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Monticello Nuclear Generating Plant
2807 W County Road 75
Monticello, MN 55362

February 7, 2011

L-MT-11-015
10 CFR 50.90

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

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

License Amendment Request: Revise Core Spray Pump Flow Rate in Specification 3.5.1, ECCS – Operating

Pursuant to 10 CFR 50.90, the Northern States Power Company – Minnesota (NSPM), proposes to revise the flow rate for the Core Spray pumps in Surveillance Requirement 3.5.1.7 within Technical Specification (TS) 3.5.1, "ECCS – Operating."

Enclosure 1 provides a description of the proposed changes and includes the technical evaluation and associated no significant hazards determination and environmental evaluation. Enclosure 2 provides a marked-up copy of the TS pages showing the proposed changes.

The Monticello Nuclear Generating Plant (MNGP) Plant Operations Review Committee has reviewed this application. In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Minnesota Official.

Summary of Commitments

This letter proposes no new commitments and does not revise any existing commitments.

NSPM requests approval of the proposed license amendment by one year from the date of submittal, with an implementation period of 120 days.

Should you have questions regarding this letter, please contact Mr. Richard Loeffler at (763) 295-1247.

A002
A001
NRR

I declare under penalty of perjury that the foregoing is true and correct.
Executed on February 7, 2011.

Timothy J. O'Connor
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures (2)

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

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## DESCRIPTION OF CHANGES

## LICENSE AMENDMENT REQUEST
## REVISE CORE SPRAY PUMP FLOW RATE IN SPECIFICATION 3.5.1, ECCS – OPERATING

## 1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, the Northern States Power Company – Minnesota (NSPM), proposes to revise the Core Spray pumps flow rate in Surveillance Requirement 3.5.1.7 within Monticello Nuclear Generating Plant (MNGP) Technical Specification 3.5.1, "ECCS – Operating," from 2800 to 2835 gallons per minute (gpm).

## 2.0 DETAILED DESCRIPTION

The margin between the 2800 gpm flow requirement of TS Surveillance Requirement (SR) 3.5.1.7 and the 2700 gpm Emergency Core Cooling System (ECCS) analysis value accounts for potential bypass leakage, i.e., flow that is diverted by several core spray piping leakage paths(1) and not delivered into the core shroud for core cooling.(2) The "B" Core Spray subsystem has had a modification in the past to address indications which could result in leakage into the reactor downcomer region. A pre-emptive modification was also made on the "A" Core Spray subsystem even though no indications were identified. Different Core Spray subsystem flow rates are not assumed for each subsystem – the same value is assumed for both Core Spray subsystems in the ECCS analyses.

A review of Inservice Test calculational results versus surveillance test criteria indicated the methodology correctly compared the measured flow against American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code alert criteria and the SR 3.5.1.7 requirement of 2800 gpm, but did not recognize the implicit 100 gpm bypass flow assumption within the ECCS analyses. Consequently, when the bypass flow for the "B" Core Spray subsystem exceeded the 100 gpm margin assumed between the ECCS analyses and the TS surveillance requirement this condition was not identified.

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1. Also, fifty (50) gpm of low pressure coolant injection jet pump slip joint leakage has been reallocated to the Core Spray System as approved by the NRC in Amendment 93 and discussed in the Monticello Updated Safety Analysis Report.
2. The design of the system resulting in the potential for flow diversion and the measures taken in the design of the Core Spray System and assumptions in the safety analyses to account for potential flow diversion are discussed in Sections 3.0 and 5.0, herein.

The margin between the Core Spray flow rate measured during surveillance testing and the ECCS analysis requirement (2700 gpm to the core) is sufficient to accommodate this increased postulated flow diversion. Both Core Spray subsystems continue to perform their required safety functions in the ECCS analyses. However, the value specified within the Technical Specifications (TS) for the required Core Spray flow rate of 2800 gpm is not conservative and requires revision.

Consequently, this condition is being treated as a non-conservative TS in accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety" (Reference 1). Accordingly, NSPM proposes to increase the required Core Spray System flow rate from 2800 to 2835 gpm in TS Surveillance Requirement 3.5.1.7 to address this potential increased flow diversion and also to provide margin in the future.

## 3.0 DESCRIPTION OF THE CORE SPRAY SYSTEM

The ECCS uses two independent methods (flooding and spraying) to cool the core during a Loss-of-Coolant Accident (LOCA). The ECCS network consists of the High Pressure Coolant Injection System, the Core Spray System, the low pressure coolant injection (LPCI) mode of the Residual Heat Removal System, and the Automatic Depressurization System. The suppression pool provides the required source of water for the ECCS and although not credited the condensate storage tanks provide an additional source of water.

The Core Spray System is composed of two independent subsystems. Each subsystem consists of a 100 percent-capacity centrifugal pump driven by an electric motor, a spray sparger in the core shroud above the core, piping within the reactor vessel annulus region between the core spray reactor vessels nozzles and the core shroud, and the piping and valves to convey water from the suppression pool to the vessel, and associated controls and instrumentation. The two 100-percent capacity core spray lines separately enter the reactor vessel through two core spray nozzles 180 degrees apart. Each internal line (pipe) then divides into a semicircular header with a downcomer at each end, which enters through the core shroud above the core. A semicircular sparger is attached to each of the four outlets to make two practically complete circles, within the core shroud, one above the other. Short elbow nozzles are spaced around the spargers to spray the water radially onto the tops of the fuel assemblies.

A leak in the piping between the reactor vessel wall and core shroud could result in a diversion of flow into the downcomer (annulus) region and out the break, resulting in this portion of the injection flow not being available for core cooling. The design of the Core Spray System and the ECCS analyses account for this possibility by two means:

- A core spray differential pressure break detection system is provided to confirm the integrity of the core spray piping between the inside of the reactor vessel and the core shroud.

- Some core spray flow is assumed to be diverted and not reach the core. This value is added to the 2700 gpm value in the ECCS analyses to establish the TS flow rate requirement. One–hundred (100) gpm is the value currently assumed for this bypass flow.

The Core Spray System is designed to provide cooling to the core when reactor pressure is low. Upon receipt of an initiation signal, the core spray pumps in both subsystems are automatically started approximately 15 seconds after AC power is available. When the reactor pressure vessel (RPV) pressure drops sufficiently, core spray flow to the RPV begins. A full flow test line is provided to route water from and to the suppression pool to allow testing of the Core Spray System without spraying water in the RPV.

The TS Bases for SR 3.5.1.7 provide information on the rationale for the surveillance. The performance requirements of the Core Spray System pumps are determined through application of the 10 CFR 50, Appendix K criteria via the ECCS analyses. This surveillance is performed in accordance with the American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code requirements to verify the Core Spray pumps develop the flow rates required by the safety analyses. The pump flow rates ensure adequate core cooling is provided to satisfy the acceptance criteria of 10 CFR 50.46. The Core Spray pump flow rates are verified against a system head equivalent to the reactor to containment pressure expected during a Loss of Coolant Accident (LOCA). The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and the reactor pressure vessel (RPV) pressure present during a LOCA. These values are established analytically.

## 4.0 PROPOSED CHANGE

MNGP proposes to revise the flow rate for the Core Spray pumps in Specification 3.5.1, "ECCS – Operating," SR 3.5.1.7 from 2800 to 2835 gpm, as shown below.[3]

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.1.7 | Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor to containment pressure.<br><br>System / Flow Rate / No. of Pumps / System Head Corresponding to a Reactor to Containment Pressure of<br>Core Spray / ≥ ~~2800~~ 2835 gpm / 1 / ≥ 130 psi<br>LPCI / ≥ 3870 gpm / 1 / ≥ 20 psi | In accordance with the Inservice Testing Program |

The TS change (mark-up) associated with this proposed change is provided in Enclosure 2. No changes are necessary to the TS Bases in conjunction with this license amendment request.

## 5.0 TECHNICAL ANALYSIS

The LOCA is analyzed in conjunction with the ECCS performance evaluation in accordance with 10CFR50.46 and Appendix K to 10CFR50. A complete spectrum of postulated break sizes and locations is considered in the evaluation of ECCS performance. The objective of the LOCA analysis is to demonstrate conformance with the ECCS acceptance criteria of 10CFR50.46 for the most limiting break size, break location and single failure combination for the plant. The SAFER/GESTR-LOCA application methodology was applied for the Monticello LOCAs evaluation. The Monticello ECCS performance evaluation for GE14 fuel supplements the ECCS-LOCA evaluation.

The Core Spray System is designed to restore and maintain the coolant in the reactor vessel in combination with other ECCSs such that the core is adequately cooled to preclude fuel damage. Section 6.2.2.1 of the Updated Safety Analysis Report (USAR) describes the pertinent licensing design basis for the Core Spray System. It states:

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3. Addition(s) are shown by dotted underlining or revision bars, and deletion(s) by strikeout.

> The Core Spray System flow requirement established by the plant safety analysis of [USAR] Section 14.7.2 resulted in a total rated flow requirement of 2,800 gpm (2,700 gpm minimum flow into the core plus 100 gpm margin for leakage).

The design rated flow for each of the two core spray subsystems is approximately 3020 gpm. The 2700 gpm flow rate assumed for each subsystem in the SAFER evaluation reflects the core spray flow assumed to be actually delivered inside the core shroud. Measurement of the core spray flow rate is performed downstream of the core spray pumps outside the reactor vessel. The core spray flow delivery curve (core spray flow inside the core shroud versus reactor vessel pressure) used in the Monticello SAFER analysis represents a quadratic fit obtained from the assumed delivery flow rate of 2700 gpm at 130 psid and the pump shutoff head of 338 psid. Postulated leakage in the core spray piping between the reactor vessel wall and core shroud results in a diversion of flow into the downcomer (annulus) region and out a recirculation line break, resulting in this portion of the injection flow being unavailable for core cooling. The flow rate specified in TS SR 3.5.1.7, currently 2800 gpm, is intended to conservatively account for postulated core spray pressure boundary leakage and low pressure coolant injection (or LPCI) jet pump slip joint leakage.

5.1 <u>MNGP Core Spray Cracking Experience and Bypass Flow Determination</u>

Core spray pipe cracking was first detected by the industry in 1978 and found to be more widespread in subsequent years. The NRC issued IE Bulletin 80-13 (Reference 2) in response, requiring visual inspections to be performed of a better quality than required by the ASME Code. MNGP is one of the plants that has been performing inspections to the IE Bulletin 80-13 requirements for many years, and in the process found core spray pipe cracking indications. An excerpt from the NRC Safety Evaluation for a 1995 MNGP license amendment revising the core spray pump flow rate within the technical specifications (Reference 3) below, summarizes a modification performed for core spray header cracking (which also introduced some additional bypass leakage flow paths) and a reallocation of flow for LPCI jet pump slip joint leakage.

> Due to the design of the core spray and LPCI systems, there are minor flow losses (bypass leakage paths) that cause the actual flow rate into the core to be slightly less than the measured discharge flow rate of the pumps. The core spray system is assumed to have a small leakage from a 1/4 inch vent hole in the T-box which is located between the inner reactor vessel wall and the core shroud. The LPCI system is assumed to have some minor leakage from slip joints on the jet pump assemblies. Also, a core spray header crack that was discovered during the 1993 refueling outage, and the licensee's modifications to repair the crack, which involved the drilling of holes through the core spray header pipe [as part of the installation of a mechanical clamping device to relieve stresses on two

indications where leakage could occur], introduce[d] additional flow losses. These flow diversions are treated as leakage paths because the associated coolant goes into the annulus region of the vessel and would flow out the postulated design-basis loss-of-coolant accident (DBA-LOCA) recirculation system suction line break.

Also, subsequent to the discovery of the core spray header crack, two separate evaluations were performed by the licensee to assess the impact of the additional leakage paths with respect to the crack and the repair of the crack. The licensee provided a 10 CFR 50.59 safety evaluation to the NRC staff as Attachment (1) to a letter dated March 8, 1993, titled "Request for NRC Review and Approval of the Evaluation of the 'B' Core Spray Header Crack Indication Discovered During the 1993 Refueling Outage." The licensee's repair plan for the crack was provided by a letter dated June 30, 1994. The NRC staff review of these licensee evaluations are documented in separate letters to the licensee dated March 19, 1993, and August 26, 1994, respectively. The staff concurred with the licensee's conclusions that there is no substantive safety concern with respect to the core spray header crack and the repair of the crack.

However, the discrepancy between the flow rates required by the TS and the values assumed in the SAFER/GESTR-LOCA analysis remains. To resolve this issue, the licensee proposes to increase the required core spray flow rate by 100 gpm (46 gpm to account for core spray leakage plus 50 gpm to account for LPCI leakage plus 4 gpm for margin) to account for all of the assumed ECCS bypass leakage paths.

This was how the present bypass leakage rate of 100 gpm assumed in the ECCS analyses was established and approved by the NRC.

5.2 Boiling Water Reactor Vessel and Internals Project

The Boiling Water Reactor Vessel and Internals Project (BWRVIP) was formed in 1994 to address boiling water reactor (BWR) vessel and internals issues. The BWRVIP developed inspection and evaluation (I&E) guidelines for the Core Spray System describing locations on the core spray piping / spargers for which inspection is needed, inspection needs for differing categories of plants, extent of inspection and reinspection for each location, and flaw evaluation procedures to determine allowable flaw sizes for each location or type of location. These I&E guidelines are followed by licensees in place of prior GE Service Information Letters and, when approved by the regulator, in the place of the requirements of NRC Bulletin 80-13.

The I&E guidelines present a baseline approach for inspections each plant will do to BWRVIP requirements for core spray components. Piping inspections can be by visual or ultrasonic testing (UT) techniques. Methods to determine the reinspection scope and frequency, taking into consideration recommended improved examination techniques, and considering the susceptibility and function of each welded and bolted core spray location are provided. Guidance on performing flaw evaluations, determining loading and stresses, and methodology to take stresses from finite element analyses of the system under loading combinations and perform limit load flaw evaluations at each weld is provided. Based on observed flaw lengths and assumed crack growth rates, a point in time is calculated at which the flaws will have grown to a size that core spray function may be impaired. Reinspection of the flaws is scheduled prior to the time at which the flaws have grown to unacceptable sizes.

For existing crack indications observed in the piping between the vessel and the core shroud, the worst case MNGP leakage has been determined with conservative assumptions for through wall conditions and crack growth over time. The predicted leakage and the SAFER evaluation flow requirements are totaled and the resulting flow rate is verified to be within the rated capability of the associated core spray pump, as described in more detail below.

5.3 Development of the Core Spray Required Flow Rate

The Monticello SAFER (ECCS) evaluation value of 2700 gpm reflects the core spray flow rate assumed within the ECCS analyses to actually inject inside the core shroud. The diverted (or bypass) leakage flow is determined as follows. A summation of the MNGP leakage sources in accordance with previously NRC approved methodologies for the MNGP and the BWRVIP core spray methodology as applied to the MNGP follows.[4]

| | |
|---|---|
| Leakage assumed from 1994 T-Box modification:<br>• Core Spray one-quarter (1/4) inch vent hole in T-box<br>• T-Box Crack<br>• Holes added by T-Box Clamp Fixture modification | 46 gpm |
| Allowance for LPCI jet pump slip joint leakage | 50 gpm |
| Assumed analysis margin | 4 gpm |
| Amendment No. 93 assumed bypass leakage – Subtotal | 100 gpm |

4. Note these are current values. The leakage projected due to indications may change with time, based upon discovery of new indications or the growth of existing indications. Leakage is identified via visual or UT inspections and quantified in the crack growth calculations. Changes in leakage are permissible provided the summation of all contributors is less then the core spray subsystem flow rate required by TS SR 3.5.1.7.

| | |
|---|---|
| Amendment No. 93 assumed bypass leakage – (carried over from previous page) | 100 gpm |
| Margin for assumed leakage associated with the P5 and P6 welds | 32 gpm |
| Additional assumed margin | 3 gpm |
| Core spray flow rate to core assumed in ECCS analysis | 2700 gpm |
| Proposed SR 3.5.1.7 flow rate — Grand Total | 2835 gpm |

Increasing the assumed bypass flow (leakage) component from 100 gpm to 135 gpm conservatively accounts for the potential bypass leakage paths (increasing the existing margin from 4 gpm to 7 gpm). Adding this value to the 2700 gpm flow rate assumed within the ECCS analyses establishes a required SR 3.5.1.7 core spray flow rate of 2835 gpm.

This increased bypass flow between the ECCS analysis value and the TS flow requirement specified in SR 3.5.1.7 accounts for expected pressure boundary leakage in the reactor annulus region (downcomer) between a core spray subsystem and the core shroud and includes LPCI jet pump slip joint leakage.

As stated previously, the existing margin between the Core Spray flow rate measured during surveillance testing and the ECCS analysis requirement is sufficient to accommodate the increased flow diversion allowing this condition to be treated as a non-conservative TS in accordance with NRC Administrative Letter 98-10 while this request is under review. Both Core Spray subsystems continue to be able to meet their required safety functions in the ECCS analyses.

Conclusion

Increasing the assumed bypass leakage conservatively accounts for potential bypass leakage paths with margin. Combining this value with the core spray flow rate assumed in the ECCS analyses establishes a required SR 3.5.1.7 core spray flow rate of 2835 gpm. This value will provide margin to minimize TS changes in the future, while maintaining the assumptions of the safety analyses.

## 6.0 REGULATORY ANALYSIS

### 6.1 No Significant Hazards Determination

In accordance with the requirements of 10 CFR 50.90, Northern States Power Company – Minnesota (NSPM) requests an amendment to facility Renewed Operating License DPR-22, for the Monticello Nuclear Generating Plant (MNGP) to revise the flow rate for the Core Spray pumps in Surveillance Requirement 3.5.1.7 within Technical Specification 3.5.1, "ECCS – Operating."

NSPM has evaluated the proposed amendment in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the MNGP in accordance with the proposed amendment presents no significant hazards. NSPM's evaluation against each of the criteria in 10 CFR 50.92 follows

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

   Response: No.

   The low pressure Emergency Core Cooling System (ECCS) subsystems are designed to inject to reflood or to spray the core after any size break up to and including a design basis Loss of Coolant Accident (LOCA). The proposed change to the Core Spray System required flow rate does not change the operating configurations or minimum amount of operating equipment assumed in the safety analysis for accident mitigation. The change does not require any change in safety analysis methods or results. Also, it does not change the amount of core spray provided to the core in the accident analyses (it increases the amount of flow assumed diverted). No changes are proposed to the manner in which the ECCS provides plant protection or which would create new modes of plant operation.

   The proposed change does not result in any new or affect the probability of any accident initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

   Therefore, the proposed change 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.

   This change does not affect the method by which any plant systems perform a safety function. It does not introduce any new equipment, or hardware changes, which could create a new or different kind of accident. No new release pathways or equipment failure modes are created. No new accident scenarios failure mechanisms or limiting single failures are introduced as a result of this request. This request does not affect the normal methods of plant operation.

   The Core Spray pumps retain their ability to function following any accident previously evaluated and provide the proper flow rate to the core. Therefore, the implementation of the proposed change will not create a possibility for an accident of a new or different type than those previously evaluated.

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

   Response: No.

   The ECCS are designed with sufficient redundancy such that if a Core Spray subsystem were unavailable, or did not provide the required flow rate, the remaining low pressure subsystems are capable of providing water and removing heat loads to satisfy the Updated Safety Analysis Report requirements for accident mitigation or unit safe shutdown.

   There is no change in the Limiting Conditions for Operation. The change in Core Spray flow rate assumed in Surveillance Requirement 3.5.1.7 is in accordance with the BWRVIP program requirements for establishing leakage and the assumptions of the Monticello ECCS analysis.

   For these reasons, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, NSPM has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(c), in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

## 6.2 Applicable Regulatory Requirements

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in the TSs. As stated in 10 CFR 50.36, the TSs will include surveillance requirements to assure that the limiting conditions for operation (LCO) (and associated remedial actions) are met. The proposed change revises Surveillance Requirement 3.5.1.7 to reflect an increased required flow rate to account for potential diversion of flow from reaching the reactor core.

MNGP was designed largely before the publishing of the 70 General Design Criteria (GDC) for Nuclear Power Plant Construction Permits proposed by the Atomic Energy Commission (AEC) for public comment in July 1967, and constructed prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50. As such, the MNGP was not licensed to the Appendix A, General Design Criteria (GDC).

MNGP USAR, Section 1.2, lists the principal design criteria (PDCs) for the design, construction and operation of the plant. USAR Appendix E provides a *plant comparative evaluation to the 70 proposed AEC design criteria.* It was concluded that the plant conforms to the intent of the GDCs. The applicable GDCs and PDCs are discussed below.

- <u>PDC 1.2.3 -- Reactor Core Cooling</u>

  c. Redundant heat removal systems are provided to preserve reactor core heat transfer geometry following various postulated design basis loss-of-coolant accidents.

- <u>PDC 1.2.11 -- Class I Equipment and Structures</u>

  Class I structures, systems and components are those whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the plant under normal or accident conditions and to the removal of decay and sensible heat from the reactor.

The Monticello ECCS analyses are performed by General Electric in accordance with the codes and methods discussed in the GESTAR licensing topical report and the following GDCs are applicable under that basis.

- <u>GDC 35 -- Emergency core cooling</u>. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

- GDC 36 -- Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

- GDC 37 -- Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

NSPM has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria. The technical analysis concludes that the proposed TS changes will continue to assure that the design requirements and acceptance criteria of MNGP ECCS LOCA analysis are met. Based on this, there is reasonable assurance that the health and safety of the public, following approval of this TS change, is unaffected.

## 7.0 ENVIRONMENTAL EVALUATION

NSPM has determined that the proposed amendment would not change a requirement with respect to installation or use of a facility or component located within the restricted area, as defined in 10 CFR 20, or change an inspection or surveillance requirement in such a way that it does not meet the following criteria. The proposed amendment does not involve (i) a no significant hazards consideration, or (ii) authorize a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for a categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, the NSPM concludes pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

## 8.0 REFERENCES

1. NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," dated December 29, 1998.

2. NRC Inspection and Enforcement Bulletin No. 80-13, "Cracking in Core Spray Spargers," dated May 12, 1980.

3. NRC Letter, "Monticello Nuclear Generating Plant — Issuance of Amendment Re: Revised Core Spray Pump Flow (TAC No. M85838)," dated July 12, 1995, (Amendment No. 93).

## ENCLOSURE 2

**MONTICELLO NUCLEAR GENERATING PLANT**

**LICENSE AMENDMENT REQUEST**
**REVISE CORE SPRAY PUMP FLOW RATE IN**
**SPECIFICATION 3.5.1, ECCS – OPERATING**

**MARKED-UP TECHNICAL SPECIFICATION PAGE**

(1 page follows)

SURVEILLANCE REQUIREMENTS (continued)

<table>
<tr><th></th><th>SURVEILLANCE</th><th>FREQUENCY</th></tr>
<tr><td>SR 3.5.1.7</td><td>Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor to containment pressure.
<table>
<tr><th>System</th><th>Flow Rate</th><th>No. of Pumps</th><th>System Head Corresponding to a Reactor to Containment Pressure of</th></tr>
<tr><td>Core Spray</td><td>2835<br>≥ 2800 gpm</td><td>1</td><td>≥ 130 psi</td></tr>
<tr><td>LPCI</td><td>≥ 3870 gpm</td><td>1</td><td>≥ 20 psi</td></tr>
</table></td><td>In accordance with the Inservice Testing Program</td></tr>
<tr><td>SR 3.5.1.8</td><td>-----NOTE-----<br>Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.<br>----------<br>Verify, with reactor steam dome pressure ≤ 1025.3 psig and ≥ 950 psig, the HPCI pump can develop a flow rate ≥ 2700 gpm against a system head corresponding to reactor pressure.</td><td>In accordance with the Inservice Testing Program</td></tr>
<tr><td>SR 3.5.1.9</td><td>-----NOTE-----<br>Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.<br>----------<br>Verify, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 2700 gpm against a system head corresponding to reactor pressure.</td><td>24 months</td></tr>
</table>