NMC
Committed to Nuclear Excellence

Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

L-PI-03- 055
10 CFR 50 Appendix A, GDC-4

June 2, 2003

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKET 50-282
LICENSE DPR-42

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING THE APPLICATION OF LEAK-BEFORE-BREAK TECHNOLOGY TO REPLACEMENT STEAM GENERATOR DESIGN (TAC NO. MB7509)

By letter dated January 29, 2003 the Nuclear Management Company, LLC (NMC) submitted a letter requesting the Nuclear Regulator Commission (NRC) provide confirmation that the replacement steam generators may be designed on the basis of the reduced loads resulting from the application of leak-before-break criteria to postulated breaks in the primary reactor coolant loop piping. By letter dated April 23, 2003, the NRC identified that additional information was needed. In accordance with the request in the NRC's April 23, 2003 letter, the NMC is providing the additional information in Attachment 1 to this letter.

In order to meet the replacement steam generator project schedule, the NMC requests that by July 11, 2003, the NRC provide confirmation that the replacement steam generators may be designed on the basis of the reduced loads resulting from the elimination of the dynamic effects of large ruptures in the main loop primary coolant piping.

This letter contains no new commitments and no revisions to existing commitments.

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Please address any comments or questions regarding this letter to Mr. H Oley Nelson at 1-651-388-1121.

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Project Manager, Prairie Island Nuclear Generating Plant, USNRC, NRR
NRC Resident Inspector - Prairie Island Nuclear Generating Plant

Attachment: Response to Request for Additional Information Regarding the Application of Leak-Before-Break Technology to Replacement Steam Generator Design

Attachment

Response to Request for Additional Information Regarding the Application of Leak-Before-Break Technology to Replacement Steam Generator Design

References:

1. United States Nuclear Regulatory Commission Safety Evaluation Related to the Elimination of Large Primary Loop Ruptures as a Design Basis, Prairie Island Nuclear Generating Plant Units 1 and 2, Docket Nos. 50-282 and 50-306, 12/22/86.
2. Northern States Power Company letter dated October 24, 1984 "Elimination of Large Reactor Coolant System Pipe Rupture from Structural Design Basis"
3. Northern States Power Company letter dated October 21, 1985 "Request for Exemption from the Requirements of 10 CFR Part 50, Appendix A, GDC-4"
4. Northern States Power Company letter dated November 5, 1985 "Supplemental Information Related to Request for Exemption from the Requirements of 10 CFR Part 50, Appendix A, GDC-4"
5. Nuclear Management Company, LLC letter L-PI-03-012 dated January 29, 2003 "Application of Leak-Before-Break Technology to Replacement Steam Generator Design for Prairie Island Nuclear Generating Plant""

Requested item 1

Does "re-analysis" mean: (a) reactor coolant loop (RCL) reanalysis and requalification? OR (b) leak-before-break (LBB) re-demonstration by reevaluation or validation of the original evaluation?

Response to item 1

1(a)

In reference 5, "re-analysis" means reactor coolant loop re-analysis. As part of the licensing support for replacement of the steam generators, Framatome ANP, Inc. performed a structural analysis of the reactor coolant system. The structural analysis determined loads at locations throughout the reactor coolant loop that are compared to the original loads. In cases where the new loads are larger than the original loads, a stress analysis is performed. This methodology demonstrates that the reactor coolant system, with the replacement steam generators remains in compliance with current design and licensing basis requirements.

The reactor coolant system structural analysis performed by Framatome ANP, Inc. employs full two loop models of the Prairie Island Unit 1 reactor coolant system containing the replacement steam generators. These models are used to determine the system displacements and loads due to pressure, deadweight, thermal expansion, operational basis earthquake, safe shutdown earthquake, and high energy line break. The results of these loading analyses are then combined using load combinations specified in the Prairie Island Updated Safety Analysis Report.

1(b)

The results of the reactor coolant loop structural evaluation will be used in demonstrating that the assessment of margins and conclusions in the leak-before-break submittal (references 2, 3, & 4) do not change. It is expected that this demonstration will show that the criteria used by the NRC in approving the original evaluation for the elimination of large reactor coolant system pipe ruptures from the Prairie Island Unit 1 structural design basis remains valid.

Requested item 2

Does NMC plan to submit results of 1(b) above for the NRC staff's review and approval whether new LBB loads are enveloped by the original LBB loads, or LBB is re-evaluated and re-demonstrated?

Response to item 2

It is expected that the validation discussed in the answer to requested item number 1 above will confirm that the assessment of margins and conclusions in the leak-before-break submittal (references 2, 3, & 4) do not change. Thus the bases of the NRC's safety evaluation (reference 1) will not change. Therefore there are no plans to submit the results of the validation of the original evaluation to the NRC.

Requested item 3

Does NMC plan to make a statement in response to (2) above that there are no new breaks due to cutting and welding of the steam generator replacement project?

Response to item 3

The plant modification process will be used to control the installation of the replacement steam generators. This process will ensure that there are no new breaks due to cutting and welding of the reactor coolant system piping during installation. This will be documented in the 10CFR50.59 evaluation(s) supporting the modification.

Requested item 4

The submittal stated that the RCL will be designed for the postulated breaks in the largest branch lines (surge, residual heat removal, safety injection system/accumulator injection). Are postulated breaks in the main feedwater and main steam lines going to be reanalyzed? Will they be reanalyzed for the same break locations and conditions?

Response to item 4

Leak-Before-Break technology is not being applied to the main feedwater and main steam lines. Therefore, double-ended guillotine breaks are postulated in the main feedwater and main steam lines at the replacement steam generator component nozzles, consistent with current licensing basis. The initial conditions used in the analysis are the most severe normal operating conditions considering the various ranges in operating temperatures and pressures.

Requested item 5

Are there any support changes associated with the steam generator replacement project?

Response to item 5

There are no planned changes for the replacement steam generator supports or the other reactor coolant system component supports. Minor changes may be made during the installation, such as the method of shimming. The plant modification process will be utilized to control such changes. If they occur, the process will ensure that the changes are evaluated to assess any impact they may have on the stiffness and stresses in the support systems and the reactor coolant loop structural analysis.

Requested item 6

Relating to the reactor coolant system (RCS) reanalysis, question 1(a) above, for thermal and seismic conditions, are there any stiffness changes which may impact the natural frequencies of the RCS and structural responses at controlling locations for LBB demonstration? Please explain how these impacts are addressed in question1(b) above.

Response to item 6

There are no planned changes made in the design of the replacement steam generators or the supports that would cause a change in the effective stiffness of the support system. As mentioned in the response to requested item number 5 above, any minor changes made to facilitate installation will be evaluated to assess any impact they may have on the stiffness and stresses in the support systems and the reactor coolant loop structural analysis.

There are no changes in the analysis of the reactor coolant system supports that would cause a change in the effective stiffness of the support system. The reactor coolant system structural analysis uses the same approach that was used in the original reactor coolant system structural analysis to determine active steam generator and reactor coolant pump supports in the thermal and seismic analyses. Thus it would be expected that the natural frequencies of the reactor coolant system be equivalent to the original analysis. However, the demonstration of the assessment of margins and conclusions in the leak-before-break submittal discussed in 1(b) above is not dependant on a comparison of the natural frequency. The demonstration is dependant on and will utilize the loads, stresses and bending moments determined in the reactor coolant loop re-analysis.

Requested Item 7

How does NMC plan to document the RCL reanalyses associated with the replacement steam generator? Will the reanalyses be governed by the same design specifications as the original steam generator?

Response to Item 7

The reactor coolant loop analysis with the replacement steam generators installed was performed and documented in accordance with Framatome ANP, Inc. Quality Assurance program. These documents are uniquely identified and will be retained in the record management systems at both Framatome ANP, Inc. and Prairie Island. These documents will also be part of the modification package(s) for the replacement steam generators and as such will be reflected in Prairie Island's Updated Safety Analysis Report as appropriate.

The original steam generators were designed and fabricated to the American Society of Mechanical Engineers Code (ASME) Section III 1965 Edition, Summer 1966 Addenda. The replacement steam generators are designed and fabricated to ASME 1995 Edition, 1996 Addenda. A reconciliation of differences in design requirements and loading conditions will be performed for the replacement steam generators. This reconciliation will include the requirements of ASME Section XI IWA-7210(c)(2) and IWA-7210(c)(3).

The structural analysis prescribed in the replacement steam generator certified design specification are consistent with the original steam generator design specification and Prairie Island's Updated Safety

Analysis Report. The replacement steam generator certified design specification is based on the original steam generator design specifications with the following improvements:

1. Design loads (to be used in the ASME design report) applied to the secondary shell, primary nozzles and support points are determined by taking the higher of the loads found in the original steam generator design specification or the loads determined through the reactor coolant loop reanalysis.
2. More detailed transient temperature and flow information is provided for the secondary side components for use in the ASME code design report to support a detailed Class 1 fatigue analysis of the replacement steam generators.