NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

January 30, 2004

L-MT-04-003
10 CFR Part 50
Section 50.90

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

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

License Amendment Request For Technical Specification Table 3.2.3 and Section 3.7/4.7

Pursuant to 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) hereby requests the following amendment. The proposed changes would revise Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS) in the following manner:

- Clarify the permissive set point for the Source Range Monitor Detector-not-fully-inserted rod block bypass.
- Correct a typographical error in the surveillance requirement for Suppression Pool Temperature Monitoring
- Clarify the set point for the Pressure Suppression Chamber-Reactor Building Vacuum Breakers instrumentation
- Clarify the operating force requirements for the Pressure Suppression Chamber-Drywell Vacuum Breakers surveillance test.
- Make corrections resulting from License Amendments 130 and 132.

NMC requests approval of the proposed amendment by January 2005. Once approved, the amendment shall be implemented within 30 days.

This letter contains no new commitments and makes no revisions to existing commitments.

The Monticello Operations Committee has reviewed this application. A copy of this submittal, including the Determination of No Significant Hazards Consideration, is being forwarded to our appointed state official pursuant to 10 CFR 50.91(b)(1).

If you have any questions or require additional information, please contact John Fields, Senior Licensing Engineer, at (763) 295-1663.

A001

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

Executed on January 30, 2004

Thomas J. Palmisano
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosures (3)

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

ENCLOSURE A

LICENSEE'S EVALUATION OF PROPOSED CHANGES

1.0 Description

This is a request to amend Operating License DPR-22 for Monticello Nuclear Generating Plant (MNGP).

The proposed changes would revise MNGP Technical Specifications (TS) in the following manner:

1) Clarify the permissive set point for the Source Range Monitor (SRM) Detector-not-fully-inserted rod block bypass.
2) Correct a typographical error in the surveillance requirement for Suppression Pool Temperature Monitoring
3) Clarify the set point for the Pressure Suppression Chamber-Reactor Building Vacuum Breakers instrumentation
4) Clarify the operating force requirements for the Pressure Suppression Chamber-Drywell Vacuum Breakers surveillance test.
5) Make corrections resulting from License Amendments (LA) 130 and 132.

No TS Bases changes are required based upon these proposed TS changes.

2.0 Proposed Change

2.1 Changes to TS Table 3.2.3

The changes to TS Table 3.2.3, "Instrumentation That Initiates Rod Block," are proposed to clarify the permissive set point for the SRM Detector-not-fully-inserted rod block bypass. Currently, TS Table 3.2.3 states four "Allowable Bypass Conditions," one of which has been deleted. Allowable Bypass Condition a. states: *"SRM Detector-not-fully-inserted rod block may be bypassed when the SRM channel rate is 100 cps or when all IRM* [Intermediate Range Monitor] *range switches are above Position 2."*

The proposed change will modify the Allowable Bypass Condition to read: *"SRM Detector-not-fully-inserted rod block may be bypassed when the SRM channel rate is* **≥** *100 cps or when all IRM range switches are above Position 2."* [Emphasis Added] The change for this statement is to add a greater-than-or-equal-to symbol to indicate that the SRM Detector-not-fully-inserted rod block may be bypassed when the SRM counts are greater than or equal to 100 cps or when all IRM range switches are above Position 2.

2.2 Changes to TS 4.7.A.1.b

The changes to TS 4.7.A.1.b, "Suppression Pool Volume and Temperature," are proposed to correct a typographical error in the text. Currently, TS 4.7.A.1.b states: *"Whenever there is indication of relief valve operation which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged ever 5 minutes until the heat addition is terminated."*

The proposed change will modify the surveillance requirement to read: *"Whenever there is indication of relief valve operation which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged* ***every*** *5 minutes until the heat addition is terminated."* [Emphasis Added] The proposed change for this statement is to modify "ever" to "every" to make the sentence read correctly.

2.3 Changes to TS 3.7.A.3

The changes to TS section 3.7.A.3 are proposed to clarify the description of the vacuum breaker instrumentation set point. TS section 3.7.A.3 provides Limiting Condition for Operation (LCO) requirements for the Pressure Suppression Chamber - Reactor Building Vacuum Breakers. This section contains the following statement: *"The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be 0.5 psi."*

The proposed change will modify the LCO requirement to read: *"The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be ≤ 0.5 psi."* [Emphasis Added] The change for this statement is to add a less-than-or-equal-to symbol to indicate that the instrument must actuate prior to surpassing a differential pressure of 0.5 psi.

2.4 Changes to TS 4.7.A.4

The change to TS section 4.7.A.4.a.(4) is proposed to clarify the description of the vacuum breaker opening force. TS section 4.7.A.4 provides Surveillance Requirements (SR) for the Pressure Suppression Chamber - Drywell Vacuum Breakers. This section contains the following statement: *"Once each operating cycle, the vacuum breakers shall be tested to determine that the force required to open each valve from fully closed to fully open does not exceed that equivalent to 0.5 psi acting on the suppression chamber face of the valve disc. (Containment access required.)"*

The proposed change will modify the surveillance requirement to read: *"Once each operating cycle, the vacuum breakers shall be tested to determine that the force required to open each valve from fully closed to fully open does not exceed that equivalent to 0.5* ***psid*** *acting on the suppression chamber face of the valve disc. (Containment access required.)"* [Emphasis Added] The proposed change for this statement is to modify "psi" to "psid" to clarify that the vacuum breaker must actuate with a differential pressure of 0.5 psi acting on the Pressure Suppression Chamber face of the valve disc.

2.5 Corrections from License Amendments 130 and 132

- A change to TS section 3.7.A.2.a (1) is required to revise the LCO wording to incorporate an omission from LA 130. TS section 3.7.A.2 provides LCO requirements for Primary Containment. This section contains the following statement: *"Primary Containment Integrity as defined in Section 1, shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel, except as specified in 3.7.A.2.a (2) or 3.7.A.2.a (3)."*

  The proposed change will modify the LCO requirement to read: *"Primary Containment Integrity as defined in Section 1, shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel, except as specified in 3.7.A.2.a (2), 3.7.A.2.a (3)* ***or 3.7.D.****"* [Emphasis Added] The proposed change will add an exception for 3.7.D.

- A change to TS 3.7.D.1 is required to revise the LCO wording to incorporate an omission from LA 130. TS section 3.7.D.1 provides LCO requirements for Primary Containment Isolation Valves. This section contains the following statement: *"During reactor power operating conditions, all Primary Containment automatic isolation valves and all primary system instrument line flow check valves shall be operable except as specified in 3.7.D.2."*

  The proposed change will modify the LCO requirement to read: *"During reactor power operating conditions, all Primary Containment automatic isolation valves and all primary system instrument line flow check valves shall be operable except as specified in 3.7.D.2* ***and 3.7.D.3.****"* [Emphasis Added] The proposed change will add an exception for 3.7.D.3.

- A change to TS 4.7.D.4 is required to revise the SR wording to remove an obsolete reference resulting from LA 132. TS section 4.7.D.4 provides SRs for Drywell And Suppression Chamber Purge and Vent Valves. This section contains the following statement: *"4, The seat seals of the drywell and suppression chamber 18-inch purge and vent valves shall be replaced at least once every six operating cycles. If periodic Type C leakage testing of the valves* ***performed per surveillance requirement 4.7.A.2.b*** *identifies a common mode test failure attributable to seat seal degradation, then the seat seals of all drywell and suppression chamber 18-inch purge and vent valves shall be replaced."* [Emphasis Added]

The proposed change will modify the SR requirement to read: *"4. The seat seals of the drywell and suppression chamber 18-inch purge and vent valves shall be replaced at least once every six operating cycles. If periodic Type C leakage testing of the valves identifies a common mode test failure attributable to seat seal degradation, then the seat seals of all drywell and suppression chamber 18-inch purge and vent valves shall be replaced."* The proposed change will modify the format to match the standard formatting of the MNGP TS and remove a surveillance requirement reference.

3.0 Background

3.1 TS Table 3.2.3

As described in the MNGP Updated Safety Analysis Report (USAR) (Reference 1), the reactor manual control system consists of the electrical circuitry, switches, indicators, and alarm devices provided for operational manipulation of the control rods and the surveillance of associated equipment. This system includes the interlocks that inhibit rod movement (rod block) under certain conditions. The reactor manual control system does not include any of the circuitry or devices used to automatically or manually scram the reactor. The objective of the reactor manual control system is to provide the operator with the means to make changes in core reactivity so that reactor power level and power distribution can be controlled. Protection is afforded to prevent inadvertent withdrawal, insertion and selection of the controls rods. This protection prevents control rod movement (rod block).

Each of the four SRM channels initiates the following rod block with the mode switch in STARTUP or REFUEL. A rod block is initiated when any SRM detector not fully inserted into the reactor core with the SRM count level below 100 cps (the retract permit level) and any Intermediate Range Monitor (IRM) range switch on either of the two lowest ranges. This assures that no control rod is withdrawn unless all SRM detectors are properly inserted when they must be relied upon to provide the operator with neutron flux level information. Mechanical switches in the SRM detector drive system provide the position signals used to indicate that a detector is not fully inserted.

An automatic bypass of the SRM detector position rod block is enabled as the neutron flux increases beyond a preset low level on the SRM instrumentation. The bypass allows the detector to be partially or completely withdrawn as a reactor startup is continued.

3.2 TS 4.7.A.1.b

No background discussion for this TS change is necessary since the proposed change is to correct a simple typrographical error.

3.3 TS 3.7.A.3

As described in the MNGP USAR (Reference 1), the Vent and Vacuum Relief System is designed to limit the negative pressure in either the Pressure Suppression Chamber or the Drywell to less than the design pressure of -2 psid. Automatic vacuum relief devices are employed to prevent the Primary Containment from exceeding the external design pressure. The Primary Containment is designed for external pressure not more than 2 psi greater than the concurrent internal pressure. The Primary Containment is periodically vented to eliminate pressure fluctuations caused by temperature changes during various operating modes. This is accomplished through ventilation purge connections, which are normally closed while the reactor is at a temperature greater than 212°F. The Pressure Suppression Chamber is vented separately. The Drywell Vacuum Relief Valves draw the atmosphere from the Pressure Suppression Chamber and the Pressure Suppression Chamber vacuum relief device draws air from the Reactor Building in the event vacuum conditions develop. (Reference 1)

The Pressure Suppression Chamber Vacuum Relief System consists of two vacuum breaker valves in series in each of two lines, which are joined into one larger line attaching to the suppression chamber.

Two vacuum breakers in series are used in each of two large vent lines, which permit air to flow from the Reactor Building to the Pressure Suppression Chamber. Vendor-supplied, flow-versus-pressure drop information was used to ensure that sufficient flow area is available to accommodate maximum obtainable vacuum relief flow conditions. Each of the Reactor Building to Pressure Suppression Chamber lines contains two valves in series, each rated at 0.5 psi differential pressure (1.0 psid total). Each of these two parallel lines was sized for 100% requirements in order to provide fully redundant capacity.

One of each pair of vacuum breakers is an air-operated butterfly valve which is AC solenoid-controlled from a differential pressure switch signal and is designed to fail open on loss of power and loss of air. A safety grade nitrogen supply system is available to close these vacuum breakers if instrument air pressure is lost. The second vacuum breaker is a self-activating swing check. The combined pressure drop at rated flow through both valves does not exceed 2 psi, the suppression chamber design external pressure.

3.4 TS 4.7.A.4

As described in the MNGP USAR (Reference 1), the Pressure Suppression Chamber-to-Drywell Vacuum Breaker Valves permit gases to flow from the Pressure Suppression Chamber to the Drywell. Eight 18-in. valves are used in parallel. These valves are sized on the results of the Bodega Bay pressure suppression system tests. Their chief purpose is to prevent excessive water level variation in the downcomers submerged in suppression pool water. The Bodega Bay tests regarding vacuum breaker sizing were conducted by simulating a small system rupture, which tended to cause downcomer water level variation, as a preliminary step in the large rupture test sequence. The vacuum breaker capacity selected on this test basis is more than adequate to limit the pressure differential between the Pressure Suppression Chamber and Drywell during post-accident drywell cooling.

The MNGP Mark I Containment Program developed methods, which could be used for determining the plant-unique vacuum breaker cyclic response, as well as the necessary size, to meet various design conditions for which the vacuum breaker must function. This analysis demonstrated that the capacity of six Drywell Vacuum Relief Valves would be sufficient to limit the pressure differential between the Pressure Suppression Chamber and Drywell to less than the design limit of 2 psi even if both drywell spray loops actuated simultaneously following a LOCA. If only one spray loop is actuated, three vacuum breakers are sufficient.

3.5 Corrections from License Amendments 130 and 132

By letter dated December 21, 2001 (Reference 2), and supplemented by letter dated April 26, 2002 (Reference 3), NMC requested a LA for the MNGP TS. The purpose of this LA was to revise the Containment Systems Section (3.7/4.7) of the TS to clarify existing requirements, make wording improvements, revise existing LCOs and SRs, and add an additional TS LCO to the Monticello TS. This submittal and supplement was reviewed and approved by the NRC and issued as Amendment Number 130 (Reference 4) to the MNGP TS.

After review of the approved amendment it has been identified that even with this approved change the LCO duration as specified in MNGP TS 3.7.A.2.a (1) would prohibit the use of the revised TS 3.7.D, "Primary Containment Isolation Valves," because TS 3.7.A.2.a (4) does not allow a penetration to be inoperable for more than 1 hour. Also, an omission was made in TS LCO 3.7.D.1 by failing to add an exception to its requirements to include TS 3.7.D.3 (inerting and deinerting operations). This addition is necessary to permit the use of the containment isolation provisions approved via LA 130.

By letter dated April 25, 2001 (Reference 5, subsequently corrected to April 25, 2002 by letter dated May 30, 2002 (Reference 6)), NMC requested a LA for the MNGP TS. The purpose of this LA was to revise the Containment Systems Section (3.7/4.7) of the TS to convert to 10 CFR 50, Appendix J, Option B for Containment Leak Rate Testing. This submittal was reviewed and approved by the NRC and issued as Amendment Number 132 (Reference 7) to the MNGP TS.

After review of the approved amendment it has been identified that even with this approved change, SR 4.7.D.4 also required revision because it contained a typographical error (4, vs 4.) and directed the operator to SR 4.7.A.2.b, which had been eliminated by the amendment.

4.0 Technical Analysis

4.1 Revision of TS Table 3.2.3

The changes to TS Table 3.2.3, "Instrumentation That Initiates Rod Block," are proposed to clarify the permissive set point for the SRM Detector-not-fully-inserted rod block bypass. The proposed change will modify the Allowable Bypass Condition to read: *"SRM Detector-not-fully-inserted rod block may be bypassed when the SRM channel rate is ≥ 100 cps or when all IRM range switches are above Position 2."* [Emphasis Added] The change for this statement is to add a greater-than-or-equal-to symbol to indicate that the SRM Detector-not-fully-inserted rod block may be bypassed when the SRM counts are greater than or equal to 100 cps or when all IRM range switches are above Position 2.

As described in Section 3.1, the SRM Detector-not-fully-inserted rod block is designed to assure that no control rod is withdrawn unless all SRM detectors are properly inserted when they must be relied upon to provide the operator with neutron flux level information. Therefore, a rod block is initiated when any SRM detector not fully inserted into the reactor core with the SRM count level below 100 cps (the retract permit level) and any IRM range switch on either of the two lowest ranges. Thus it follows, that an automatic bypass of the SRM detector position rod block is enabled as the neutron flux increases beyond a preset low level (≥ 100 cps) on the SRM instrumentation. The bypass allows the detector to be partially or completely withdrawn as a reactor startup is continued.

The proposed TS changes are consistent with the design of the rod block and the bypass of the rod block. Therefore, the changes are acceptable.

4.2 Revision of TS 4.7.A.1.b

The changes to TS 4.7.A.1.b, "Suppression Pool Volume and Temperature," are proposed to correct a typographical error in the text. The proposed change will modify the surveillance requirement to read: *"Whenever there is indication of relief valve operation which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged* ***every*** *5 minutes until the heat addition is terminated."* [Emphasis Added] The proposed change for this statement is to modify "ever" to "every" to make the sentence read correctly.

NMC considers this to be a typographical error, which is administrative in nature and therefore, considers the change acceptable.

4.3 Revision of TS 3.7.A.3

This proposed change to the TS would clarify the description of the vacuum breaker instrumentation set point. TS section 3.7.A.3 provides LCO requirements for the Pressure Suppression Chamber - Reactor Building Vacuum Breakers. The proposed change will modify the LCO requirement to read: *"The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be* $\leq$ *0.5 psi."* [Emphasis Added] The change for this statement is to add a less-than-or-equal-to symbol to indicate that the instrument must actuate prior to surpassing a differential pressure of 0.5 psi.

As identified in section 3.3 above, the Pressure Suppression Chamber-Reactor Building Vacuum Breakers are designed in concert with the Pressure Suppression Chamber-Drywell Vacuum Breakers to limit the negative pressure in either the suppression chamber or the drywell to less than the design pressure of -2 psid. Operation of these systems will maintain the pressure differential less than 1 psi.

NMC has performed an analysis which demonstrated that actuation of the instrumentation associated with the Pressure Suppression Chamber-Reactor Building Vacuum Breakers at 0.5 psid will ensure that the Pressure Suppression Chamber and drywell will not exceed their design pressure. To account for instrument inaccuracies, repeatability and other instrumentation uncertainties, the set point for the actuation instrumentation must be set less than or equal to 0.5 psid. Therefore, NMC considers this acceptable to assure plant safety, by ensuring that each instrument will actuate prior or equivalent to a differential pressure 0.5 psi.

4.4 Revision of TS 4.7.A.4

The change to TS section 4.7.A.4.a.(4) is proposed to clarify the description of the vacuum breaker opening force. TS section 4.7.A.4 provides Surveillance Requirements (SR) for the Pressure Suppression Chamber - Drywell Vacuum Breakers. The proposed change will modify the surveillance requirement to read: *"Once each operating cycle, the vacuum breakers shall be tested to determine that the force required to open each valve from fully closed to fully open does not exceed that equivalent to 0.5* ***psid*** *acting on the suppression chamber face of the valve disc. (Containment access required.)"* [Emphasis Added] The proposed change for this statement is to modify "psi" to "psid" to clarify that the vacuum breaker must actuate with a differential pressure of 0.5 psi acting on the Pressure Suppression Chamber face of valve disc.

As described above in sections 3.3 and 3.4 the Pressure Suppression Chamber to Drywell Vacuum Breakers are designed to ensure that the Primary Containment external pressure is not more than 2 psi greater than the concurrent internal pressure. Analysis has demonstrated that the capacity of the vacuum breakers will ensure that this design requirement is met. The purpose of TS surveillance requirement 4.7.A.4.a.(4) is to demonstrate that the vacuum breakers are functioning adequately to achieve the design requirement.

The current TS surveillance requirement 4.7.A.4.a.(4) could be considered unclear as to the operating force requirements for the vacuum breaker testing. Specifying the requirements for operating force on one side of the valve disc does not ensure that a differential pressure exists across the valve. To demonstrate that the vacuum breaker will function as designed, the proposed TS change clarifies the surveillance requirement to indicate that the vacuum breaker will open with an operating force not to exceed 0.5 psid across the Pressure Suppression Chamber face of the valve. Therefore, NMC finds this acceptable.

4.5 Miscellaneous changes from LA 130 and 132

The proposed changes described in section 2.5 above will enhance the Monticello TS in the following ways:

Adding an exception to TS 3.7.A.2.a (1) LCO, to allow the requirements of TS 3.7.D to be included as an exception, will provide the operators the ability to use the LCO requirements of TS 3.7.D to allow a 4-hour time interval to restore an inoperable valve to operable status or isolate a valve in a penetration with two Primary Containment Isolation Valves (PCIVs) and one PCIV inoperable. Revising TS 3.7.A.2.a (1) to add an additional exception to allow the use of TS 3.7.D is acceptable because it corrects a previously inadvertently omitted exception to allow the use of an approved TS change. Providing the ability for operations personnel to enter an available LCO condition does not impact the safe operation of the facility, but enhances the overall ability of the operators to enter the specifically required LCO condition when appropriate. This change is needed to eliminate confusion and clarify the conditions for the use of TS 3.7.D.

Adding an exception to TS 3.7.D.1 will allow operators the ability to use the requirements of TS 3.7.D.3 for purge and vent valve operation and isolation. Revising TS 3.7.D.1 to add an exception to allow the use of TS 3.7.D.3 is acceptable because it corrects a previously inadvertently omitted exception to allow the use of an approved TS change. Providing the ability for operations personnel to enter an available LCO condition does not impact the safe operation of the facility, but enhances the overall ability of the operators to enter the specifically required LCO condition when appropriate. This change is needed to eliminate confusion and clarify the conditions for the use of TS 3.7.D.3.

Removing a reference to TS 4.7.A.2.b will eliminate potential confusion when operators use TS 4.7.D.4. Revising TS 4.7.D.4 to remove reference to TS 4.7.A.2.b, which does not exist, is acceptable because it corrects an inadvertent oversight, which occurred during the relocation of the TS under LA 132. This change is needed to eliminate confusion and clarify the conditions for the use of TS 4.7.D.4. This change also corrects the format of the TS, which is considered a typographical error.

5.0 Regulatory Safety Analysis

5.1 No Significant Hazards Consideration

The proposed changes would revise Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS) in the following manner:

1) Clarify the permissive set point for the Source Range Monitor Detector-not-fully-inserted rod block bypass.
2) Correct a typographical error in the surveillance requirement for Suppression Pool Temperature Monitoring
3) Clarify the set point for the Pressure Suppression Chamber-Reactor Building Vacuum Breakers instrumentation
4) Clarify the operating force requirements for the Pressure Suppression Chamber-Drywell Vacuum Breakers surveillance test.
5) Make corrections resulting from License Amendments (LA).

Nuclear Management Company, LLC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The SRM Detector-not-fully-inserted rod block bypass set point, the Pressure Suppression Chamber - Reactor Building Vacuum Breakers actuation instrumentation set point requirement and the Pressure Suppression Chamber - Drywell Vacuum Breakers surveillance test requirements are being clarified in the MNGP TS to ensure these functions will adequately support safe operation of the facility. Typographical errors are being corrected along with corrections resulting from omissions and an oversight from previous LAs. The proposed TS changes do not introduce new equipment or new equipment operating modes, nor do the proposed changes alter existing system relationships. The changes do not affect plant operation, design function or any analysis that verifies the capability of a SSC to perform a design function. Further, the proposed changes do not increase the likelihood of the malfunction of any structure, system or component (SSC) or impact any analyzed accident. Consequently, the probability of an accident previously evaluated is not affected.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The SRM Detector-not-fully-inserted rod block bypass set point, the Pressure Suppression Chamber - Reactor Building Vacuum Breakers actuation instrumentation set point requirement and the Pressure Suppression Chamber - Drywell Vacuum Breakers surveillance test requirements are being clarified in the MNGP TS to ensure these functions will adequately support safe operation of the facility. Typographical errors are being corrected along with corrections resulting from omissions and an oversight from previous LAs. The changes do not create the possibility of new credible failure mechanisms, or malfunctions. These changes do not modify the design function or operation of any SSC. Further the changes do not involve physical alterations of the plant; no new or different type of equipment will be installed. The proposed changes do not introduce new accident initiators. Consequently, the changes cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

3. Does the proposed amendment involve a significant reduction in the margin of safety?

The SRM Detector-not-fully-inserted rod block bypass set point, the Pressure Suppression Chamber - Reactor Building Vacuum Breakers actuation instrumentation set point requirement and the Pressure Suppression Chamber - Drywell Vacuum Breakers surveillance test requirements are being clarified in the MNGP TS to ensure these functions will adequately support safe operation of the facility. Typographical errors are being corrected along with corrections resulting from omissions and an oversight from previous LAs. These changes do not exceed or alter a design basis or a safety limit for a parameter established in the MNGP Updated Safety Analysis Report (USAR) or the MNGP facility license. Consequently, the changes do not result in a significant reduction in the margin of safety.

Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

Based on the above, NMC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10CFR50.36 requires that technical specification limiting conditions for operation of a nuclear reactor must be established for installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. In addition, 10CFR50.36 requires that surveillances relating to test, calibration, or inspection be completed to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The proposed changes would revise Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS) to clarify the set point for the SRM Detector-not-fully-inserted rod block bypass, the set point for the Pressure Suppression Chamber-Reactor Building Vacuum Breakers instrumentation and the operating force requirements for the Pressure Suppression Chamber-Drywell Vacuum Breakers surveillance, correct typographical errors and make corrections resulting from omissions and an oversight from previous license amendments. Each of the changes as evaluated above has been demonstrated to be in accordance with NRC regulations.

In conclusion, based on the considerations discussed above, (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.

6.0 Environmental Consideration

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve:

(i) <u>a significant hazards consideration,</u>
The proposed amendment does not involve a significant hazard as evaluated previously in section 5.1.

(ii) <u>a significant change in the type or significant increase in the amounts of any effluent that may be released offsite, or</u>
The proposed amendment is consistent with and does not change the design basis of the plant. The proposed amendment will not result in an increase in power level, will not increase the production of radioactive waste and byproducts, and will not alter the flowpath or method of disposal of radioactive waste or byproducts. Therefore, the proposed amendment does not involve any change in the type or amount of any effluent that may be released offsite.

(iii) <u>a significant increase in individual or cumulative occupational radiation exposure.</u>
The proposed amendment does not result in changes in the level of control or methodology used for processing radioactive effluents or handling of solid radioactive waste. There will be no change to the normal radiation levels within

the plant. Therefore, the amendment does not involve an increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 References

1. Monticello Nuclear Generating Plant, Updated Safety Analysis Report, Revision 20

2. Letter from NMC to NRC, "License Amendment Request for Containment Systems Technical Specification Changes," dated December 21, 2001.

3. Letter from NMC to NRC, "Response to NRC Request For Additional Information and Supplemental License Amendment Request for Previously Submitted Containment Systems License Amendment Request (TAC No. MB3706)," dated April 26, 2002.

4. Letter from NRC to NMC, "Monticello Nuclear Generating Plant – Issuance of Amendment Relating to Containment Systems Technical Specification Revisions (TAC No. MB3706)," dated September 23, 2002.

5. Letter from NMC to NRC, "License Amendment for conversion to Option B for Containment Leak Rate Testing," dated April 25, 2001.

6. Letter from NMC to NRC, "Correction of Typographical Error in Submittal Date for Previously Submitted License Amendment Request (TAC # MB4975)," dated May 30, 2002.

7. Letter from NRC to NMC, "Monticello Nuclear Generating Plant – Issuance of Amendment RE: License Amendment Request for Conversion to Option B for Containment Leak Rate Testing (TAC No. MB4975)," dated February 4, 2003.

ENCLOSURE B

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

This enclosure consists of current Technical Specification pages marked up with the proposed changes. The pages included in this exhibit are as listed below:

<u>Pages</u>
58a
156
158
163
164
170
171a

Table 3.2.3 Continued
Instrumentation That Initiates Rod Block

Notes:

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

A. Reactor in Shutdown mode.

B. No rod withdrawals permitted while in Refuel or Startup mode.

C. Reactor in Run mode.

D. No rod withdrawals permitted while in the Run mode.

E. Power on IRM range or below and reactor in Startup, Refuel, or Shutdown mode.

**Allowable Bypass Conditions

a. SRM Detector-not-fully-inserted rod block may be bypassed when the SRM channel count rate is ≥ 100 cps or when all IRM range switches are above Position 2.

b. IRM Downscale rod block may be bypassed when the IRM range switch is in the lowest range position.

c. (deleted)

d. SRM Upscale block may be bypassed when associated IRM range switches are above Postion 6.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment.

1. *Suppression Pool Volume and Temperature*

When irradiated fuel is in the reactor vessel and either the reactor water temperature is greater than 212°F or work is being done which has the potential to drain the vessel, the following requirements shall be met, except as permitted by Specification 3.5.E.2:

a. Water temperature during normal operating shall be ≤90°F.

b. Water temperature during test operation which adds heat to the suppression pool shall be ≤100°F and shall not be >90°F for more than 24 hours.

c. If the suppression chamber water temperature is >110°F, the reactor shall be scrammed immediately. Power operation shall not be resumed until the pool temperature is ≤90°F.

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. Suppression Pool Volume and Temperature

a. The suppression chamber water temperature shall be checked once per day.

b. Whenever there is indication of relief valve operation which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. A visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. Primary Containment Integrity	2. Primary Containment Integrity
a. (1) Primary Containment Integrity as defined in Section 1, shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel, except as specified in 3.7.A.2.a.(2), ~~or~~ 3.7.A.2.a.(3), or 3.7.D	a. Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.
(2) Primary Containment Integrity is not required when performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).	
(3) Primary Containment Integrity is not required when performing reactor vessel hydrostatic or leakage tests with the reactor not critical.	
(4) If requirements of 3.7.A.2.a.(1) cannot be met, restore Primary Containment Integrity within one hour or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.	

3.0 LIMITING CONDITIONS FOR OPERATION

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

   a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when the primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be ≤ 0.5 psi.

   b. From and after the ~~date that~~ one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

   c. If requirements of 3.7.A.3 cannot be met, the reactor shall be placed in a Cold Shutdown condition within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

   a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentation including set point shall be checked for proper operation every three months.

3.0 LIMITING CONDITIONS FOR OPERATION

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

   a. When primary containment integrity is required, all eight drywell-suppression chamber vacuum breakers shall be operable and positioned in the closed position as indicated by the position indication system, except during testing and except as specified in 3.7.A.4.b through 3.7.A.4.d below.

   b. Any drywell-suppression chamber vacuum breaker may be nonfully closed as indicated by the position indication and alarm system provided that drywell to suppression chamber differential pressure decay does not exceed that shown on Figure 3.7.1

   c. Up to two drywell-suppression chamber vacuum breakers may be inoperable provided that: (1) the vacuum breakers are determined to be fully closed and at least one position alarm circuit is operable or (2) the vacuum breaker is secured in the closed position or replaced by a blank flange.

   d. Drywell-suppression chamber vacuum breakers may be cycled, one at a time, during containment inerting and deinerting operations to assist in purging air or nitrogen from the suppression chamber vent header.

4.0 SURVEILLANCE REQUIREMENTS

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

   a. Operability and full closure of the drywell-suppression chamber vacuum breakers shall be verified by performance of the following:

      (1) Monthly each operable drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle.

      (2) Once each operating cycle, drywell to suppression chamber leakage shall be demonstrated to be less than that equivalent to a one-inch diameter orifice and each vacuum breaker shall be visually inspected. (Containment access required)

      (3) Once each operating cycle, vacuum breaker position indication and alarm systems shall be calibrated and functionally tested. (Containment access required)

      (4) Once each operating cycle, the vacuum breakers shall be tested to determine that the force required to open each valve from fully closed to fully open does not exceed that equivalent to 0.5 psi acting on the suppression chamber ~~face~~ of the valve disc. (Containment access required.)

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

reactor core, operations with a potential for reducing the shutdown margin below that specified in specification 3.3.A, and handling of irradiated fuel or the fuel cask in the secondary containment are to be immediately suspended if secondary containment integrity is not maintained.

D. Primary Containment Isolation Valves (PCIVs)

1. During reactor power operating conditions, all Primary Containment automatic isolation valves and all primary system instrument line flow check valves shall be operable except as specified in 3.7.D.2 and 3.7.D.3.

D. Primary Containment Isolation Valves (PCIVs)

1. The primary containment automatic isolation valve surveillance shall be performed as follows:

   a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

   b. At least once per operating cycle the primary system instrument line flow check valves shall be tested for proper operation.

   c. All normally open power-operated isolation valves shall be tested in accordance with the Inservice Testing Program. Main Steam isolation valves shall be tested (one at a time) with the reactor power less than 75% of rated.

   d. At least once per week the main steam-line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

3.0 LIMITING CONDITIONS FOR OPERATION

3. a. The inerting and deinerting operations permitted by TS 3.7.A.5.b shall be via the 18-inch purge and vent valves (equipped with 40-degree limit stops) aligned to the Reactor Building plenum and vent. All other purging and venting, when primary containment integrity is required, shall be via the 2-inch purge and vent valve bypass line and the Standby Gas Treatment System.

   b. In the event one or more penetration flow paths with one or more containment purge and vent valves not within purge and vent valve leakage limits, reactor operation in the run mode may continue provided that within the subsequent 24 hours, restore the valve(s) to within leakage limits, or at least one valve in each line having a purge and vent valve not within leakage limits is deactivated in the isolated position. This requirement may be satisfied by use of one closed and deactivated automatic valve, closed manual valve, or blind flange. (Deactivated means electrically or pneumatically disarm or otherwise secure the valve.)

4. If Specification 3.7.D.1, 3.7.D.2 and 3.7.D.3 cannot be met, initiate normal orderly shutdown and have reactor in the Cold Shutdown condition within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

3. Whenever containment purge and vent valves are isolated to meet the requirements of TS 3.7.D.3.b, the position of the deactivated and isolated valves outside primary containment shall be recorded monthly.**

4~~x~~. The seat seals of the drywell and suppression chamber 18-inch purge and vent valves shall be replaced at least once every six operating cycles. If periodic Type C leakage testing of the valves ~~performed per surveillance requirement 4.7.A.2.b~~ identifies a common mode test failure attributable to seat seal degradation, then the seat seals of all drywell and suppression chamber 18-inch purge and vent valves shall be replaced.

** Isolated valves in high radiation areas may be verified by use of administration means.

ENCLOSURE C

PROPOSED TECHNICAL SPECIFICATION CHANGES (RETYPED)

This enclosure consists of revised Technical Specification pages that incorporate the proposed changes. The pages included in this exhibit are as listed below:

<u>Pages</u>
58a
156
158
163
164
170
171a

Table 3.2.3 Continued
Instrumentation That Initiates Rod Block

Notes:

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

A. Reactor in Shutdown mode.

B. No rod withdrawals permitted while in Refuel or Startup mode.

C. Reactor in Run mode.

D. No rod withdrawals permitted while in the Run mode.

E. Power on IRM range or below and reactor in Startup, Refuel, or Shutdown mode.

**Allowable Bypass Conditions

a. SRM Detector-not-fully-inserted rod block may be bypassed when the SRM channel count rate is ≥100 cps or when all IRM range switches are above Position 2.

b. IRM Downscale rod block may be bypassed when the IRM range switch is in the lowest range position.

c. (deleted)

d. SRM Upscale block may be bypassed when associated IRM range switches are above Postion 6.

3.0 LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment.

1. Suppression Pool Volume and Temperature

When irradiated fuel is in the reactor vessel and either the reactor water temperature is greater than 212°F or work is being done which has the potential to drain the vessel, the following requirements shall be met, except as permitted by Specification 3.5.E.2:

a. Water temperature during normal operating shall be ≤90°F.

b. Water temperature during test operation which adds heat to the suppression pool shall be ≤100°F and shall not be >90°F for more than 24 hours.

c. If the suppression chamber water temperature is >110°F, the reactor shall be scrammed immediately. Power operation shall not be resumed until the pool temperature is ≤90°F.

4.0 SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. Suppression Pool Volume and Temperature

a. The suppression chamber water temperature shall be checked once per day.

b. Whenever there is indication of relief valve operation which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. A visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. Primary Containment Integrity	2. Primary Containment Integrity
a. (1) Primary Containment Integrity as defined in Section 1, shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel, except as specified in 3.7.A.2.a.(2), 3.7.A.2.a.(3) or 3.7.D.	a. Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.
(2) Primary Containment Integrity is not required when performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).	
(3) Primary Containment Integrity is not required when performing reactor vessel hydrostatic or leakage tests with the reactor not critical.	
(4) If requirements of 3.7.A.2.a.(1) cannot be met, restore Primary Containment Integrity within one hour or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.	

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

3. *Pressure Suppression Chamber - Reactor Building Vacuum Breakers*

   a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when the primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be ≤ 0.5 psi.

   b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

   c. If requirements of 3.7.A.3 cannot be met, the reactor shall be placed in a Cold Shutdown condition within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

3. *Pressure Suppression Chamber - Reactor Building Vacuum Breakers*

   a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentation including set point shall be checked for proper operation every three months.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

   a. When primary containment integrity is required, all eight drywell-suppression chamber vacuum breakers shall be operable and positioned in the closed position as indicated by the position indication system, except during testing and except as specified in 3.7.A.4.b through 3.7.A.4.d below.

   b. Any drywell-suppression chamber vacuum breaker may be nonfully closed as indicated by the position indication and alarm system provided that drywell to suppression chamber differential pressure decay does not exceed that shown on Figure 3.7.1

   c. Up to two drywell-suppression chamber vacuum breakers may be inoperable provided that: (1) the vacuum breakers are determined to be fully closed and at least one position alarm circuit is operable or (2) the vacuum breaker is secured in the closed position or replaced by a blank flange.

   d. Drywell-suppression chamber vacuum breakers may be cycled, one at a time, during containment inerting and deinerting operations to assist in purging air or nitrogen from the suppression chamber vent header.

4.0 SURVEILLANCE REQUIREMENTS

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

   a. Operability and full closure of the drywell-suppression chamber vacuum breakers shall be verified by performance of the following:

      (1) Monthly each operable drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle.

      (2) Once each operating cycle, drywell to suppression chamber leakage shall be demonstrated to be less than that equivalent to a one-inch diameter orifice and each vacuum breaker shall be visually inspected. (Containment access required)

      (3) Once each operating cycle, vacuum breaker position indication and alarm systems shall be calibrated and functionally tested. (Containment access required)

      (4) Once each operating cycle, the vacuum breakers shall be tested to determine that the force required to open each valve from fully closed to fully open does not exceed that equivalent to 0.5 psid acting on the suppression chamber face of the valve disc. (Containment access required.)

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

reactor core, operations with a potential for reducing the shutdown margin below that specified in specification 3.3.A, and handling of irradiated fuel or the fuel cask in the secondary containment are to be immediately suspended if secondary containment integrity is not maintained.

D. Primary Containment Isolation Valves (PCIVs)

1. During reactor power operating conditions, all Primary Containment automatic isolation valves and all primary system instrument line flow check valves shall be operable except as specified in 3.7.D.2 and 3.7.D.3.

D. Primary Containment Isolation Valves (PCIVs)

1. The primary containment automatic isolation valve surveillance shall be performed as follows:

   a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

   b. At least once per operating cycle the primary system instrument line flow check valves shall be tested for proper operation.

   c. All normally open power-operated isolation valves shall be tested in accordance with the Inservice Testing Program. Main Steam isolation valves shall be tested (one at a time) with the reactor power less than 75% of rated.

   d. At least once per week the main steam-line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

3. a. The inerting and deinerting operations permitted by TS 3.7.A.5.b shall be via the 18-inch purge and vent valves (equipped with 40-degree limit stops) aligned to the Reactor Building plenum and vent. All other purging and venting, when primary containment integrity is required, shall be via the 2-inch purge and vent valve bypass line and the Standby Gas Treatment System.

   b. In the event one or more penetration flow paths with one or more containment purge and vent valves not within purge and vent valve leakage limits, reactor operation in the run mode may continue provided that within the subsequent 24 hours, restore the valve(s) to within leakage limits, or at least one valve in each line having a purge and vent valve not within leakage limits is deactivated in the isolated position. This requirement may be satisfied by use of one closed and deactivated automatic valve, closed manual valve, or blind flange. (Deactivated means electrically or pneumatically disarm or otherwise secure the valve.)

4. If Specification 3.7.D.1, 3.7.D.2 and 3.7.D.3 cannot be met, initiate normal orderly shutdown and have reactor in the Cold Shutdown condition within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

3. Whenever containment purge and vent valves are isolated to meet the requirements of TS 3.7.D.3.b, the position of the deactivated and isolated valves outside primary containment shall be recorded monthly.**

4. The seat seals of the drywell and suppression chamber 18-inch purge and vent valves shall be replaced at least once every six operating cycles. If periodic Type C leakage testing of the valves identifies a common mode test failure attributable to seat seal degradation, then the seat seals of all drywell and suppression chamber 18-inch purge and vent valves shall be replaced.

** Isolated valves in high radiation areas may be verified by use of administration means.