September 28, 2005

Mr. John T. Conway
Site Vice President
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE MONTICELLO NUCLEAR GENERATING PLANT LICENSE RENEWAL APPLICATION (TAC NO. MC6440)

Dear Mr. Conway:

By letter dated March 16, 2005, Nuclear Management Company, LLC, (NMC or the applicant) submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54) to renew the operating license for Monticello Nuclear Generating Plant (MNGP), for review by the U.S. Nuclear Regulatory Commission (NRC). The NRC staff is reviewing the information contained in the license renewal application (LRA) and has identified, in the enclosure, areas where additional information is needed to complete the review.

These questions were discussed with your staff, Mr. Patrick Burke, and a mutually agreeable date for this response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-3777 or e-mail at DXM2@nrc.gov.

Sincerely,

/RA/

Daniel J. Merzke, Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No.: 50-263

Enclosure: As stated

cc w/encl: See next page

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Dear Mr. Conway:

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Daniel J. Merzke, Project Manager
License Renewal Section A
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DISTRIBUTION: Letter to J. Conway, Re: RAI for Monticello, Dated: September 28, 2005

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MONTICELLO NUCLEAR GENERATING PLANT
LICENSE RENEWAL APPLICATION (LRA)
REQUEST FOR ADDITIONAL INFORMATION (RAI)

STRUCTURES, SCOPING AND SCREENING

RAI 2.4.2-1

The two component groups: (1) Concrete in air/gas (foundation, walls, slabs) and (2) Concrete in air/gas (foundation, walls, slabs, grout) have identical intended functions in Tables 2.4.2-1, 2.4.3-1, 2.4.4-1, 2.4.7-1, 2.4.8-1, 2.4.11-1, 2.4.12-1, 2.4.14-1, 2.4.15-1, 2.4.16-1, and 2.4.17-1. Explain the difference in the foundations, walls, and slabs between the two component groups. If there is no difference, explain the need for having the first component group since the second component group has already covered the first component group.

In addition to the two component groups mentioned, concrete in air/gas (walls, slabs) is another component group listed with the same intended function as the previous two component groups in Tables 2.4.3-1, 2.4.4-1, 2.4.8-1, 2.4.12-1, and 2.4.17-1. Explain the difference in the walls and slabs listed in this component group and the previous two component groups. If there is no difference, explain the need for having this component group since other component groups have already covered walls and slabs.

RAI 2.4.4-1

Table 2.4.4-1 lists two identical component groups, "Concrete in atmosphere/weather (walls, slab)," with identical intended functions. Explain the need for listing the same component group twice.

RAI 2.4.5-1

Table 2.4.5-1 lists three identical component groups, "Non-metallic fire proofing in air/gas (...cementitious fireproofing, ...)," with identical intended functions. Explain the need for listing the same component group three times.

RAI 2.4.6-1

After the "System Function Listing," the applicant provides reference to Sections 12.2.1.2 and 12.2.1.3 of the Updated Safety Analysis Report (USAR) for additional hangers and supports commodity group details. A review of the SAR sections indicates that the systems, structures and components (SSCs) are classified as Class I and Class II with the definitions of Class I and Class II noticeably different from Criteria 1, 2, and 3 in 10 CFR 54.4. The applicant is requested to provide a discussion of how the current licensing basis (CLB) classification has been reconciled in the system function listing in Section 2.4.6.

RAI 2.4.6-2

In Table 2.4.6-1 line item "carbon steel, low-alloy steel in air/gas," the applicant identified a number of supports/anchorages as ASME Class MC supports and some are identified as non-ASME support components. The applicant is requested to provide information regarding the classification of component supports inside the torus (some may be non-ASME), and the supports outside the torus, specifically, the classification of the support system supporting the torus. It appears that the torus support system is classified as Class MC supports, and all its components are and will be inspected by the requirements of subsection IWF. After reviewing LRA Table 3.5.2-6, it was not obvious how the applicant treated these supports. The applicant is requested to provide clarification.

RAI 2.4.6-3

Table 2.4.6-1 lists "Carbon steel, low-alloy steel in atmosphere/weather (bolted connections and anchorage)" as a component group and "Carbon steel, low-alloy steel in atmosphere/weather (bolted connections and support anchorage)" as another component group. The only difference between the two component groups is that one group lists "anchorage" and the other "support anchorage." Explain the difference between "anchorage" and "support anchorage." The staff is uncertain of the nature of anchorage and support anchorage, and therefore requests the applicant to provide examples of each.

RAI 2.4.13-1

It appears that the supports and components included in Code PCT-04 of the system function listing are not within the scope of license renewal. The applicant is requested to provide a summary listing of these supports and components, and a confirmation that their failure under earthquake induced loads will not affect the functioning of the safety-related SSCs.

RAI 2.4.13-2

The second and third line items in Table 2.4.13-1 list almost identical components subjected to the same material and environment combination, and identical intended functions. A similar redundancy is noted on the first two line items on p. 2-257. The applicant is requested to clarify these redundancies.

RAI 2.4.13-3

Table 2.4.13-1 lists the component group as "lubrite in air/gas," with the drywell head included as a component. In the description of drywell head, the applicant states, "The head is held in place by bolts and sealed with a double gasket arrangement." The applicant is requested to clarify where the lubrite bearings are used in the drywell head.

RAI 2.4.17-1

Table 2.4.17-1 lists "Carbon steel, low-alloy steel in air/gas (fire rated doors)" as a component group with the intended function being fire barrier, and "Carbon steel, low-alloy steel in air/gas (... doors,...)" as another component group, with one of the intended functions also being fire barrier. Explain the difference in the doors between the two component groups.

Structures, Aging Management

RAI 3.5.2-1

Concrete in below grade (Foundation, Walls) is listed as requiring no AMP in Tables 3.5.2-2, 3.5.2-3, 3.5.2-4, 3.5.2-6, 3.5.2-7, 3.5.2-9, 3.5.2-11, 3.5.2-12, 3.5.2-13, 3.5.2-15, 3.5.2-16, and 3.5.2-17. The applicant states that an AMP is not required to manage aging because, as described in Note 501, plant documents confirm that the concrete had an air content between 3 percent and 6 percent, and inspections performed on concrete in accessible areas did not exhibit degradation related to freeze-thaw. It is the staff's position that the existing concrete also have a water-to-cement ratio of 0.35 - 0.45 to ensure there is no aging degradation related to freeze-thaw. The applicant is requested to verify the water-to-cement ratio is 0.35 - 0.45, or provide an appropriate AMP for this component group.

The same question applies to Concrete in below grade (Foundations, Walls, Lean Concrete) listed in Table 3.5.2-8.

RAI 3.5.2-2

Concrete in below grade (Foundation, Walls) is listed as requiring no AMP in Tables 3.5.2-4, 3.5.2-6, 3.5.2-7,3.5.2-9, 3.5.2-11, 3.5.2-12, 3.5.2-13, 3.5.2-15, 3.5.2-16, and 3.5.2-17. The applicant states that an AMP is not required to manage aging because, as described in Note 506: (1) the plant initial licensing basis did not include a program to monitor settlement, (2) no significant settlement has been observed, and (3) de-watering systems are not used. The applicant's claim for not requiring an AMP is inconsistent with ISG-03, which requires a "Structural Monitoring Program" based on the requirement of 10 CFR 50.65 (Maintenance Rule) for accessible areas and a "plant-specific program for inaccessible areas." Therefore, the applicant is requested to provide an appropriate AMP for this component group.

The same question applies to Concrete in below grade (Foundation, Walls, Lean Concrete) listed in Table 3.5.2-8.

RAI 3.5.2-3

Concrete in below grade (Diesel Fuel Oil Storage Tank Deadmen) is listed as requiring no AMP in Table 3.5.2-6. The applicant states that an AMP is not required to manage aging because, as described in Note 552, "NUREG-1801 lists inside or outside containment as the environment. Consider that this environment includes atmosphere/weather and below grade". The applicant's claim for not requiring an AMP is inconsistent with ISG-03, which requires a "Structural Monitoring Program" based on the requirement of 10 CFR 50.65 (Maintenance Rule)

for accessible areas and a "plant-specific program for inaccessible areas." Therefore, the applicant is requested to provide an appropriate AMP for this component group.

RAI 3.5.2-4

Concrete in below grade (Pedestal) is listed as requiring no AMP in Table 3.5.2-10. The applicant states that an AMP is not required to manage aging because, as described in Notes 501 and 506: (1) the plant initial licensing basis did not include a program to monitor settlement, (2) no significant settlement has been observed, (3) de-watering systems are not used, and (4) plant documents confirm that the concrete had an air content between 3 percent and 6 percent and inspection performed on concrete in accessible areas did not exhibit degradation related to freeze-thaw. It is the staff's position that the existing concrete also have a water-to-cement ratio of 0.35 - 0.45 to ensure there is no aging degradation related to freeze-thaw. The applicant is requested to verify the water-to-cement ratio is 0.35 - 0.45.

The applicant's claim for not requiring an AMP is inconsistent with ISG-03, which requires a "Structural Monitoring Program" based on the requirement of 10 CFR 50.65 (Maintenance Rule) for accessible areas and a "plant-specific program for inaccessible areas." Therefore, the applicant is requested to provide an appropriate AMP for this component group.

The same question applies to Concrete in below grade (Foundation, Walls, Slabs, Grout) listed in Table 3.5.2-18.

RAI 3.5.2-5

In describing the intended functions of the three items in Table 3.5.2-13, related to concrete in air/gas, the applicant states one of the intended functions is "non-safety support." The applicant is requested to clarify this characterization in terms of the CLB safety classification as well as in terms of 10 CFR 54.4 definitions, and provide examples of how the components are providing non-safety support.

RAI 3.5.2-6

In Table 3.5.2-13, under the component type "concrete in air/gas," a number of structural components (e.g.,drywell equipment foundation, bioshield wall, RPV pedestal) are listed. Section 3.5.2.2.1.3 and Note 508 describe the elevated temperature situation around the reactor vessel, and justify the existence of the elevated temperatures in these areas, based on the estimated temperatures in the MNGP drywell. The applicant is requested to provide the following information related to this component type:

a. Please provide a summary description of the cooling system installed to control the temperatures inside the drywell.

b. Please provide the operating experience related to the effectiveness of the cooling system. Are the shield wall temperatures, or any other parameter monitored that would detect the malfunctioning of the cooling system?

c. Based on the discussion of the elevated temperature condition in and around the bioshield wall in Section 3.5.2.2.1.3, the staff agrees that the concrete properties will not be significantly affected, if the actual temperatures around the shield wall remain within the estimated limits. However, additional shrinkage and loss of moisture due to radiation could degrade the concrete on a long term basis. In this context, please provide a summary of the results of the last two inspections performed for: (1) the bioshield wall, (2) RPV pedestal, (3) anchorages of seismic stabilizer frame, and (4) masonry walls (if any) inside the drywell.

RAI 3.5.2-7

The applicant identifies two line items in Table 3.5.2-06 related to carbon steel and low-alloy steel embedded in concrete as not requiring aging management. Note 549 states: "Requirements specified in NUREG-1801 for concrete quality, inspections and housekeeping are satisfied for steel elements in inaccessible areas." Based on the industry-wide experience related to corrosion of drywell shell in sand pocket region, the applicant is requested to provide more information regarding the inspections and housekeeping, and why these activities should not be part of an existing AMP. The purpose of including the embedded items in an existing program is to look for evidence of environment change (e.g., sand drains not working properly) in accessible areas that would indicate potential degradation in inaccessible areas.

The same question applies to carbon steel and low-alloy steel embedded in concrete listed in Table 3.5.2-13.

RAI 3.5.2-8

Recent experience with torus cracking at Fitzpatrick indicates high-pressure coolant injection (HPCI) discharge configuration in the torus as one of the reasons for the cracking. The applicant is requested to provide information regarding the HPCI configuration at MNGP that could affect torus integrity during the period of extended operation.

RAI 3.5.2-9

Three line items in Table 3.5.2-6 indicate that lubrite plates have been used at several locations in hangers and supports. It is the staff's position that an inspection of the accessible portion of the bearing is needed to ensure proper functioning during postulated environmental conditions. Therefore, the applicant is requested to incorporate an examination of the accessible portion of the lubrite bearings in an appropriate AMP.

RAI 3.5.2-10

Recent experience concerning breakage of T-Quencher support bolts at Hatch indicates that the Plant Chemistry Program (PCP) that controls the chemistry of treated water may not be adequate for managing the aging of submerged support components. The applicant is requested to discuss the adequacy of PCP, by itself, to manage the aging degradation of the submerged supports. This RAI is applicable to all line items in Table 3.5.2-13, where PCP has been identified as the only AMP.

RAI 3.5.2-11

For a number of items in Table 3.5.2-13, 10 CFR 50, Appendix J has been identified as the AMP to manage aging. Option B of Appendix J would permit the applicant to conduct Type B leakage rate tests of penetrations at 10 year intervals. The applicant is requested to discuss the plant-specific process (e.g., test frequency and operating experience) that is credited to manage degradation and leak tightness of the pressure boundary penetrations, including vent bellows.

RAI 3.5.2-12

Three line items in Table 3.5.2-13 indicate that lubrite plates have been used at several locations in the MNGP primary containment. In Note 556, the applicant states that graphite plate material is not used for drywell head and downcomers. The applicant is requested to clarify how lubrite plates are associated with drywell head and downcomers. In Note 559, the applicant states that beam seats in the drywell consist of carbon steel plate over a bronze plate lubricated with graphite packed into trepanned depressions. The steel plate covers the graphite packing and protects it from particulate contaminants. The staff believes that, if the lubrite bearing, in general, is qualified for use in the sustained temperatures and radiation existing in the drywell, the inspection should consist of the examination of the accessible part of the bearing to ensure proper functioning during postulated environmental conditions. In view of this discussion, the applicant is requested to incorporate an examination of the accessible portion of the lubrite bearings in an appropriate AMP.

Reactor Vessel Surveillance

RAI B2.1.29-1

The scope of the Reactor Vessel Surveillance Program indicates that the Monticello Nuclear Generating Plant (MNGP) Reactor Vessel Surveillance Program monitors radiation embrittlement of the reactor pressure vessel (RPV). MNGP is part of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) integrated surveillance program (ISP) that uses data from boiling water reactor (BWR) member surveillance programs to select the "best" representative material to monitor radiation embrittlement for a particular plant. The BWRVIP ISP monitors capsule test results from various member plants. MNGP plans to use the BWRVIP ISP during the period of extended operation by implementing the requirements of BWRVIP-116. BWRVIP-116 indicates the "best" representative plate material for the MNGP RPV is the plate (C2220) in the MNGP RPV surveillance capsules. Table 2-3 in BWRVIP-116 indicates the last capsule to be removed from MNGP RPV will be at 40 effective full power years (EFPY) where the estimated neutron fluence for the capsule will be 1.98×10^{18} n/cm^2 (E>1.0 MeV).

a. Provide the lead or lag factors (the ratio of the neutron fluence for the surveillance capsule to the peak neutron fluence for the reactor vessel at the 1/4 thickness depth {1/4T}) for all the surveillance capsule locations in the Monticello reactor vessel. Identify the lead or lag factor for the surveillance capsule that is planned to be removed at 40 EFPY.

b. Explain why the neutron fluence for the surveillance capsules lags the peak neutron fluence for the MNGP RPV.

c. Can the surveillance capsule that is planned to be removed at 40 EFPY be moved to another location in the MNGP RPV so that it receives a greater amount of neutron fluence? Explain: If it can be moved, what is the impact of moving it on the estimated neutron fluence for the capsule?

d. Since BWRVIP-116 is presently under review by the staff, the staff requests that the applicant add to Commitment 41 the following:

"The Reactor Vessel Surveillance Program will be enhanced to ensure that any additional requirements that result from the NRC review of BWRVIP-116 will be addressed prior to the period of extended operation."

Reactor Coolant System

RAI 3.1.2-1

Table 3.1.2-2 of the MNGP LRA indicates that the Top Head Dollar Plate, Top Head Flange and the Top Head Torus are susceptible to Crack Initiation and Growth due to Stress Corrosion Cracking and Intergranular Stress Corrosion Cracking, and their intended function is pressure boundary. The material is identified as alloy steel (A533-65 Grade B Class and A508 Class 2) and clad (308/309). Identify which materials provide the pressure boundary function and provide your basis, including any operating experience, for concluding that these materials are susceptible to Crack Initiation and Growth due to Stress Corrosion Cracking and Intergranular Stress Corrosion Cracking.

Neutron Embrittlement of the Reactor Pressure Vessel and Internals

RAI 4.2-1

Section 4.2.1 of the LRA indicates that neutron fluence was calculated for the MNGP RPV for the extended 60-year (54 EFPY) licensed operating period based on 3.90×10^{8} megawatt hour (MWh) through Cycle 22 at 1775 megawatts thermal (MWt) plus 4.76×10^{8} MWh at 1880 MWt. This results in a peak neutron fluence of 5.17×10^{18} n/cm^2 (E>1.0 MeV) and a peak 1/4T fluence of 3.82×10^{18} n/cm^2 (E>1.0 MeV) for the RPV and a neutron fluence at the inside of the shroud of 3.84×10^{21} n/cm^2 (E>1.0 MeV) at the end of the extended operating period. To support the projected MNGP neutron fluence values, provide the following information:

a. Identify the current operating cycle for MNGP. Clarify your intent with respect to uprating the power level of MNGP and how any intended future power uprate corresponds to the assumptions noted above regarding MNGP operating cycles beyond Cycle 22.

b. Provide the MNGP capacity factor at which MNGP has operated over the current and three preceding operating cycles.

c. State the capacity factor and neutron flux that were assumed in the MNGP LRA for all future MNGP operating cycles through the end of the period of extended operation. If different capacity factors were assumed for future operating cycles at 1775 MWt and at 1880 MWt, provide the assumed capacity factors for each power level.

d. If the capacity factors from the response to (b) are different than those in response to (c), provide justification for using the capacity factors from (c) for determining the projected neutron fluence in the MNGP LRA neutron embrittlement analyses.

RAI 4.2-2

Tables 4.2.1-1 and 4.2.1-2 contain the equivalent margins analysis (EMA) for the limiting MNGP plate and weld. These tables do not include an evaluation of surveillance plate and weld data. Surveillance data was submitted to the NRC in a letter dated December 21, 1998. The letter contains report SIR-97-003, Revision 2, "Review of the Results of Two Surveillance Capsules, and Recommendations for the Materials Properties and Pressure-Temperature Curves to be Used for the Monticello Reactor Pressure Vessel," which indicates unirradiated upper shelf energy (USE) data was available for surveillance plates, but not available for surveillance welds. Therefore, USE evaluations using surveillance data could be performed for the plates but not the welds. Based on the surveillance plate data in SIR-97-003, Revision 2, determine the impact of the surveillance plate data on the limiting beltline plate USE and evaluate what impact, if any, this data has on validity of the plate EMA.

RAI 4.2-3

Using the generic weld and plate data in Appendix B of EPRI Topical Report (TR) -113596, "BWR Vessel and Internals Project, Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)," determine the projected Charpy USE for the limiting weld and plate in the reactor vessel beltline at the 1/4T depth using the neutron fluence at the end of the period of extended operation.

RAI 4.2-4

The peak fluence at the RPV wall for the MNGP RPV is 5.17×10^{18} n/cm^2 (E>1.0 MeV) for 54 EFPY of operation. Based on this fluence value, the previous Reflood Thermal Shock Analysis of the RPV is not bounding for the period of extended operation. The original analysis has been superseded by an analysis for BWR-6 RPVs that is applicable to the MNGP BWR-3 RPV. The BWR-6 RPV analysis is applicable to MNGP because it uses a bounding main steam line break event and an RPV thickness similar to the MNGP RPV. This analysis assumes end-of-license (EOL) material toughness, which in turn depends on the EOL adjusted reference temperature (ART). The critical location for the fracture mechanics analysis is at 1/4 of the RPV thickness (from the inside, 1/4T). For the main steam line break event, the peak stress intensity occurs at approximately 300 seconds after initiation of the event. The analysis shows that at 300 seconds into the thermal shock event, the temperature of the vessel wall at 1.5 inches deep (which is the 1/4T depth for the BWR-6 RPV) is approximately 400°F. For the MNGP vessel, the 1/4T depth is 1.26 inches. Provide the fracture toughness (peak stress intensity value) required to prevent fracture of the MNGP RPV due to reflood thermal shock.

Stress Relaxation of Rim Holddown Bolts

RAI 4.8-2

The Time-Limited Aging Analysis (TLAA) for Stress Relaxation of Rim Holddown Bolts is discussed in Section 4.8 of the LRA. To more accurately address Stress Relaxation of Rim Holddown Bolts at MNGP, a plant-specific calculation was performed that incorporated the MNGP core plate geometry, an operating temperature of 288° C (550° F), and an MNGP fluence calculation that was performed in accordance with guidance provided in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001, (LRA Section 4.2.1). The maximum fluence applicable to the bolts of the core plate was determined to be 2.2 x 10^{19} n/cm^2 (E>1.0 MeV) at the end of the 60-year operating license. The resultant relaxation was determined to be based on GE Design Documents. The analysis assumed that all of the bolts were at this fluence even though many bolts experience a lower fluence depending on their specific location. This plant-specific analysis is bounded by the original analysis that conservatively assumed a higher value of 19 percent relaxation.

a. Provide the stress-relaxation curves and their supporting data that were utilized to determine the percent relaxation of the rim holddown bolts at MNGP. Explain how this data was utilized to establish the curves.

b. The staff requests that the applicant provide information regarding the material type, (i.e., type 304 or type 316, etc.), values of neutron flux and temperature related to the data cited in response to RAI 4.8-2(a) and compare them to the type of material, neutron flux and temperature values of the rim holddown bolts at MNGP. If the type of material, neutron flux or temperature values for the rim holddown bolts at MNGP are different than that for the data, evaluate the impact of these differences on the predicted stress relaxation values of the rim holddown bolts at MNGP.

c. Describe the load on the specimens used to develop the stress relaxation curves in response to RAI 4.8-2(a). Explain why these specimens are applicable for use in determining the axial stess relaxation in the rim holddown bolts.

d. Based on the stress relaxation curves provided in response to RAI 4.8-2(a) describe how the value of 8 percent stress relaxation was determined for the rim holddown bolts at MNGP.

e. The bending stresses in the holddown bolts result from the horizontal loads acting on the core plate. Some of these loads may depend on the preloading of the holddown bolts. The core plate is also subjected to vertical loads, which could cause portions of the core plate rim to separate from the shroud support as a result of smaller bolt preloads.

Show that the axial and bending stresses for the mean and highest loaded holddown bolts will not exceed the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III allowable stresses for P_m and P_m+P_b, as a result of the loss of preload at the end of the period of extended operation. State clearly the assumptions on which the analysis was based.