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Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

July 21, 2005

L-MT-05-082
10 CFR Part 54

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

Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

Response to Request for Additional Information Regarding the Monticello License Renewal Application (TAC NO. MC6440)

References: 1) NMC letter to NRC, "Application for Renewed Operating License," dated March 16, 2005.

2) NRC Request for Additional Information Regarding the License Renewal Application for MNGP, June 21, 2005 (ADAMS Accession No. ML0517205930)

Pursuant to 10 CFR 54, the Nuclear Management Company, LLC, (NMC) submitted a License Renewal Application (LRA) (Reference 1) to renew the operating license for the Monticello Nuclear Generating Plant (MNGP).

By letter dated June 21, 2005, the Nuclear Regulatory Commission (NRC) issued a Request for Additional Information (RAI) regarding the LRA for MNGP (Reference 2).

Enclosure 1 provides the NMC response to this RAI with the exception of Item 4.5-1. Item 4.5-1 requests additional information regarding the effects of environment on fatigue as well as identification of F_{en} multipliers. The calculations are currently being reviewed to determine the effect of an error discovered in the industry use of service temperatures for carbon steel and low alloy steel F_{en} values. NMC expects to respond to this item with revised F_{en} values within 30 days.

This letter contains no new regulatory commitments.

A113

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I declare under penalty of perjury that the foregoing is true and correct.

Executed on July 21st 2005.

John T. Conway

John T. Conway
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
License Renewal Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

ENCLOSURE 1
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
DATED JUNE 21, 2005

Plant Level Scoping Results

NRC RAI 2.2-1

The Control Rod Velocity Limiters are described in Section 6.4 of the USAR. The Control Rod Velocity Limiters are provided as an integral part of each control rod. They provide hydraulic damping to reduce the free fall velocity of the rod and thereby reduce the consequences in the event the control rod became detached from its drive and dropped from the core. The LRA does not mention this component, nor does it appear to refer to USAR Section 6.4 in the text. Please provide justification.

NMC Response

The Control Rod Velocity Limiters are plant engineered safety features (ESFs) as described in Chapter 6 of the Monticello Nuclear Generating Plant (MNGP) Updated Safety Analysis Report (USAR). Since the control rod velocity limiters are an integral part of each control rod, the velocity limiters are within the scope of license renewal. However, the control rods (including the velocity limiters) screen-out from the Aging Management Review (AMR) process since they are periodically replaced.

As explained below, all of the MNGP ESFs are within the scope of license renewal. The ESFs are in addition to the safety features included in the design of the reactor, reactor primary system, plant and reactor control systems and other instrumentation or process systems. Most of the ESFs serve no function during normal plant operation but are included for the sole purpose of reducing the consequences of design basis accidents described in Section 14 of the MNGP USAR.

The following list of MNGP ESFs describes where these features are covered in the MNGP License Renewal Application (LRA):

Containment Systems

The containment systems are described in Section 5 of the MNGP USAR. Those containment systems considered engineered safety features are:

(a) Primary Containment (LRA Sections 2.3.2.5 & 2.4.13)

(b) Secondary Containment including the Standby Gas Treatment System (LRA Sections 2.3.2.8 & 2.4.15)

(c) Containment Isolation System (LRA Section 2.5.1.10)

Emergency Core Cooling System (ECCS)

The ECCS provides for continuity of reactor core cooling over the entire range of postulated breaks in the reactor primary system. The design of the cooling systems assumes that, although highly improbable, there may be a double-ended break in one of the pipes connected to the primary system. Systems are provided to maintain cooling whether the assumed break is in the largest or smallest pipe and whether it is a steam or water containing line. The systems included in the ECCS are:

(a) Core Spray System (LRA Section 2.3.2.3)
(b) Low Pressure Coolant Injection System (LRA Section 2.3.2.7)
(c) High Pressure Coolant Injection System (LRA Section 2.3.2.4)
(d) Automatic Depressurization System (LRA Section 2.3.2.1)

Main Steam Line Flow Restrictors

A flow restricting venturi is installed in each main steam line. Its purpose is to protect the fuel barrier by limiting the flow of steam, and therefore the loss of reactor coolant from the reactor vessel in the postulated case of a complete severance of a main steam line. (LRA Section 2.3.4.4)

Control Rod Velocity Limiters

Control rod velocity limiters are provided as an integral part of each control rod. They provide hydraulic damping to reduce the free fall velocity of the rod and thereby reduce the consequences in the event the control rod became detached from its drive and dropped from the core. (LRA Section 2.3.3.4)

Control Rod Drive Housing Supports

Control rod drive housing supports prevent the blowout of a control rod in the event a control rod drive housing breaks or separates from the bottom of the reactor vessel. (LRA Section 2.4.6)

Standby Liquid Control System

The Standby Liquid Control System is designed to bring the reactor from full power to a cold, xenon free, shutdown assuming that none of the withdrawn control rods can be inserted. The system is started by the control room operator and injects a neutron absorbing solution into the reactor primary system. (LRA Section 2.3.3.17)

Main Control Room, Emergency Filtration Train Building, and Technical Support Center Habitability Systems

Heating, ventilating and air conditioning equipment is provided to maintain a suitable environment in the Control Room and certain other essential areas. Radiological protection is also provided for personnel in these areas. (LRA Sections 2.3.3.7, 2.4.4, & 2.4.12)

NRC RAI 2.2-2

The Safety Parameter Display System is described in Section 7.13 of the MNGP USAR. The purpose of the Safety Parameter Display System (SPDS) is to provide a concise display of critical plant variables to control room operators, to aid them in rapidly and reliably determining the safety status of the plant. The LRA does not mention this system, nor does it appear to refer to USAR Section 7.13 in the text. Please provide justification.

NMC Response

Section 2.1.4.1 of the LRA states: "System identifier codes are used to sort and track plant systems and components in the plant equipment database. This identification scheme supports plant needs with respect to maintenance work, but is not sufficient to identify license renewal system functional boundaries. For this reason, revision or the combination of some plant equipment database system identifiers was necessary for license renewal purposes. License renewal systems were defined to account for all of the plant equipment database systems that contain permanently installed equipment consistent with the system descriptions in the MNGP USAR."

The Process Computer System (PCS) was included in the license renewal system under "Computer" (CMP). The LRA lists the computer system in Table 2.2-1, Plant Level Scoping Results, Page 2-47, which states that the system is not within the scope of license renewal. Input to the SPDS is provided by the PCS and the SPDS is considered a subsystem of the PCS. Both the PCS and SPDS were evaluated in the scoping and screening reports for the CMP system as follows:

> The purpose of the PCS system is to aid the operator in timely determination of plant operability status during all plant conditions by providing a real time presentation of operational data pertaining to the reactor core and other plant equipment. This includes providing input to the SPDS. The USAR (Sect. 7.13) states that the SPDS is not essential to the safe operation of the plant; it is not essential to the prevention of events that endanger public health and safety; nor is it essential to the mitigation of the consequences of an accident. The PCS also records plant operational data, which can be recalled for evaluation of abnormal and unusual events.

The USAR discusses the PCS in relation to various topics such as the NUMAC Rod Worth Minimizer, Accident Monitoring Instrumentation, and the SPDS. The Process Computer is not safety-related and its failure will not cause a safety-related function to fail (USAR Section 7.8.3). Safety related isolation devices, between the PCS and the

Neutron Monitoring and the Plant Protection Systems signal inputs, are part of these other systems for License Renewal purposes.

The Process Computer is not on the Monticello Q-list. The Process Computer is not required for any of the regulated events. The 250 Vdc System battery that powers the process computer is not required to function during a Station Blackout Event. The Fire Protection System does not rely on the computer for processing fire detection and alarm signals. The required Anticipated Transients Without Scram (ATWS) monitoring instrumentation does not rely on the PCS. Therefore, the PCS, including the SPDS, was scoped out of consideration for license renewal.

Metal Fatigue TLAAs

NRC RAI 4.3.1-1

Section 4.3.1 of the license renewal application (LRA) discusses the evaluation of the reactor pressure vessel (RPV) components. The LRA indicates that a reanalysis of the fatigue usage of RPV components was performed as part of the 1998 power uprate. The LRA also indicates that revised fatigue usage factors were determined from the MNGP fatigue monitoring program (FMP). Describe how these revised fatigue usage factors were calculated. Provide a copy of SASR 89-77, "Accumulated Fatigue Usage for the Monticello Nuclear Generating Station Reactor Pressure Vessel," December 1989.

NMC Response

General Electric Document SASR 89-77 is an internal engineering document that is available on site for NRC review. This document provides a tabulation of thermal cycles experienced by MNGP through July of 1989. In addition, it provides fatigue usage calculations for selected reactor pressure vessel locations, extrapolated to forty years based on experienced cycles. As described in LRA Section 4.3, the MNGP fatigue monitoring program was subsequently implemented which periodically (once per refueling cycle) requires review of plant operating records to determine transient cycles experienced during the respective period. These updated cycles are then used to project the cycles expected at the end of the license renewal period of extended operation (60 years) and determine the corresponding cumulative fatigue that has been reported in Table 4.3.1-1 of the LRA. All relevant operating data such as Control Room Logs, Monthly Core Performance Reports, and Plant Information System data are used to ensure that a comprehensive review and accounting of all relevant cycles is obtained. A key point of conservatism in a cycle counting approach is that all cycles within a transient grouping are assumed to occur at the maximum heatup/cooldown profile, thereby resulting in the maximum stress range for fatigue calculation purposes.

As expected, the rate of occurrence of many transients experienced by MNGP has been decreasing due to operational improvements. This has resulted in corresponding decreases in the cumulative fatigue usages. This trend and the expected transition to two year fuel cycles provides further assurance that the fatigue usages reported in LRA Section 4.3 are conservative.

NRC RAI 4.3.2-1

Section 4.3.2 of the LRA discusses the evaluation of the reactor vessel internals (RVI). The LRA indicates that the 60-year fatigue usage was estimated by multiplying original fatigue usage by a factor of 1.5. Confirm that this extrapolation bounds the number of startup/shutdown design cycles listed in Section 4.3.1 of the LRA.

NMC Response

As described in LRA Section 4.3.2, the only significant contributor to fatigue for the jet pump diffuser to baffle plate weld (the most significant fatigue loading location) is the transient that includes improper recirculation loop startup and post Design Basis Accident (DBA) flooding at a point in time that maximizes the shroud and shroud support plate through-wall gradients. All other events analyzed, including startup and shutdown, were negligible with respect to their effect on cumulative fatigue usage. It should be noted that while this evaluation was done specifically for Dresden Unit 2, it was determined to be applicable to MNGP due to similarity in design. The MNGP shroud support plate is thicker than the Dresden Unit 2 shroud support plate. This has the effect of reducing the maximum strain, thereby resulting in conservatism for MNGP.

Although the table of MNGP transient cycles in Section 4.3.1 of the LRA identifies 10 design cycles (8 projected for the license renewal extended period of operation) for improper recirculation loop startup, the net effect of 10 improper recirculation loop startup cycles only increases the cumulative usage factor (CUF) to approximately 0.40 which is less than the CUF of 0.5 identified in LRA Section 4.3.2 using a 1.5 multiplication factor to demonstrate compliance with acceptance criteria for 60 years.

NRC RAI 4.3.3-1

Section 4.3.3 of the LRA discusses the evaluation of reactor coolant pressure boundary (RCPB) piping. The LRA indicates that portions of the RCPB were required to be analyzed for fatigue in accordance with the ASME Code Section III for Nuclear Class 1 piping. Provide the basis for this requirement. The LRA further indicates that the design fatigue usage at the limiting location for RCPB core spray piping is less than 0.65 (core spray valve joint). The LRA estimates the 60-year fatigue usage by multiplying the design value by 1.5. Table 4.3.1-1 of the LRA indicates that the projected 60-year fatigue usage of the core spray nozzle is 0.65 based on the FMP. Indicate whether number of thermal transient cycles used to estimate the 60-year fatigue usage of the core spray valve joint is consistent with the number of thermal transient cycles, obtained from the FMP, used to estimate the 60-year fatigue usage of the core spray nozzle.

NMC Response

As noted in LRA Section 4.3.3, the American Society of Mechanical Engineers (ASME) Code Section III fatigue analysis guidelines were implemented for replaced portions of the reactor recirculation, core spray, and residual heat removal systems. This was done for the general purpose of attaining a higher quality level and to provide more detailed analyses confirming reactor coolant system integrity.

The fatigue usage value of 0.645 for the core spray nozzle described in LRA Table 4.3.1-1 is based on a projection of relevant cycles experienced through September 30, 2004, and comparing those cycles to 180 allowable cycles. This allowable was determined by analysis of the replacement core spray nozzle safe ends, which were installed in 1986 for intergranular stress corrosion cracking (IGSCC) mitigation. Reanalysis of the nozzle/safe end region determined a maximum alternating stress intensity of approximately 190 ksi, which corresponds to 180 allowable cycles. Extrapolation of the actual cycles experienced at MNGP through September 30, 2004, to the end of the license renewal extended period of operation (2030) yields an expected 116 cycles. This results in a cumulative fatigue usage of 0.645 at the end of the extended period of operation for the core spray nozzle/safe end.

As part of the IGSCC mitigation program in response to NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," portions of safety related piping inside of the drywell were replaced with IGSCC resistant materials. These replacements included core spray piping (1986). The replacement design included a Nuclear Class 1 fatigue analysis in accordance with Subsection NB-3600 of the ASME Section III Boiler and Pressure Vessel Code (1980 Edition including Addenda through Summer 1982). This analysis included 100 startup/shutdown cycles and 5 operating basis earthquakes (OBEs) with 10 maximum stress cycles per earthquake for 50 OBE cycles. To arrive at the 60-year cumulative fatigue described in the LRA, the fatigue resulting from both of these events was conservatively multiplied by 1.5 to show that margin to allowable was maintained through the extended period of operation.

Although different analysis approaches are used, the core spray nozzle and the core spray valve joint analyses both use the same thermal transient basis. Consequently, they are consistent.

NRC RAI 4.6.1-1

Section 4.6.1 of the LRA discusses the evaluation of the suppression chamber. The LRA indicates that the maximum fatigue usage for the torus shell was calculated to be 0.98. Provide the number of safety relief valve (SRV) lifts used in this analysis.

NMC Response

The number of SRV lifts used in the evaluation of the suppression chamber was 852 single valve lifts and 325 multiple valve lifts. This includes an evaluation of the effects of power rerate that was implemented in 1998. Major conservatisms in this application include the fact that all load cycles are assumed to occur in the same suppression chamber bay, and that the power rerate SRV lift factor was based on an analysis at a power level of 1880 MWth as opposed to the actual rerate power level of 1775 MWth.

TRANSMITTAL MANIFEST
NUCLEAR MANAGEMENT COMPANY, LLC
MONTICELLO NUCLEAR GENERATING PLANT
REGULATORY AFFAIRS

Response to Request for Additional Information Regarding the Monticello License Renewal Application (TAC No. MC6440)

Correspondence Date: July 21, 2005 Manifest Date: July 21, 2005

Monticello Special Instructions

Todd Hurrle ....USAR File................ Yes__ No_x__
Ron Baumer.............NRC Commitment..... Yes__ No_x__
Linda Christianson ....Monti OC Sec .... Yes___ No_x_
Chuck Hoglin............Monti OSRC Engr..... Yes_x_ No___

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Italics – letter only, electronic