Docket No. 50-263

March 31, 1989

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

Dear Mr. Musolf:

SUBJECT: AMENDMENT NO.62 TO FACILITY OPERATING LICENSE NO. DPR-22: (TAC NO. 65572)

The Commission has issued the enclosed Amendment No.62 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. This amendment consists of changes to the Technical Specifications in response to your application dated December 5, 1986.

The amendment revises the plant Technical Specifications to reflect logic changes made to implement the requirements of NUREG-0737, Item II.K.3.18, and lower the Safety/Relief Valve Discharge Pipe Pressure Switch setpoints. Technical Specification changes also proposed in your letter dated December 5, 1986, relative to NUREG-0737, Item II.F.2, are still being evaluated and will be addressed in a separate license amendment action. This completes out action under TAC No. 65572.

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

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

Enclosures:
1. Amendment No. 62 to License No. DPR-22
2. Safety Evaluation

DF01
1/1

cc w/enclosures:
See next page

LA/PD31:DRSP	PM/PD31:DRSP	(A)D/PD31:DRSP	SRXB:DEST	SICB:DEST	OGC
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 31, 1989

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John J. Stefano, Project Manager
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cc w/enclosures:
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. 62
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 December 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. 62 , 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

Dominic C. DiDanni/TQ

Theodore R. Quay, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 31, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 62

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
52	52
53	53
60b	60b
63	63
71	71
110	110
127	127

Table 3.2.2
Instrumentation that Initiates Emergency Core Cooling Systems

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

Table 3.2.2
Instrumentation that Initiates Emergency Core Cooling Systems

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

TABLE 3.2.7

Instrumentation for Safety/Relief Valve Low-Low Set Logic

Function	Trip Setting	Min. No. of Operable or Operating Trip Systems	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System	Required Conditions
Reactor Scram Detection		2(2)	2	2	A or B or C
Reactor Coolant System Pressure for Opening/ Closing (1)	1072±3/992±3 psig 1062±3/982±3 psig 1052±3/972±3 psig	2(2)	2	2	A or B or C
Discharge Pipe Pressure Inhibit and Position Indication	30±1 psid(3)	2(2)	2	2	A or B or C
Inhibit Timers	10±1 sec	2(2)	2	2	A or B or 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)
SAFEGUARDS BUS VOLTAGE			
1. Degraded Voltage Protection	Note 1	Quarterly	Not applicable
2. Loss of Voltage Protection	Note 1	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	-

	Trip Function	Deviation
Instrumentation That Initiates Emergency Core Cooling Systems Table 3.2.2	*Low-Low Reactor Water Level	-3 Inches
	Reactor Low Pressure (Pump Start) Permissive	-10 psi
	Reactor Low Pressure (Pump Start) Permissive Bypass Timer	>10 min <24 min
	High Drywell Pressure	+1 psi
	Low Reactor Pressure (Valve Permissive)	-10 psi
Instrumentation That Initiates Rod Block Table 3.2.3	IRM Downscale	-2/125 of Scale
	IRM Upscale	+2/125 of Scale
	APRM Downscale	-2/125 of Scale
	APRM Upscale	See Basis 3.2
	RBM Downscale	-2/125 of Scale
	RBM Upscale	+2/125 of Scale
	Scram Discharge Volume-High Level	+ 1 gallon
Instrumentation That Initiates Recirculation Pump Trip	High Reactor Pressure	+ 12 psi
	*Low Reactor Water Level	-3 Inches
Instrumentation for Safeguards Bus Protection	Degraded Voltage	$\geq$3897 volts (trip) $\leq$3975 volts (reset) $\geq$5 sec $\leq$10 sec (delay)
	Loss of Voltage	<3000 volts >2000 volts

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
1. Except as specified in 3.5.E.2 and 3.5.E.3 below, the entire automatic pressure relief system shall be operable at any time the reactor pressure is above 150 psig and irradiated fuel is in the reactor vessel	1. Testing:
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.	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.
4. If the requirements of 3.5.E.1-3 cannot be met, an orderly reactor	

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

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

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
E. Safety/Relief Valves	E. Safety/Relief Valves
1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F:	1. a. A minimum of seven safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. The nominal self-actuation setpoints are specified in Section 2.4.B.
a. The safety valve function (self-actuation) of seven safety/relief valves shall be operable.	b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.
b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.	c. The integrity of the safety/relief valve bellows shall be continuously monitored.
c. The Low-Low Set function for three non-Automatic Pressure Relief Valves shall be operable.	d. The operability of the bellows monitoring system shall be demonstrated at least once every three months.
	2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 62 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated December 5, 1986, the Northern States Power Company (the licensee) requested amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed amendment would (a) reflect modifications made to the Auto Pressure Relief System (APRS) in response to the requirements set forth in NUREG-0737, Item II.K.3.18; and (b) revise the pressure setpoint value to the current TS for the Safety/Relief Valve Discharge Piping System pressure switches. The APRS changes result from the elimination of the high drywell pressure permissive, the addition of a bypass timer for the low reactor vessel pressure permissive of the ECCS pump start, and the addition of a manual inhibit switch for the APRS. These were plant changes previously accepted by the staff and documented in a letter to the licensee dated January 29, 1985. The Safety/Relief Valve Discharge Piping System changes reflected in the proposed TS changes are predicated on the results of the low reactor pressure tests performed by the licensee, which indicated that the original setpoint of 50 psid, specified in the current TS for the pressure switches, should be reduced in order to enhance the accident monitoring of the system. Other changes are also proposed to clarify the TS and to delete inconsistencies.

2.0 DISCUSSION AND EVALUATION

2.1 APRS(1) Logic Modification, NUREG-0737, Item II.K.3.18

The licensee proposed a revision to the TS to reflect logic changes made to implement the requirements of NUREG-0737, Item II.K.3.18. A bypass timer has been added to the reactor low pressure permissive start switch contacts for each Core Spray and Low Pressure Coolant Injection (LPCI) system, and the high pressure initiation signal has been removed from the APRS.

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(1) The APRS at Monticello is the same as the Automatic Depressurization System or ADS at other Boiling Water Reactor plants.

The following revisions to the TS are proposed:

a. On page 52, revise Table 3.2.2, "Instrumentation That Initiates Emergency Core Cooling Systems," revise Item A.1.b to include the Reactor Low Pressure Permissive and Bypass Timer. Specify a trip setting of 20 minutes plus or minus one minute for the setting of the bypass timer.

b. On Page 53, under Item C of Table 3.2.2, "Automatic Depressurization," delete item (1), "High Drywell Pressure (1)" and the word "and" between item (1) and item (2), "Low-Low Reactor Water Level." Renumber items (2) through (4) as items (1) through (3) and insert the logic connector "and" between them.

c. On page 71, add an entry in the allowable deviation table for the new reactor low pressure permissive bypass times. The minimum timer setting is specified as 10 minutes. The maximum specified timer setting is 24 minutes.

d. On page 110, add the requirement to test the automatic depressurization system inhibit switch each operating cycle.

The APRS design modification was previously accepted by the staff and its acceptance documented in a letter to the licensee (D.B. Vassallo to D. M. Musolf) dated January 29, 1985. The design modification allows the APRS to respond to a wider range of small pipe break accidents, some of which do not increase drywell pressure. Design features intended to reduce unwanted low pressure ECCS pump starts on water level transients are preserved. Spurious operation is prevented by the addition of a new timer in the logic design. Response to analyzed accidents requiring reactor depressurization is improved.

The TS changes proposed will accommodate the APRS logic modifications made in response to NUREG-0737, Item II.K.3.18, and include an additional surveillance test of the system inhibit switch to ensure operability and reduce the probability of spurious operation. We, therefore, conclude that the TS changes proposed for the modified APRS are acceptable.

2.2 Reduction in Safety/Relief Valve Discharge Pipe Pressure Switch Setpoints

Tests performed by the licensee at low reactor pressures (near 150 psig) have indicated that the original setpoint for the discharge pipe pressure switches of 50 psid may be too high for reliable detection of valve actuation. The licensee has, therefore, proposed to change this setting so that the switches will detect a safety/relief valve actuation at 30 psid in the discharge line instead of 50 psid. Because there will be no change in the operation of the low-low set logic, and the pressure switches will be more responsive at low reactor pressures for valve position indication purposes, the licensee believes that this setting change, and the related TS changes which follow item a, will enhance the accident monitoring function of the system.

The following revisions to the TS have been proposed accordingly (note that items b, c, and d are made to clarify the TS and/or to correct TS duplications/inconsistencies):

a. Change the setpoint specified on Table 3.2.7 for "Discharge Pipe Pressure Inhibit" from "50 ± 1 psid (3)" to "30 psid ± 1 psid (3)." Revise the function description by changing "Discharge Pipe Pressure Inhibit" to read, "Discharge Pipe Pressure Inhibit and Position Indication."

b. On Table 4.2.1, page 63, "SAFETY/RELIEF VALVE LOW-LOW SET LOGIC" add reference to Note 5 for items 2 through 5. Change "(8)" under "Test" for item 1 to read, "(Note 8)."

c. Add a reference to Table 4.14.1, "Minimum Test and Calibration Frequency for Accident Monitoring Instrumentation," for the "Calibration" and "Sensor Check" in lieu of the currently specified frequencies for "Discharge Pipe Pressure."

d. Revise Specification 3.6.E.2 to read:

"c. The Low-Low Set function for three non-Automatic Pressure Relief Valves shall be operable."

We have reviewed the TS changes proposed, and the licensee's basis for the changes, and find that the changes meet the requirements and guidelines set forth in Section 7.2 of the Standard Review Plan (NUREG-0800), and that the pressure switch setpoint change will ensure that the safety limits presented in Chapter 15 of the Monticello USAR will be maintained. Therefore, we conclude that the TS changes are acceptable as proposed.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes an inspection or surveillance requirement. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

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

Principal Contributors: F. Skopec, NRR
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Dated: March 31, 1989