February 15, 1991

Docket No. 50-263

Mr. T. M. Parker, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Parker:

SUBJECT: AMENDMENT NO. 77 TO FACILITY OPERATING LICENSE NO. DPR-22: (TAC NO. 79286)

The Commission has issued the enclosed Amendment No. 77 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. This amendment consists of changes to the Technical Specifications in response to your application dated December 13, 1990.

The amendment revises pump and valve surveillance testing requirements to be consistent with ASME code requirements. The amendment also eliminates requirements for immediate and more frequent surveillance testing of redundant equipment when equipment is found or made inoperable.

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

Sincerely,

/s/

William O. Long, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 77 to License No. DPR-22
2. Safety Evaluation

cc w/enclosures:
See next page

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

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Mr. T. M. Parker, Manager
Northern States Power Company

Monticello Nuclear Generating Plant

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
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 November 13, 1990, 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. 77, 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for William O. Long

L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 15, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 77

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 marginal lines indicating the area of change.

Remove	Insert
ii	ii
93	93
94	94
95	95
96	96
99	99
100	100
101	101
102	102
103	103
104	104
105	105
106	106
107	107
108	108
110	110
111	111
111a	
112	112
113	113
116	116
117	117
118	118
119	119
120	120
166	166
167	167
170	170
171	171
201	201
204	204
225	225
226	226
228b	228b
229ff	229ff
229g	229g

Amendment No. 9, 35, 47, 71, 77

3.0 LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the standby liquid control system.

Objective

To assure the availability of an independent reactivity control mechanism.

SPECIFICATION:

A. System Operation

1. The standby liquid control system shall be operable at all times when fuel is in the reactor and the reactor is not shut down by control rods, except as specified in 3.4.A.2.

2. From and after the date that a redundant component is made or found to be inoperable, reactor operation is permissible only during the following 7 days provided that the redundant component is operable.

4.0 SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirements for the standby liquid control system.

Objective:

To verify the operability of the standby liquid control system.

SPECIFICATION

A. The operability of the standby liquid control system shall be verified by performance of the following tests:

1. At least once per month -

   Demineralized water shall be recycled to the test tank. Pump minimum flow rate of 24 gpm shall be verified against a system head of 1275 psig. Comparison of the measured pump flow rate against equation 2 of paragraph 3.4.B.1 shall be made to demonstrate operability of the system in accordance with the ATWS Design Basis.

2. At least once during each operating cycle -

   a. Manually initiate one of the two standby liquid control systems and pump demineralized water into the reactor vessel. This test checks explosion of the charge associated with the tested system, proper operation of the valves and pump capacity. Both systems shall be tested and inspected, including each explosion valve in the course of two operating cycles.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	b. Explode one of two primer assemblies manufactured in the same batch to verify proper function. Then install, as a replacement, the second primer assembly in the explosion valve of the system tested for operation. c. Test that the setting of the system pressure relief valves is between 1350 and 1450 psig.

B. Boron Solution Requirements

At all times when the Standby Liquid Control System is required to be operable:

1. The liquid poison tank shall contain a boron bearing solution that satisfies the volume, concentration and enrichment requirements of Figure 3.4.1, or compliance can be demonstrated by satisfying the following equations:

Equation 1 (Original Design Basis):

$$V \geq \left(\frac{71.18}{0.0051xC + 0.998}\right)\left(1+\frac{4821}{1101-E}\right)\left(\frac{19.8}{E}\right)\left(\frac{100}{C}\right) + 128 \text{ gal}$$

Equation 2 (ATWS Design Basis):

$$C \geq 8.28\left(\frac{86}{Q}\right)\left(\frac{19.8}{E}\right)$$

where:

V - indicated Boron solution tank volume (gal)
E - measured Boron solution enrichment (atom%)
C - measured Boron solution concentration (wt%)
Q - measured pump flow rate (gpm) at 1275 psig

If Equation 1 is satisfied but Equation 2 cannot be met, continued plant operation is permissible, provided that:

a. Compliance with Equation 2 is demonstrated within 7 days or

b. The Commission shall be notified and a special report provided outlining the actions taken and the plans and schedule for demonstrating compliance with the ATWS Design Basis.

2. The temperature shall not be less than the solution temperature presented in Figure 3.4.2.

3. The heat tracing on the pump suction lines shall be operable whenever the room temperature is less than the solution temperature presented in Figure 3.4.2.

B. Boron Solution Surveillance

The availability of the proper boron bearing solution shall be verified by performance of the following tests:

1. At least once per cycle -

Boron enrichment shall be determined. The laboratory analysis to determine enrichment shall be obtained within 30 days of sampling or chemical addition.

2. At least once per month -

Boron concentration shall be determined. In addition, the boron concentration shall be determined any time water or boron are added or if the solution temperature drops below the limits specified by Figure 3.4.2.

3. At least once per day -

a. Solution volume shall be checked.

b. The solution temperature shall be checked.

c. The room temperature shall be checked in the vicinity of the standby liquid control system pumps.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
C. If Specification 3.4.A through B are not met, an orderly shutdown shall be initiated and the reactor shall be in Hot Shutdown within 12 hours.	

Basis 3.4 and 4.4:

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of boron in the reactor core in less than 125 minutes sufficient to bring the reactor from full power to a 3% delta k subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional 25% boron concentration margin for possible imperfect mixing of the chemical solution in the reactor water and dilution from the water in the cooldown circuit.

The time requirement (125 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak.

The ATWS Rule (10CFR50.62) requires the addition of a new design requirement to the generic SLC System design basis. Changes to flow rate, solution concentration or boron enrichment to meet the ATWS Rule do not invalidate the original system design basis. Paragraph (c)(4) of 10CFR50.62 states that:

> "Each boiling water reactor must have a Standby Liquid Control System (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution" (natural boron enrichment).

The described minimum system parameters (equivalent to 24 gpm, 10.7% concentration and 55 atom percent Boron-10 enrichment) will ensure an equivalent injection capability that meets the ATWS rule requirement.

Boron enrichment concentration, solution temperature, and volume (including check of tank heater and pipe heat tracing system) are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. A reliability analysis indicates that the plant can be operated safely in this manner for ten days. For additional margin, the allowable out of service time has been reduced to seven days.

The only practical time to test the standby liquid control system is during a refueling outage and by initiation from local stations. Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of less than once each refueling outage unnecessary. A test of explosive charges from one manufacturing batch is made to assure that the replacement charges for the tested system are satisfactory. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

The relief valves in the standby liquid control system protect the system piping and positive displacement pumps which are nominally designed for 1500 psi from overpressure. The pressure relief valves discharge back to the standby liquid control solution tank.

B. The solution saturation temperature varies with the concentration of sodium pentaborate. The solution will be maintained at least 5°F above the saturation temperature within the tank and suction piping to guard against precipitation. The 5°F margin is included in Figure 3.4.2. Temperature and liquid level alarms for the system are annunciated in the control room.

Pump operability is checked on a frequency to assure a high reliability of operation of the system should it ever be required.

Once the solution has been made up, boron concentration will not vary unless more boron or more water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Boron Enrichment will not vary unless more Boron is added. No deterioration of the Boron-10 enrichment level should occur during system standby operation. Considering these factors, the test intervals have been established.

3.0 LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT 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:

Low Pressure Core Cooling Capability

A. Core Spray System

1. Except as specified in 3.5.A.2., 3.5.A.3., and 3.5.A.5. below, both core spray subsystems shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant water temperature is greater than 212°F.

4.0 SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to periodic testing of the emergency cooling systems.

Objective:

To verify the operability of the emergency cooling systems.

Specification:

Low Pressure Core Cooling Capability

A. Surveillance of the core spray system shall be performed as follows:

1. Testing

Item	Frequency
Simulated automatic actuation test	Each refueling outage

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

Item	Frequency
Pump Operability	Pursuant to Specification 4.15.B
Valve operability	Pursuant to Specification 4.15.B

Core Spray header Δp instrumentation

Check	Once/day
Test	Once/month
Calibrate	Once/3 months

2. From and after the date that one of the core spray systems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days unless such system is sooner made operable, provided that during such 7 days all active components of the other core spray system and the LPCI mode of the RHR system and the diesel generators required for operation of such components (if no external source of power were available) shall be operable.

3. From and after the date that both core spray systems are made or found to be inoperable for any reason, reactor

operation is permissible only during the succeeding 72 hours unless at least one of such systems is sooner made operable, provided that during such 72 hours all active components of the LCPI mode of RHR system and the diesel generators required for operation of such components (if no external source of power were available) shall be operable.

4. Each core spray system shall be capable of delivering 3,020 gpm against a reactor pressure of 130 psig. If this rate of delivery requirement cannot be met, the system shall be considered inoperable.

5. If the requirements of 3.5.A.1-3 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

B. Low Pressure Coolant Injection (LPCI) Subsystem (LPCI Mode of RHR System)

1. Except as specified in 3.5.B.2 and 3.5.B.3 below, the LPCI shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that one of the LPCI pumps or admission valves is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 30 days for an inoperable LPCI pump or the succeeding 7 days for an inoperable admission valve, unless such pump or admission valve is sooner made operable, provided that during such 30 days for an inoperable LPCI pump or during such 7 days for an inoperable admission valve the remaining active components of the LPCI and containment cooling subsystem and all active components of both core spray systems and the diesel generators required for operation of such components (if no external source of power were available) shall be operable.

B. Surveillance of the Low Pressure Coolant Injection (LPCI) Subsystem (LPCI Mode of RHR System) shall be performed as follows:

1. Testing

Item	Frequency
Pump Operability	Pursuant to Specification 4.15.B
Valve operability	Pursuant to Specification 4.15.B
Cycling of RHR Intertie Line Valves	Once/Quarter
Simulated automatic actuation test	Every refueling outage

2. Deleted

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. From and after the date that two of the LPCI pumps or admission valves are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days for two inoperable LPCI pumps or the succeeding 72 hours for two inoperable addmission valves unless such pumps or admission valves are made operable sooner, provided that during such 7 days for two inoperable LPCI pumps or during such 72 hours for two inoperable admission valves all active components of both core spray systems, the containment cooling subsystem (including 2 LPCI pumps) and the diesel generators required for operation of such components (if no external source of power were available) shall be operable.	3. Deleted
4. A maximum of one drywell spray loop (containment cooling mode of RHR) may be inoperable for 7 days when the reactor water temperature is greater than 212°F. If the loop is not returned to service within 7 days, the orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F.	4. During each five year period, an air test shall be performed on the drywell spray headers and nozzles.
5. Each LPCI subsystem (RHR) pump shall be capable of delivering 4,000 gpm ±10% against a system head corresponding to three pumps delivering 12,000 gpm at a reactor pressure of 20 psi above the suppression chamber pressure. If this rate of delivery requirement cannot be met, the pump shall be considered inoperable.	

3.0 LIMITING CONDITIONS FOR OPERATION

Containment Cooling Capability

C. Residual Heat Removal (RHR) Service Water System

1. Except as specified in 3.5.C.2 and 3.5.C.3 below, both RHR service water system loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that one of the RHR service water system pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days all other active components of the RHR service water system are operable.

4.0 SURVEILLANCE REQUIREMENTS

Containment Cooling Capability

C. Surveillance of the RHR service water system shall be performed as follows:

1. Testing

Item	Frequency
Pump and valve operability	Pursuant to Specification 4.15.B

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. From and after the date that one of the RHR service water systems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days for 2 inoperable pumps powered from different divisions or during the succeeding 72 hours for 2 inoperable pumps powered from the same division unless such system is sooner made operable, provided that during such time all active components of the operable RHR service water system system shall be operable. 4. To be considered operable, a RHR service water pump shall be capable of delivering 3500 gpm against a head of 500 feet. 5. If the requirements of 3.5.C.1-3 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.	

3.0 LIMITING CONDITIONS FOR OPERATION

High Pressure Core Cooling Capability

D. High Pressure Coolant Injection (HPCI) System

1. Except as specified in 3.5.D.2 below, the HPCI system 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.

2. From and after the date that the HPCI system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, provided that during such 14 days all of the Automatic Pressure Relief systems, the RCIC system, both of the core spray systems, and the LPCI subsystem and containment cooling mode of the RHR system are operable.

4.0 SURVEILLANCE REQUIREMENTS

High Pressure Core Cooling Capability

D. Surveillance of HPCI System shall be performed as follows:

1. Testing

Item	Frequency
Pump operability	Pursuant to Specification 4.15.B
Valve operability	Pursuant to Specification 4.15.B
Simulated automatic actuation test (testing valve operability)	Each refueling outage

3.0 LIMITING CONDITIONS FOR OPERATION

E. Automatic Pressure Relief System

1. Except as specified in 3.5.E.2 and 3.5.E.3 below, the entire automatic pressure relief system shall be operable whenever the reactor pressure is above 150 psi and irradiated fuel is the reactor vessel, except during reactor vessel hydrostatic or leakage tests.

2. From and after the date that one of the automatic pressure relief system valves is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 14 days unless such valve is sooner made operable, provided that during such 14 days both remaining automatic relief system valves and the HPCI system are operable.

3. From and after the date that more than one of the automatic pressure relief valves are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 12 hours unless repairs are made and provided that during such time the HPCI system is operable.

4. If the requirements of 3.5.E.1-3 cannot be met, an orderly reactor

4.0 SURVEILLANCE REQUIREMENTS

E. Surveillance of the Automatic Pressure Relief System shall be performed as follows:

1. Testing:

Item	Frequency
Valve operability	Each operating cycle
Simulated automatic actuation test	Each operating cycle
ADS Inhibit Switch	Each operating cycle

NOTE: Safety/relief valve operability is verified by cycling the valve and observing a compensating change in turbine bypass valve position.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

shutdown shall be initiated immediately and the reactor pressure shall be reduced to 150 psig within 24 hours thereafter.

F. Reactor Core Isolation Cooling System (RCIC)

1. Except as specified in 3.5.F.2 below, the RCIC system 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. To be considered operable, the RCIC system shall meet the following conditions:

   a. The RCIC shall be capable of delivering 400 gpm into the reactor vessel at 150 psig.

   b. The controls for automatic transfer of the RCIC pump suction from the condensate storage tank to the suppression chamber shall be operable.

   c. The controls for automatic restart on subsequent low reactor level after it has been terminated by a high reactor level signal shall be operable.

4.0 SURVEILLANCE REQUIREMENTS

F. Surveillance of Reactor Core Isolation Cooling System (RCIC)

Surveillance of the RCIC System shall be performed as follows:

1. Testing

Item	Frequency
Pump operability	Pursuant to Specification 4.15.B
Valve operability	Pursuant to Specification 4.15.B
Simulated automatic actuation, transfer of suction to suppression pool, and automatic restart on subsequent low reactor water level	Once/Operating Cycle

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. From and after the date that the RCIC system is made or found to be inoperable for any reason, except automatic transfer of pump suction, reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable. With the controls for automatic transfer of pump suction inoperable, operation for up to 30 days is permissible if the pump suction is aligned to the suppression pool. If these conditions cannot be met, an orderly shutdown shall be initiated and the reactor pressure reduced to 150 psig within 24 hours.	

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
G. Minimum Core and Containment Cooling System Availability 1. When irradiated fuel is in the reactor vessel and reactor coolant temperature is less than 212°F, all low pressure core and containment cooling subsystems may be inoperable provided no work is being done which has the potential for draining the reactor vessel except as allowed by specification 3.5.G.2 below. 2. When irradiated fuel is in the reactor vessel and the vessel head is removed, the suppression chamber may be drained completely and no more than one control rod drive housing or instrument thimble opened at any one time provided that the spent fuel pool gates are open and the fuel pool water level is maintained at a level of greater than or equal to 33 feet.	

Bases Continued:

3.5 The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in reference (1). Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems constitute a 1 out of 3 system; however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 is pursuant to Specification 4.15.B, Inservice Testing, and Specification 4.15.B references ASME Code Section XI which is 3 months. Therefore, an allowable repair period which maintains the basic risk should be less than 45 days and this specification is within this period. Although it is recognized that the information given in reference (1) provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the need for core cooling arise. Based on judgments of the reliability of the remaining systems, i.e., the core spray and LPCI, a 7 day repair period was obtained.

If both core spray subsystems become inoperable, only the LPCI is available for low pressure cooling. Based on the fact that the LPCI is available, a 72 hour repair period was obtained.

Should the loss of one LPCI pump occur, a nearly full complement of core and containment cooling equipment is available. Three LPCI pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, a 7 day repair period is justified. If the LPCI subsystem is not available, at least 2 LPCI pumps must be available to fulfill the containment cooling function. The 72 hour repair period was set on this basis.

---

(1) APED 5736 - Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards - April, 1969, I. M. Jacobs and P. W. Marriott.

Bases Continued 3.5:

C. RHR Service Water

The containment heat removal portion of the RHR system is provided to remove heat energy from the containment in the event of a loss of coolant accident. For the flow specified, the containment longterm pressure is limited to less than 5 psig and, therefore, is more than ample to provide the required heat removal capability. Reference Section 6.2.3.2.3. FSAR. The repair periods specified were arrived at as in 3.5.B above.

The containment cooling subsystem consists of two sets of 2 service water pumps, 1 heat exchanger, and 2 RHR pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability as two of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a 30 day repair period is adequate. Loss of 1 containment cooling subsystem leaves one remaining system to perform the containment cooling function. Based on the fact that when one containment cooling subsystem becomes inoperable only one system remains, a 72 hour repair period was specified.

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. High Pressure Coolant Injection

The high pressure coolant injection system is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of off-site AC power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled and thus no clad damage occurs. Reference Section 6.2.4.3 FSAR.

The HPCI system is backed up by the automatic pressure relief system and either of two core spray systems or the LPCI system. Therefore, when the HPCI system is out of service, the automatic pressure relief and core spray systems and LPCI system are required to be operable. For additional margin, the RCIC system (a non-safeguard system) has been required to be operable during this time, since the RCIC system is capable of supplying significant water makeup to the reactor (400 gpm).

E. Automatic Pressure Relief

The relief valves of the automatic pressure relief subsystem are a backup to the HPCI subsystem. They enable the core spray system or LPCI to provide protection against the small pipe break in the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core sprays or LPCI. Either of the two core spray systems or LPCI provide sufficient flow of coolant to limit fuel clad temperatures to well below clad melt and to assure that core geometry remains intact. Three safety/relief valves are included in the automatic pressure relief system. Of these three, only two are required to provide sufficient capacity for the automatic pressure relief system. See section 4.4 and 6.2.5.3 FSAR.

F. 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 the HPCI system to be operable should the RCIC system be found to be inoperable.

G. Emergency Cooling Availability

The purpose of Specification G 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 cooling subsystems may be out of service. Specification 3.5.G.1 allows all core and containment 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.G.2 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be 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.

H. Recirculation System

Specification 3.5.H.1 is based upon providing assurance that neutron flux limit cycle oscillations, which have a small probability of occurring in the high power/low flow corner of the operating domain, are detected and suppressed. Under certain high power/low flow conditions that could occur during a recirculation pump trip and subsequent Single Loop Operation (SLO) where reverse flow occurs in inactive jet pumps, a hydraulic/reactor kinetic feedback mechanism can be enhanced such that sustained limit cycle oscillations of flow noise with peak to peak levels several times normal values are exhibited. Although large margins to safety limits are maintained when these limit cycle oscillations occur, they are to be monitored for, and suppressed when flux noise exceeds the three time baseline valve by inserting rods and/or increasing coolant flow. The line in Figure 3.5.1 is based on the 80% rod line below which the probability of limit cycle oscillations occurring is negligible.

APRM and/or LPRM oscillations in excess of those specified in Specification 3.5.H.1.e could be an indication that a condition of thermal hydraulic instability exists and that appropriate remedial action should be taken. By restricting core flow to greater than or equal to 39% of rated, which corresponds to the core flow at the 80% rod line with 2 recirculation pumps running at minimum speed, the region of the power/flow map where these oscillations are most likely to occur is avoided (Ref. 1).

Above 45% of rated core flow in Single Loop Operation there is the potential to set up high flow-induced noise in the core. Thus, surveillance of core plate ΔP noise is required in this region of the power/flow map to alert the operators to take appropriate remedial action if such a condition exists.

Specification 3.6.A.2 governs the restart of the pump in an idle recirculation loop. Adherence to this specification limits the probability of excessive flux transients and/or thermal stresses.

I. Deleted

References: 1. General Electric Service Information Letter No. 380, Rev. 1, February 10, 1984

Bases 4.5:

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 450 psig; thus, during operation even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel, which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The pumps and motor-operated valves are tested pursuant to Specification 4.15.B, Inservice Testing, to assure their operability. The combination of a simulated automatic actuation test each refueling cycle and Section XI testing of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by periodic testing of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the periodic pump and valve operability checks assure the reliability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
c. Except for inerting and deinerting operations permitted in (b) above, all containment purging and venting above cold shutdown shall be via a 2-inch purge and vent valve bypass line and the Standby Gas Treatment System. Inerting and deinerting operations may be via the 18-inch purge and vent valves (equipped with 40-degree limit stops) aligned to the Reactor Building plenum and vent.	
6. If the specifications of 3.7.A cannot be met, the reactor shall be placed in a cold shutdown condition within 24 hours.	
B. Standby Gas Treatment System	B. Standby Gas Treatment System
1. Two separate and independent standby gas treatment system circuits shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1.(a) and (b).	1. At least once per month, initiate from the control room 3500 cfm (±10%) flow through both circuits of the standby gas treatment system.
a. After one of the standby gas treatment system circuits is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment system are operable. Within 36 hours following the 7 days, the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.2.(a) through (d).	

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

b. If both standby gas treatment system circuits are not operable, within 36 hours the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.2.(a) through (d).

2. Performance Requirements

a. Periodic Requirements

(1) The results of the in-place DOP tests at 3500 cfm (±10%) on HEPA filters shall show ≤1% DOP penetration.

(2) The results of in-place halogenated hydrocarbon tests at 3500 cfm (±10%) on charcoal banks shall show ≤1% penetration.

(3) The results of laboratory carbon sample analysis shall show ≥90% methyl iodine removal efficiency when tested at 130°C, 95% R.H.

4.0 SURVEILLANCE REQUIREMENTS

2. Performance Requirement Tests

a. At least once per 720 hours of system operation; or once per operating cycle, but not to exceed 18 months, whichever occurs first; or following painting, fire, or chemical release in any ventilation zone communicating with the system while the system is operating that could contaminate the HEPA filters or charcoal absorbers, perform the following:

(1) In-place DOP test the HEPA filter banks.

(2) In-place test the charcoal adsorber banks with halogenated hydrocarbon tracer.

(3) Remove one carbon test canister from the charcoal adsorber. Subject this sample to a laboratory analysis to verify methyl iodine removal efficiency.

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 Automatic Isolation Valves

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.

D. Primary Containment Automatic Isolation Valves

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 pursuant to Specification 4.15.B. Main Steam isolation valves shall be tested (one at a time) with the reactor power less than 75% of rated.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	d. At least once per week the main steam-line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.
2. In the event any Primary Containment automatic isolation valve becomes inoperable, reactor operation in the run mode may continue provided at least one valve in each line having an inoperable valve is closed.	2. Whenever a Primary Containment automatic isolation valve is inoperable, the position of at least one fully closed valve in each line having an inoperable valve shall be recorded daily.
3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, initiate normal orderly shutdown and have reactor in the cold shutdown condition within 24 hours.	3. Deleted
	4. The seat seals of the drywell and suppression chamber 18-inch purge and vent valves shall be replaced at least once every five years.

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service providing both the emergency diesel generators are operable.

2. Reserve Transformers

If offsite power sources are made or found to be inoperable for any reason such that Specification 3.9.A.1 is not met, reactor operation is permissible only during the succeeding 72 hours unless such offsite sources are sooner made operable, provided that either 1R or 2R Transformer is operable.

3. Standby Diesel Generators

a. From and after the date that one of the diesel generators is made or found to be inoperable, reactor operation is permissible only during the succeeding 7 days provided that the operable diesel generator is demonstrated to be operable within 24 hours.

This test is required to be completed regardless of when the inoperable diesel generator is restored to operability.

The operability of the other diesel generator need not be demonstrated if the diesel generator inoperability was due to preplanned preventative maintenance or testing.

b. If both diesel generators become inoperable during power operation, the reactor shall be placed in the cold shutdown condition.

B. 3. Standby Diesel Generators

a. Each diesel generator shall be manually started, loaded and operated at approximately rated load for at least 60 minutes once every month to demonstrate operational readiness.

b. During the monthly generator test, the diesel starting air compressor shall be checked for operation and their ability to recharge air receivers.

Bases 3.9:

The general objective is to assure an adequate supply of power with at least one active and one standby source of power available for operation of equipment required for a safe plant shutdown, to maintain the plant in a safe shutdown condition, and to operate the required engineered safeguards equipment following an accident.

AC for shutdown requirements and operation of engineered safeguards equipment can be provided by either of the two standby sources of power (the diesel generators) or any of the three active sources of power (No. 1R, No. 2R, or No. 1AR transformers). Refer to Section 8 of the USAR.

To provide for maintenance and repair of equipment and still have redundancy of power sources, the requirement of one active and one standby source of power was established. The plant's main generator is not given credit as a source since it is not available during shutdown.

The plant 250 V dc power is supplied by two batteries. Most station 250 V loads are supplied by the original station 250 V battery. A new 250 V battery has been installed for HPCI loads and may be used for other station loads in the future. Each battery is maintained fully charged by two associated chargers which also supply the normal dc requirements with the batteries as a standby source during emergency conditions. The plant 125 V dc power is normally supplied by two batteries, each with an associated charger. Backup chargers are available.

The minimum diesel fuel supply of 32,500 gallons will supply one diesel generator for a minimum of seven days of full load operation. The diesel fuel oil requirement of 32,500 gallons ensures that one emergency diesel generator can run for seven days at full load (2500 KW). The amount of fuel oil necessary to run one emergency diesel generator for seven days is 31,248 gallons. The difference between these two volumes allows for instrument inaccuracy, tank volume uncertainties, and the location of the suction pipe. Additional diesel fuel can normally be obtained within a few hours. Maintaining at least seven days supply is therefore conservative

In the normal mode of operation, power is available from the off-site sources. One diesel may be allowed out of service based on the availability of off-site power provided that the remaining diesel generator is demonstrated to be operable within 24 hours. This test is required even if the inoperable diesel is restored to operability within 24 hours. Thus, though one diesel generator is temporarily out of service, the off-site sources are available, as well as the remaining diesel generator. Based on a monthly testing period (Specification 4.9), the seven day repair period is justified. (1)

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(1) "Reliability of Engineered Safety Features as a Function of Testing Frequency", I.M. Jacobs, Nuclear Safety, Volume 9, No. 4, July - August 1968.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. It is permissible to have one of the pumps required by Specification 3.13.B.1.a inoperable provided that the redundant pumps are operable. Restore the inoperable pump to operable status within seven days or provide a special report to the Commission within 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in the Fire Suppression Water system. 3. With the fire suppression water system otherwise inoperable: a. Establish a backup fire Suppression Water System within 24 hours. b. Provide a special report to the Commission within 14 days outlining the actions taken and the plans and schedule for restoring the system to operable status.	e. Every three months verify that a sample of fuel from the diesel oil storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water, and sediment. f. Every 18 months subject the diesel-driven fire pump engine to an inspection in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service. g. A simulated automatic actuation of each fire pump and the screen wash/fire pump, including verification of pump capability, shall be conducted every 18 months. h. The yard main and the reactor building and turbine building headers shall be flushed every 12 months. i. System flow tests shall be performed every three years. j. Valves in flow paths supplying fire suppression water to safety related structures, systems, and component shall be cycled every 12 months.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	k. Each valve (manual, power operated, or automatic) in the flow path that is not electrically supervised, locked, sealed or otherwise secured in position, shall be verified to be in its correct position every month.
C. <u>Hose Stations</u>	C. <u>Hose Stations</u>
1. Whenever equipment protected by hose stations in the following areas is required to be operable, the hose station(s) protecting equipment required to be operable in those areas shall be operable: a. Diesel generator rooms b. Safety related areas of the turbine building c. Safety related areas of the screen-house d. Reactor building e. Reactor building addition f. Safety related areas of the Administration building	1. The hose stations specified in 3.13.C.1 shall be demonstrated operable as follows: a. Each month a visual inspection shall be conducted to assure all equipment is available. b. Every 18 months the hose shall be removed for inspection and re-racking and all gaskets in the couplings shall be inspected and replaced if necessary. c. Every 3 years each hose station valve shall be partially opened to verify valve operability and no flow blockage. d. Every 3 years each hose shall be hydrostatically tested at a pressure at least 50 psig greater than the maximum pressure available at any hose station.

4.13 BASES:

Fire detectors are tested in accordance with the manufacturer's recommendations. All tests and inspections are performed by the plant staff. Every six months each detector is functionally tested. Combustion generated smoke is not used in these tests. Alarm circuits are functionally checked every six months. In addition, all circuitry is automatically supervised for open wiring and ground faults.

Fire pumps are tested each month to verify operability. Test starting of the screen wash/fire pump is not required since it is normally in service. Each fire pump is manually started and operated for at least 15 minutes with pump flow directed through the recirculation test line. Every 18 months the operability of the automatic actuation logic for the fire pumps and the screen wash/fire pump is verified and the performance of each pump is verified to meet system requirements. The specified flush and valve checks provide assurance that the piping system is capable of supplying fire suppression water to all safety related areas.

A system flow test is specified every three years. This test verifies the hydraulic performance of the fire suppression fire water header system. The testing will be performed using Section II, Chapter 5 of the Fire Protection Handbook, 14th Edition, as a procedural guide. This test is generally performed in conjunction with a visit from insurance company inspectors.

Hose stations and yard hydrant hose houses are inspected monthly to verify that all required equipment is in place. Gaskets in hose couplings are inspected periodically and the hose is pressure tested. Pressure testing of outdoor hose is conducted more frequently than indoor hose because of the less favorable storage conditions. Operability of hose station isolation valves is verified every three years by partially opening each valve to verify flow. All of these tests provide a high degree of assurance that each hose station and yard hydrant hose house will perform satisfactorily after periods of standby service.

Simulated automatic actuation tests are conducted each 18 months to confirm the operability of the sprinkler and Halon systems. These tests consist of verification that all valves, dampers (Halon system only), alarms, and flow paths are functional.

Plant fire barrier walls are provided with seals for pipes and cables where necessary. Where such seals are installed, they must be maintained intact to perform their function. Visual inspection of each installed seal is required every 18 months and after seal repair. A visual inspection following repair of a seal is sufficient to assure that seal integrity will be within acceptable limits.

Bases 3.15 and 4.15

The inservice inspection program for the Monticello plant conforms to the requirements of 10 CFR 50, Section 50.55a(g). Where practical, the inspection of components classified into NRC Quality Groups A, B, and C conforms to the requirements of ASME Code Class 1, 2, and 3 components, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code. A program of inservice testing of Quality Group A, B, and C pumps and valves is also in effect at the Monticello plant, that conforms to the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code or where alternate testing is justified in accordance with Generic Letter 89-04. If a Code required inspection is impractical for the Monticello facility, a request for a deviation from that requirement is submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i).

Deviations which are needed from the procedures prescribed in Section XI of the ASME Code and applicable Addenda will be reported to the Commission prior to the beginning of each 10-year inspection period if they are known to be required at that time. Deviations which are identified during the course of inspection will be reported quarterly throughout the inspection period.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	2. Welds in austenitic stainless steel piping four inches or larger in diameter containing reactor coolant at a temperature above 200 degrees F during power operation, including reactor vessel attachments and appurtenances, shall be included in an augmented inspection program meeting the requirements of Generic Letter 88-01. B. Inservice Testing 1. Inservice Testing of Quality Group A, B, and C pumps and valves shall be performed in accordance with the requirements for ASME Code Class 1, 2 and 3 pumps and valves, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g) except where relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55(a)(g)(6)(i), or where alternate testing is justified in accordance with Generic Letter 89-04. 2. Nothing in the ASME Boiler and Pressure Vessel code shall be construed to supersede the requirements of any Technical Specification.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 77 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 December 13, 1990, the Northern States Power Company (the licensee) requested an amendment to the Technical Specifications appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed amendment would revise pump and valve surveillance testing requirements to be consistent with ASME Section XI code requirements. The proposed amendment would also eliminate requirements for immediate and more frequent surveillance testing of redundant equipment when equipment is found or made inoperable.

A discussion of each requested change and the NRC staff's evaluation and findings relative to each are addressed in Section 2.0 of this Safety Evaluation Report.

2.0 DISCUSSION AND EVALUATION

Addition of General Requirement for Inservice Testing: A new subsection "Inservice Testing" would be added to Section 4.15 on Page 229ff. This change would provide separate paragraphs referencing (a) Inservice Inspection (ISI) requirements which apply to pressure vessels and (b) piping and Inservice Testing (IST) requirements which apply to pumps and valves. The new paragraph is similar to the existing ISI requirement but expands and clarifies the IST requirements to incorporate staff-approved features of Generic Letter 89-04 and acknowledges that ASME Code requirements are not to be construed as superseding any Technical Specification.

The addition of the proposed new subparagraph is editorial and clarificational, providing a distinction between ISI and IST. It does not, in itself, modify any operability or testing requirements but serves as a cross-reference applicable to systems, structures and equipment as invoked elsewhere in the Technical Specifications. The proposed change is therefore acceptable.

Standby Liquid Control System: Technical Specifications 3/4.4, Pages 93, 94, 95, and 96 would be revised to (1) renumber and retitle paragraphs and make other editorial corrections, and (2) eliminate a requirement that, upon determination that a component is inoperable, additional (immediate and daily thereafter) testing of redundant components and support equipment.

Item (1) would have no effect on safety and is acceptable on that basis. Item (2) is acceptable on the basis that the normal surveillance test interval and the 7-day allowable out-of-service time (AOT) provide adequate assurance of system reliability and that excessive testing has the potential of reducing degrading equipment and reducing system reliability. Monticello's AOT and test interval are consistent with those prescribed by NUREG-0123 Revision 2 "Standard Technical Specifications for General Electric Boiling Water Reactors" (NUREG-0123). The proposed changes are therefore acceptable.

Core Spray System: Technical Specifications 3/4.5.A, (Pages 101, 102 and 103), which specify core spray system operability and test requirements, would be changed to (1) include a cross-reference to the new IST general requirement (see above) invoking ASME Section XI requirements, (2) delete statements pertaining to post-maintenance testing, (3) reduce the AOTs to a value consistent with NUREG-0123 guidance, (4) eliminate a requirement that, upon determination that one of the two subsystems is inoperable, the redundant core spray subsystem be demonstrated as operable immediately and daily thereafter and the low pressure coolant injection (LPCI) mode of RHR and the diesel generators be demonstrated operable immediately, and (5) eliminate a requirement that, upon determination that both core spray subsystems are inoperable, the LPCI and LPCI support systems be demonstrated operable immediately and daily thereafter.

Item (1) would permit the maximum test interval for certain valves to be increased from monthly to every three months. This is consistent with ASME code practice and is acceptable on that basis. Item (2) eliminates a redundant statement, since post-maintenance testing is specified by ASME Section XI and is acceptable on that basis. Item (3) reduces the period of time permitted for operating with one core spray subsystem inoperable from 15 days to 7 days, and for operating with both subsystems inoperable from seven days to 72 hours. This is consistent with NUREG-0123 guidance and serves to maintain an acceptable level of confidence in system availability in view of the other changes. Items (4) and (5) would eliminate additional (immediate and daily thereafter) testing currently required only when related equipment is known to be inoperable. This is based on the position that the normal surveillance test schedule provides a high level of confidence in system operability, and that excessive testing has the potential to degrade equipment performance. The proposed changes are acceptable based on consistency with ASME Section XI and NUREG-0123 Revision 2 (Standard Technical Specifications).

Low Pressure Coolant Injection System: Technical Specification 3/4.5.B Pages 104 and 105 would be changed in a manner similar to the above core spray changes. The proposed changes include (1) reducing the AOT for an inoperable LPCI injection valve from 30 days to 7 days, (2) reducing the AOT for two inoperable LPCI injection valves from 7 days to 72 hours (3) reducing the AOT for an inoperable drywell spray loop from 30 days to 7 days, (4) elimination of text which is redundant to ASME Section XI pertaining to post-maintenance testing, (5) addition of a cross-reference to the new Section 4.15.B paragraph discussed above, and (6) elimination of additional (immediate and daily thereafter) testing currently required only when related equipment is known to be inoperable. The AOT applicable when two LPCI pumps are inoperable is currently seven days and would not be changed. The corresponding AOT prescribed by NUREG-0123 is 72 hours. The longer AOT for Monticello is acceptable based on the provision of an existing additional limitation that both core spray subsystems be operable in this condition. Otherwise, these changes are similar to those for the core spray system as described above and are acceptable on the same basis.

Containment Heat Removal System: The containment heat removal function is provided by the residual heat removal (RHR) system using coolant provided by the residual heat removal service water (RHRSW) system. Technical Specification 3/4.5.C, Pages 106 and 107, which specifies the Limiting Conditions for Operation and the Surveillance Requirements for the RHRSW system would be changed to (1) provide a cross-reference to the new Section 4.15.B paragraph discussed above, (2) eliminate text which is redundant to ASME Section XI pertaining to post-maintenance testing, and (3) eliminate additional (immediate and daily thereafter) testing currently required only when related equipment is known to be inoperable. The current AOT applicable when one RHRSW subsystem is inoperable is seven days. The licensee's application indicates a desire to retain this AOT. However, it is the staff position that a 7-day AOT is appropriate when one pump in either or both subsystems is inoperable, but when two pumps in the same subsystem are inoperable a 72-hour AOT should be applied. On December 28, 1990, the NRC staff contacted the licensee's staff (T. Parker and M. Vik) and discussed this issue. The licensee's representatives readily accepted the staff position and verbally amended the application accordingly.

Items (1) and (2) are editorial changes having no safety significance. Item (3), together with the reduced AOT applicable for the condition when two RHRSW pumps in the same ESF division are inoperable, will provide greater assurance that core and containment cooling systems will be available if required. The applicant's proposed RHRSW technical specification revisions (as amended in the telecon) are acceptable based on consistency with NUREG-0123 guidance.

High Pressure Core Cooling Capability: Technical Specification 3/4.5.D, Page 108 would be revised to (1) invoke by cross-reference the new specification 4.15.B discussed above, (2) eliminate a requirement for immediate operability tests of the core spray systems, reactor coolant injection (RCIC) system and LPCI subsystems when the high pressure core injection system (HPCI) is determined to be inoperable, and (3) revise the AOT for the HPCI system from seven days to 14 days.

Item (1) will result in certain valve ISI/operability test intervals being changed from monthly to every three months. This is consistent with the requirements of ASME Section XI and is acceptable. Item (2) is similar to changes proposed for the core spray system described above and is similarly acceptable based on the position that the normal test intervals provide a high level of confidence in system operability, and that excessive testing has the potential to degrade equipment performance. Item (3) allows the licensee a longer AOT for periods of HPCI inoperability. The 14-day AOT is acceptable based on consistency with NUREG-0123. A 14-day AOT represents the staff position for HPCI AOTs applicable to facilities in which the total operable ECCS capability is composed of two trains of two redundant core spray systems, two redundant (or loop-select type) LPCI systems, and an automatic depressurization system (ADS), with the additional provision of an independent turbine-driven RCIC system.

Automatic Pressure Relief System: The proposed amendment would revise Section 3/4.5.E relating to the ADS. The AOT for inoperability of one valve would be changed from seven to 14 days. The AOT for the case of more than one inoperable valves would be changed from 24 hours to 12 hours. A requirement to immediately, and weekly thereafter, demonstrate HPCI operability would be deleted.

These changes are consistent with NUREG-0123. The proposed AOTs and elimination of additional HPCI testing are acceptable on a similar basis as those changes proposed to related systems as described above.

Reactor Core Isolation Cooling System: The proposed amendment would revise 3/4.5.F, Pages 111, 111a (to be eliminated) and 112), relating to the RCIC system. Proposed changes include (1) pump operability and flow rate test requirements would be changed to cross-reference the new 4.15.B described above, (2) the AOT for the RCIC system would be changed from 15 days to (a more restrictive) 14 days, (3) a statement that the AOT applies only if the HPCI system is operable would be deleted and (4) a requirement to immediately, and daily thereafter, demonstrate HPCI system operability would also be deleted.

Item (1) is similar to changes proposed for other systems described above and is acceptable on similar basis. Item (2) brings the AOT into conformance with current staff guidance (NUREG-0123) and is also acceptable. Item (3) deletes superfluous wording and is also acceptable (If both the HPCI and RCIC systems are inoperable, Technical Specification 3.5.D applies). Item (4) is similar to changes proposed for other systems described above and is acceptable on similar basis.

Minimum Requirements for Core and Containment Cooling Capability: Section 3/4.5.G, Pages 112 and 113 (with 112 to become part of 3/4.5.F) would be revised to (1) delete a condition that the 7-day AOT for an inoperable standby diesel generator applies only if all the low pressure core and containment cooling systems connected to the operable standby diesel generator are operable, (2) delete a requirement that no combination of inoperable components in the core and containment cooling systems shall defeat the capability of remaining components to fulfill the core and containment cooling function, and (3) delete a requirement to, upon determining that a standby diesel generator is inoperable, verify operability immediately and daily thereafter, of the other standby diesel generator and all of its associated low pressure core and containment cooling systems. This statement conflicts with changes proposed to Section 3/4.9.3.a.

Item (1) is acceptable based on the fact that the requirements contained in the text to be deleted are already encompassed by the individual equipment operability requirements invoked by the definition of "operability" contained in Section 1.0.L of the Technical Specifications. Under the terms of 1.0.L, if a standby diesel generator power source is inoperable, no associated system, component or train is considered operable unless the corresponding redundant system, component or train served by the operable standby diesel power source is operable. Thus, should a standby diesel power source be determined to be inoperable, while at the same time, there is an inoperable low pressure core or containment cooling system, component or train in the division served by the operable standby diesel generator, the entire core or containment cooling function provided by both divisions of the low pressure core or containment cooling systems, components or trains is considered inoperable and an equal or more restrictive system, component or train AOT applies (typically 72-hours). Item (2) is also acceptable, on the basis that the definition of "operability" requires that necessary support systems be operable in order for a core or containment cooling system to be considered operable. Item (3) is acceptable based on the need to maintain consistency with other Technical Specifications. See "Standby Diesel Generators" below.

Standby Gas Treatment System: Section 3/4.7.B, Pages 166 and 167, of the facility Technical Specifications would be revised to (1) delete the requirement to, upon determining that one train of the SGTS is inoperable, test within two hours, and daily thereafter, the other SGTS train, and (2) delete a statement (4.7.1.b) that, in event both trains of SGTS are inoperable for seven days, within 36 hours place the facility in a condition for which secondary containment integrity is not required. Item (1) is acceptable consistent with the previously stated position that normal surveillance test intervals provide a high level of confidence in systems operability, and that additional testing has the potential to degrade equipment performance. Item (2) is acceptable on the basis that it eliminates a redundancy. The AOT specified by 4.7.1.b, to be deleted, is redundant to statements specified in 3.7.B.1.

Primary Containment Automatic Isolation Valves: Section 3/4.7.D would be revised to (1) replace requirements specifying quarterly full-stroke testing of normally open power operated isolation valves with requirements that these valves be tested pursuant to the new 4.15.B, and (2) eliminate the specification for post-maintenance testing. These changes are acceptable on the basis that the Section 3/4.7.D requirements are redundant to the ASME Section XI code requirements invoked by 4.15.B. Section 4.15.B specifies appropriate operability and post-maintenance testing requirements.

Standby Diesel Generators: Section 3/4.9.3.a, Page 201, presently reads:

> From and after the date that one of the diesel generators is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided that during such seven days the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter.

This would be revised to read:

> From and after the date that one of the diesel generators is made or found to be inoperable, reactor operation is permissible only during the succeeding 7 days, provided that the operable diesel generator is demonstrated to be operable within 24 hrs.
>
> This test is required to be completed regardless of when the inoperable diesel generator is restored to operability. The operability of the other diesel generator need not be demonstrated if the diesel generator inoperability was due to preplanned preventative maintenance or testing.

These changes would (1) delete the requirement for additional testing of a diesel generator solely because the other diesel generator has been deliberately made inoperable for the purpose of planned maintenance or testing, (2) allow, in event a diesel generator is found inoperable, a 24-hour period before performing the required testing the other diesel generator. Surveillance test intervals and AOTs would not be changed.

Item (1) is acceptable on the basis that the current surveillance test intervals provide a high level of confidence in systems operability, and that the Monticello diesel generators have demonstrated high reliability. Item (2) is acceptable on the basis of staff policy which has adopted the concept that when a system, structure or component has been intentionally made inoperable for purposes of preplanned maintenance or testing activities, there is no basis to suspect that the redundant component is thereby made less reliable. It is noted that the 7-day AOT (to be retained) for an inoperable diesel generator is inconsistent NUREG-0123 guidance (72-hours). However, based on diesel generator reliability information provided by the licensee in response to the staff's previous Generic Letter 84-15 diesel generator reliability review effort, there is insufficient basis for upgrading to the requirements of current policy (72-hour AOT and staggered test intervals) as a condition for acceptance of the proposed changes. The licensee's proposed changes are acceptable.

Section 4.9.B.3.a, Page 201, would also be revised. This section presently specifies:

> Each diesel generator shall be manually started and loaded once every month to demonstrate operational readiness. The test shall continue until both the diesel engine and the generator are at equilibrium conditions of temperature while full load output is maintained.

This would be replaced by:

> Each diesel generator shall be manually started, loaded and operated at approximately rated load for at least 60 minutes once every month to demonstrate operational readiness.

Changing the monthly operability test duration from the period of time its takes for conditions to stabilize to 60 minutes is consistent with NUREG-0123 guidance, provides an adequate time period for conditions to stabilize and will provide a more clearly defined test termination criterion. This change is therefore acceptable.

Fire Pumps: The licensee's application requests changes to Section 3.13.B, Pages 225 and 226. The proposed changes would eliminate the requirement for daily testing of the remaining two fire suppression pumps, when one of the pumps is determined to be inoperable. This is acceptable based on the previously stated position that normal surveillance test intervals provide a high level of confidence in systems operability, and that additional testing has the potential to degrade equipment performance.

Table of Contents and Bases: The licensee's application includes proposed changes to (a) the Table of Contents and (b) Bases included in the Technical

Specifications. These reflect the "Limiting Conditions for Operation" and "Surveillance Requirements" changes proposed above and are acceptable.

Additional Staff Comment: In conducting its review of the above proposed changes, the staff has used NUREG-0123 extensively as guidance. This is consistent with Section 16.0 of the Standard Review Plan. Where Monticello has unusual design features that would invalidate the use of NUREG-0123 for evaluating proposed changes to its Technical Specifications the differences have been noted and evaluated accordingly.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes an inspection or surveillance requirement. We have 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 published a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

4.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: W. Long

Date: February 15, 1991