NMC
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Prairie Island Nuclear Generating Plant
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U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42

Application of Leak-Before-Break Technology to Replacement Steam Generator Design for Prairie Island Nuclear Generating Plant

References:

1. NRC Safety Evaluation Related to the Elimination of Large Primary Loop Ruptures as a Design Basis, PINGP Units 1 and 2, Docket NOs. 50-282 and 50-306, 12/22/86.
2. NRC Safety Assessment, Methodology for Analysis of the Reactor Coolant Loops for Steam Generator Replacement, Oconee Units 1, 2, and 3, Docket NOs. 50-269, 50-270, and 50-287, 09/06/01
3. Letter from NRC to Rochester Gas & Electric, "Application of Leak-Before-Break Technology, R.E. Ginna Nuclear Power Plant (TAC NO. M86376), 7/23/93.
4. NRC Safety Evaluation Related to Issuance of Amendments RE: Steam Generator Replacements, Docket Nos. 50-348 and 50-364, 12/29/99.

Nuclear Management Corporation (NMC) is currently in the design and fabrication phase of the Steam Generator Replacement (SGR) Project for Unit 1 of the Prairie Island Nuclear Generating Plant (PINGP). The modification will be performed under 10CFR50.59. NMC seeks confirmation that the replacement steam generator (RSG) can be designed on the basis of the reduced loads resulting from the application of leak-before-break (LBB) criteria to postulated breaks in the primary reactor coolant loop (RCL) piping.

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The elimination of large primary loop ruptures as a design basis for PINGP Units 1 and 2 was approved by the NRC in Reference 1. The re-analysis of the RCL being performed for the SGR Project uses current methodology and is expected to demonstrate that the LBB considerations previously reviewed by the NRC continue to be valid after the RSGs have been installed. Framatome ANP, responsible for the PINGP RSG design engineering and fabrication, is conducting the RCL structural analysis in accordance with the approach and methodology previously reviewed and approved by the NRC in Reference 2. The methodologies are identical except:

A. For PINGP, the reactor building interior concrete structure is not included in the model. Instead, enveloped response spectrum seismic analyses are performed using applicable reactor building floor response spectra.

B. For PINGP, the surge line and pressurizer are excluded from the model.

These are the methods used in the original design basis PINGP Unit 1 RCL structural analysis. To further ensure that the reduced loads on SG internals and externals are solely due to the elimination of large primary loop ruptures as a design basis, no special methods were employed in the seismic or pipe rupture structural analyses that would significantly reduce the loads on the RSG internal components and external shell (i.e., the structural damping, modal combination method and directional combination method are the same as those referenced in the PINGP USAR). As in Reference 2, ruptures in the largest RCL pipes not qualified by leak-before-break methodologies were considered (10" surge line, 12" safety injection accumulator line and the 8" residual heat removal line).

The NRC has specified during a previous SGR project that RSG internal components can be designed on the basis of LBB-reduced loads provided that the cutting and welding of the RCL piping during the replacement process will not introduce any new pipe break locations not covered by the LBB criteria (Reference 3). This proviso will be adhered to during the PINGP SGR Project. The NRC has more recently approved the application of LBB considerations to the structural design basis of the Farley RSGs (Reference 4), SG replacements for Westinghouse Model 51 SGs similar to those installed at PINGP.

Based on the above, NMC is applying LBB-reduced loads resulting from the Framatome ANP structural analysis to all aspects of RSG design, including the following:

- Externals

  1. SG shell
  2. SG primary nozzles

- Internals, including

  1. SG tube support plates, anti-vibration bars, divider plate
  2. SG tube integrity analysis in accordance with RG 1.121
  3. Determination of SG tube crush penalty

NMC requests NRC confirmation that the application of LBB-reduced loads to RSG design is acceptable. In order to meet the project schedule, NMC requests that the NRC provide confirmation by March 14, 2003.

In this letter we have made no new Nuclear Regulatory Commission commitments. Please contact Terry Higgins (651-388-1121, ext. 4189) if you have any questions related to this letter.

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c: Regional Administrator - Region III, NRC
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