment (PRA) techniques with traditional engineering analyses into ISI programs for piping, and addresses risk-informed approaches that are consistent with the basic elements identified in RG 1.174.

The licensee has proposed to use an RI-ISI program for ASME Class 1 and 2 piping (Examination Categories B-F, B-J, C-F-1, and C-F-2 welds) as an alternative to the ASME Code, Section XI requirements. The licensee states that this proposed program was developed using the RI-ISI methodology described in EPRI TR-112657 and Code Case N–578. The NRC staff's safety evaluation (SE) of October 28, 1999, approving the methodology described in the TR, concluded that the methodology conforms to guidance provided in RGs 1.174 and 1.178, and that no significant risk increase should be expected from the changes to the ISI program resulting from applying the methodology.

The transmittal letter for this SE stated that an RI-ISI program as described in the TR utilizes a sound technical approach and will provide an acceptable level of quality and safety. It also stated that, pursuant to 10 CFR 50.55a, any RI-ISI program meeting the requirements of the TR provides an acceptable alternative to the piping ISI requirements with regard to (1) the number of locations, (2) the locations of inspections, and (3) the methods of inspection.

3.0 TECHNICAL EVALUATION

Pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff has reviewed and evaluated the licensee's proposed RI-ISI program based on guidance and acceptance criteria provided in the following documents:

- RGs 1.174 and 1.178
- NRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 3.9.8
- EPRI TR-112657 and its NRC SE

3.1 Proposed Changes to the ISI Program

The scope of the licensee's proposed changes to its ISI program is limited to ASME Code Class 1 piping welds in the Examination Categories B-F for pressure retaining dissimilar metal welds in piping and B-J for pressure retaining welds in piping, and to Code Class 2 piping welds in the Examination Categories C-F-1 and C-F-2. The RI-ISI program is proposed as an alternative to the existing ISI requirements of the ASME Code, Section XI, 1998 Edition with Addenda through 2000.

The program changes would result in the number and locations of non-destructive examination (NDE) inspections based on ASME Code Section XI requirements being replaced by the number and locations of these inspections based on RI-ISI guidelines. The ASME Code requires, in part, that for each successive 10-year ISI interval, 100 percent of Category B-F welds and 25 percent of Category B-J welds for the Code Class 1 non-exempt piping be selected for volumetric and/or surface examination based on existing stress analyses and cumulative usage factors. Categories C-F-1 and C-F-2 require examination of 7.5 percent of the total number of non-exempt and exempt welds, but not less than 28 welds. RI-ISI selects welds for inspection based on high, medium, and low risk-significance as opposed to ASME Code Class. The proposed RI-ISI program selects 22 out of 85 piping welds in high risk region, 88 out of 830 piping welds in medium risk region of Unit 1, 24 out of 90 in high risk region, and 89 out of 838 welds in medium risk region of Unit 2 for NDE and will be implemented during the first period of the fourth 10-year inspection interval. Stated in terms of ASME Code Class, 69 out of 690 Class 1 piping welds and 41 out of 933 Class 2 piping welds in Unit 1 and 73 out of 705 Class 1 piping welds and 40 out of 951 Class 2 piping welds in Unit 2 are selected for examination. The augmented inspection program for flow accelerated corrosion (FAC) per Generic Letter 89-08 at both units of Prairie Island will not be affected or changed as a result of the proposed RI-ISI program. All locations within the Class 1 and 2 systems will receive a system pressure test and VT-2 visual examination in accordance with the applicable ASME Code, Section XI regardless of its risk classification.

3.2 Engineering Analysis

In accordance with the guidance provided in RGs 1.174 and 1.178, the licensee provided the results of an engineering analysis of the proposed changes, using a combination of traditional engineering analysis and supporting insights from the PRA. The licensee performed an evaluation to determine the susceptibility of components (i.e., a piping weld) to a particular degradation mechanism that may be a precursor to leak or rupture, and then performed an independent assessment of the consequence of a failure at that location. The results of this analysis assure that the proposed changes are consistent with the principles of defense-in-depth because the EPRI TR-112657 methodology requires that the population of welds with high consequences following failure will always have some weld locations inspected regardless of the failure potential. No changes to the evaluation of design-basis accidents in the updated final safety analysis report (UFSAR) are being made by the RI-ISI program. Therefore, sufficient safety margins will be maintained.

3.2.1 Failure Potential

Piping systems within the scope of the RI-ISI program are divided into piping segments. Pipe segments are defined as lengths of pipe whose failure (anywhere within the pipe segment) would lead to the same consequence and which are exposed to the same degradation mechanisms, i.e., some lengths of pipe whose failure would lead to the same consequence may be split into two or more segments when two or more regions are exposed to different degradation mechanisms. The licensee states that the failure potential assessment, summarized in Tables 3.3-1 and 3.3-2 of the December 29, 2004, submittal, was accomplished utilizing industry failure history, plant-specific failure history, and other relevant information using the guidance provided in the TR.

The NRC staff concludes that the licensee has met the Standard Review Plan (SRP) 3.9.8 guidelines to confirm that a systematic process was used to identify the component's (i.e., pipe segments) susceptibility to common degradation mechanisms, and to categorize these degradation mechanisms into the appropriate degradation categories with respect to their potential to result in a postulated leak or rupture.

3.2.2 Consequence Analysis

The licensee states that the consequences of pressure boundary failures were evaluated and ranked based on their impact on core damage and containment performance (isolation, bypass, and large early release). Also, the licensee indicated that impact on the above measures due to both direct and indirect effects was considered. The licensee reports no deviations from the approved consequence evaluation guidance provided in the TR. Therefore, the NRC staff considers the consequence analysis performed by the licensee for this application to be acceptable.

3.2.3 PRA

The original PINGP, Units 1 and 2 individual plant examination (IPE), was submitted to the NRC in February 1994. The IPE estimated a core damage frequency (CDF) of 5E-05/yr. The NRC's evaluation of the IPE, issued in May 1997, did not identify any significant weaknesses, and concluded that the IPE met the intent of Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities."

As stated in the December 29, 2004, submittal, the licensee used the PINGP, Units 1 and 2 living PRA for evaluation of the consequences of pipe ruptures. The licensee stated that the specific PRA model used for this application was the PRA results based on the Revision 1.2 update. The licensee states that the baseline CDF estimated from Revision 1.2 of its PRA covers both Level 1 and Level 2 consequences, and addresses accidents initiated at full power.

A Westinghouse Owners Group (WOG) PRA Peer Certification Review was conducted on the 1999 update PRA model (Revision 1.1) in September 2000. In general, the review team concluded that the PINGP PRA could effectively be used to support applications involving risk significance determinations supported by deterministic analysis once the Facts and Observations (F&Os) noted in the report were addressed. A majority of the F&Os (including all of the Level A findings) have been addressed in the Revision 1.2 model. The licensee stated that while addressing the remaining “Facts and Observations” would take time to resolve completely, this is not expected to result in model changes that could significantly affect the overall results or conclusions of the Risk-Informed ISI consequence evaluation.

In Reference 2, the licensee provided a table of all closed Levels A and B F&Os, including the manner in which they have been resolved, and the open Level B F&Os. Several of the open F&Os would have no impact on the RI-ISI program if they had been resolved. Several open F&Os observed that common cause parameter estimates and component failure parameter estimates have not been updated with more current generic data and with plant-specific data. The final open F&O suggests improving the methodology for estimating pre-initiator human error probabilities (HEPs). The licensee stated that these data parameters and HEPs had not been updated before the RI-ISI program development was completed, but that they will be updated as part of an update project. Any impact of this update on the RI-ISI program will be incorporated as part of the living program aspect of the RI-ISI program. The RI-ISI methodology assigns segments into three broad consequence categories and the staff expects any changes arising from the parameter and pre-initiator HEPs updates will have a minimal effect on the placement of the segments, and that minor changes are expected as part of the long-term implementation of RI-ISI. Therefore, the staff finds that resolving the remaining F&Os as part of the implementation of the living program requirements for RI-ISI programs is acceptable.

The NRC staff did not review the PRA models to assess the accuracy of their quantitative estimates. The NRC staff recognizes that the quantitative results of the PRA model are used as order of magnitude estimates to support the assignment of segments into three broad consequence categories. Inaccuracies in models or in assumptions large enough to invalidate the broad categorizations developed to support the RI-ISI should have been identified during the NRC staff’s review of the IPE, and by the licensee’s model update control program, which included peer review/certification of the PRA model. Minor errors, inappropriate assumptions, and resolving the remaining F&Os are expected to affect only the consequence categorization of a few segments and will not invalidate the general results or conclusions.

As required by Section 3.7 of EPRI TR-112657, the licensee has evaluated the change in risk expected from replacing the current Section XI ISI program with the RI-ISI program.
The analysis estimates the net change in risk due to the positive or negative influence of adding or removing locations from the inspection program. The expected change in risk is quantitatively evaluated using the “Simplified Risk Quantification Method” described in Section 3.7 of EPRI TR-112657 and is shown in the table below.

Unit Number	Change in Risk ([a]CDF)		Change in Risk ([a]LERF)	
	With increased POD	Without increased POD	With increased POD	Without increased POD
Unit 1	-2.19E -07	-7.52E -08	-2.19E -08	-7.52E -09
Unit 2	-2.16E -07	-6.05E -08	-2.16E -08	-6.05E -09

POD = probability of detection

The NRC staff finds the licensee's process to evaluate and bound the potential change in risk reasonable because it accounts for the change in the number and location of elements inspected, recognizes the difference in degradation mechanism related to failure likelihood, and considers the effects of enhanced inspection. All system level and aggregate estimates of the changes in CDF and LERF are less than the corresponding guideline values in EPRI TR-112657. The NRC staff finds that redistributing the welds to be inspected with consideration of the safety-significance of the segments provides assurance that segments whose failure has a significant impact on plant risk receive an acceptable and often improved level of inspection. Therefore, the staff finds that the change in risk estimate is appropriate and the results provide assurance that the fourth key principle in RG 1.178 (any risk increase in CDF or risk should be small and consistent with the intent of the Commission's Safety Goal Policy Statement) is met.

3.2.4 Integrated Decisionmaking

The licensee used an integrated approach in defining the proposed RI-ISI program by considering in concert the traditional engineering analysis, the risk evaluation, the implementation of the RI-ISI program, and performance monitoring of piping degradation. This is consistent with the guidelines given in RG 1.178 and is, therefore, acceptable.

3.2.4.1 Risk Characterization

The licensee states in its December 29, 2004, submittal, that pipe segments (and ultimately the elements within, which are defined as all having the same degradation susceptibility) are ranked in accordance with definitions given in the TR and is, therefore, acceptable.

3.2.4.2 Selection of Element Population for Inspection

The licensee has opted to use the element selection guidance provided in EPRI TR-112657 under Section 3.6.4.2 "ASME Code Case N–578." The NRC staff notes that the specific requirement in the TR requires that at least 25 percent of the locations in each high risk category and that at least 10 percent of the locations in each medium risk category must be selected for NDE.

The licensee provides detailed information on the results of the evaluation in the following tables of its submittal:

- Tables 3.1-1 and 3.1-2 identify on a per system basis, the number of segments and number of elements (welds) for PINGP, Units 1 and 2.

- Tables 3.3-1 and 3.3-2 “Failure Potential Assessment Summary” provide the degradation mechanism assessment summary for each unit of Prairie Island.

- Tables 3.4-1 and 3.4-2 identify on a per system basis, the number of segments by risk category for each unit of Prairie Island.

- Tables 3.5-1 and 3.5-2 identify on a per system basis, the number of elements selected for inspection by risk category excluding the impact of FAC for each unit of Prairie Island.

- Tables 3.6-1 and 3.6-2 provide the risk impact analysis results for each system, as well as a summary comparing the number of inspections performed under the 1989 ASME Code, Section XI, ISI program with that of the alternative RI-ISI program for each unit of Prairie Island.

- Tables 5-1-1 and 5-1-2 provide a comparison of selection of inspection locations between the ASME Code, Section XI and the EPRI TR-112657 by risk region.

- Tables 5-2-1 and 5-2-2 provide a comparison of selection of inspection locations between the ASME Code, Section XI and the EPRI TR-112657 by risk category.

In reviewing the tables above, the NRC staff concludes that EPRI TR-112657's requirement that at least 25 percent of the locations in each high risk category, and that at least 10 percent of the locations in each medium risk category must be selected for NDE has been met.

In its December 29, 2004, submittal, the licensee reported that 10 percent of Unit 1 and 10.4 percent of Unit 2 Class 1 piping welds were selected for RI-ISI NDEs. Section 3.6.4.2 of EPRI TR-112657 states that if the percentage of Class 1 piping locations selected for examination falls substantially below 10 percent, then the basis for selection needs to be investigated. The licensee has met this expectation of the TR, and no investigation is required.

Based on the NRC staff’s review of the above tables (containing the results of element selection), the staff concludes that the element selection results are consistent with the described process and with EPRI TR-112657 guidelines. Hence, the licensee’s selection of element locations, which includes consideration of degradation mechanisms in addition to those covered by augmented inspection programs, is judged to be acceptable.

3.2.4.3 Examination Methods

As noted in Section 2.0 of this SE, the objective of ISI is to identify conditions (i.e., flaw indications) that are precursors to leaks and ruptures in the pressure boundary that may impact plant safety. To meet this objective, the RI-ISI location selection process, per EPRI TR-112657, employs an “inspection for cause” approach. To address this approach, Section 4 of the same TR provides guidelines for the areas and/or volumes to be inspected, as well as the examination method, acceptance standard, and evaluation standard for each degradation mechanism. Based on its review and acceptance of the TR, the NRC staff concluded that these examination methods are appropriate since they are selected based on specific degradation mechanisms, pipe sizes, and materials of concern. The licensee states that

Section 4 of the TR was used as guidance in determining the examination methods and requirements for these locations.

Based on these considerations, the NRC staff concludes that the licensee's determination of examination methods is acceptable.

3.2.4.4 Relief Requests for Examination Locations and Methods

The licensee states that the examination locations are selected such that a minimum of >90 percent volumetric coverage (i.e., Code Case N–460 criteria) is attainable. In instances where greater than 90 percent volumetric coverage cannot be obtained at an examination location, the process outlined in EPRI TR-112657 will be followed.

As required by Section 6.4 of EPRI TR-112657, the licensee has completed an evaluation of existing relief requests to determine if any should be withdrawn or modified due to changes that occur from implementing the RI-ISI program. The licensee states that no existing relief requests are being withdrawn due to the RI-ISI application.

The NRC staff finds that the licensee's proposed treatment of existing relief requests to be acceptable.

3.2.5 Implementation and Monitoring

Implementation and performance monitoring strategies require careful consideration by the licensee and are addressed in Element 3 of RG 1.178 and the SRP 3.9.8. The objective of Element 3 is to assess performance of the affected piping systems under the proposed RI-ISI program by utilizing monitoring strategies that confirm the assumptions and analyses used in the development of the RI-ISI program. The licensee states that upon approval of the RI-ISI program, procedures that comply with EPRI TR-112657 guidelines will be prepared to implement and monitor the RI-ISI program.

The licensee indicates in Section 4 of the December 29, 2004, submittal, that the RI-ISI program is a living program and its implementation will require feedback of new relevant information to ensure the appropriate identification of safety significant piping locations. The licensee also states that, as a minimum, risk ranking of piping segments will be reviewed and adjusted on an ASME period basis and that significant changes may require more frequent adjustment as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant-specific feedback. This periodic review and adjustment of the risk-ranking of segments ensure that changes to the PRA that the licensee will make will also be incorporated into the RI-ISI program as necessary. These changes should include those caused by resolution of the remaining F&Os from the WOG PRA Peer Certification Review.

The licensee addresses additional examinations in Section 3.5.1 of the December 29, 2004, submittal, which states that examinations performed that reveal flaws or relevant conditions exceeding the applicable acceptance standards shall be extended to include additional examinations. These additional examinations shall include piping structural elements with the same root cause conditions or degradation mechanisms. The additional examinations will include high risk significant elements and medium risk significant elements, if needed, up to a number equivalent to the number of elements initially required to be inspected on the segment or segments during the current outage. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined during the current outage. No additional examinations will be performed if there are no additional elements identified as being susceptible to the same root cause conditions. The staff finds the licensee's approach acceptable since the additional examinations, if required, will be performed during the outage that the indications or relevant conditions are identified.

The NRC staff finds that the proposed process for RI-ISI program implementation, monitoring, feedback, and update meets the guidelines of RG 1.174 which states that risk-informed applications should include performance monitoring and feedback provisions. Hence, the licensee's proposed process for program implementation, monitoring, feedback, and update is judged to be acceptable.

4.0 CONCLUSIONS

Pursuant to 10 CFR 50.55a(a)(3)(i), alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if the licensee demonstrates that the proposed alternatives will provide an acceptable level of quality and safety. In this case, the licensee has proposed an alternative to use the risk-informed process described in NRC-approved EPRI TR-112657.

The methodology in EPRI TR-112657 also provides for implementation and performance monitoring strategies to ensure a proper transition from the current ISI program, and to assure that changes in plant performance, and new information from the industry and/or from the NRC, is incorporated into the licensee's ISI program as needed.

Other aspects of the licensee's ISI program, such as system pressure tests and visual examination of piping structural elements will continue to be performed on all Class 1, 2, and 3 systems in accordance with the ASME Code, Section XI. This provides a measure of continued monitoring of areas that are being eliminated from the NDE portion of the ISI program. As required by EPRI TR-112657 methodology, the existing ASME Code performance measurement strategies will remain in place. In addition, EPRI TR-112657 methodology provides for increased inspection volumes for those locations that are included in the NDE portion of the program.

The NRC staff concludes that the licensee's development of its RI-ISI program is consistent with the methodology described in the TR. Hence, the NRC staff concludes that the licensee's proposed program is consistent with the methodology as described in the TR, and will provide an acceptable level of quality and safety pursuant to 10 CFR 50.55a(a)(3)(i) for the proposed alternative to the piping ISI requirements with regard to (1) the number of locations, (2) the locations of inspections, and (3) the methods of inspection.

The NRC staff concludes that the licensee's proposed RI-ISI program is an acceptable alternative to the current ISI program for Class 1 and 2 piping welds at Prairie Island, Units 1 and 2. Therefore, the proposed RI-ISI program is authorized for the fourth 10-year ISI interval pursuant to 10 CFR 50.55a(a)(3)(i) on the basis that this alternative will provide an acceptable level of quality and safety. 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.

5.0 REFERENCES

1. Letter from Joseph M. Solymossy, Nuclear Management Company, dated December 29, 2004, to U.S. Nuclear Regulatory Commission, Relief Request to Implement Risk-Informed Inservice Inspection (ISI) Scheduling for the Fourth 10-Year Inspection Interval for Prairie Island Units 1 and 2.

2. Letter from Thomas J. Palmisano, Nuclear Management Company, dated August 30, 2005, to U.S. Nuclear Regulatory Commission, Response to NRC Questions Regarding the "Relief Request to Implement Risk-Informed Inservice Inspection (ISI) Scheduling for the Fourth 10-Year Inspection Interval for Prairie Island Units 1 and 2".

3. EPRI TR-112657, Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Final Report, December 1999.

4. Safety Evaluation Report Related to "Revised Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)," October 28, 1999.

5. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.

6. NRC Regulatory Guide 1.178, Revision 1, "An Approach for, Plant-Specific, Risk-Informed Decisionmaking for Inservice Inspection of Piping," September 2003.

7. NRC NUREG-0800, Chapter 3.9.8, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, "Risk-Informed Inservice Inspection of Piping," September 2003.

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Administrator
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Prairie Island Indian Community
ATTN: Environmental Department
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