April 18, 1989

Docket No. 50-263

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

Dear Mr. Musolf:

SUBJECT: AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. DPR-22: MISCELLANEOUS TECHNICAL SPECIFICATION IMPROVEMENTS (TAC NO. 61661)

The Commission has issued the enclosed Amendment No. 63 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 5, 1986.

The amendment revises the Plant TSs to include changes resulting from a detailed review of the TSs conducted following the 1985 refueling and recirculation piping replacement outage. Several of the changes are administrative in nature and/or clarify the interpretation of the existing TSs. The specific changes are delineated and discussed in the enclosed Safety Evaluation.

Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/
John J. Stefano, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. Amendment No. 63 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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See next page

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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. 63
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 May 5, 1986, 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. 63, 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

John J. Stefano

John J. Stefano, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 18, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 63

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

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

REMOVE	INSERT
v	v
1	1
-	17
28	28
29	29
30	30
32	32
33	33
41	41
42	42
43	-
44	-
60	60
61	61
62	62
63	63
63a	63a
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73	-
74	-
75	-
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-	82a
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INTRODUCTION

These Technical Specifications are prepared in accordance with the requirements of 10CFR50.36 and apply to the Monticello Nuclear Generating Plant, Unit No. 1. The bases for these Specifications are included for information and understandability purposes.

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the Specifications may be achieved.

A. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud with the vessel head removed and fuel in the reactor vessel. (Normal operating functions such as control rod movement using the normal drive mechanism, tip scans, SRM and IRM detector movements, etc., are not to be considered core alterations.)

B. Hot Standby - Hot Standby means operation with the reactor critical in the startup mode at a power level just sufficient to maintain reactor pressure and temperature.

C. Fire Suppression Water System - The fire suppression water system consists of: water sources; pumps; and distribution piping with associated sectionalizing isolation valves. Such valves include yard hydrant valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

D. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

E. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the primary sensor to verify proper instrument channel response, alarm, and/or initiating action.

Bases Continued:

Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position and the associated APRM is not downscale. This switch occurs when reactor pressure is greater than 850 psig.

The operator will set the APRM neutron flux trip setting no greater than that stated in Specification 2.3.A.1. However, the actual setpoint can be as much as 3% greater than that stated in Specification 2.3.A.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 39.

B. Deleted

TABLE 3.1.1
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Limiting Trip Settings	Modes in which function must be Operable or Operating** Refuel (3)	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels per Trip System (1)	Required Condition*
1. Mode switch in Shutdown		X	X	X	1	1	A
2. Manual Scram		X	X	X	1	1	A
3. Neutron Flux IRM (See Note 2) a. High-High b. Inoperative	≤ 120/125 of full scale	X	X		4	3	A
4. Flow Referenced Neutron Flux APRM (See Note 5) a. High-High b. Inoperative	See Specifications 2.3A.1			X	3	2	A or B
5. High Reactor Pressure (See Note 9)	≤ 1075 psig	X	X(f)	X(f)	2	2	A
6. High Drywell Pressure (See Note 4)	≤ 2 psig	X	X(e,f)	X(e,f)	2	2	A
7. Reactor Low Water Level	≥ 7 in.(6)	X	X(f)	X(f)	2	2	A
8. Scram Discharge Volume High Level							
a. East	≤ 56 gal.(8)	X(a)	X(f)	X(f)	2	2	A
b. West	≤ 56 gal.(8)	X(a)	X(f)	X(f)	2	2	A
9. Turbine Condenser Low Vacuum	≥ 23 in. Hg	X(b)	X(b,f)	X(f)	2	2	A or C

TABLE 3.1.1 - CONTINUED

Trip Function	Limiting Trip Settings	Modes in which function must be Operable or Operating** Refuel(3)	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. Operable or Operating Instrument Channels Per Trip System(1)	Required Conditions*
10. Main Steamline High Radiation (See Note 9)	≤ 10 X Normal background at rated power	X	X(f)	X(f)	2	2	A
11. Main Steamline Isolation Valve Closure	≤ 10% Valve Closure	X(b)	X(b)	X	8	8	A or C
12. Turbine Control Valve Fast Closure	(See Note 7)			X(d,f)	2	2	D
13. Turbine Stop Valve Closure	≤ 10% Valve Closure			X(d)	4	4	D

NOTES:

1. There shall be two operable or tripped trip systems for each function.

2. For an IRM channel to be considered operable, its detector shall be fully inserted.

3. In the refueling mode with the reactor subcritical and reactor water temperature less than 212°F, only the following trip functions need to be operable: (a) Mode Switch in Shutdown, (b) Manual Scram, (c) High Flux IRM, (d) Scram Discharge Volume High Level.

4. Not required to be operable when primary containment integrity is not required.

5. To be considered operable, an APRM must have at least 2 LPRM inputs per level and at least a total of 14 LPRM inputs, except that channels 1, 2, 5, and 6 may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

Table 3.1.1 - Continued

6. Seven inches on the water level instrumentation is 10'6" above the top of the active fuel at rated power.

7. Trips upon loss of oil pressure to the acceleration relay.

8. Limited trip setting refers to the volume of water in the discharge volume receiver tank and does not include the volume in the lines to the level switches.

9. High reactor pressure and main steam line high radiation are not required to be operable when the reactor vessel head is unbolted.

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

A. All operable control rods fully inserted within 8 hours.

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

C. Reactor in Startup or Refuel mode and pressure below 600 psig.

D. Reactor power less than 45% (751.5 MWt.).

** Allowable Bypass Conditions

It is permissible to bypass:

a. The scram discharge volume High Water Level scram function in the refuel mode to allow reactor protection system reset. A rod block shall be applied while the bypass is in effect.

b. The Low Condenser vacuum and MSIV closure scram functions in the Refuel and Startup modes if reactor pressure is below 600 psig.

c. Deleted.

d. The turbine stop valve closure and fast control valve closure scram functions when the reactor thermal power is $\leq$ 45%(751.5 MWt).

TABLE 4.1.1

SCRAM INSTRUMENT FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENTATION AND CONTROL CIRCUITS

INSTRUMENTATION CHANNEL	GROUP*	FUNCTIONAL TEST	MINIMUM FREQUENCY (4)
High Reactor Pressure	A	Trip Channel and Alarm	Once each month
High Drywell Pressure	A	Trip Channel and Alarm	Once each month
Low Reactor Water Level (2)	A	Trip Channel and Alarm	Once each month
High Water Level in Scram Discharge	A	Trip Channel and Alarm	Once each month
Condenser Low Vac	A	Trip Channel and Alarm	Once each month
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once each month
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Once each month
Manual Scram	A	Trip Channel and Alarm	Once each month
Turbine Control Valve Fast Closure	A	Trip Channel and Alarm	Once each month
APRM/Flow Reference (5)	B	Trip Output Relays	Once each week
IRM (5)	C	Trip Channel and Alarm	Note 3
High Steam Line Rad. (5)	B	Trip Channel and Alarm	Once each week
Mode Switch in Shutdown	C	Place mode switch in shutdown	Each refueling outage

TABLE 4.1.1 (Continued)

Note 1: Deleted.

Note 2: A sensor check shall be performed on low reactor water level once per day and on high steam line radiation once per shift.

Note 3: Perform functional test prior to every startup and normal shutdown.

Note 4: Functional tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

Note 5: A functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response, alarm, and/or initiating action.

*GROUPS

A. On-Off sensors that provide a scram function.
B. Analog devices coupled with bi-stable trips that provide a scram function.
C. Devices which only serve a useful function during some restricted mode of operation, such as startup or shutdown, or for which the only practical test is one that can be performed at shutdown.

Bases:

4.0 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operations are met and will be performed during the periods when the Limiting Conditions for Operation are applicable.

A tolerance for performing surveillance activities beyond the nominal interval is provided to allow operational flexibility because of scheduling and performance considerations. The plant uses a fixed surveillance program that prevents repetitive addition of the allowable 25% extension. Each surveillance test is completed within plus or minus 25% of each scheduled fixed date. Scheduled dates are based on dividing each calendar year into four 13-week "surveillance" quarters consisting of 3 4-week "surveillance" months and one "catch up" week. This method of scheduling permits certain tests always to be scheduled on certain days of the week.

The specification ensures that surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into a plant condition for which the Limiting Condition for Operation is applicable. Under the terms of this specification, for example, during-initial plant startup or following extended plant outage, the surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment to Operable status.

4.1 The 13 scram sensor channels listed in Table 4.1.1 are divided into three groups (A., B., and C.) and are defined on Table 4.1.1.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. The probability of success is primarily a function of the sensor failure rate and the test interval. A one month test interval is specified for group (A) sensors. This is in keeping with good operating practice, and exceeds the design goal for the logic configuration utilized in the Reactor Protection System.

4.1 Group (B) devices utilize an analog sensor followed by an amplifier and a bi-stable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An "as-is" failure is one that "sticks" mid-scale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. The test frequency of once per week has developed principally on the basis of past practice and good judgment, and nothing has developed to indicate that the frequency should change.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided in two groups as defined on Table 4.1.2.

Experience with passive type instruments indicates that a yearly calibration is adequate. Where possible, however, quarterly calibration is performed. For those devices which employ amplifiers, etc., drift specifications call for a drift to be less than 0.5%/month; i.e., in the period of a month a drift of 0.5% would occur and thus provide for adequate margin. For the APRM system, drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every three days. Calibration on this frequency assures plant operation at or below thermal limits.

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Table 3.2.5
Instrumentation that Initiates a Recirculation Pump Trip
and Alternate Rod Injection

Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (1)	Total No. of Instrument Channels per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Conditions*
1. High Reactor Dome Pressure	≤1150 psig	2	2	2	A
2. Low-Low Reactor Water Level	≥6'6" above the top of the active fuel	2	2	2	A

NOTE:

1. When one of the two trip systems is made or found to be inoperable, restore the inoperable trip svstem to operable status within 14 days or place the plant in the specified required condition within the next eight hours. When both trip sytems are inoperable, place the plant in the specified required condition within eight hours unless at least one trip system is sooner made operable.

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

A. Reactor in Startup, Refuel, or Shutdown Mode.

Table 4.2.1
Minimum Test and Calibration Frequency For Core Cooling
Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
ECCS INSTRUMENTATION			
1. Reactor Low-Low Water Level	Once/month	Once/3 months	Once/Shift
2. Drywell High Pressure	Once/month	Once/3 months	None
3. Reactor Low Pressure (Pump Start)	Once/month	Once/3 months	None
4. Reactor Low Pressure (Valve Permissive)	Once/month	Once/3 months	None
5. Undervoltage Emergency Bus	Refueling Outage	Refueling Outage	None
6. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	Once/month	Once/3 months	None
7. Loss of Auxiliary Power	Refueling Outage	Refueling Outage	None
8. Condensate Storage Tank Level	Refueling Outage	Refueling Outage	None
9. Reactor High Water Level	Once/month	Once/3 months	Once/day
ROD BLOCKS			
1. APRM Downscale	Once/month (Note 5)	Once/3 months	None
2. APRM Flow Variable	Once/month (Note 5)	Once/3 months	None
3. IRM Upscale	Notes (2,5)	Note 2	Note 2
4. IRM Downscale	Notes (2,5)	Note 2	Note 2
5. RBM Upscale	Once/month (Note 5)	Once/3 months	None
6. RBM Downscale	Once/month (Note 5)	Once/3 months	None
7. SRM Upscale	Notes (2,5)	Note 2	Note 2
8. SRM Detector Not-Full-In Position	Notes (2,9)	Note 2	None
9. Scram Discharge Volume-High Level	Once/3 months	Refueling outage	None
MAIN STEAM LINE (GROUP I) ISOLATION			
1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	None
2. Steam Line High Flow	Once/month	Once/3 months	Once/Shift

Table 4.2.1 - Continued

Minimum Test and Calibration Frequency For Core Cooling Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
3. Steam Line Low Pressure	Once/month	Once/3 months	None
4. Steam Line High Radiation	Once/week (Note 5)	Note 6	Once/shift
<u>CONTAINMENT ISOLATION (GROUPS 2 & 3)</u>			
1. Reactor Low Water Level (Note 10)	-	-	-
2. Drywell High Pressure (Note 10)	-	-	-
<u>HPCI (GROUP 4) ISOLATION</u>			
1. Steam Line High Flow	Once/month	Once/3 months	None
2. Steam Line High Temperature	Once/month	Once/3 months	None
<u>RCIC (GROUP 5) ISOLATION</u>			
1. Steam Line High Flow	Once/month	Once/3 months	None
2. Steam Line High Temperature	Once/month	Once/3 months	None
<u>REACTOR BUILDING VENTILATION</u>			
1. Radiation Monitors (Plenum)	Once/month	Once/3 months	Once/day
2. Radiation Monitors (Refueling Floor)	Once/month	Once/3 months	Note 4
3. Wide Range Gas Monitors	-	See Table 4.8.2	-
<u>RECIRCULATION PUMP TRIP AND ALTERNATE ROD INJECTION</u>			
1. Reactor High Pressure	Once/month	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/Day
2. Reactor Low Low Water Level	Once/month	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/shift
<u>SHUTDOWN COOLING SUPPLY ISOLATION</u>			
1. Reactor Pressure Interlock	Once/month	Once/3 Months	None

Table 4.2.1 - Continued

Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
SAFEGUARDS BUS VOLTAGE			
1. Degraded Voltage Protection	Once/month	Quarterly	Not applicable
2. Loss of Voltage Protection	Once/month	Once/Operating Cycle	Not applicable
SAFETY/RELIEF VALVE LOW-LOW SET LOGIC			
1. Reactor Scram Sensing	Once/Shutdown (Note 8)	-	-
2. Reactor Pressure - Opening	Once/3 months (Note 5)	Once/Operating Cycle	Once/day
3. Reactor Pressure - Closing	Once/3 months (Note 5)	Once/Operating Cycle	Once/day
4. Discharge Pipe Pressure	Once/3 months (Note 5)	See Table 4.14.1	See Table 4.14.1
5. Inhibit Timer	Once/3 months (Note 5)	Once/Operating Cycle	-

Table 4.2.1 - Continued

Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

NOTES:

(1) (Deleted)

(2) Calibrate prior to normal shutdown and start-up and thereafter check once per shift and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed, as necessary, in the APRM/IRM overlap region.

(3) Functional tests, calibrations and sensor checks are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

(4) Whenever fuel handling is in process, a sensor check shall be performed once per shift.

(5) A functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action.

(6) This instrument will be calibrated every three months by means of a built in current source, and each refueling outage with a known radioactive source.

(7) (Deleted)

(8) Once/shutdown if not tested during previous 3 month period.

(9) Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position.

(10) Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2.

Bases:

4.2 The instrumentation in this section will be functionally tested and calibrated at regularly scheduled intervals. Although this instrumentation is not generally considered to be as important to plant safety as the Reactor Protection System, the same design reliability goals are applied. As discussed in Section 4.1 Bases, monthly testing is generally specified unless the testing must be conducted during refueling outages. Quarterly calibration is specified unless the calibration must be conducted during refueling outages. Where applicable, sensor checks are specified on a once/shift or one/day basis.

NEXT PAGE IS 76

3.0 LIMITING CONDITIONS FOR OPERATION

Any four rod group may contain a control rod which is valved out of service provided the above requirements and Specification 3.3.A are met.

D. Control Rod Accumulators

Control rod accumulators shall be operable in the Startup, Run, or Refuel modes except as provided below.

1. In the Startup or Run Mode, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

   (a) Inoperable accumulator, or

   (b) Directional control valve electrically disarmed while in a non-fully inserted position.

   If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.

4.0 SURVEILLANCE REQUIREMENTS

D. Control Rod Accumulators

Once a shift check the status in the control room of the required Operable accumulator pressure and level alarms.

Amendment No. ~~5~~,~~11~~,~~13~~,~~54~~,63

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. In the Refuel Mode, a rod accumulator may be inoperable provided: (a) All fuel is removed from the cell containing the associated control rod, or (b) The one-rod-out refuel interlock for the associated rod drive is operable.	

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 seven days unless such system is sooner made operable, provided that during such seven days all of the Automatic Pressure Relief systems, the RCIC system, both of the core spray systems, and LPCI subsystem and containment cooling mode of the RHR system are operable.

High Pressure Core Cooling Capability

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

1. Testing

Item	Frequency
Pump operability	Once/month
Motor operated Valve operability	Once/month
Flow rate test	After major pump maintenance and every three months
Simulated automatic actuation test (testing valve operability)	Each refueling outage

2. When it is determined that HPCI system is inoperable, the RCIC system, the LPCI subsystem, and both of the core spray systems shall be demonstrated to be operable immediately.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. To be considered operable, the HPCI system shall meet the following conditions: a. The HPCI shall be capable of delivering 3,000 gpm into the reactor vessel for reactor pressure range of 1120 psig to 150 psig. b. The condensate storage tanks shall contain at least 75,000 gallons of condensate water. c. The controls for automatic transfer of the HPCI pump suction from the condensate storage tank to the suppression chamber shall be operable. 4. If the requirements of 3.5.D.1-2 cannot be met, an orderly reactor shutdown shall be initiated immediately and the reactor pressure shall be reduced to 150 psig within 24 hours thereafter.	

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 psig 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 seven days unless such valve is sooner made operable, provided that during such seven 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 24 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.

2. When it is determined that one or more automatic pressure relief valves of the Automatic Pressure Relief System is inoperable, the HPCI system shall be demonstrated to be operable immediately and weekly thereafter.

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.

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	Once/month
Motor operated valve operability	Once/month
Flow rate test	After major pump maintenance and every three months
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

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.G.4:

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.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

C. Secondary Containment

1. Except as specified in 3.7.C.2 and 3.7.C.3, Secondary Containment Integrity shall be maintained during all modes of plant operation.

2. Secondary Containment Integrity is not required when all of the following conditions are satisfied:

   a. The reactor is subcritical and Specification 3.3.A is met.

   b. The reactor water temperature is below 212°.

   c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A

   d. The fuel cask or irradiated fuel is not being moved within the reactor building.

3. With an inoperable secondary containment isolation damper, restore the inoperable damper to operable status or isolate the affected duct by use of a closed damper or blind flange within eight hours.

4. If Specifications 3.7.C.1 through 3.7.C.3 cannot be met, initiate a normal orderly shutdown and have the reactor in the Cold Shutdown condition within 24 hours. Alterations of the

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

   a. Secondary containment capability to maintain at least a 1/4 inch of water vacuum under calm wind ($2 < u <$ mph) conditions with a filter train flow rate of $\leq$4,000 scfm, shall be demonstrated at each refueling outage prior to refueling. Verification that each automatic damper actuates to its isolation position shall be performed at each refueling outage and after maintenance, repair or replacement work is performed on the damper or its associated actuator, control circuit, or power circuit.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
E. Extended Core and Control Rod Drive Maintenance More than one control rod may be withdrawn from the reactor core during outages provided that, except for momentary switching to the Startup mode for interlock testing, the reactor mode switch shall be locked in the Refuel position. The refueling interlock signal from a control rod may be bypassed after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core.	

3.14 ACCIDENT MONITORING INSTRUMENTATION

Applicability:

Applies to plant instrumentation which does not perform a protective function, but which provides information to monitor and assess important parameters during and following an accident.

Objective:

To assure that sufficient information is available to operators to determine the effects of and determine the course of an accident to the extent required to carry out required manual actions.

Specification:

Whenever irradiated fuel is in the reactor vessel and reactor coolant water temperature is greater than 212°F, the limiting conditions for operation for accident monitoring instrumentation given in Table 3.14.1 shall be satisfied.

4.14 ACCIDENT MONITORING INSTRUMENTATION

Applicability:

Applies to the surveillance requirements for accident monitoring instrumentation.

Objective:

To specify the type and frequency of surveillance to be applied to accident monitoring instrumentation.

Specification:

The accident monitoring instrumentation shall be functionally tested and calibrated in accordance with Table 4.14.1.

Table 3.14.1

Instrumentation for Accident Monitoring

Function	Total No. of Instrument Channels	Minimum No. of Operable Channels	Required Conditions*
Reactor Vessel Fuel Zone Water Level	2	1	A, B
Safety/Relief Valve Position (One Channel Pressure Switch and One Channel Thermocouple Position Indication per Valve)	2	1	A, C
Drywell Wide Range Pressure	2	1	A, B
Suppression Pool Wide Range Level	2	1	A, B
Suppression Pool Temperature	2	1	A, D
Drywell High Range Radiation	2	1	A, D
Drywell and Suppression Pool Hydrogen and Oxygen Monitor	2	1	A, B
Offgas Stack Wide Range Radiation	2	1	A, D
Reactor Bldg Vent Wide Range Radiation	2	1	A, D

* Required Conditions

A. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the total number of channels, either restore the inoperable channels to operable status within seven days, or prepare and submit a special report to the Commission pursuant to Technical Specification 6.7.B.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

Table 3.14.1 (continued)

Instrumentation for Accident Monitoring

* Required Conditions (continued)

B. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the minimum number of channels shall be restored to operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.

C. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, the torus temperature shall be monitored at least once per shift to observe any unexplained temperature increase which might be indicative of an open SRV; the minimum number of channels shall be restored to operable status within 30 days or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.

D. When the number of channels made or found to be inoperable is such that the number of operable channels is less than the minimum number of operable channels shown, initiate the preplanned alternate method of monitoring the appropriate parameters in addition to submitting the report required in (A) above.

Table 4.14.1

Minimum Test and Calibration Frequency for
Accident Monitoring Instrumentation

Instrument Channel	Test (Note 1)	Calibration (Note 1)	Sensor Check (Note 1)
Reactor Vessel Fuel Zone Water Level Monitor	-	Once/Operating Cycle	Once/month (Note 3)
Safety/Relief Valve Position (Pressure Switches)	-	Once/Operating Cycle	Once/month (Notes 2 & 4)
Safety/Relief Valve Position (Thermocouples)	-	Once/Operating Cycle	Once/month (Note 4)
Drywell Wide Range Pressure Monitors	-	Once/Operating Cycle	Once/month
Suppression Pool Wide Range Level Monitors	-	Once/Operating Cycle	Once/month
Suppression Pool Temperature	-	Once/Operating Cycle	Once/month
Drywell High Range Radiation Monitors	-	Once/Operating Cycle	Once/month
Drywell and Suppression Pool Hydrogen and Oxygen Monitors	-	Once/Operating Cycle	Once/month
Offgas Stack Wide Range Radiation Monitors	-	Once/Operating Cycle	Once/month
Reactor Bldg Wide Range Radiation Monitors	-	Once/Operating Cycle	Once/month

Notes:

(1) Functional tests, calibrations, and sensor checks are not required when the instruments are not required to be operable. If tests are missed, they shall be performed prior to returning the instruments to an operable status.

(2) Once/month sensor check will consist of verifying that the pressure switches are not tripped.

(3) Once/month sensor check will consist of verifying that the fuel zone level indicates off scale high.

(4) Following every Safety/Relief Valve actuation it will be verified that recorder traces or computer logs indicate sensor responses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.63TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated May 5, 1986, the Northern States Power Company (NSP or the licensee) proposed changes to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed amendment included changes resulting from a detailed review of the TSs that occurred following the 1985 plant refueling and recirculation piping replacement outage. Several of the proposed changes are administrative in nature or are proposed to clarify the application of the existing TSs. The specific changes and our evaluation of each change are presented below.

2.0 DISCUSSION AND EVALUATION

The TS changes proposed by the licensee, including our assessment of the acceptability of each change, follows:

a. Definition of Core Alteration

Revise the definition of "Alteration of the Reactor Core" in Section 1.0.A by adding the words, "with the vessel head removed and fuel in the vessel," to the end of the first sentence.

According to the licensee, a literal interpretation of the existing definition would require movements to be considered core alterations even when fuel is not present within the vessel. This is not necessary, and it creates conflicts with other Specifications. For example, Specification 3.10.B, which uses this definition, would require SRM operability when no fuel is in the vessel. We agree that the proposed change will clarify the definition of "Core Alteration."

b. 2.3 Bases Correction

Delete the partial sentence in the first line of the first paragraph of the Section 2.3 Bases on page 17.

These words should have been deleted with a previous license amendment request, but were left through an oversight.

c. Table 3.1.1, Startup Mode Operability Requirements

Move the reference to Note 4 from the "Refuel" column to the "Trip Function" column so the note is applicable to all modes, and add the following new note to the table:

9. High reactor pressure and main steam line high radiation are not required to be operable when the reactor vessel head is unbolted.

Add a reference to Note 9 to the table entries for high reactor pressure and main steam line high radiation.

We agree with the licensee that the startup mode operability requirements listed for high drywell pressure, high reactor pressure, and main steam line high radiation are unnecessarily restrictive for activities such as low power physics tests and that it is desirable to eliminate such unnecessary requirements from the TSs. For example, with the vessel head unbolted, high reactor pressure and steam line radiation functions are not necessary. High drywell pressure functions are not necessary when containment integrity is not required.

d. Table 4.1.1 and Table 4.2.1, Variable Surveillance Frequencies, and Associated Bases

Delete Note 1 of Table 4.1.1 and Note 1 of Table 4.2.1. Delete Figure 4.1.1 and correct the List of Figures to reflect deletion of this figure. Delete all references to Note 1 on both tables and replace with a requirement for monthly surveillance. Delete those portions of the 4.1 and 4.2 Bases which refer to variable surveillance frequencies.

Note 1 of these tables allows certain surveillance intervals to be lengthened up to a maximum of three months by application of Figure 4.1.1. Lengthening of surveillance intervals is based on the number of unsafe failures that are experienced over a period of time. Several years ago NRC asked the licensee not to use TS Figure 4.1.1 to lengthen surveillance intervals, and the licensee is reluctant to do so also since this would require periodic changes in test intervals and records. As a result, monthly testing always has been conducted even though the TSs would permit less frequent testing. Because this is a potentially confusing situation, we agree that the TSs should be revised as proposed to eliminate the option to extend surveillance intervals.

e. SRM Not-Full-In Rod Block Interlock Conflicts

Add a new Note 9 to Table 4.2.1 as follows:

9. Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position.

Add a reference to Note 9 on Table 4.2.1 under item 8 of Rod Blocks. Change the item to read, "SRM Detector Not-Full-In Position" instead of, "... not in Start-Up Position." Change the sensor check requirement from "Note 2" to "None."

The existing testing requirement for the SRM Not-Full-In rod block interlock conflicts with normal CRD maintenance work. The specific application of the test requirements could be inconsistent with the normal and prudent practices of rerouting the SRM cables to allow CRD maintenance and securing the SRM detectors in the full-in position. Also, it is not possible to perform the required sensor check of the interlock. It is an on-off device, not an analog signal subject to sensor checks.

Because of the need to reroute the drive cables to allow normal CRD work and the lack of space under the vessel, it is preferable not to perform detector withdrawals for testing. Such testing would damage the reconfigured cable. Restoration of original cable configuration to allow testing would result in additional personnel exposure, additional wear, and risk of damage to cable and connector assemblies. Verifying that the detectors are full-in and securing the detector drive power in "off" enforces the condition under which the interlock is satisfied.

NRC inspectors have accepted this procedure in the past for the reasons stated, and as such, it is desirable to revise the TS wording to agree with this practice.

f. Clarification of Containment Isolation Instrumentation Surveillance on Table 4.2.1

Expand the headings for main steam, HPCI, and RCIC isolation by adding a reference to the containment isolation group and add a new category for Group 2 and Group 3 containment isolation. Delete Note 7 and all references to Note 7 in the table. Add a new Note 10 as follows:

10. Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2.

Add a reference to Note 10 for containment isolation Groups 2 and 3 reactor low water level and drywell high pressure surveillance.

We agree with the licensee that the existing TS surveillance requirements for containment isolation functions are misleading and imprecise. The proposed changes would clarify and expand Table 4.2.1 to more clearly list the containment isolation logic inputs and also note that some of the signals for Group 2 are derived from the scram logic.

Note 7 was added as part of an earlier TS change accepted by the NRC. The intent was to have Table 4.2.1 cover containment isolation surveillance. The earlier change was not as precise as it should have been, and this further change provides that precision.

g. Bases for Specification 4.0

Revise the Bases section to explain the surveillance testing requirements in Section 4.0 of the TSs and add information to assist in understanding and applying this section.

The proposed wording is derived from the NRC Standard TSs. Additional clarification of the surveillance interval tolerance is derived from clarifying information contained in NRC Inspection Report 50-263/85012(DRP) dated July 19, 1985.

This addition to the Bases will help in understanding Section 4.0 of the TSs. This section was recently added to provide general requirements for the Surveillance Program. The proposed wording summarizes the application related to surveillance intervals and surveillance scheduling established with NRC inspectors over a period of many years.

h. Table 3.2.5, Note 1, ATWS Instrumentation Requirements

Revise Note 1 of Table 3.2.5 to read:

1. When one of the two trip systems is made or found to be inoperable, restore the inoperable trip system to operable status within 14 days or place the plant in the specified required condition within the next eight hours. When both trip systems are inoperable, place the plant in the specified required condition within eight hours unless at least one trip system is sooner made operable.

The existing Note 1 is inconsistent with the requirements for minimum number of operable or operating trip systems in Table 3.2.5. A loss of one trip system would require a plant shutdown since it is not possible to place a trip system in a tripped condition without actually causing actuation of the logic (this is 1 of 2 logic). As long as the remaining trip system is operable, this is an unnecessary requirement. One operable trip system is sufficient to initiate the required protective action. We believe this was an oversight on the part of the licensee when this TS change was first developed. However, it should be noted that the staff is presently reviewing ATWS requirements to determine wheter, and to what extent, Technical Specfciations are appopriate. The staff will provide guidance regarding the Technical Specification requirements for ATWS at a later date.

i. Control Rod Accumulator Operability Clarification

Delete the last paragraph of Specification 3.3.D and redesignate items 1 and 2 under 3.3.D as items 3.3.D.1(a) and (b). Reword the opening paragraph as follows:

Control rod accumulators shall be operable in the Startup, Run, or Refuel modes except as provided below.

Add Specification 3.3.D.2 as follows:

2. In the Refuel Mode, a rod accumulator may be inoperable provided:

   (a) All fuel is removed from the cell containing the associated control rod, or

(b) The one-rod-out refuel interlock for the associated rod drive is operable.

We agree with the licensee that a specific application of the last paragraph of this specification could preclude normal CRD maintenance where the rod out refuel interlock is not bypassed and fuel is not removed from the cell. This could force cell unloading for all drive changeouts. This was not the intent of the specification. The specification was intended to apply only to the situation of multiple CRD removal for extended core and control rod drive maintenance, controlled by Specification 3.10.E, where the rod out interlock is bypassed for withdrawn rods. It is not impractical to unload fuel cells during actual refueling with the vessel head removed. It is impractical, however, to require cell unloading for situations requiring drive maintenance after the core is fully reloaded and the head has been replaced. Accordingly, the proposed change is acceptable.

j. High Pressure Coolant Injection (HPCI), Automatic Pressure Relief System (APRS), and Reactor Core Isolation Cooling (RCIC) Operability Conditions

Revise Specifications 3.5.D.1, 3.5.E.1, and 3.5.F.1 so that operability of these systems is not required above 150 psig during reactor coolant system leakage and hydrostatic tests by revising the Operability condition to read, "...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." Also, reformat pages 109 and 110 to move the headings for the APRS sections to the top of page 110.

We agree with the licensee that a specific application of the current requirements for HPCI, APRS, and RCIC system operability conflicts with the requirement to perform reactor coolant system leakage tests following each refueling outage, and reactor coolant system hydrostatic tests at ten-year intervals and following major system repairs or modifications. The TSs currently require these systems to be operable when irradiated fuel is in the vessel and reactor pressure is greater than 150 psig. Since these systems are designed to operate from a source of steam, they cannot be made operable during leakage or hydrostatic tests when the vessel is flooded and reactor coolant temperature is below saturation temperature.

The NRC staff was contacted during the last refueling outage to obtain concurrence with the logical understanding that these systems are in fact not required to be operable during leakage and hydrostatic tests. The TSs for many other boiling water reactor plants contain similar conflicts. The proposed wording change eliminates the need for this interpretation.

k. Clarification of Primary Containment Requirements

Reword Specification 3.7.A.1 as follows: "When irradiated fuel is in the reactor vessel and either the reactor coolant 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.G.4...."

We agree with the licensee that a specific application of existing TS 3.7.A.1 would not allow draining of the suppression chamber when irradiated fuel was not in the reactor vessel and work which would, or had the potential, to drain the reactor vessel was in progress. This is due to the ommission, in this case, of the standard wording, "when irradiated fuel is in the reactor vessel." All other Monticello primary containment and ECCS TSs contain this provision.

l. Reactor Coolant System Venting and Requirements for Secondary Containment

In Specification 3.7.C.2.b, delete the phrase, "...and the reactor coolant system is vented."

The requirement for the reactor to be vented as a condition for not requiring secondary containment conflicts with normal and reasonable activities during outages. For example, reactor vents must be closed to perform vessel leakage and hydrostatic testing. At other times, it is prudent to close reactor vents for radiological protection purposes.

There is no basis for relating secondary containment requirements to reactor venting. Closing reactor vents during an outage when secondary containment was not established could be considered as a violation of the TSs even though the event would have no safety implications. In addition, the standard TSs do not include this or similar limitations on reactor venting.

m. Extended Core and CRD Maintenance TS Conflicts

Delete Specification 3.10.E.2 and redesignate Specifications 3.10.E.1 and 3.10.E. Reword the first portion of the Specification to read, "More than one control rod may be withdrawn from the reactor core during outages provided that, except for momentary switching to the Startup mode for interlock testing, the reactor mode switch is locked in the Refuel position. The refueling interlock...." Change "withdrawn control rod" to "control rod" in two locations.

We agree with the licensee that the existing Specification 3.10.E.2 is totally redundant to Specification 3.10.B and therefore unnecessary and possibly confusing.

We also agree that there is a conflict between the existing Specification 3.10.E.1 and Specification 4.10.A and that a literal interpretation of the existing specification would prohibit rod withdrawal for normal operation and testing.

Specification 4.10.A requires weekly checks of the refueling interlocks until core alterations are completed and they are no longer required. Core alterations, as defined in Section 1.0, occur throughout most of an outage. During this time it is normal to have control rods removed from the core for maintenance in accordance with Specification 3.10.E.1. The conflict is that Specification 3.10.E.1 requires the mode switch to be locked in "Refuel," but the weekly check of refueling interlocks requires switching momentarily to the "Startup" mode.

The lead-in statement of the existing TS does not limit its applicability. The current wording, taken literally, would require that the mode switch be locked in refuel for any rod withdrawal from the core. Although obviously not the intent, this would prohibit more than one rod from ever being withdrawn.

n. Accident Monitoring Instrumentation Operability Conditions

Revise Specification 3.14 to require operability of accident monitoring instrumentation, "...whenever irradiated fuel is in the reactor vessel and reactor coolant water temperature is greater than 212° F...." Revise the notes in Table 3.14.1 to require placing the plant in the cold shutdown condition within 24 hours when required conditions of instrument operability are not satisfied.

We agree with the licensee that the existing wording for Specification 3.14 requires operability of this instrumentation in the startup and run modes. During outages, the mode switch must often be placed in the startup position to perform tests or other normal operations. However, accident monitoring instrumentation may be inoperable during an outage due to normal activities or conditions. For example, SRV removal would render SRV position indication inoperable and vessel draining would render vessel level instrumentation inoperable.

There is no reason for accident instrumentation operability to be based on mode switch position. A more desirable wording for the TS is to make it consistent with other accident mitigation system operability requirements (i.e., above 212° F) and the NRC Standard TSs.

o. Tables 3.14.1 and 4.14.1, Suppression Pool Temperature Monitoring Instrumentation

Add the Suppression Pool Temperature Monitoring instrumentation to Tables 3.14.1 and 4.14.1.

This instrumentation was added as a result of the Mark I Containment Long-Term Improvement Program to accurately monitor suppression pool average temperature. We find that the operability and surveillance requirements proposed for this instrumentation are consistent with other TSs for accident monitoring instrumentation.

The Suppression Pool Temperature Monitoring System (SPOTMOS) installed at Monticello was described in Volume 1, Section 5, of the Monticello Plant Unique Analysis Report submitted to NRC on December 15, 1982. This design was reviewed and approved by the NRC, and a safety evaluation was issued on September 11, 1985.

In the event of inoperability of SPOTMOS, alternate methods of monitoring suppression pool temperature are available until operability of SPOTMOS can be restored. These methods include use of an alternate multipoint recorder and temperature sensors in the suppression pool (the original temperature monitoring system).

p. Clarification of Table 4.14.1 Sensor Checks

Provide additional notes in Table 4.14.1 to clarify sensor check requirements for reactor water level, SRV valve position pressure switches, and SRV valve position thermocouples as follows:

(2) Once/month sensor check will consist of verifying that the pressure switches are not tripped.

(3) Once/month sensor check will consist of verifying that the fuel zone level indicates off scale high.

(4) Following every Safety/Relief Valve actuation it will be verified that recorder traces or computer logs indicate sensor responses.

Add a reference to Note 2 for SRV position pressure switches. Add a reference to Note 3 for reactor vessel fuel zone water level, and add a reference to Note 4 for SRV position pressure switches and thermocouples.

We agree with the licensee that the specific application of the requirements for sensor checks of the SRV position pressure switches would require operation of the SRV's once per month. It would also require establishing an abnormal reactor level to perform a sensor check of the fuel zone level instrument.

Table 14.1.1 notes have been revised to clarify the intent of these sensor checks, which is to require a verification of sensor operation following each SRV actuation, and once each month to verify that the fuel zone indicator is off scale high and the SRV pressure switches are not tripped.

These sensor checks do not comply literally with the definition of "Sensor Check" in Section 1.0 of the TSs, and it is important that notes to Table 14.1.1 clearly specify the intended requirements.

To summarize, the proposed TS changes itemized above either improve clarity and logic, provide some relief from some of the restrictions found to be either unnecessary or impracticable to perform, add a new requirement considered desirable by the NRC staff, and/or involve an administrative change, as follows:

Item a. - clarifies the definition of "Core Alteration."

Item b. - corrects a typographical error in the Section 2.3 Bases.

Item c. - corrects and clarifies the Startup Mode operability requirements for high drywell pressure, high reactor pressure, and main steam line high radiation.

Item d. - deletes the obsolete provision of the TSs which permits surveillance intervals to be extended.

Item e. - corrects conflicts with the SRM-Not-Full-In rod block interlock and CRD maintenance.

Item f. - corrects and clarifies the surveillance requirements for containment isolation instrumentation.

Item g. - adds a new section to the Bases explaining general surveillance requirements.

Item h. - corrects the action statements for ATWS instrumentation to correspond with the acceptable as-installed logic.

Item i. - clarifies CRD accumulator operability requirements.

Item j. - corrects the HPCI, RCIC and APRS operability requirements to permit reactor coolant system leakage and hydrostatic testing.

Item k. - clarifies the requirements for containment integrity when no fuel is in the reactor.

Item l. - corrects and clarifies the relationship between secondary containment requirements and reactor venting.

Item m. - clarifies the requirements for extended CRD maintenance.

Item n. - corrects and clarifies the operability conditions for facility monitoring instrumentation.

Item o. - adds TS LCOs and surveillance requirements for suppression pool temperature monitoring instrumentation.

Item p. - clarifies the meaning of sensor checks for safety/relief valve position pressure switches and reactor fuel zone water level instrumentation.

With the exception of Items b. and o., all of the changes have the intent of eliminating conflicts and problems in understanding the TSs.

These items were identified during a detailed review of the TSs by senior reactor operator licensed members of the Monticello technical staff. This review was made to fulfill a commitment made to NRC management following the discovery during the 1985 refueling and maintenance outage of a number of conflicts in the TSs. While some relief from impossible or unreasonable restrictions is granted in several instances (e.g. HPCI will no longer be required operable during hydrostatic tests - but because the vessel is filled solid with subcooled water during these tests it is an impossible condition to impose), the requested changes will not, in any significant way, change the way the plant is operated or maintained. Item o. adds new requirements for an instrumentation system installed to meet the requirements of the NRC approved Mark I Containment Long-Term Program and NRC Regulatory Guide 1.97, Revision 2. This new and improved instrumentation system will enhance the information available to plant operators during normal and postulated accident conditions.

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

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, 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.

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