NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

January 30, 2002

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

10 CFR Part 50
Section 50.90

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Response to NRC Request for Additional Information Regarding License Amendment Request for Revised Reference for Reactor Vessel Level Setpoints, Simplification of Safety Limits, and Improvements to the Bases (TAC NO. MB2246)

Reference 1: Nuclear Management Company, LLC, "License Amendment Request dated June 18, 2001, Changes to the Technical Specifications, Revised Reference Point for Reactor Vessel Level Setpoints, Simplification of Safety Limits, and Improvements to the Bases"

Reference 2: Nuclear Regulatory Commission letter, "Monticello Nuclear Generating Plant – Request for Additional Information Related to License Amendment Request (TAC NO. MB2246)," dated December 21, 2001

By letter dated June 18, 2001, Nuclear Management Company, LLC (NMC) submitted a License Amendment Request for changes to the Technical Specifications (TS) related to a revised reference point for reactor vessel level setpoints, simplification of Safety Limits, and improvements to the Bases (Reference 1). By letter dated December 21, 2001 (Reference 2), NRC requested NMC to provide additional information related to this License Amendment Request.

The attached Exhibit A contains NMC's response to the NRC's Request for Additional Information contained in Reference 2. Exhibit B contains revised Monticello TS and TS Bases pages which are being submitted to reflect changes that have been made since the submittal of the Reference 1 License Amendment Request.

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The Reference 1 submittal was evaluated in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92, and determined not to involve any significant hazards considerations. Additionally, the Reference 1 submittal was determined to meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9) and pursuant to 10 CFR 51.22(b) an Environmental Assessment was not required. The attached information does not impact either of these evaluations, therefore, the No Significant Hazards Consideration Determination and the Environmental Assessment submitted by the MNC letter dated June 18, 2001, is also applicable to this submittal.

The Exhibit B cover sheet provides remove and insert instructions for the Reference 1 submittal.

Please contact Mr Doug Neve, Licensing Project Manager, at 763-295-1353 if you require additional information related to this submittal.

Jeffrey S. Forbes
Site Vice President
Monticello Nuclear Generating Plant

Subscribed to and sworn before me this 30th day of January, 2002

Kathryn D Kleine
Notary

KATHRYN I. KLEINE
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2005

Attachments: Exhibit A: Response to Request for Additional Information
Exhibit B: Revised Monticello Technical Specification Pages

c: Regional Administrator-III, NRC
NRR Project Manager, NRC
Sr. Resident Inspector, NRC
Minnesota Department of Commerce
J Silberg, Esq.

Exhibit A
Response to NRC Request for Additional Information
License Amendment for Revised Reference for Reactor Vessel Level Setpoints,
Simplification of Safety Limits, and Improvements to the Bases (TAC NO. MB2246)

1. Exhibit A, Section II.1, "Reactor Vessel Level Instrumentation Reference Point" discusses removing the upper limit specification on reactor vessel level.

Please provide your safety basis for requesting this change to the TS. This section also states the removal makes the specification consistent with NUREG-1433, Standard Technical Specification (STS), and that the instrument calibration will conform to standard setpoint methodology. Please support or clarify these statements in your analysis and discuss how the STS and the setpoint methodologies, if they were evaluated for application to Monticello, are related to the proposed change.

Response

The Monticello Technical Specifications (TS) establish the Low Low Reactor Water Level trip setting as ≥ 6' 6", ≤ 6'10" Above Top of Active Fuel (ATOAF). This is equivalent to trip settings based on the reactor vessel instrument zero reference point of ≥ -48", ≤ -44".

The –48" TS lower limit of the trip setting was chosen to be high enough to initiate emergency core cooling system (ECCS) operation and Group 1 containment isolation to meet the acceptance criteria of 10CFR50, 10CFR50.46, 10CFR50 Appendix K, and 10CFR100. As explained in the Section 3.2 of the TS Bases, an allowable deviation of –3" in the lower limit is permitted, and is taken into account in the transient and accident analyses.

Monticello continues to use a custom TS format which is essentially unchanged since initial licensing. The current Monticello TS setpoints for reactor water level are not based on allowable values used in the Standard TS, NUREG-1433. However, actual lower limit nominal setpoints have been demonstrated to be conservative using the General Electric Setpoint Methodology, NEDC-31336. The actual nominal setpoint is more conservative than the setpoint obtained by applying a Licensee Event Report (LER) avoidance factor to the most conservative allowable value obtained from:

TS setting plus TS allowable deviation,

or,

Low Low Reactor Water Level analytical limit specified on General Electric Transient Analysis Protection Parameter (TRAPP) form plus calculated setpoint margin.

Historical performance of this setpoint gives sufficient confidence that spurious trips will not occur.

The current TS –44" upper limit of the trip setting was chosen simply to be low enough to prevent spurious operation of the ECCS initiation and containment isolation logic. It has no basis in the safety analyses.

It is therefore concluded that the upper limit trip setting is unnecessary since (1) the established conservative low limit trip setting does not give rise to a spurious trip concern, and (2) it is not used in the safety analyses.

2. Exhibit A, Section II.2, "Simplify the Safety Limits and Limiting Safety System Settings (LSSSs)," discusses incorporating LSSSs into Section 3 of the TS (Reactor Protection System (RPS) table).

The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. This is achieved by specifying LSSS in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance. *The LSSSs are defined as the Allowable Values*, which, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including Safety Limits during design-basis accidents.

The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The allowable values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g. drift).

Trip setpoints are those predetermined values of output at which an action should take place. A channel is inoperable if its actual trip setpoint is not within its required allowable value.

The STS, Section B 3.3.1.1, explicitly explain why the allowable value is acceptable to be the LSSS. It is inadvisable to make the trip setpoint the same as the LSSS. If the trip setpoint is the same as the LSSS and the setpoint drifts, the licensee must report to the Commission and preclude it from happening again, which would be impossible. Alternatively, both a nominal trip setpoint and allowable value can be included in the TS. If it is your intention to define the trip setpoint as the LSSS, it should be clearly stated in the TS Bases. Please clarify your intentions in this regard.

Response

As noted above, Monticello continues to use a custom TS format which is essentially unchanged since initial licensing. Nominal trip setpoints and allowable deviation values are used in the TS with the exception of several new instrument setpoints added to the TS in recent years where allowable values are specified.

While the current Monticello TS setpoints for reactor water level are not based on allowable values used in the NUREG-1433, the actual setpoints used have been demonstrated to be conservative using the General Electric Setpoint Methodology described in NEDC-31336.

Adoption of the NRC Standard TS at Monticello is currently under consideration by NMC management. If the Standard TS format is adopted in the future, all setpoints will be converted to the technically superior allowable value format.

3. Exhibit A, Section II.2, “Simplify the Safety Limits and Limiting Safety System Settings,” discusses a 12-inch reduction in the Safety Limit reference for reactor vessel water level.

Please discuss any impact on accident analyses affected by this changes.

Response

The reactor vessel water level Safety Limit is not used in any transient or accident analyses.

During power or startup operation, all required ECCS and containment isolation safety features are automatically initiated six or more feet above the top of the core. The Safety Limit is not approached during normal operation and is challenged only in the event of a loss of coolant accident. Core cooling in this case is assured by restoring reactor vessel water level to 2/3 core height and by two redundant core spray subsystems.

The reactor vessel level Safety Limit is more directly applicable to shutdown or refueling operations. During this period reactor decay heat is removed by the Residual Heat Removal (RHR) System. Either of two redundant RHR subsystems takes suction from a reactor recirculation loop, located well below the top of the core, passes the coolant through one or both RHR heat exchangers, and returns the coolant to a reactor recirculation loop. Covering all of the active fuel elements in the core with water is sufficient to ensure the RHR system is effective in cooling the core.

During in-vessel inspection and maintenance activities, it may be necessary to reduce reactor vessel water level below normal levels. During these periods vessel level is closely monitored and redundant sources of backup makeup water may be provided. In the event of loss of both RHR trains and coincident failure of all means of makeup to the reactor vessel during these periods, an additional 12" of coolant inventory above the top of the active fuel could delay the onset of boiling in the core and possible core uncovery by several minutes, but this sequence of events is extremely unlikely.

Therefore, a reactor vessel Safety Limit based on covering all of the active fuel with coolant is sufficient to ensure the core is adequately cooled and fuel damage does not occur.

4. Exhibit A, Section III.2, "Simplify the Safety Limits and Limiting Safety System Settings," discusses converting TS units to psig from psia by subtracting "15 psi" from current TS values.

Exhibit A
Response to NRC Request for Additional Information
License Amendment for Revised Reference for Reactor Vessel Level Setpoints,
Simplification of Safety Limits, and Improvements to the Bases (TAC NO. MB2246)

Please discuss any impact the "0.3 psi" change in margin to atmospheric pressure (14.7 psi) has on accident analyses and TS limits.

Response

The parameter values proposed for conversion from units of psia to psig are:

Thermal Power and MCPR Applicability

From:	To
Reactor pressure (800 psia)	Reactor steam dome pressure (785 psig)

Safety Limit Bases

From:	To:
Reactor pressure (800 psia)	Reactor pressure (785 psig)
ATLAS test pressure (14.7 – 800 psia)	ATLAS test pressure (0 – 785 psig)
Reactor pressure (1400 psia)	Reactor pressure (1385 psig)

The conversion from psia units to psig units was done by subtracting 15 psi from the psia values to achieve consistency with the NRC Standard TS. The value of 15 psi is appropriate. Applying a more precise conversion value, such as 14.7 psi (average barometric pressure at sea level), would imply a higher degree of precision in the converted value than it deserves. That is, specifying a value of 785.3 psig suggests the accuracy of the result is known to one more additional decimal place than 785 psig.

A reactor thermal power Safety Limit of $\leq$ 25% of rated thermal power at reactor pressure below 785 psig is essentially as conservative as $\leq$ 25% of rated thermal power at reactor pressure below 785.3 psig. As noted in Bases Section 2.1.B of the current TS Bases (Section 2.1.A.1 of proposed Bases Section) there is a factor of approximately two in the margin for this limit. That is, the Safely Limit is established at 25% core average thermal power while the actual critical power of concern is equivalent to approximately 50% core average thermal power.

Applying the MCPR Safety Limit at the lower pressure of 785 psig vs. 785.3 psig is conservative.

Changes in the Bases from 800 psia to 785 psig and 1400 psia to 1385 psig in lieu of 800 psia to 785.3 psig and 1400 psia to 1385.3 psig are appropriate for the reasons given above.

5. Please cite the NRC-approved instrumentation trip setpoint and allowable value methodologies utilized in your calculations.

Response

As noted above, Monticello uses a custom TS format. With few exceptions, the current Monticello TS setpoints are not based on allowable values derived from setpoint methodology calculations used in the Standard TS. General Electric Setpoint methodology calculations have been used, however, to confirm the conservatism of setpoints specified in instrument calibration procedures.

The setpoints proposed in our License Amendment Request dated June 18, 2001, are based on the existing TS custom format. TS setpoints are in the form of trip settings and allowable deviations.

The only changes proposed are in reactor vessel level instrument "zero" reference point. No actual changes of any significance in instrument setpoints will result from the proposed TS changes.

As noted above, if the Standard TS format is adopted at Monticello in the future, all TS setpoints will be converted to allowable values.

Exhibit B

Response to NRC Request for Additional Information
License Amendment for Revised Reference for Reactor Vessel Level Setpoints, Simplification of Safety Limits, and Improvements to the Bases (TAC NO. MB2246)

Revised Monticello Technical Specification Pages

Remove pages	Insert pages
-	iv
6	6
12	12
60d	60d
127	127
151	151
249b	250

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2.0 SAFETY LIMITS

2.1 SAFETY LIMITS

A. Reactor Core Safety Limits

1. With the reactor steam dome pressure < 785 psig or core flow $< 10\%$ rated core flow:

   Thermal power shall be $\leq 25\%$ Rated Thermal Power

2. With the reactor steam dome pressure ≥ 785 psig and core flow $\geq 10\%$ rated core flow:

   MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

3. Reactor vessel water level shall be greater than the top of active irradiated fuel.

B. Reactor Coolant System Pressure Safety Limit

Reactor steam dome pressure shall be ≤ 1332 psig.

LIMITING SAFETY SYSTEM SETTINGS

Limiting Safety System Settings are incorporated into Section 3 of the Technical Specifications.

Bases 2.2:

Exceeding a Safety Limit may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," guidelines. Therefore, it is required to insert all insertable control rods and restore compliance with the Safety Limits within 2 hours. The 2 hour completion time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal. Other required actions are delineated in 10 CFR 50.36, 10 CFR 50.72, and 10 CFR 50.73

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Table 3.2.8 Other Instrumentation					
Function	Trip Setting	Minimum No. of Operable or Operating Trip System (1) (2)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1) (2)	Required Conditions*
A. RCIC Initiation					
1. Low-Low Reactor Level	≥ -48″	1	4	4	B
B. HPCI/RCIC Turbine Shutdown					
1. High Reactor Level	≤ 48″	1	2	2	A
C. HPCI/RCIC Turbine Suction Transfer					
1. Condensate Storage Tank Low Level Allowable Values	≥ 2′ 3″ above tank bottom (Two Tank Operation)	1	2	2	C
	≥ 6′ 9″ above tank bottom (One Tank Operation)	1	2	2	C

NOTE:

1. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated as follows:
   a. With one required instrument channel inoperable per trip function, place the inoperable channel or trip system in the tripped condition within 12 hours, or
   b. With more than one instrument channel per trip system inoperable, immediately satisfy the requirements by placing the appropriate channels or systems in the tripped condition, or
   c. Place the plant under the specified required condition using normal operating procedures.
2. A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other operable channel in the same trip system is monitoring that parameter.

* Required conditions when minimum conditions for operation are not satisfied:

A. Comply with Specification 3.5.A.
B. Comply with Specification 3.5.D.
C. Align HPCI and RCIC suction to the suppression pool.

3.0 LIMITING CONDITIONS FOR OPERATION

E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F the safety valve function (self actuation) of seven safety/relief valves shall be operable (note: Low-Low Set and ADS requirements are located in Specification 3.2.H. and 3.5.A, respectively).

   Valves shall be set as follows:

   8 valves at $\leq$1120 psig

2. If Specification 3.6.E.1 is not met, initiate an orderly shutdown and have reactor coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

E. Safety/Relief Valves

1. a. Safety/relief valves shall be tested or replaced each refueling outage in accordance with the Inservice Testing Program.

   b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.

   c. The integrity of the safety/relief valve bellows shall be continuously monitored.

   d. The operability of the bellows monitoring system shall be demonstrated each operating cycle.

2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

Bases 3.6/4.6 (Continued):

The safety/relief valves have two functions; 1) over-pressure relief (self-actuation by high pressure), and 2) Depressurization/ Pressure Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation).

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9, Section N-911.4(a)(4) of the ASME Pressure Vessel Code Section III Nuclear Vessels (1965 and 1968 editions) requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

Testing of the safety/relief valves in accordance with ANSI/ASME OM-1-1981 each refueling outage ensures that any valve deterioration is detected. An as-found tolerance value of 3% for safety/relief valve setpoints is specified in ANSI/ASME OM-1-1981. Analyses have been performed with the valves assumed to open at 3% above their setpoint of 1109 psig. The 1375 psig Code limit is not exceeded in any case. When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system once per cycle provides assurance of bellows integrity.

I. Deleted

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7. Core Operating Limits Report

a. Core operating limits shall be established and documented in the Core Operating Limits Report before each reload cycle or any remaining part of a reload cycle for the following:

Rod Block Monitor Operability Requirements (Specification 3.2.C.2a)
Rod Block Monitor Upscale Trip Settings (Table 3.2.3, Item 4.a)
Recirculation System Power to Flow Map Stability Regions (Specification 3.5.F)
Maximum Average Planar Linear Heat Generation Rate Limits (Specification 3.11.A)
Linear Heat Generation Rate Limits (Specification 3.11.B)
Minimum Critical Power Ratio Limits (Specification 3.11.C)
Power to Flow Map (Bases 3.1)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (the approved version at the time the reload analyses are performed)

NSPNAD-8608-A, "Reload Safety Evaluation Methods for Application to the Monticello Nuclear Generating Plant" (the approved version at the time the reload analyses are performed)

NSPNAD-8609-A, "Qualification of Reactor Physics Methods for Application to Monticello" (the approved version at the time the reload analyses are performed)

NEDO-31960, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," June 1991 (the approved version at the time the reload analyses are performed)

NEDO-31960, Supplement 1, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," March 1992 (the approved version at the time the reload analyses are performed)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

d. The Core Operating Limits Report, including any mid-cycle revisions or supplements, shall be supplied upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.