NMC
Committed to Nuclear Excellence

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

FEB 2 3 2004

L-PI-04-027
10 CFR 50.90

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

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKETS 50-282 AND 50-306
LICENSE Nos. DPR-42 AND DPR-60

SUPPLEMENT TO LICENSE AMENDMENT REQUEST (LAR) DATED MARCH 25, 2003, SAFETY ANALYSES TRANSITION
(TAC NOs. MB8128 and MB8129)

By letter dated March 25, 2003, the Nuclear Management Company, LLC (NMC) submitted an LAR titled, "Safety Analyses Transition," which proposes Technical Specification (TS) changes associated with transition to Westinghouse performance of safety analyses for the Prairie Island Nuclear Generating Plant (PINGP).

By letter dated January 14, 2004, NMC provided supplemental information in support of the subject LAR. The NRC, by letter dated February 12, 2004 provided follow-up questions on the supplemental information provided by the NMC letter dated January 14, 2004. Attachment 1 to this letter, "NMC Response to NRC Follow-up Questions," provides the responses to the NRC follow-up questions. NMC submits this supplement in accordance with the provisions of 10 CFR 50.90.

The information provided in this supplement does not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the original March 25, 2003 submittal as supplemented June 16, 2003, January 14, 2004 and February 11, 2004.

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

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In this letter NMC has not made any new or revised any Nuclear Regulatory Commission commitments. Please address any comments or questions regarding this LAR supplement to Mr. Dale Vincent at 1-651-388-1121.

I declare under penalty of perjury that the foregoing is true and accurate.
Executed on FEB 2 3 2004

Richard Graham for

Joseph M. Solymossy
Site Vice-President, Prairie Island Nuclear Generating Plant

CC Regional Administrator, USNRC, Region III
Project Manager, Prairie Island Nuclear Generating Plant, USNRC, NRR
NRC Resident Inspector – Prairie Island Nuclear Generating Plant
Glenn Wilson, State of Minnesota

Attachment:

1. NMC Response to NRC Follow-up Questions

ATTACHMENT 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Letter L-PI-04-027, Supplement to
License Amendment Request dated March 25, 2003

NMC Response to NRC Follow-up Questions

1. Steam Generator Measurement Errors

Table 5.1.2 (page 5-21) specifies the initial steam generator (SG) water level assumed in the non-loss-of-coolant accident (LOCA) analysis for Prairie Island. Table 5.1.5 (page 5-26) indicates that the low-low SG water level trip is credited in the analysis of the loss of normal feedwater (LONF) and loss of all AC power (LOAC) events to trip the reactor and to actuate auxiliary feedwater (AFW). Section 5.1.15.2 (page 5-274) indicates that the low SG wide range level signal is used in the anticipated transient without scram (ATWS) analysis to trip the reactor and the turbine, and actuate the AFW. However, insufficient information was provided for how the SG water level measurement errors are considered in the safety analysis.

For the Westinghouse-manufactured SGs, Westinghouse issued a number of Nuclear Service Advisory Letters (NSALs) addressing the SG water level measurement errors. NSAL-02-3 and the revision to this NSAL deal with the uncertainty in the SG water level measurement caused by the placement of the mid-deck plate between the upper and lower pressure taps. This results in a delayed actuation of the SG water level low-low trip signal that trips the reactor and actuates the auxiliary feedwater. NSAL-02-4 deals with uncertainties in the SG water level measurement due to the void content of the two-phase mixture above the mid-deck plate that is not reflected in the SG water level setpoint calculation. This results in a premature actuation of the SG water level high-high trip signal that isolates the feedwater system. NSAL-02-5 deals with potential inaccuracies in the initial conditions assumed in safety analyses affected by SG water level. The safety analyses may not be bounding because the velocity head under some conditions may increase in the uncertainties in the SG water level control system. NSAL-03-09 indicates that Westinghouse has developed a program for the Westinghouse Owners Group that evaluates the effects on the SG water level control system uncertainties from various items. These items include the mid-deck plate, feedwater ring and feedwater ring supports, lower-deck plate supports, non-recoverable losses due to carryunder, decrease in subcooling due to carryunder, and transient conditions due to events such as the single loop LONF, etc. Under the program, Westinghouse evaluated the design features of Westinghouse-designed SGs and other phenomena associated with Westinghouse SGs as they affect uncertainties with respect to the SG water level control system, and the SG water level low, low-low and high-high reactor trip functions. Westinghouse has documented the results of its program in WCAP-16115, "Steam Generator Level Uncertainties Program." Westinghouse recommends that all licensees with plants using Westinghouse SGs review the WCAP-16115 results to determine the impact on plant-specific SG level uncertainty, in addition to the effects identified in NSALs 02-3, -4 and -5.

Please provide the following information:

(a) Discuss how [Prairie Island Nuclear Generating Plant] PINGP accounts for the applicable uncertainties documented in the Westinghouse NSALs and the guidance specified in WCAP-16115 in determining the initial SG water level and SG water level (low and low-low) setpoints.

Nuclear Management Company (NMC) Response:

Instrumentation uncertainties at the Prairie Island Nuclear Generating Plant (PINGP) are considered in setpoint calculations in accordance with the PINGP Engineering Manual Section 3.3.4.1, "Engineering Design Standard for Instrument Setpoint/Uncertainty Calculations", which was accepted by NRC SER dated July 26, 2002, Section G.2.1.

PINGP has an Operating Experience program under which it is required to address vendor information such as that provided by Westinghouse in Nuclear Advisory Safety Letters (NSALs). Specifically, NSAL-02-3, 02-4 and 02-5 have been fully evaluated and the effects have been accounted for in PINGP's SG water level setpoint calculations. NSAL-03-9 and associated WCAP-16115-P, issued late 2003, have also been evaluated within the PINGP Operating Experience Program; this evaluation found that the new information presented in WCAP-16115-P had no significant impact on PINGP SG water level safety system setpoint calculations, and that no plant setpoints or Technical Specification Allowable Values would be affected. Incorporation of the new SG level measurement uncertainty information presented in WCAP-16115-P into remaining PINGP SG level setpoint and uncertainty calculations is being completed in accordance with the PINGP Corrective Action Process.

(b) Provide information to demonstrate that the effects of water level uncertainties discussed in the NSALs and WCAP-16115 are appropriately considered in the analyses of the non-LOCA transients and ATWS presented in Attachment 5.

NMC Response:

With respect to the non-LOCA transient analyses, steam generator water level uncertainties impact the assumptions of initial steam generator water level, and low-low and high-high level setpoints. As identified in Note 7 of Table 5.1-2 on page 5-21 (Attachment 5), the conservative initial steam generator level uncertainties applied in the non-LOCA transient analyses are +11% narrow range span (NRS) / -15% NRS. For example, in the loss of normal feedwater (LONF) transient analysis, it is conservative to assume a high initial steam generator level because it delays the time of reactor trip due to a low-low steam generator level. Thus, as specified in Section 5.1.11.2, an initial steam generator level of 55% NRS (the programmed full-power value of 44% NRS plus 11% NRS

uncertainty) is assumed in the LONF transient analysis. The assumed initial condition uncertainties of +11% NRS and -15% NRS conservatively bound the uncertainties calculated by NMC.

In relation to the low-low and high-high steam generator level setpoints, the bottom and top of the narrow range span (0% NRS and 100% NRS) are assumed, respectively, in the non-LOCA safety analyses. As the most limiting setpoints are modeled in the safety analyses, applicable uncertainties are accounted for in the establishment of the Technical Specification allowable values.

Finally, the assumed low steam generator level AMSAC/DSS setpoint is 35% wide range level (WRL), which includes allowance for instrumentation uncertainties; the actual plant setpoint is 42.5% WRL.

2. Reactor Coolant Pump Coastdown Delay Times

Section 5.1.1.2.2 (page 5-195) and 5.1.15.5 (page 5-292) indicate that in the analysis of the LOAC event, the reactor coolant pumps (RCPs) are assumed to lose power and begin coasting down two seconds after the reactor trip. Assumption 2 on pages 5-209 and 5-209 for the steam line break (SLB) analysis indicates that the RCPs are assumed to begin coasting down three seconds after the SLB initiation for the case without offsite power.

(a) Please justify the adequacy of the RCP coastdown delay times assumed in the analysis of the LOAC and SLB events in predicting a minimum departure from nucleate boiling ratio (DNBR), maximum pressurizer pressure and the minimum margin to overfill the pressurizer.

NMC Response:

For the LOAC and SLB analyses it is assumed that a loss of offsite power occurs as a consequence of instability on the power grid caused by the unit trip. It is typically assumed that RCP coastdown begins 2 or 3 seconds following the turbine trip, which occurs as a result of the reactor trip. This is a reasonable assumption in that the sequence of turbine trip–generator trip–grid instability–loss of power to RCPs would not occur instantaneously. However, for these specific transients, the RCP coastdown delay time is not an important or critical parameter. The transient results, including DNBR, pressurizer pressure, and peak pressurizer water volume would be negligibly affected if a zero delay time were assumed in the analysis. Note also that the LOAC results are significantly less limiting than the results for the Loss of Normal Feedwater (LONF) event in which continuous operation of the RCPs is assumed. Similarly, the SLB case with loss of offsite power is less limiting than the SLB case with offsite power available (no RCP coastdown).

3. Anticipated Transient Without Scram (ATWS) Analysis

(a) List the values of the moderator temperature coefficient (MTC) assumed in each of the analyzed ATWS cases listed in Table 5.1.15-1 (page 5-273), and show that the the values of MTC used in the ATWS analysis bound at least 95 percent of the fuel cycle time.

NMC Response:

The most negative (least conservative) MTC assumed in the AMSAC/DSS analyses of Section 5.1.15 (Attachment 5) is -4.2 pcm/°F, which, as indicated on page 5-319, is bounding for 95% of a representative fuel cycle. Transient-specific MTCs assumed are as follows:

- Partial Loss of Reactor Coolant Flow (Section 5.1.15.3), 0.0 pcm/°F,
- Loss of Normal Feedwater (Section 5.1.15.4), -2.0 pcm/°F,
- Loss of AC Power (Section 5.1.15.5), -2.0 pcm/°F,
- Loss of Load/Turbine Trip (Section 5.1.15.6), -4.1 pcm/°F,
- RCCA Bank Withdrawal at Power (Section 5.1.15.7), -2.0 pcm/°F,
- Uncontrolled Boron Dilution (Section 5.1.15.8), -4.2 pcm/°F.

(b) Figures 5.1.15.4-1 through 5.1.15.4-7 (pages 5-284 through 5-290) show the plant responses following an ATWS- LONF accident. Provide a table listing the sequence of events for the ATWS - LONF accident and explain the causes of changes in reactor coolant system temperatures (Figure 5.1.15.4-2 (page 5-285)), pressurizer and SG pressures (Figure 5.1.15.4-3 and –6 (pages 5-286 and –289)), and pressurizer and SG water volumes (Figure 5.1.15.4-4 and –7 (pages 5-287 and –290)).

NMC Response:

The following table provides the time sequence of events for the ATWS-LONF event analysis.

Event	Time (seconds)
Main Feedwater Flow Ceases	0.0
Initial Pressurizer Spray Flow Actuation	11.0
Backup Heaters Actuated due to High Pressurizer Level Deviation of 10% span (remain on for duration of transient).	50.1
Minimum DNBR (1.87) Occurs	64.0
Steam Generator Level Reaches the AMSAC/DSS Setpoint of 35% Wide Range Level (WRL)	66.8
Reactor Trip Signal Generated	71.8
Rods Begin to Drop (Rod Motion Initiated)	74.8
Turbine Trip Occurs (Turbine Control/Stop Valve Closed)*	76.3
MSSV with the Lowest Setting on Each Loop Opens	82.5
AFW Flow Initiated to Both Steam Generators	131.8
Peak Cold Leg Temperature (586.5°F) Occurs	313.0
Peak Pressurizer Mixture Volume (762 ft^3) Occurs	365.0
Peak Pressurizer Pressure (2305.8 psia) Occurs	1774.0

*Note that although the text of Section 5.1.15.4 (Attachment 5) indicates a turbine trip is assumed to initiate 5 seconds after the reactor trip (see major assumption item 7), the turbine trip was actually modeled to initiate 4 seconds after the reactor trip, with an additional 0.5-second valve closure delay. Having the turbine trip occur slightly sooner is conservative because it decreases the heat removal rate of the secondary-side more rapidly.

Figure 5.1.15.4-1 – Nuclear Power - Initially, the nuclear power decreases slightly due to reactivity feedback from the RCS heatup (with a negative MTC) caused by the reduction in the secondary side heat removal rate. The nuclear power decreases rapidly at ~75 seconds as a result of rod motion initiated after a diverse scram is actuated from the steam generator level reaching 35% WRL.

Figure 5.1.15.4-2 – Reactor Coolant Temperatures – Initially, the hot leg (HL) and cold leg (CL) temperatures increase due to the loss of feedwater. At the time of reactor trip, the HL temperature decreases and the CL temperature increases. Following reactor trip, the HL and CL temperatures converge and the average temperature increases as the primary-side heat generation rate (decay heat and reactor coolant pump (RCP) heat) exceeds the heat removal capacity of the secondary-side (AFW was initiated at ~132 seconds). The average

temperature turns around at ~313 seconds when the heat removal capacity of the secondary-side exceeds the decay heat and RCP heat.

Figure 5.1.15.4-3 – Pressurizer Pressure – As a result of the primary-side heatup, the pressurizer pressure tends to increase. However, the pressurizer sprays and heaters are modeled, and they act to maintain pressurizer pressure. Competing effects between the backup pressurizer heaters, actuated due to high pressurizer level deviation at ~50 seconds, and the pressurizer sprays and proportional heaters cause the oscillatory behavior seen in the pressurizer pressure plot. After the primary-side temperature turns around at ~313 seconds, the pressurizer pressure drops rapidly, but is recovered by the action of the proportional heaters.

Figure 5.1.15.4-4 – Pressurizer Water Volume – As a result of the primary-side heatup, the reactor coolant expands and causes an insurge into the pressurizer, thus increasing the pressurizer water volume. In addition, the action of the pressurizer heaters further expands the water in the pressurizer. The rate of water volume increase is slightly attenuated when the AFW is initiated at ~132 seconds. Within a minute after the primary-side temperature turns around at ~313 seconds (when the heat removal capacity of the secondary-side exceeds the decay heat and RCP heat), the pressurizer water volume peaks at 762 ft^3 (365 seconds) and begins to decrease.

Figure 5.1.15.4-5 – Reactor Vessel Lower Plenum Pressure – The reactor vessel lower plenum pressure varies for the same reasons described above for the pressurizer pressure (Figure 5.1.15.4-3).

Figure 5.1.15.4-6 – Steam Generator Pressure – Initially, the steam generator pressure increases as a result of the loss of normal (subcooled) feedwater. After the diverse scram signal trips the reactor, a turbine trip is actuated, and the steam generator pressure rapidly increases to the setpoint of the first bank of main steam safety valves (MSSVs). Pressure is then maintained essentially constant for the remainder of the transient.

Figure 5.1.15.4-7 – SG Wide Range Indicated Level – As a result of the loss of normal feedwater, the steam generator mixture level decreases to the diverse scram setpoint of 35% wide range level (setpoint reached at ~67 seconds and rod motion initiated at ~75 seconds). Following reactor trip, the level in each steam generator continues to decrease until both steam generators dry out at ~230 seconds. The AFW acts to cool the plant and mass begins to recover at ~380 seconds, although the wide-range indicated level doesn't begin to recover until after ~980 seconds.