Docket No. 50-263 August 18, 1992

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

Dear Mr. Parker:

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - AMENDMENT NO. 83 TO FACILITY OPERATING LICENSE NO. DPR-22 (TAC NO. M82783)

The Commission has issued the enclosed Amendment No. 83 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. This amendment consists of three changes to the Technical Specifications (TS) in response to your application dated February 14, 1992. The first change revises reactor protection system technical specifications to eliminate the main steam line high radiation scram and associated reactor vessel isolation function. The second change revises the description of the average power range monitor scram trip function Bases to clarify when bypasses are permissible. The third change relates to the impracticality of performing intermediate range monitor functional testing during Mode 1 operation.

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

Sincerely,

Original signed by

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. 83 to DPR-22
2. Safety Evaluation

cc w/enclosures:
See next page *SEE PREVIOUS CONCURRENCE
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OFFICE	LA:PD31	PM:PD31	SRXB*	SICB	OGC	D:PD31
NAME	MShuttleworth	WLong: jkd	RJones	SNewberry	RBachmann	LMarsh
DATE	5/29/92	7/13/92	3/5/92	8/6/92	8/18/92	8/18/92

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

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cc w/enclosures:
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DATED: August 18, 1992

AMENDMENT NO. 83 TO FACILITY OPERATING LICENSE NO. DPR-22-MONTICELLO

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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. 83
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (the licensee) dated February 14, 1992, 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. 83 , 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Ledyard 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: August 18, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 83

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
29	29
30	30
32	32
33	33
34	34
36	36
37	37
49	49
62	62
63a	63a
66	66

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 Isolation Valve Closure	≤ 10% Valve Closure	X(b)	X(b)	X	8	8	A or C
11. Turbine Control Valve Fast Closure	(See Note 7)			X(d,f)	2	2	D
12. 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. A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other operable channel in the same trip system is monitoring that parameter.

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.

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 is 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 functionin 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 ≤ 45% (751.5 MWt).

TABLE 4.1.1

SCRAM INSTRUMENT FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENTATION AND CONTROL CIRCUITS

INSTRUMENTATION CHANNEL	FUNCTIONAL TEST	MINIMUM FREQUENCY (4)
High Reactor Pressure	Trip Channel and Alarm	Quarterly
High Drywell Pressure	Trip Channel and Alarm	Quarterly
Low Reactor Water Level (2, 5)	Trip Channel and Alarm	Quarterly
High Water Level in Scram Discharge Volume	Trip Channel and Alarm	Quarterly
Condenser Low Vac	Trip Channel and Alarm	Once each month
Main Steam Line Isolation Valve Closure	Trip Channel and Alarm	Quarterly
Turbine Stop Valve Closure	Trip Channel and Alarm	Quarterly
Manual Scram	Trip Channel and Alarm	Weekly
Turbine Control Valve Fast Closure	Trip Channel and Alarm	Quarterly
APRM/Flow Reference (5)	Trip Output Relays	Quarterly
IRM (5)	Trip Channel and Alarm	Note 3
Mode Switch in Shutdown	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 demonstrate that the IRM and APRM channels overlap at least 1/2 decade prior to every 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.

TABLE 4.1.2

SCRAM INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

INSTRUMENT CHANNEL	GROUP	CALIBRATION METHOD	MINIMUM FREQUENCY (2)
APRM	B	Heat Balance	Once every 3 days (4)
IRM	B	Heat Balance	See Note 1
High Reactor Pressure	A	Pressure Standard	Every 3 months
High Drywell Pressure	A	Pressure Standard	Every 3 months
Low Reactor Water	B	Pressure Standard	Every Operating Cycle - Transmitter Every 3 months - Trip Unit
High Water Level in Scram Discharge	A or B	Water Level	Every 3 months
Condenser Low Vacuum	A	Vacuum Standard	Every 3 months
Main Steamline Isolation Valve Closure	A	Observation	Every Operating Cycle
Turbine Control Valve Fast Closure	A	Pressure Standard	Every 3 months
Turbine Stop Valve Closure	A	Observation	Every Operating Cycle
Recirculation Flow Meters & Flow Instrumentation	-	Pressure Standard	Every 3 months

Notes:

1. Perform calibration test during every startup and normal shutdown.
2. Calibration 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.
3. (Deleted).
4. This calibration is performed by taking a heat balance and adjusting the APRM to agree with the heat balance. Alarms and trips will be verified and calibrated if necessary during functional testing.

*GROUPS:

A. Passive type devices.
B. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Bases Continued:

3.1 Three APRM instrument channels are provided for each protection trip system. APRM's #1 and #3 operate contacts in one subchannel, and APRM's #2 and #3 operate contacts in the other subchannel. APRM's #4, #5, and #6 are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required. This allows the bypassing of one APRM per protection trip system. Additional IRM channels have also been provided to allow for bypassing of one such channel in each trip system.

The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, turbine control valve fast closure, and turbine stop valve closure are discussed in Specifications 2.3 and 2.4.

Instrumentation (pressure switches) in the drywell are provided to detect a loss of coolant accident and initiate the emergency core cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by the scram can be accommodated in the discharge piping. Part of this piping consists of two instrument volumes which accommodate in excess of 56 gallons of water each and is the low point in the piping. During normal operation the discharge volumes are empty; however, should they fill with water, the water discharge to the piping from the reactor could not be accommodated which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volumes which alarm and scram the reactor when the volume of water in either of the discharge volume receiver tanks reaches 56 gallons. At this point there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of

Bases Continued:

3.1 condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass. Reference FSAR Section 14.5.1.2.2 and supplemental information submitted February 13, 1973. The condenser low vacuum scram is a back-up to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 23" Hg vacuum, stop valve closure occurs at 20" Hg vacuum, and bypass closure at 7" Hg vacuum.

The main steamline isolation valve closure scram is set to scram when the isolation valves are ≤10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting the resultant transient is insignificant. Reference Section 14.5.1.3.1 FSAR and supplemental information submitted February 13, 1973.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.7.1 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system provides protection against excessive power levels and short reactor periods in the

Table 3.2.1
Instrumentation That Initiates Primary Containment Isolation Functions

Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1,2)	Required Conditions*
1. Main Steam and Recirc Sample Lines (Group 1)				
a. Low Low Reactor Water Level	≥6'-6" ≤6'10"	2	2	A
b. High Flow in Main Steam Line	≤140% rated	8	8	A
c. High temp. in Main Steam Line Tunnel	≤200°F	8	2 of 4 in each of 2 sets	A
d. Low Pressure in Main Steam Line (3)	≥825 psig	2	2	B
2. RHR System, Head Cooling, Drywell, Sump, TIP (Group 2)				
a. Low Reactor Water Level	≥10'6" above the top of the active fuel	2	2	C

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/3 months	Once/3 months	None
4. Reactor Low Low Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Once/3 months - Trip Unit	Once/shift
CONTAINMENT ISOLATION (GROUPS 2 & 3)			
1. Reactor Low Water Level (Note 10)	-	-	-
2. Drywell High Pressure (Note 10)	-	-	-
HPCI (GROUP 4) ISOLATION			
1. Steam Line High Flow	Once/month	Once/3 months	None
2. Steam Line High Temperature	Once/month	Once/3 months	None
RCIC (GROUP 5) ISOLATION			
1. Steam Line High Flow	Once/month	Once/3 months	None
2. Steam Line High Temperature	Once/month	Once/3 months	None
REACTOR BUILDING VENTILATION			
1. Radiation Monitors (Plenum)	Once/month	Once/3 months	Once/day
2. Radiation Monitors (Refueling Floor)	Once/month	Once/3 months	Note 4
RECIRCULATION PUMP TRIP AND ALTERNATE ROD INJECTION			
1. Reactor High Pressure	Once/month (Note 5)	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/Day
2. Reactor Low Low Water Level	Once/month (Note 5)	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/shift
SHUTDOWN COOLING SUPPLY ISOLATION			
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

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) (Deleted)

(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 Continued:

3.2 instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst case accident, main steamline break outside the drywell, this trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steamline valve closure, limit the mass inventory loss such that fuel is not uncovered, fuel clad temperatures remain less than 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Sections 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a back-up to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

Pressure instrumentation is provided which trips when main steamline pressure drops below 825 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "refuel" and "Startup" mode this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 825 psig inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500°F; thus, there are no fission products available for release other than those in the reactor water. Reference License Amendment Request Dated December 1, 1975 from L. O. Mayer (NSP) to R. S. Boyd (USNRC).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 83 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated February 14, 1992, Northern States Power Company (the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed amendment consists of three change requests to the Technical Specifications (TS). The first change would revise the reactor protection system TS to eliminate the main steam line (MSL) high radiation scram and associated reactor vessel isolation function. The second change would revise the description of the average power range monitor (APRM) scram trip function Bases to clarify when bypasses are permissible. The third change relates to the impracticality of performing intermediate range monitor functional testing during Mode 1 operation.

2.0 EVALUATION

2.1 DELETION OF MSL SCRAM/ISOLATION FUNCTION

Discussion: The license's first proposed change is based on NEDO-31400. By letter dated July 9, 1987, the BWR Owners' Group (BWROG) submitted licensing topical report "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," May 1987, General Electric Company for NRC approval. BWRs are equipped with radiation monitors (MSLRMs) which are located on the main steam lines downstream of the main steam isolation valves (MSIVs). The MSLRMs were provided to detect moderate to large fuel failures and close the MSIVs to stop the release of radioactivity into the steam lines. Typically, an alarm sounds if the radiation level exceeds 1.5 X nominal full power background level. If the radiation level exceeds 3 X nominal full power background level, an isolation signal is generated isolating the MSLs. Switches on the MSIVs sense MSIV closure and initiate a reactor scram signal. The MSL high radiation scram signal is not associated with or credited for in any of the facility analyzed design basis accidents or transients. In control rod drop accident (CRDA) analyses, the MSIVs are assumed to close on the MSLRM signal, but the radioactive source term is assumed to already have been transferred to the condenser and turbine prior to MSIV closure. In NEDO-31400, the applicant reevaluated the role of the MSLRM in the CRDA

analysis and confirmed that removal of the MSLRM scram/isolation features would not compromise CRDA consequences. The topical report also evaluated the potential effect on occupational exposure in the event of a sudden release of radioactivity from the fuel and concluded that the elimination of the scram/isolation features would have no adverse effect.

The staff evaluated NEDO-31400 and issued its Safety Evaluation (SE) on May 15, 1991. The SE concluded that the topical report was acceptable for referencing in plant specific licensing actions for listed facilities, including Monticello, subject to the following conditions:

(1) The applicant demonstrates that the assumptions with regard to input values that are used in the NEDO-31400 generic analyses bound those for the plant.

(2) The applicant includes sufficient evidence to provide reasonable assurance that increases in significant levels of radioactivity in the MSLs will be controlled expeditiously to limit occupational doses and environmental releases.

(3) The applicant standardizes the MSLRM and offgas radiation monitor alarm set point at 1.5 times the nominal nitrogen-16 full power background level and commits to promptly sample the reactor coolant in the event the MSLRM of offgas monitors exceed their alarm set points.

Regarding (1) above: The licensee's application indicates that Monticello is bounded by the NEDO-31400 generic analysis with the exception that Monticello offgas system charcoal filter capacity is considerably smaller than that assumed in the generic analysis. The licensee, therefore, reanalyzed the CRDA taking no credit for charcoal adsorption of iodine in the offgas system and found that the accident consequences would be less than 0.01% of 10 CFR Part 100 criteria for whole body and thyroid doses.

Regarding (2) above: The licensee indicated that existing plant procedures direct operators to confirm high radiation alarms and reduce power or shut down in event of high radiation level, and to implement appropriate emergency actions and system isolation.

Regarding (3) above: The licensee takes exception to a fixed 1.5 X full power nitrogen-16 background set point. Monticello utilizes hydrogen water chemistry (HWC) which has a significant effect on background radiation levels in the MSLs. Occasionally the need to modify the hydrogen injection program may arise. The licensee therefore intends to adjust the radiation monitor setpoints as necessary to account for the background effects of HWC changes. Set points will be selected to maintain a margin at least equivalent to a 1.5 X full power background level setpoint. The exception is therefore acceptable.

Conclusion: Based on conformance to the generic SE of May 15, 1991, with exceptions noted and justified as described above the MSL high radiation scram/isolation features may be eliminated at Monticello.

2.2 Average Power Range Monitor (APRM) BASES

Discussion: Monticello was designed and constructed to have two reactor protection system primary channels (A & B), with each channel further divided into two subchannels to provide a one-out-of-two-twice scram logic. Each subchannel receives inputs from a dedicated APRM. In addition, each primary channel is served by a third APRM that is connected to both subchannels. This third APRM serves, in effect, as an installed spare and is used to maintain the one-out-of-two-twice logic when one of the other APRMs becomes inoperable or is removed from service for other reasons. To facilitate this, each primary channel is provided with a joystick type switch that permits bypass of one (but only one) APRM. The reactor protection system (RPS) is also configured such that some APRMs in Channel A share local power range monitors (LPRMs) with APRMs in Channel B. Specifically, APRM 1 (Channel A) receives inputs from the same LPRMs as APRM 5 (Channel B). Similarly, APRM 2 (Channel A) receives inputs from the same LPRMS as APRM 6 (Channel B). As a result, a single LPRM failure could trip both channels of the RPS and cause a scram. This has happened twice at Monticello (scram #53 on July 27, 1978 and scram #55 on August 24, 1978). For this reason, unless otherwise needed, one APRM per primary channel (APRMs 2 & 5, or 1 & 6) is bypassed during operation to preclude unnecessary scrams. The current 3.1 Bases wording can be construed as being all inclusive, implying that the only permissible reasons for bypassing an APRM are those listed (i.e., for maintenance, testing, or calibration). In fact, there are other valid reasons for bypassing an APRM, such as the case described above and the system is designed to accommodate these situations. The proposed change to the Technical Specification 3.1 Basis section is intended to clarify that APRM bypass need not be limited to the indicated situations and are acceptable in other cases provided the TS requirements regarding the minimum number of operable channels are maintained. Plant operation with one APRM bypassed in each RPS subsystem is consistent with the system design basis that no one instrument (i.e., LPRM) failure should result in a trip or failure to trip.

Conclusion: Based on conformance to the APRM system design basis requirements Updated Safety Analysis Report (USAR) Section 7.3.5.2.1.e, consistency with the Acceptance Criteria of Standard Review Plan Section 7.2, regarding protection system redundancy, testability, and independence, the proposed change is acceptable.

2.3 IRM SCRAM FUNCTIONAL TEST REQUIREMENT

Discussion: The Monticello TS Table 4.1.1 currently specifies that an IRM Scram functional test be performed prior to each startup and normal shutdown. Such a test requires "lifted leads." The licensee proposes to delete the requirement for such test prior to normal shutdown. It is the licensee's position, and the staff acknowledges the fact (i.e., not required by Standard Technical Specifications), that such a test is impractical during power operation.

As a result of discussions with the staff, the licensee revised its proposed amendment to add a new specification to verify 1/2-decade of APRM/IRM overlap during normal shutdown. This change reflects a staff position of the Standard Technical Specifications, and is also acceptable.

Conclusion: Based on consistency with the acceptance criteria of the Standard Review Plan, Section 16, the proposed change is acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes in surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 9447). 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.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of 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: August 18, 1992