June 2, 2004

Mr. Thomas J. Palmisano
Site Vice President
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT — ISSUANCE OF AMENDMENT RE: REVISED ANALYSES OF LONG-TERM CONTAINMENT RESPONSE AND NET POSITIVE SUCTION HEAD (TAC NO. MB7185)

Dear Mr. Palmisano:

The Commission has issued the enclosed Amendment No. 139 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment revises the operating license to change Monticello's design bases and its Updated Safety Analysis Report in response to your application of December 6, 2002, as supplemented September 24, 2003, and March 12, 2004.

The amendment revises existing analyses for the following:

- Long-term containment response to the design-basis loss-of-coolant accident (LOCA).

- Containment overpressure required for adequate available net positive suction head (NPSH) for the low-pressure emergency core cooling system pumps following a LOCA, reactor vessel isolation, and Appendix R fire.

Nuclear Management Company performed long-term containment analyses assuming a service water temperature of 94 °F. The NPSH calculations assume a service water temperature of 90 °F. The lower service water temperature, 90 °F, would be operationally controlling (i.e., exceeding a service water temperature of 90 °F would exceed the Monticello licensing basis since the NPSH calculations would no longer be valid).

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

L. Mark Padovan, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures: 1. Amendment No. 139 to DPR-22
2. Safety Evaluation

cc w/encls: See next page

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T. Palmisano

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

L. Mark Padovan, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-263

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
License No. DPR-22

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated December 6, 2002, as supplemented September 24, 2003, and March 12, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 139, Facility Operating License No. DPR-22 is hereby amended to change Monticello's design bases and its Updated Safety Analysis Report in response to the license amendment application of December 6, 2002, as supplemented September 24, 2003, and March 12, 2004, and evaluated in the associated safety evaluation by the Commission's Office of Nuclear Reactor Regulation. The licensee shall submit the changes authorized by this amendment with the next update of the Safety Analysis Report in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of its date of issuance and shall be implemented as stated in 2 above.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA JLamb for/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: June 2, 2004

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 139 TO FACILITY OPERATING LICENSE NO. DPR-22

NUCLEAR MANAGEMENT COMPANY, LLC

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

The Nuclear Management Company, LLC's (NMC's) application of December 6, 2002, as supplemented September 24, 2003, and March 12, 2004, requested changes to the Monticello Nuclear Generating Plant (Monticello) operating license to change its design bases and the Updated Safety Analysis Report (USAR). The supplemental letters provided clarifying information that was within the scope of the initial notice and did not expand the scope of the original *Federal Register* notice or change the initial proposed no significant hazards consideration determination.

The proposed amendment revises the existing analyses for the following:

- Long-term containment response to the design-basis loss-of-coolant accident (LOCA).
- Containment overpressure (the pressure above the initial containment pressure) required for adequate available net positive suction head (NPSH) for the low-pressure emergency core cooling system (ECCS) pumps following a LOCA, isolation event or Appendix R fire.

In addition, NMC intends to use these analyses to justify revising the service water temperature licensing basis. NMC administratively limits the service water temperature to 85 °F, instead of its current licensing basis value of 90 °F, because the results of analyses of a new scenario (reactor vessel isolation with high-pressure coolant injection [HPCI] unavailable) showed that the design temperature for the piping attached to the wetwell would be exceeded. A license amendment is required since NMC used different methods of evaluation in the updated containment analyses from those currently described in the Monticello USAR and previously approved by the Nuclear Regulatory Commission (NRC). NMC's submittal of December 6, 2002, demonstrates acceptable results for the long-term containment LOCA response with a service water temperature of 94 °F. The NPSH analyses were performed using a service water temperature of 90 °F.

Appendix A to this safety evaluation (SE) contains relevant design and licensing bases information for Monticello. NMC's proposed changes to the Monticello design-basis containment response analysis methodology and resulting containment response are as follows:

1. Reanalyze at 102 percent of the licensed rated thermal power of 1775 megawatt thermal (MWt), and calculate decay heat using American National Standards Institute / American Nuclear Society (ANSI/ANS) 5.1-1979 with two standard deviation (2σ) uncertainty.

2. Increase reactor decay heat to account for additional nuclides as shown in General Electric Company's (GE's) Service Information Letter [(SIL)] 636, Revision 1, "Additional Terms Included in Reactor Decay Heat Calculations," dated June 6, 2001.

3. Increase the residual heat removal (RHR) heat exchanger heat transfer coefficient (K-value) from 143.1 British thermal units (BTUs)/sec-°F to 147 BTUs/sec-°F.

4. Analyze the effects of increasing the time to initiate long-term containment cooling on long-term containment response for reactor isolation and small-break LOCA conditions.

5. Analyze reactor vessel isolation and small-break LOCA events as part of determining the worst-case suppression pool heat-up event.

6. Update the long-term containment analysis for a design-basis LOCA with direct wetwell pool cooling.

7. Evaluate the effect of delaying the realignment of an RHR pump (from injection to containment cooling) on long-term containment response.

8. Increase the design temperature for the piping connected to the wetwell from 195 °F to 196.7 °F, and determine the maximum service water temperature that will keep the peak wetwell temperature below 196.7 °F. NMC determined this service water temperature to be 94 °F.

9. Perform short- and long-term containment analyses for the design-basis LOCA for NPSH calculations using values which minimize the containment pressure and maximize suppression pool temperature. These analyses used a service water temperature of 90 °F.

10. Determine the amount of containment overpressure necessary to ensure adequate available NPSH for the ECCS low-pressure pumps using the values of containment pressure and suppression pool temperature obtained in proposed change 9.

This requested license amendment proposes the following two licensing-basis values of service water temperature:

- 94 °F is the design-basis service water temperature for long-term LOCA calculations.

- 90 °F is the design-basis service water temperature for available NPSH calculations, including the determination of the required containment overpressure.

The lower service water temperature, 90 °F, would be operationally controlling (i.e., exceeding a service water temperature of 90 °F would exceed the Monticello licensing basis).

In addition to these changes, NMC's letter of December 6, 2002, said it will use the revised, long-term wetwell temperature response to update the plant environmental qualification program, as necessary.

2.0 REGULATORY EVALUATION

Monticello was licensed prior to the NRC issuing the General Design Criteria of Title 10, *Code of Federal Regulations* (10 CFR) Part 50, Appendix A. Section 1.2 of the Monticello USAR contains principal design criteria specific to Monticello. Section 1.2.4.b of the USAR says the following:

> Provision is made both for the removal of energy from within the primary containment and/or such other measures as may be necessary to maintain integrity of the primary containment system as long as necessary following the various postulated design-basis loss-of-coolant accidents.

Regulatory Guide (RG) 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," dated November 2, 1970, says that no increase in containment pressure from that present before a postulated LOCA should be considered in calculating available NPSH. However, RG 1.1 does not apply to some of the older operating reactors, including Monticello. These plants need some credit for the increase in containment pressure during a postulated design-basis LOCA in order to demonstrate that the available NPSH is greater than the required NPSH for the ECCS pumps. The NRC recently updated RG 1.1 to reflect the NRC's review of industry responses to NRC Generic Letter (GL) 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps." The provisions of RG 1.1 are included in RG 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident." RG 1.82, Revision 3 recognizes that it may not be practicable to alter the design of an operating reactor and, therefore, some overpressure may be needed to assure adequate available NPSH. Although RG 1.82, Revision 3, is not part of Monticello's licensing basis, the NRC staff used it as guidance for this review. Namely, that the containment accident pressure should be conservatively calculated and the amount of credit given for containment overpressure should be minimized.

NUREG-0800, "Standard Review Plan," Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," and Section 6.3, "Emergency Core Cooling System," give guidelines for evaluating the mass and energy releases reported in NMC's proposed change.

NMC's June 19, 1997, letter to the NRC provided NMC's long-term power uprate containment analysis and its long-term NPSH analyses (discussed in Appendix A of this SE). This submittal and the NRC's July 25, 1997, letter to NMC approving its analyses of June 19, 1997, establish the existing Monticello licensing basis for NPSH. NMC's license amendment request of December 6, 2002, as supplemented, proposes to revise its existing NPSH licensing basis, as discussed in the Introduction section of this SE.

3.0 TECHNICAL EVALUATION

The following contains the NRC staff's evaluations of the ten proposed changes listed in the Introduction.

3.1 Reanalyze at 102 Percent of the Licensed Rated Thermal Power of 1775 MWt, and Calculate Decay Heat Using ANSI/ANS 5.1-1979 with 2σ Uncertainty.

NMC's proposed analyses use Monticello's actual licensed, rated thermal power of 1775 MWt. The minimum-pressure containment analyses approved in the NRC staff's letter of July 25, 1997, assumed 102 percent of 1880 MWt power, and used the ANSI/ANS 5.1-1979 standard for calculating decay heat without 2σ uncertainty. As discussed in Appendix A of this SE, the previous analyses without 2σ uncertainty were acceptable because the thermal power used for the analyses was greater than the rated thermal power to account for the omission. However, the NRC staff's position is that the 2σ uncertainty should be included to give a conservative decay heat result.

The regulatory position of RG 1.49 states that accident analyses should be done at 102 percent of the licensed power level. NMC did this using ANSI/ANS 5.1-1979 with 2σ uncertainty. This is acceptable to the NRC staff since ANSI/ANS 5.1-1979 uses an accurate model which includes all significant phenomena related to decay heat and the 2σ uncertainty provides an acceptable level of conservatism. Use of ANSI/ANS 5.1-1979 is common industry practice and has been accepted by the NRC staff many times for NPSH calculations and other applications. Thus, the NRC staff finds NMC's use of ANSI/ANS 5.1-1979 with 2σ uncertainty acceptable.

3.2 Increase Reactor Decay Heat to Account for Additional Nuclides as Recommended in GE's SIL 636, Revision 1, "Additional Terms Included in Reactor Decay Heat Calculations," Dated June 6, 2001.

GE's SIL 636, Revision 1, discusses failing to account for the cumulative effects of radionuclides, other than U^{239} and Np^{239}, as well as activation products of structural materials in calculating shutdown power. NMC's submittal of December 6, 2002, says that this omission causes nonconservatisms that can potentially impact the results of the current licensing-basis, minimum containment pressure and NPSH calculations.

SIL 636, Revision 1, highlights that ANSI/ANS 5.1-1979 specifies that decay heat from activation products in structural materials and from actinides, other than those specifically included in the standard, need to be evaluated for inclusion in calculating shutdown power. SIL 636, Revision 1 states that previous evaluations of these decay heat sources found them to be negligible. However, a reassessment found that, although separately their contributions may be negligible, summed together their effect may not be negligible. Therefore, SIL 636, Revision 1 recommends actions to be taken to correct this deficiency. NMC states that its

NPSH calculations are consistent with these recommendations. NMC's use of SIL 636, Revision 1 corrects a known deficiency and is, thus, acceptable to the NRC staff.

3.3 Increase the RHR Heat Exchanger K-value from 143.1 BTU/sec-°F to 147 BTU/sec-°F.

The temperature of the suppression pool, and hence the NPSH margin, is sensitive to the value of K assumed for the RHR heat exchangers. NMC said that the original RHR heat exchanger K-value of 143.1 BTU/sec-°F was based on original vendor heat exchanger design calculations. The updated value is based on the state-of-the-art heat exchanger performance predictions under accident conditions. NMC said it performed the updated calculations using the computer code HTRI, and that NMC independently verified the calculations using the PROTO-HX code developed by the Proto-Power Company.

In addition to using recognized industry methods to predict K, NMC's letter of September 24, 2003, described how NMC ensures that the K-value will not decrease below the value of 147 BTU/sec-°F used in its safety analyses. NMC said that it checks the K-value annually to determine if there has been any deterioration in the heat removal capability. The following recent test results show that the K-values are greater than the proposed value of 147 BTU/sec-°F:

- 170.97 BTU/sec-°F for the #11 heat exchanger (measured on November 19, 2002)

- 159.9 BTU/sec-°F for the #12 heat exchanger (measured on February 16, 2003)

As NMC determined the K-value using industry-recognized methods, and since NMC annually verifies that it is greater than the value assumed in the safety analyses, the NRC staff finds that NMC's use of a K-value of 147 BTU/sec-°F in its safety analyses is acceptable.

3.4 Analyze the Effects of Increasing the Time to Initiate Long-Term Containment Cooling on Long-Term Containment Response for Reactor Isolation and Small-Break LOCA Conditions.

NMC used the SAFER/GESTR computer codes to analyze (1) the reactor vessel isolation event and (2) 0.01 and 0.1 ft^2 breaks to predict the time when the automatic depressurization system (ADS) activates and the time to initiate containment cooling. NMC submitted containment mass and energy calculations performed by GE Nuclear Energy, and documented in GE-NE-0000-0002-8817-01, Revision 1, "Monticello Nuclear Generating Plant Long-Term Containment Analysis," dated September 2002. In response to the NRC staff's request for additional information (RAI) of August 19, 2003, NMC submitted a revised set of analyses (GE-NE-0000-0002-8817-01, Revision 2, dated August 2003) on September 24, 2003. The NRC staff reviewed the analyses and their underlying assumptions and input data (e.g., decay heat generation and break flow area) as described below.

The design and sizing of the containment systems are largely based upon the pressure and temperature conditions which result from release of reactor coolant following a postulated LOCA. The containment design basis includes the effects of stored energy in the reactor coolant system, decay heat, and energy from other sources such as the secondary system and metal-water reactions.

The containment pressure and temperature response to a postulated LOCA is calculated with the SHEX code, based upon mass and energy releases that are determined with the SAFER/GESTR code. The NRC staff has accepted SAFER/GESTR LOCA methodology for use in licensing analyses. The NRC staff has previously evaluated the SHEX code in several license applications, including applications from NMC for Monticello. These include an evaluation of benchmarking studies for the Monticello Mark I containment documented in the following NRC letters to NMC:

- Amendment No. 98 to Facility Operating License No. DPR-22, “Monticello Nuclear Generating Plant - Issuance of Amendment Re: Updated Analysis of DBA [Design-Basis Accident] Containment Temperature and Pressure Response and Reliance on Containment Pressure to Compensate for Potential Deficiency in NPSH for ECCS Pumps During DBA,” TAC No. M97781, dated July 25, 1997.

- Amendment No. 102 to Facility Operating License No. DPR-22, “Monticello Nuclear Generating Plant - Issuance of Amendment Re: Power Uprate Program,” TAC No. M96238, dated September 16, 1998.

The SHEX code has been found to yield satisfactory results. Results of SHEX benchmarking studies were also presented in Appendices B of GE reports GE-NE-0000-0002-8817-01, Revisions 1 and 2.

In March 2001, GE modified the definition of the limiting event for the purposes of NPSH calculation. The limiting event became a large-break (4.095 ft^2) DBA LOCA with a 48.54-minute time delay for initiating containment cooling. This time delay was determined from the results of a SAFER/GESTR analysis of the reactor isolation event. This hybrid limiting event was deemed to be more conservative than the DBA LOCA event it replaced.

With containment cooling initiation time assumed to be 48.54 minutes, the resulting suppression pool temperature exceeded the 195 °F attached-piping design limit. Reducing the ultimate heat sink (UHS) temperature to 85 °F (from 90 °F) reduced the suppression pool temperature to below the 195 °F limit. Therefore, NMC administratively imposed an 85 °F maximum UHS temperature limit for Monticello. In February 2002, GE performed an updated DBA-LOCA containment analysis for Monticello. Some of the objectives of this analysis were as follows:

- Investigate the effect of increasing the maximum UHS temperature limit.

- Update the long-term containment analysis with decay heat input from the ANSI/ANS 5.1-1979 decay heat model, including a 2σ addition for uncertainty and incorporating the decay heat from activation products in structural materials and certain actinides.

- Evaluate a range of small-break LOCAs and reactor isolation events with respect to their effects upon long-term containment response.

- Update the RHR heat exchanger K-value to 147 BTU/sec-°F.

- Evaluate the effects of increasing the design temperature of the piping attached to the suppression pool.

- Evaluate the effect of delay in realigning RHR injection flow to containment cooling with flow through the RHR heat exchangers.

NMC's license amendment request of December 6, 2002, submitted its analysis results to the NRC. In response to the NRC staff's RAI of August 19, 2003, NMC's letter of September 24, 2003, submitted a revised GE report. The calculated mass and energy release rates in the revised report did not differ from those in NMC's original license amendment request.

The NRC staff followed the guidelines of Standard Review Plan Sections 6.2.1.5 and 6.3 to evaluate the mass and energy releases reported in NMC's proposed change. The mass and energy release calculations for long-term containment pressure response analysis were performed with accepted SAFER/GESTR methodology and with input assumptions that represented reasonable, incremental changes from prior analyses. The most notable change was in the conservative direction, when GE applied a 2σ uncertainty to the ANSI/ANS 5.1-1979 nominal decay heat model and incorporated the additional terms described in SIL 636. Another conservative assumption was the use of a longer delay time for initiating containment cooling, derived from the results of the reactor isolation case.

NMC used this methodology to evaluate a range of LOCA and reactor isolation event cases in order to determine the worst-case scenario (i.e., minimum containment pressure) to calculate the minimum NPSH available to the ECCS pumps. The NRC staff agrees with this approach, with the input assumptions that NMC used, and the methodology that it applied. The NRC staff has reviewed NMC's calculation of the mass and energy releases that are used to determine containment pressure response, including the methods and key underlying input assumptions (e.g., decay heat generation), and has determined that they are acceptable.

3.5 Analyze Reactor Vessel Isolation and Small-Break LOCA Events as Part of Determining the Worst-Case Suppression Pool Heat-up Event.

NMC considered several scenarios in order to assure that it used worst-case conditions to calculate available NPSH. NMC previously calculated a design-basis, large-break LOCA with several single-failure assumptions to determine the worst single failure. For this case, NMC assumed that the ECCS pumps operate at runout flow for 10 minutes after the pipe break without operator action. After 10 minutes, the operator reduces the pump flow rate to the design flow rate and initiates suppression pool cooling. NMC has now analyzed a scenario involving reactor vessel isolation with HPCI being unavailable. In this case, plant operators do not start suppression pool cooling until 48.5 minutes after reactor vessel isolation. NMC also assumed two, small piping breaks occurred to ensure that the reactor isolation was the limiting event for suppression pool cooling and hence suppression pool temperature. Reactor vessel isolation is the limiting case of a small break since more energy is held in the vessel.

The assumptions chosen for the design-basis, large-break LOCA overestimate the suppression pool temperature while underestimating containment pressure. For instance, suppression pool temperature is conservatively calculated by assuming the following:

- an upper bound to the decay heat
- technical specification minimum suppression pool water volume
- a single failure of a diesel generator
- using containment sprays rather than suppression pool cooling

These assumptions both maximized suppression pool temperature and minimized wetwell pressure. In addition to using containment sprays, NMC underestimated the initial air mass in the drywell and wetwell to further minimize the calculated pressure. Exhibit C of NMC's December 6, 2002, letter gives more detail on the input assumptions.

The reactor vessel isolation and small-break LOCA analyses also had conservative assumptions. For example, NMC assumed that containment (suppression pool) cooling didn't begin until after the ADS had depressurized the reactor coolant system. NMC said that this is conservative since suppression pool cooling could be started before ADS initiation occurs. The single failure assumed for this evaluation was a battery failure that eliminated the division of ECCS that contains HPCI. The remaining components were two, low-pressure coolant injection (LPCI) / RHR pumps and one core spray pump. One RHR pump must be shed when the service water pump is started to initiate containment cooling.

The NRC staff reviewed NMC's analyses assumptions for both the design-basis, large-break LOCA and the reactor vessel isolation with HPCI unavailable and finds the assumptions conservative and, therefore, acceptable. Table 1 below summarizes the cases NMC considered in its analyses.

Table 1 — Cases NMC Considered in its Analyses

Break Size	RHR Heat Exchanger K	SW Temp °F	Peak Suppression Pool Temp	Comment
Large break*	143.1***	90	195.6	Direct suppression pool cooling
Large break	147	90	194.1	Direct suppression pool cooling
Large break	147	90	194.2	Containment spray cooling
Large break	147	94**	196.5	Direct suppression pool cooling
Large break	147	94	196.2	Containment spray cooling
Rx vessel isolation	143.1	90	194.0	One RHR loop, HPCI unavailable, Direct suppression pool cooling
Rx vessel isolation	143.1	90	167.0	Two RHR loops with HPCI unavailable, Direct suppression pool cooling
.01 ft^2 small break	143.1	90	190.0	One RHR loop with HPCI unavailable, Direct suppression pool cooling
.1 ft^2 small break	143.1	90	191.2	One RHR loop with HPCI unavailable. Direct suppression pool cooling

* A study of single failures in NMC's letter of June 19, 1997, showed the failure of one emergency diesel generator with loss of offsite power to be most limiting.

** Suppression pool water temperature remains at 90 °F.

*** The original K-value for the RHR heat exchanger was 143.1 BTU/sec-°F while the updated value is 147 BTU/sec-°F.

There are several important conclusions from these calculations, as follows:

- The limiting event with respect to peak suppression pool temperature was a large-break LOCA. For this case, the current suppression pool temperature limit of 195 °F was exceeded when K was assumed to be at its current value of 143.1 BTU/sec-°F. When the proposed RHR heat exchanger K-value of 147 BTU/sec-°F was used with a service water temperature of 90 °F, the suppression pool temperature was less than the current suppression pool temperature limit.

- For the large-break LOCA with an RHR heat exchanger K-value of 147 BTU/sec-°F, service water temperature at 94 °F, and suppression pool temperature at 90 °F, the peak suppression pool temperature was slightly less than the proposed suppression pool (wetwell) temperature limit of 196.7 °F. NMC performed this calculation to determine the maximum service water temperature that would keep peak suppression pool temperature below the limit of 196.7 °F.

- The reactor vessel isolation events required a longer time to establish suppression pool cooling but were less limiting than the large-break LOCA. These analyses were performed using the SHEX containment computer code to predict containment response. The GESTR/SAFER computer codes gave the ADS activation time and the RHR suppression pool cooling initiation time for the reactor vessel isolation events. The ADS activation time was determined by assuming the automatic initiation is inhibited and that the operator initiates ADS when the water level outside the core shroud decreases to the top of the active fuel. This was conservative. The total time to initiate containment cooling was determined from among several possible procedures. NMC selected the procedure that resulted in the longest time.

- The Monticello design-basis service water temperature for the long-term LOCA analyses is 94 °F. The Monticello design basis service water temperature for the determination of NPSH is 90 °F. Both service water temperatures are supported by appropriate analyses.

SHEX uses a coupled model of the reactor vessel and the containment. The NRC staff has previously accepted the SHEX computer code for Monticello containment analyses, as well as containment analyses for other boiling-water reactors. The NRC staff has obtained acceptable comparisons of calculated results from SHEX and CONTAIN 2.0 (the NRC's containment computer code) in previous audit calculations. Therefore, the NRC staff considers SHEX to be acceptable for these Monticello calculations.

NMC pointed out in Exhibit C, Section 4.2.1, of its December 6, 2002, letter that previous analyses showed that the reactor vessel isolation event would be more limiting. This was NMC's basis for administratively limiting service water temperature to 85 °F rather than the licensing basis value of 90 °F. This was the case since the peak suppression pool temperature for the isolation event was obtained by analyzing the design-basis, large-break LOCA assuming containment cooling initiation was delayed from 10 minutes (the containment cooling initiation time for a LOCA) to 48.54 minutes (the containment cooling initiation time for the reactor vessel isolation). This is unrealistic, and more conservative than required, since it combines the conservative features from two distinct scenarios; the large-break LOCA and the reactor vessel isolation. Exhibit C, Section 4.2.1 explains that for the more realistic calculation, the design-basis, large-break LOCA is more limiting because more sensible energy is transferred from the vessel to the suppression pool for this event than for the reactor vessel isolation.

NMC's letter of June 19, 1997, analyzed other single failures since there was a question that other failures, while not resulting in the highest suppression pool temperature, might give lower available NPSH values because of higher flow rates and, therefore, higher frictional resistance. NMC found that the single failure used in its December 6, 2002, analyses was most limiting with respect to NPSH.

Therefore, the NRC staff finds NMC's determination that the design-basis, large-break LOCA is the most limiting event in terms of suppression pool temperature to be acceptable since NMC did the analyses with acceptable methods and conservative assumptions. The use of two different service water temperatures (94 °F for long-term LOCA analyses and 90 °F for available NPSH calculations) is acceptable. As stated previously, the lower service water temperature, 90 °F, would be operationally controlling (i.e., exceeding a service water temperature of 90 °F would exceed Monticello's licensing basis).

3.6 Update the Long-Term Containment Analysis for a Design-Basis LOCA with Direct Wetwell Pool Cooling.

As discussed in Section 3.3 of this SE, NMC updated the K-value of the RHR heat exchanger from 143.1 BTU/sec-°F to 147 BTU/sec-°F. NMC used the updated RHR heat exchanger K-value to calculate the Monticello long-term response to the limiting design-basis LOCA. NMC based its calculations on the following:

- thermal power was 102% of 1775 MWt
- service water temperature was 90 °F
- added 2σ uncertainty to the decay heat calculated with the ANSI/ANS 5.1-1979 model
- included the adjustments of GE's SIL 636, Revision 1
- maximum initial containment pressure
- minimum initial drywell humidity
- direct suppression pool cooling using one loop of RHR (one pump and one heat exchanger)

NMC assumed maximum initial containment pressure and minimum initial drywell humidity to maximize peak containment pressure. Exhibit C in Section 4.4.1 of NMC's December 6, 2002, submittal gives more details on the inputs NMC used. The peak suppression pool temperature was 194.1 °F and the peak pressure was 18.3 psig. The corresponding limits are, respectively, 281 °F and 56 psig. Since NMC did the calculations with conservative assumptions and with the SHEX computer code, the NRC staff finds these results acceptable.

3.7 Evaluate the Effect of Delaying the Realignment of an RHR Pump from Injection to Containment Cooling on Long-Term Containment Response.

NMC's analyses supporting the power uprate considered the case of long-term containment cooling. In this case, the RHR pumps remain in the LPCI injection mode (injecting into the vessel and exiting through the break) with the flow routed through the RHR heat exchangers rather than the containment cooling mode (direct cooling of the suppression pool water). NMC estimated the difference in suppression pool temperature between these two cooling modes to be about 1 °F. Thus, no limits are exceeded if the LPCI injection mode is continued rather than switching to the RHR suppression pool cooling mode.

3.8 Increase the Design Temperature for the Piping Connected to the Wetwell from 195 °F to 196.7 °F, and Determine the Maximum Service Water Temperature That Will Keep the Peak Wetwell Temperature Below 196.7 °F.

NMC's letter of September 24, 2003, describes the following method it used to determine the revised design temperature for the piping connected to the wetwell:

> The piping and supports affected by the postulated long-term LOCA temperature of 196.7 °F were evaluated by direct ratio of the original rerate thermal stress results at

195 °F. Recalculation of the stresses and [American Society of Mechanical Engineers] Code compliance checks for the governing load combinations followed. Accordingly, consistency with the original licensing basis analysis methodology was preserved.

Based on the small increase in the temperature limit and the linearity of the stresses in the piping and supports, the NRC staff finds this method acceptable.

3.9 Perform Short- and Long-Term Containment Analyses for the Design-Basis LOCA for NPSH Calculations Using Values Which Minimize the Containment Pressure and Maximize Suppression Pool Temperature.

NMC performed these calculations with conservative inputs that maximize suppression pool temperature and minimize wetwell pressure. The long-term calculations assumed a 90 °F service water temperature and an RHR heat exchanger K-value of 147 BTU/sec-°F. The short-term calculations are independent of these quantities since there is no suppression pool cooling in the short term. NMC assumed different worst-single failures for each case.

NMC assumed that drywell temperature and humidity were 150 °F and 100 percent, respectively. These values result in a lower air mass than with lower values of temperature and humidity and, therefore, result in a lower pressure, which is conservative.

NMC assumed the wetwell air space was in thermal equilibrium with the wetwell water during the early blowdown period. Thereafter, mechanistic heat and mass transfer were calculated to determine the pool and wetwell air temperature. NMC said that the change from a thermal equilibrium assumption to a mechanistic calculation is responsible for the change in necessary containment overpressure in the time interval between 2000 and 4000 seconds, as shown in Table 2 of this SE.

The long-term calculations take containment leakage into account. NMC said that for the 12-day (long-term) calculation, containment leakage reduces the calculated containment pressure by 2 psi. NMC calculated the containment conditions for both the short-term and long-term with conservative assumptions using the SHEX code which is acceptable to the NRC staff. Therefore, these results are acceptable.

3.10 Determine the Amount of Containment Overpressure Necessary to Ensure Adequate Available NPSH for the ECCS Low-Pressure Pumps Using the Values of Containment Pressure and Suppression Pool Temperature Obtained in Proposed Change.

NMC calculated available NPSH values as a function of time for the RHR and core spray pumps using the containment pressure and temperature results discussed in Section 3.9 of this SE (see Exhibit F in NMC's letter of December 6, 2002). NMC's submittal said that the calculations took into account head loss due to LOCA-generated debris transported to, and accumulated on, the ECCS suction strainers. The submittal further said that the new analyses of the December 6, 2002, letter did not change the licensing basis for LOCA-generated debris described in the USAR. The debris loading is used to calculate the head loss across the ECCS strainers, which is an input to the NPSH calculations.

The containment pressure as a function of time required for adequate NPSH is shown in proposed USAR Figure 5.2-15c for the short-term, and Figure 5.2-15d for the long-term

(Exhibit A of NMC's December 6, 2002, letter). Table 2 below, from NMC's letter of December 6, 2002, gives values of containment pressure needed to ensure adequate NPSH for the low-head ECCS pumps.

Table 2 — Current and Requested Containment Pressure
Needed for Adequate NPSH for Low Head ECCS Pumps

Time (Seconds)	Current USAR Containment Pressure (psig)	Proposed Containment Pressure (psig)
10 to 600	2.0	2.0
600 to 2,000	2.0	2.0
2,000 to 4,000	4.0	3.25
4,000 to 10,000	4.0	4.0
10,000 to 16,000	5.3	5.3
16,000 to 55,000	6.1	6.1
55,000 to 69,000	5.6	5.6
69,000 to 85,000	5.0	5.0
85,000 to 110,000	4.2	4.2
110,000 to 140,000	3.3	3.3
140,000 to 200,000	2.3	2.3
200,000 to 330,000	1.0	1.0

Since credit is taken for a portion of the accident pressure to ensure adequate NPSH for the ECCS pumps, a possible concern is venting of the containment, which is allowed by plant emergency operating procedures under certain conditions. NMC discussed this in a July 16, 1997, letter and committed to add a caution in the emergency operating procedures to include NPSH considerations while venting to control primary containment pressure.

Venting for hydrogen control following an accident is also allowed by the Monticello emergency operating procedures. This could occur only for an accident beyond the design basis. However, offsite releases are to be maintained below technical specification values. Since the generation of hydrogen would necessarily involve fuel failure, venting is effectively precluded. Therefore, NPSH would be maintained to the ECCS pumps even in considerations involving primary containment venting.

NMC's July 21, 1997, letter discussed conservatism in the NPSH analyses due to initial conditions such as the following:

- initial containment pressure and humidity
- containment leakage rate during the accident
- suppression pool water level
- service water (UHS) temperature

By using conservative inputs for these parameters, as was done for the analyses described in NMC's letter of December 6, 2002, the containment pressure is underestimated and the suppression pool temperature is overestimated. This is conservative for NPSH calculations.

The containment pressure necessary for adequate available NPSH is ensured by (1) assuring that it is less than a calculated containment pressure which is underestimated, and (2) emergency operating procedures which ensure that containment accident pressure will be maintained when required during the accident. NMC's containment pressure and NPSH calculations are, therefore, acceptable to the NRC staff.

3.11 Summary of Conclusions

NMC has re-analyzed the containment conditions used to calculate NPSH of the Monticello ECCS pumps. Results were then used to recalculate the NPSH for these pumps. NMC determined the limiting case and performed the calculations using methods previously approved for Monticello or otherwise evaluated in this SE. The NRC staff finds that NMC used conservative assumptions which underestimate the containment pressure and overestimate the suppression pool water temperature. Some overpressure is necessary to ensure sufficient available NPSH. The conservative assumptions used in NMC's calculations and the cautions in the Monticello emergency operating procedures ensure that this pressure will be available. For these reasons, as discussed in more detail in the sections above, the NRC staff finds the proposed license amendment to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact has been prepared and published in the *Federal Register* on January 14, 2004 (69 FR 2163). A revised Environmental Assessment and Finding of No Significant Impact has been prepared and published in the *Federal Register* on May 26, 2004 (69 FR 29983) as a result of NMC's supplemental letter of March 12, 2004. Accordingly, based on the environmental assessments, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Appendix A — Monticello Design and Licensing Bases Information

The Monticello Nuclear Generating Plant (Monticello) is a General Electric-designed boiling-water reactor (BWR) with a Mark I containment. The Mark I containment system consists of a leak-tight drywell which encloses the reactor vessel, the reactor coolant system, and other branch connections to the reactor coolant system. A toroidal-shaped pressure suppression chamber (torus) surrounds the lower part of the drywell. The torus is partially filled with a large volume of water, the suppression pool, above which is an air space. The drywell atmosphere is connected by a vent system to the suppression pool. The suppression pool and the air space above it inside the torus comprise the wetwell. In the case of a high-energy line break in the drywell, the vent system directs steam and drywell air to the suppression pool where the steam is condensed. The suppression pool is also the source of water for the emergency core cooling system (ECCS) pumps.

The Monticello low-head ECCS pumps include the core spray pumps and the residual heat removal (RHR) pumps. The RHR system pumps have the following functions:

- shutdown cooling
- low-pressure coolant injection (LPCI) ECCS pumps
- transferring heat to the Monticello ultimate heat sink to cool the suppression pool water following a loss-of-coolant accident (LOCA)

A July 26, 1996, Northern States Power Company (the Monticello licensee at that time) submittal requested an increase in Monticello's licensed rated thermal power from 1670 megawatts thermal (MWt) to 1775 MWt. This submittal was later superseded by a December 4, 1997, submittal. The Nuclear Regulatory Commission (NRC) staff approved this power increase in its letter of September 16, 1998. As part of the power uprate, Nuclear Management Company, LLC (NMC) calculated the response of the containment to a design-basis LOCA. This included the peak containment pressure and temperature in the drywell and the wetwell in both the short and the long term. The short term is defined as less than 10 minutes after initiation of the event. No operator action is credited during this time. The long term is the time period greater than 10 minutes after initiation of the event. Operator actions are credited during this time. The short-term analysis determined the drywell pressure response during the initial blowdown of the reactor vessel inventory into the containment. The long-term analysis determined the peak suppression pool temperature considering the addition of heat (core decay heat, fission power, fuel relaxation energy, metal-water reaction heat, and heat stored in the vessel structure). Separate analyses, with different assumptions, were performed for both the short and long term to demonstrate adequate available NPSH for the low-head ECCS pumps. These analyses assumed a reactor power of 1880 MWt which is greater than the requested power increase. The analyses used the nominal American National Standards Institute / American Nuclear Society (ANSI/ANS) 5.1-1979 decay heat model without accounting for uncertainty in the model. The NRC staff's position is that a 2σ uncertainty should be included for conservatism in calculating decay heat using the ANSI/ANS 5.1-1979 model where σ is the standard deviation in the ANSI/ANS 5.1-1979 decay heat model. However, NMC's use of the higher power level compensates for omitting the 2σ (see NMC's letter to the NRC, response to Question 51, dated September 5, 1997).

The limiting short-term event for net positive suction head (NPSH) considerations is the design-basis, large-break LOCA with LPCI loop-select logic failure analyzed in NMC's June 19, 1997, letter. The limiting long-term event is a design-basis LOCA with containment spray cooling, loss-of-offsite power, and failure of one emergency diesel generator.

The June 19, 1997, letter from Northern States Power Company supplemented the July 26, 1996 power rerate request by requesting a license amendment to clarify the technical specification requirements for the minimum RHR and RHR service water pumps required to be operable for post-accident containment heat removal. The letter also requested credit for containment pressure greater than the initial containment pressure (containment overpressure) in the calculation of available NPSH for the ECCS pumps following a LOCA. These issues were raised as a result of an NRC System Operational Performance Inspection which was completed on January 8, 1997. A July 25, 1997, letter from the NRC (License Amendment No. 98) to NMC approved NMC's license amendment request of June 19, 1997. The amendment also authorized an update to Monticello's USAR to reflect the reliance upon containment pressure in the analyses. These analyses supplemented the results of the power uprate analysis.

The NRC issued Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water reactors," on May 6, 1996. NRC Bulletin 96-03 requested that licensees of BWRs, including Monticello, implement procedures and plant modifications to minimize the potential for clogging of the ECCS suction strainers located in the suppression pool. The licensee responded in a November 1, 1996, letter to the NRC that large capacity passive strainers would be installed in the 1998 refueling outage. This was done. The head loss due to ECCS suction strainer clogging is an important input to NPSH calculations.

When credit is taken for containment pressure in calculating available NPSH during a LOCA, the containment pressure is conservatively underestimated and the suppression pool water temperature is overestimated. This ensures adequate margin between the available and the required NPSH. NMC's letter of June 19, 1997, used the following assumptions for the containment calculations used for determining available NPSH:

- Reactor thermal power level was increased 5.92% (from 1775 MWt to 1880 MWt) to account for not adding a 2σ uncertainty to the ANSI/ANS 5.1-1979 decay heat model.

- Calculated decay heat using the ANSI/ANS 5.1-1979 nominal value with no uncertainty included.

- ECCS pumps automatically started and ran for 10 minutes at the maximum possible flow rates (runout flow).

- At 10 minutes after pump start, the operator reduced the ECCS pump flow rate to the design value used for long-term core and containment cooling.

- RHR service water temperature was at the licensing basis value of 90 °F.

- RHR heat exchanger K-value [heat transfer coefficient] was 143.1 BTU/sec-°F.

- Initial suppression pool water temperature was 90 °F.

- Suppression chamber airspace temperature was 90 °F.
- Design temperature of the piping attached to the suppression pool was 195 °F.
- Initial drywell temperature was 135 °F.
- Design-basis LOCA was the limiting event for calculation of peak suppression pool temperature.

NMC's letter of December 6, 2002, requested a license amendment to update both Monticello's long-term power uprate containment analysis and its long-term NPSH analyses contained in its June 19, 1997, letter. NMC's letter of December 6, 2002, also reported that NMC had recently administratively lowered the allowable service water temperature limit from 90 °F to 85 °F. This was based on the calculated results of a postulated reactor vessel isolation event with delayed initiation of suppression pool cooling. One purpose of NMC's December 6, 2002, license amendment request is to justify an increase the service water temperature. NMC proposes a service water temperature licensing basis of 94 °F for the long-term LOCA calculations and 90 °F for the ECCS pump NPSH calculations. The lower service water temperature, 90 °F, would be operationally controlling (i.e., exceeding a service water temperature of 90 °F would exceed Monticello's licensing basis).