July 12, 1995

Mr. Roger O. Anderson, Director
Licensing and Management Issues
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: REVISED CORE SPRAY PUMP FLOW (TAC NO. M85838)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 93 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated February 12, 1993, as supplemented by letters dated March 22, 1993, and August 25, 1994.

The amendment increases the minimum core spray pump flow to more conservatively account for emergency core cooling systems bypass leakage paths. In addition, the amendment makes editorial and administrative changes to correct branching errors, typographical errors, and similar discrepancies which existed in various sections of the TS.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

T. J. Kim, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures: 1. Amendment No. 93 to DPR-22
2. Safety Evaluation

cc w/encl: See next page

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DATED: July 12, 1995

AMENDMENT NO. 93 TO FACILITY OPERATING LICENSE NO. DPR-22-MONTICELLO

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 93
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (the licensee) dated February 12, 1993, as supplemented March 22, 1993, and August 25, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Facility Operating License No. DPR-22 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 93 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 12, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 93

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
52	52
53	53
54	54
55	55
60d	60d
101	101
107	107
110	110
113	113
127	127
151	151
156	156

Table 3.2.2
Instrumentation That Initiates Emergency Core Cooling Systems

Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems(3)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (3)	Required Conditions*
A. Core Spray and LPCI					
1. Pump Start					
a. Low Low Reactor Water Level and	≥6'6"≤6'10"	2	4(4)	4	A.
b. i. Reactor Low Pressure Permissive or	≥450 psig	2	2(4)	2	A.
ii. Reactor Low Pressure Permissive Bypass Timer	20±1 min	2	1	1	B.
c. High Drywell Pressure (1)	≤2 psig	2	4(4)	4	A.
2. Low Reactor Pressure (Valve Permissive)	≥450 psig	2	2(4)	2	A.
3. Loss of Auxiliary Power	-----	2	2(2)	2	A.

Table 3.2.2
Instrumentation That Initiates Emergency Core Cooling Systems

Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (3)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (3)	Required Conditions*
B. HPCI System					
1. High Drywell Pressure (1)	≤2 psig	1	4	4	A.
2. Low-Low Reactor Water Level	≥6'6"≤6'10"	1	4	4	A.
C. Automatic Depressurization					
1. Low-Low Reactor Water Level and	≥6'6"≤6'10"	2	2	2	B.
2. Auto Blowdown Timer and	≤120 seconds	2	1	1	B.
3. Low Pressure Core Cooling Pumps Dis-Charge Pressure Interlock	≤100 psig	2	12(4)	12(4)	B.

Table 3.2.2 - Continued

Instrumentation That Initiates Emergency Core Cooling System

Function	Trip Setting	Min. No. of Operable or Operating Trip Systems(3)	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (3)	Required Conditions*
D. Diesel Generator					
1. Degraded or Loss of Voltage Essential Bus (5)					
2. Low Low Reactor Water Level	≥6'6"≤6'10"	2	4(4)	4	C.
3. High Drywell Press	≤2 psig	2	4(4)	4	C.

NOTES:

1. High drywell pressure may be bypassed when necessary only by closing the manual containment isolation valves during purging for containment inerting or de-inerting. Verification of the bypass condition shall be noted in the control room log. Also need not be operable when primary containment integrity is not required.

2. One instrument channel is a circuit breaker contact and the other is an undervoltage relay.

Table 3.2.2 - Continued

Notes:

3. Upon discovery that minimum requirements for the number of operable or operating trip systems, or instrument channels are not satisfied action shall be initiated to:

   (a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

   (b) Place the plant under the specified required conditions using normal operating procedures.

4. All instrument channels are shared by both trip systems.

5. See table 3.2.6.

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

   A. Comply with Specification 3.5.A.

   B. Reactor pressure ≤150 psig.

   C. Comply with Specification 3.9.B.

Table 3.2.8
Other Instrumentation

Function	Trip Setting	Minimum No. of Operable or Operating Trip System (1)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Conditions*
A. RCIC Initiation					
1. Low-Low Reactor Level	≥6'6"& ≤6'10" above top of active fuel	1	2	2	B
B. HPCI/RCIC Turbine Shutdown					
a. High Reactor Level	≤14'6" above top of active fuel	1	2	2	A
C. HPCI/RCIC Turbine Suction Transfer					
a. Condensate Storage Tank Low Level	≥2'0" above tank bottom	1	2	2	C

NOTE:

1. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channel are not satisfied, action shall be initiated to:

   a. Satisfy the requirements by placing the appropriate channels or systems in the tripped condition (Turbine/Feedwater Trip only), or
   b. Place the plant under the specified required condition using normal operating procedures.

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

   A. Reactor in Startup, Refuel, or Shutdown Mode.
   B. Comply with Specification 3.5.D.
   C. Align HPCI and RCIC suction to the suppression pool. Restore channels to operable status within 30 days or place the plant in Required Condition A.

3.0 LIMITING CONDITION FOR OPERATION

3.5 CORE AND CONTAINMENT SPRAY/COOLING SYSTEMS

Applicability:

Applies to the operational status of the emergency cooling systems.

Objective:

To insure adequate cooling capability for heat removal in the event of a loss of coolant accident or isolation from the normal reactor heat sink.

Specification:

A. ECCS Systems

1. Except as specified in section 3.5.A.3, both Core Spray subsystems and the Low Pressure Coolant Injection (LPCI) Subsystem (LPCI Mode of RHR System) shall be operable whenever irradiated fuel is in the reactor vessel and the reactor water temperature is greater than 212°F.

2. Except as specified in section 3.5.A.3, the High Pressure Coolant Injection (HPCI) System and the Automatic Depressurization System (ADS) shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel except during reactor vessel hydrostatic or leakage tests.

4.0 SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT SPRAY/COOLING SYSTEMS

Applicability:

Applies to the periodic testing of the emergency cooling systems.

Objective:

To verify the operability of the emergency cooling systems.

Specification:

A. ECCS Systems

1. Demonstrate the Core Spray Pumps develop a 2,800 gpm flow rate against a system head corresponding to a reactor pressure of 130 psi greater than containment pressure, when tested pursuant to Specification 4.15.B.

2. Demonstrate the LPCI Pumps develop a 3,870 gpm flow rate against a system head corresponding to two pumps delivering 7,740 gpm at a reactor pressure of 20 psi greater than containment pressure, when tested pursuant to Specification 4.15.B.

3. Demonstrate the HPCI Pump develops a 2700 gpm flow rate against a reactor pressure range of 1120 psig to 150 psig, when tested pursuant to Specification 4.15.B.

3.0 LIMITING CONDITION FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

F. Recirculation System

1. The reactor may be started and operated, or operation may continue with only one recirculation loop in operation provided that:

   a. The following changes to setpoints and safety limit settings will be made within 24 hours after initiating operation with only one recirculation loop in operation.

      1. The Operating Limit MCPR (MCPR) will be changed per Specification 3.11.C.

      2. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) will be changed as noted in Table 1 of the Core Operating Limits Report.

      3. The APRM Neutron Flux Scram and APRM Rod Block setpoints will be changed as noted in Specification 2.3.A and Table 3.2.3.

   b. Total core flow will be maintained greater than 39% when core thermal power is above the limit specified in Figure 3.5.1.

F. Recirculation System

1. See Specification 4.6.G

2. The following baseline noise levels will be obtained prior to operation with only one recirculation pump in operation at a core thermal power greater than that specified in Figure 3.5.1 or with a core flow greater than 45% provided that baseline values have not been established since the last core refueling. Baseline values will be taken with only one recirculation pump running.

   a. Establish a baseline core plate ΔP noise level.

   b. Establish a baseline APRM and LPRM neutron flux noise level.

3. With only one recirculation loop in operation at a core thermal power greater than that specified in Figure 3.5.1 or with a core flow greater than 45%, determine the following noise levels at least once per 8 hour period and within 30 minutes after a core thermal power increase of greater than 5% of rated thermal power.

   a. Core plate ΔP noise levels.

   b. APRM and LPRM neutron noise levels.

Bases 3.5/4.5

A. ECCS Systems

The core spray system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and, together with the LPCI mode of the RHR system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the automatic depressurization system (ADS).

The Core Spray System is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining. The Core Spray pump is designed to deliver greater than or equal to 3020 gpm (the SAFER/GESTR-LOCA safety analysis assumed a Core Spray Pump flow of 2,800 gpm, or 2,700 gpm flow into the core + 100 gpm to account for ECCS bypass leakage) against a system head corresponding to a reactor pressure of 130 psi greater than containment pressure.

The surveillance requirements provide adequate assurance that the Core Spray System will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Four pumps are available to provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS. LPCI Loop Selection Logic determines which Recirculation loop the four RHR pumps will pump into. Each RHR pump was designed to deliver greater than or equal to 4000 gpm (the safety analysis assumed two pumps delivering 7,740 gpm) against a system head corresponding to a reactor pressure of 20 psi greater than containment pressure.

The allowed out-of-service conditions (Section 3.5.A.3) are determined from ECCS analysis cases analyzed. Only one of these conditions is permitted to exist. If more than one condition exists, an orderly shutdown shall be initiated. A LPCI injection path consists of the two motor operated injection valves on that path.

The surveillance requirements provide adequate assurance that the LPCI system will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The

Bases 3.5/4.5 Continued:

The RHR service water system provides cooling for the RHR heat exchangers and can thus maintain the suppression pool water within limits. With the flow specified, the pool temperature limits are maintained as specified in Specification 3.7.A.1.

D. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. The pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therefore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

The surveillance requirements provide adequate assurance that the RCIC system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

E. Cold Shutdown and Refueling Requirements

The purpose of Specification 3.5.E is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment spray/cooling subsystems may be out of service. This specification allows all core and containment spray/cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.E.2 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may by necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

3.0 LIMITING CONDITION FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
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 Specifications 3.2.H and 3.5.A, respectively). 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.	E. Safety/Relief Valves 1. a. Safety/relief valves shall be tested or replaced each refueling outage pursuant to Specification 4.15.B. The nominal self-actuation setpoints are specified in Section 2.4.B. 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 at least once every three months. 2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

The safety/relief valves have two functions; 1) over-pressure relief (self-actuated 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 Low-Low Set and ADS functions are discussed further in Sections 3.2 and 3.5.

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 of the ASME Pressure Vessel Code Section III Nuclear Vessels requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system quarterly provisions assurance of bellows integrity.

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.

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.

I. Deleted

3.0 LIMITING CONDITION 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 operation 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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 93 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated February 12, 1993, as supplemented March 22, 1993, and August 25, 1994, the Northern States Power Company (NSP, the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed amendment would make the following changes to the Monticello TS:

TS Page	Section	Description of Change
52	Table 3.2.2	Revise core spray and low pressure coolant injection trip function A.1.b.ii to delete the word "and" from the function description (Reactor Low Pressure Permissive Bypass Timer); change Required Condition "C" to Required Condition "B"; and correct the spelling of the word "Channels" in the heading for the second column from the right.
53	Table 3.2.2	Revise HPCI system trip functions B.1 (High Drywell Pressure) and B.2 (Low-Low Reactor Water Level) to refer to Required Condition "A" instead of "B." Revise automatic depressurization system trip functions C.1 (Low-Low Reactor Water Level), C.2 (Auto Blowdown Timer), and C.3 (Low Pressure Core Cooling Pumps Discharge Pressure Interlock) to refer to Required Condition "B" instead of "C."
54	Table 3.2.2	Revise diesel generator trip functions D.2 (Low-Low Reactor Water Level) and D.3 (High Drywell Pressure) to refer to Required condition "C" instead of "D."
55	Table 3.2.2	Delete the existing Required Condition "B", and re-identify remaining Required Conditions "C" and "D" as "B" and "C", respectively.

60d — Table 3.2.8 — Revise Required Condition "B" to refer to Specification 3.5.D instead of 3.5.F.2.

Delete redundant "status" from the description of Required Condition "C." near bottom of page.

101 — 4.5.A.1 — Revise the minimum required flow rate of the core spray pumps upwards from 2,700 gpm to 2,800 gpm.

107 — 3.5.F.1.a.2 — The specification currently reads:

> "The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) will be changed as noted in Table 3.11.1."

Revise this specification to read:

> "The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) will be changed as noted in Table 1 of the Core Operating Limit Report."

110 — 3.5/4.5 Bases Part A. — The second sentence of the second paragraph of the ECCS Bases currently reads:

> "The Core Spray pump is designed to deliver greater than or equal to 3,020 gpm (safety analysis assumed 2,700 gpm) against a system head corresponding to a reactor pressure 130 psi greater than containment pressure."

Revise the above sentence to read:

> "The Core Spray pump is designed to deliver greater than or equal to 3,020 gpm (the SAFER/GESTR-LOCA safety analysis assumed a Core Spray pump flow of 2,800 gpm, or 2,700 gpm flow into the core + 100 gpm to account for ECCS bypass leakage) against a system head corresponding to a reactor pressure 130 psi greater than containment pressure."

Also, in the fifth paragraph on page 110, reference Specification 3.5.A.3 instead of 3.5.A.2.

113 — 3.5/4.5 Bases Part E — Revise the last paragraph of this page to refer to Specification 3.5.E.2 instead of 3.5.E.4.

127 — 3.6.E.1 — The specification currently reads:

> "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:

"a. The safety valve function (self-actuation) of seven safety/relief valves shall be operable.

"b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.

"c. The Low-Low Set Function for three non-Automatic Pressure Relief Valves shall be operable as required by Specification 3.2.H."

Revise this specification to read as follows:

"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 Specifications 3.2.H and 3.5.A, respectively)."

127 3.6.E.2 — Revise specification to refer to Specification 3.6.E.<u>1</u> instead of 3.6.E.<u>1.a</u>.

151 3.6/4.6 Bases — The first paragraph on this page currently reads:

"The safety/relief valves have two functions; i.e., power relief or self-actuated by high pressure. The solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate opening of the valves. This function is discussed in Specification 3.5.E. In addition, the valves can be operated manually."

Revise this paragraph to read:

"The safety/relief valves have two functions; (1) over pressure relief (self-actuated 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 Low-Low Set and ADS functions are discussed further in Sections 3.2 and 3.5."

156 3.7.A.1 Revise this specification to refer to Specification 3.5.E.2 instead of 3.5.G.4.

The licensee states that the portion of the change related to increasing the required core spray pump flow from 2,700 gpm to 2,800 gpm is intended to account for the flow losses (bypass leakage paths) inherent to the emergency core cooling systems (ECCS) design. Increasing the required flow rate for the core spray pumps will assure that the total flow entering the core (ECCS pump flow minus bypass leakage) during a loss of coolant accident (LOCA) is consistent with the value assumed in the Monticello SAFER/GESTR-LOCA analysis.

In addition, NSP's changes to the 3.6/4.6 Bases discussion on page 151 are intended to clarify and correct existing statements that are both confusing and misleading. The current wording states, incorrectly, that coincident high drywell pressure and low-low water level signals initiate automatic actuation of the safety relief valves. The licensee has indicated that this is no longer true because of a modification performed in response to NUREG-0737, Item II.K.3.18 (Reference: License Amendment No. 62 dated March 31, 1989). The correct discussion of this function is provided in Section 3.2 of the TS. The proposed change will address this discrepancy and reference the proper information.

Similarly, NSP's changes to Specification 3.6.E are intended to clarify the intent of the specification with respect to automatic depressurization system (ADS) and low-low set system requirements. As presently written, Specifications 3.2.H, 3.5.A, and 3.6.E cross reference each other in a manner that could lead to misinterpretation of the governing requirements for these systems. The language of the proposed change is intended to alleviate this concern.

The licensee states that the remaining changes are editorial in nature and are intended primarily to correct branching errors that occurred in previous license amendments. Most of these errors resulted from License Amendment No. 79 (SAFER/GESTR), dated April 9, 1991, in which Section 3.5/4.5 (Core and Containment Cooling Systems) was substantially rewritten and reorganized. Several specifications were either deleted or re-numbered at the time and related changes to associated cross-references were missed.

The August 25, 1994 letter provided clarifying information within the scope of the original submittal and did not change the staff's initial proposed no significant hazards considerations determination.

2.0 EVALUATION

2.1 Increase in Core Spray Pump Required Flow Rate

Technical Specification 4.5.A.1 currently requires that the core spray pumps develop a flow rate of 2,700 gpm against a system head corresponding to a reactor pressure of 130 psi greater than the containment pressure. Technical Specification 4.5.A.2 requires that the low pressure coolant injection (LPCI) pumps develop a flow rate of 3,870 gpm, corresponding to two pumps delivering 7,740 gpm, at a reactor pressure of 20 psi greater than containment pressure. The SAFER/GESTR-LOCA analysis prepared for Monticello by General Electric incorrectly utilized the above flow rates to represent actual flow into the core.

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, introduce 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.

An evaluation was performed by the licensee (Reference: Nonconforming Item Report 92-037) which confirmed that the actual flow rates for individual ECCS pumps minus assumed leakage was adequate to meet the flow rates assumed in the SAFER/GESTR-LOCA analysis; therefore, there were no immediate operability concerns. 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. The LPCI flow rate currently required by the TS (3,870 gpm per pump/7,740 gpm total) would remain unchanged.

The licensee discussed this issue with General Electric, who performed the Monticello SAFER/GESTR-LOCA analysis. General Electric has concluded that with respect to the analysis, it is of no significance whether the assumed ECCS bypass leakage of 100 gpm is accounted for by increasing core spray flow, LPCI flow, or both. However, when the trade-off between increasing core spray or LPCI flow is considered, increased core spray flow is preferred for the following reasons:

a. In addition to replenishing vessel water inventory lost during the DBA-LOCA, core spray flow (which is injected into the vessel above the core) is more effective in collapsing any steam bubble that might form in the vessel.

b. The core spray pumps deliver flow to the reactor vessel at higher reactor pressures than the residual heat removal (RHR) pumps operating in the LPCI mode, which is beneficial in mitigating a postulated DBA-LOCA.

An additional factor in NSP's decision to account for all ECCS assumed bypass leakage by increasing core spray flow involves the relative capacities of the core spray and RHR pumps. Each of the four RHR pumps (which provide LPCI flow) is currently capable of consistently meeting the existing TS flow rate requirement of 3,870 gpm. A review of recent surveillance test results by the licensee has confirmed that the pumps are also capable of meeting the slightly higher flow rate assumed by the SAFER/GESTR-LOCA analysis (3,895 gpm, which equates to an additional 25 gpm per operating pump assuming only two pumps are running, to account for the total LPCI bypass leakage of 50 gpm). However, the higher value (3,895 gpm) is very near the upper limit of RHR pump capacity, and there is insufficient margin remaining to ensure the pumps would consistently achieve this higher flow in the future.

Conversely, core spray pump performance is such that the minimum required flow could be increased by 100 gpm to 2,800 gpm without difficulty. The current test criteria for the core spray pumps conservatively specifies an acceptance criteria of 3,020 gpm against a system head corresponding to 130 psi greater than containment pressure. Thus, the current test criteria provides a margin of more than 200 gpm over the proposed new TS criteria.

The combination of ECCS pumps available for each single failure evaluated for a DBA-LOCA by the SAFER/GESTR-LOCA analysis includes a core spray pump whenever two LPCI (RHR) pumps are available. Therefore, a core spray pump would always be available to provide the additional flow necessary to offset the assumed LPCI bypass leakage.

As discussed above, the proposed change would adequately resolve the discrepancy between the current TS ECCS pump flow rates and the flow into the core assumed by the SAFER/GESTR-LOCA analysis. The change is primarily administrative and has no impact on plant safety, since the basic assumptions supporting the SAFER/GESTR-LOCA analysis, and therefore the conclusions of the analysis, remain unchanged. Therefore, the staff finds the changes acceptable.

2.2 Editorial Changes

The proposed change to TS page 53, Table 3.2.2 corrects a typographical error introduced in Amendment 62 which indicated two instrument channels per trip system instead of one. The proposed change is necessary to correct the number of instrument channels for the Auto Blow-down Timer, C.2, to one. The staff finds this change acceptable.

The changes to the TS 3.6/4.6 Bases discussion should have been included as part of License Amendment No. 62, dated March 31, 1989. Amendment No. 62 reflected modifications to the automatic depressurization system logic that, among other things, removed the high drywell pressure interlock in response to NUREG-0737 Item II.K.3.18. Other portions of the TS affected by the modification were updated appropriately, but the necessary changes to page 151 were missed. Safety considerations associated with the automatic depressurization system logic change were fully addressed at the time Amendment No. 62 was processed and the proposed correction does not present any new safety questions or concerns. The proposed change is necessary to ensure the 3.6/4.6 Bases discussion is consistent with the intent of the remainder of the TS and is therefore acceptable to the staff.

The remaining changes are editorial in nature and do not change the intent of the existing TS. Most of these changes serve to correct internal branching and cross reference errors that occurred during previous license amendments. One of the requested typographical changes has already been corrected. The remaining changes clarify, but do not change, the intent of existing specifications. These changes have no impact on plant safety. The staff finds these editorial changes acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (58 FR 41508). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR

51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

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

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