NMC
Committed to Nuclear Excellence

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

JUN 2 2004

L-PI-04-068
10 CFR 50.90

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

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Supplement To License Amendment Request (LAR) Dated March 25, 2004, Main Steam Line Break Mass and Energy Release Using BAW-10169-A, (TAC Nos. MC-2439/MC-2440)

By letter dated March 25, 2004, the Nuclear Management Company, LLC (NMC) submitted an LAR titled, "Main Steam Line Break Mass and Energy Release Using BAW-10169-A." This LAR proposed Prairie Island Nuclear Generating Plant (PINGP) licensing basis changes to allow use of the methodology described in Framatome-ANP Topical BAW-10169-A, "RSG Plant Safety Analysis – B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," dated October 1989. Topical BAW-10169-A utilizes the RELAP5/MOD2-B&W code described in Topical BAW-10164-A, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient Analysis", Revision 3 dated October 1996. NMC submits this supplement in accordance with the provisions of 10 CFR 50.90.

By phone call May 19, 2004, the NRC Staff requested additional information on the use of BAW-10169-A for the PINGP as follows:

1. *Explain the choice of 0.6 ft^2, discussed on page 5 of Exhibit A, LAR dated March 25, 2004, for the lower bound break size for the sensitivity study. Are split breaks considered?*

   The minimum break size was chosen based on Framatome-ANP experience. Typically, the peak containment temperature and pressure decrease as the break sizes decrease below 0.6 ft^2. The minimum break size of 0.6 ft^2 was part of the initial matrix of break sizes and single failures developed prior to evaluating any of the cases. The transient results showed a trend toward increasing containment response as the break size was lowered to 0.6 ft^2. As a result additional breaks of 0.4 ft^2 were analyzed. The limiting break size for containment response was 0.6 ft^2.

   All cases below 1.4 ft^2 (the area of the steam exit nozzle venturis) were analyzed as split breaks (i.e. communication remains between the two sides of the break).

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2. *Verify that both cases with and without loss of offsite power will be considered to determine the limiting case.*

   Loss of offsite power (LOOP) was considered in determining the limiting case. LOOP would result in immediate loss of reactor coolant and main feedwater pumps, which would reduce the energy transfer from the reactor coolant system to the steam generators (SGs) as well as the mass transfer from the main feedwater system to the SGs. Consequently, LOOP may result in significant reductions in the rates and integrated values of mass and energy transfer to containment, lessening the severity of the transient. Thus no LOOP cases were explicitly analyzed.

3. *Describe or provide a reference for the treatment of entrainment in the break flow. Was this included in the core response calculations?*

   The liquid swell within the steam generator and entrainment in the break flow was explicitly calculated by the RELAP5/MOD2-B&W code for each case. Therefore there is not a particular reference for the treatment of entrainment. Modeling of the swell was included in the sample core calculations contained in BAW-10169-A, see Figures 7.5.28 through 7.5.30.

4. *The statement is made on page 5 of Exhibit A, LAR dated March 25, 2004, that liquid inventory is maximized. But this corresponds to zero power. How is this consistent with sensitivity study to select limiting power?*

   The liquid inventory at any given power level is maximized by assuming a maximum initial water level that bounds all power levels. Thus, the water level remains constant while the mass changes with the temperature of the liquid inventory.

5. *On page 5 of Exhibit A, LAR dated March 25, 2004, why isn't a single failure of the auxiliary feedwater pump automatic run-out protection analyzed?*

   Previous analyses performed by NMC's Nuclear Analysis and Design Department showed that failure of the auxiliary feedwater pump automatic run-out protection was not the limiting failure.

6. *Is the description of the modeling of the steam system, discussed on page 5 of Exhibit A, LAR dated March 25, 2004, consistent with the current licensing basis?*

   The modeling of the steam system is consistent with the modeling requirements of BAW-10169-A. It was generated using Prairie Island specific data.

The proposed changes in this supplement do not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the original March 25, 2004 submittal.

In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR by transmitting a copy of this letter to the designated State Official.

Please address any comments or questions regarding this LAR supplement to Mr. Dale Vincent at 1-651-388-1121.

Summary of Commitments

In this letter NMC has not made any new or revised any Nuclear Regulatory Commission commitments.

I declare under penalty of perjury that the foregoing is true and accurate.
Executed on JUN 2 2004 .

Richard Graham for

Joseph M. Solymossy
Site Vice President, Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
Minnesota Department of Commerce