MAR 2 1981

Distribution
Docket
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RTedesco
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SNorris
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TBarnhardt (4)
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Docket No. 50-263

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MAR 11 1981
U.S. NUCLEAR REGULATORY COMMISSION

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Mayer:

The Commission has issued the enclosed Amendment No. 2 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 12, 1980 and subsequent discussions between the NRC staff and your staff.

These changes to the Technical Specifications involve incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. These requirements concern (1) Emergency Power Supply/Inadequate Core Cooling, (2) Valve Position Indication, (3) Containment Isolation, (4) Shift Technical Advisor, (5) Systems Integrity Measurements Program, and (6) Improved Iodine Measurements Capability. This amendment also includes additions to the table of safety-related hydraulic snubbers as proposed in your December 12, 1980 application.

Copies of the Safety Evaluation and a related Notice of Issuance are also enclosed.

Sincerely,

Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 2 to DPR-22
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

P No legal objection to amendment + FR notice

8103130660

OFFICE	ORB #2	ORB #2	AD:OR	OELD	ORB #2		
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 2, 1981

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cc w/enclosures:
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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. 2
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 12, 1980, 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:

   2. Technical Specifications

      The Technical Specifications contained in Appendices A and B as revised through Amendment No. 2 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8103130666

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 2, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 2

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
iv	iv
vi	vi
50	50
61	61
62	62
63	63
65	65
131	131
132	132
174	174
-	132a
-	229a
-	229b
-	229c
-	229d
232	232
236	236
245	245

	Page
3.13 and 4.13 Fire Detection and Protection Systems	223
A. Fire Detection Instrumentation	223
B. Fire Suppression Water System	224
C. Hose Stations	226
D. Fire Barrier Penetration Fire Seals	227
3.13 Bases	228
4.13 Bases	229
3.14 and 4.14 Accident Monitoring Instrumentation	229a
3.14 and 4.14 Bases	229d
5.0 DESIGN FEATURES	230
5.1 Site	230
5.2 Reactor	230
5.3 Reactor Vessel	230
5.4 Containment	230
5.5 Fuel Storage	231
5.6 Seismic Design	231
6.0 ADMINISTRATIVE CONTROLS	232
6.1 Organization	232
6.2 Review and Audit	237
6.3 Special Inspections and Audits	243
6.4 Action to be taken if a Safety Limit is Exceeded	243
6.5 Plant Operating Procedures	244
6.6 Plant Operating Records	246
6.7 Reporting Requirements	248
6.8 Environmental Qualification	254

LIST OF TABLES

Table No.		Page
3.1.1	Reactor Protection System (Scram) Instrument Requirements	28
4.1.1	Scram Instrument Functional Tests - Minimum Functional Test Frequencies for Safety Instrumentation and Control Circuits	32
4.1.2	Scram Instrument Calibration - Minimum Calibration Frequencies for Reactor Protection Instrument Channels	34
3.2.1	Instrumentation that Initiates Primary Containment Isolation Functions	49
3.2.2	Instrumentation that Initiates Emergency Core Cooling Systems	52
3.2.3	Instrumentation that Initiates Rod Block	56
3.2.4	Instrumentation that Initiates Reactor Building Ventilation Isolation and Standby Gas Treatement System Initiation	59
3.2.5	Instrumentation that Initiates a Recirculation Pump Trip	60
4.2.1	Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation	61
3.2.6	Trip Functions and Deviations	70
3.6.1	Safety Related Snubbers	131
4.6.1	In-Service Inspection Requirements for Monticello	138
3.7.1	Primary Containment Isolation	172
4.8.1	Monticello Nuclear Plant - Environmental Monitoring Program Sample Collection and Analysis	193
3.11.1	Maximum Average Planar Linear Heat Generation Rate vs. Exposure	214
3.14.1	Instrumentation for Accident Monitoring	229b
4.14.1	Minimum Test and Calibration Frequency for Accident Monitoring Instrumentation	229c
6.1.1	Minimum Shift Crew Composition	236

TABLE 3.2.1 - Continued

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
b. High Drywell Pressure (5)	$\leq$ 2 psig	2	2	D
3. Reactor Cleanup System (Group 3)				
a. Low Reactor Water Level	$\geq$ 10'6" above the top of the active fuel	2	2	E
b. High Drywell Pressure	$\leq$ 2 psig	2	2	E
4. HPCI Steam Lines				
a. HPCI High Steam Flow	$\leq$150,000 lb/hr with $\leq$60 second time delay	2(4)	2	F
b. HPCI High Steam Flow	$\leq$300,000 lb/hr	2(4)	2	F
c. HPCI Steam Line Area High Temp.	$\leq$200°F	16(4)	16	F
5. RCIC Steam Lines				
a. RCIC High Steam Flow	$\leq$45,000 lb/hr	2(4)	2	G
b. RCIC Steam Line Area	$\leq$200°F	16(4)	16	G

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 (Note 7)	once/month	Once/3 months	Once/Shift
2. Drywell High Pressure (Note 7)	once/month	Once/3 months	None
3. Reactor Low Pressure (Pump Start)	Note 1	Once/3 months	None
4. Reactor Low Pressure (Valve Permissive)	Note 1	Once/3 months	None
5. Undervoltage Emergency Bus	Refueling Outage	Refueling Outage	None
6. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	Note 1	Once/3 months	None
7. Loss of Auxiliary Power	Refueling Outage	Refueling Outage	None
ROD BLOCKS			
1. APRM Downscale	Notes (1,5)	Once/3 months	None
2. APRM Flow Variable	Notes (1,5)	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	Notes (1,5)	Once/3 months	None
6. RBM Downscale	Notes (1,5)	Once/3 months	None
7. SRM Upscale	Notes (2,5)	Note 2	Note 2
8. SRM Detector not in Start-up Position	Note 2	Note 2	Note 2
MAIN STEAM LINE ISOLATION			
1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	None
2. Steam Line High Flow	Note 1	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 (5)	Note 6	Once/shift
HPCI ISOLATION			
1. Steam Line High Flow	once/month	Once/3 months	None
2. Steam Line High Temperature	once/month	Once/3 months	None
RCIC ISOLATION			
1. Steam Line High Flow	once/month	Once/3 months	None
2. Steam Line High Temperature	Note 1	Once/3 months	None
REACTOR BUILDING VENTILATION			
1. Radiation Monitors (Plenum)	Note 1	Once/3 months	Once/shift
2. Radiation Monitors (Refueling Floor)	Note 1	Once/3 months	(4)
OFF-GAS ISOLATION			
1. Radiation Monitors (Air Ejectors)	Notes (1,5)	Note 6	Once/shift
RECIRCULATION PUMP TRIP			
1. Reactor High Pressure	Note 1	Once/Operating Cycle- Transmitter Once/3 Months- Trip Unit	Once/Day
2. Reactor Low Water Level (Note 7)	once/month	Once/Operating Cycle- Transmitter Once/3 Months- Trip Unit	Once/Shift

NOTES:

(1) Initially once per month until exposure hours (M as defined on Figures 4.1.1) is 2.0×10^5, thereafter according to Figure 4.1.1 with an interval not greater than three months.

Table 4.2.1 - Continued

NOTES:

(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) Surveillance also to be performed on containment isolation function of this instrumentation at the specified intervals.

Bases Continued:

3.2 This trip setting level was chosen to be low enough to prevent spurious operation but high enough to initiate ECCS operation and primary [illegible] so that no melting of the fuel cladding will occur and so that post accident cooling can [illegible] and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential [illegible] a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Section 6.2.7 and 14.6.3 FSAR. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria. Reference Section 6.2.7 FSAR.

The high drywell pressure instrumentation is a back-up to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 and Group 3 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 and Group 3 isolation valves include the drywell vent, purge, sump isolation, and RWCU system valves.

Two pressure switches are provided on the discharge of each of the two core spray pumps and each of the four RHR pumps. Two trip systems are provided in the control logic such that either trip system can permit automatic depressurization. Each trip system consists of two trip logic channels such that both trip logic channels are required to permit a system trip.

Division I core spray and RHR pump discharge pressure permissives will interlock one trip system and Division II permissives will interlock the other trip system. One pressure switch on each pump will interlock one of the trip channels and the other pressure switch will interlock the other trip channel within their respective trip system.

The pump pressure permissive control logic is designed such that no single failure (short or open circuit) will prevent auto-blowdown or allow auto-blowdown when not required.

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow,

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	HIGH RADIATION AREA	DIFFICULT TO REMOVE	ACCESSIBLE -A INACCESSIBLE-I
PS1-H2	MAIN STEAM	DRYWELL	955	071			I
PS1-H3	MAIN STEAM	DRYWELL	950	148			I
PS2-H2	MAIN STEAM	DRYWELL	950	120			I
PS3-H2	MAIN STEAM	DRYWELL	950	240			I
PS4-H3	MAIN STEAM	DRYWELL	950	212			I
RV24-H3	SAFETY-RELIEF	DRYWELL	950	110			I
RV24-H4	SAFETY-RELIEF	DRYWELL	935	100			I
RV24-H4A	SAFETY-RELIEF	DRYWELL	935	100			I
RV24-H5	SAFETY-RELIEF	DRYWELL	935	110			I
RV24-N1	SAFETY-RELIEF	DRYWELL	955	090		X	I
RV24A-H4A	SAFETY-RELIEF	DRYWELL	947	048		X	I
RV24A-H7	SAFETY-RELIEF	DRYWELL	953	115			I
RV24A-H8	SAFETY-RELIEF	DRYWELL	939	032			I
RV24A-N1	SAFETY-RELIEF	DRYWELL	956	086		X	I
RV25-H1	SAFETY-RELIEF	DRYWELL	953	180			I
RV25-H1A	SAFETY-RELIEF	DRYWELL	953	180	X		I
RV25-H2	SAFETY-RELIEF	DRYWELL	948	190		X	I
RV25-H2A	SAFETY-RELIEF	DRYWELL	948	190		X	I
RV25-H3	SAFETY-RELIEF	DRYWELL	934	180	X		I
RV25A-H2	SAFETY-RELIEF	DRYWELL	945	120	X	X	I
RV25A-H2A	SAFETY-RELIEF	DRYWELL	945	120	X	X	I
RV25A-H7	SAFETY-RELIEF	DRYWELL	953	135			I
RV26-H1	SAFETY-RELIEF	DRYWELL	953	200	X		I
RV26-H1A	SAFETY-RELIEF	DRYWELL	953	200			I
RV26-H2	SAFETY-RELIEF	DRYWELL	947	200		X	I
RV26-H2A	SAFETY-RELIEF	DRYWELL	947	200			I
RV26-N1	SAFETY-RELIEF	DRYWELL	956	200	X	X	I
RV26A-H2	SAFETY-RELIEF	DRYWELL	940	250			I
RV26A-H2A	SAFETY-RELIEF	DRYWELL	935	250			I
RV26A-N1	SAFETY-RELIEF	DRYWELL	950	250	X	X	I
RV26A-N2	SAFETY-RELIEF	DRYWELL	951	250	X	X	I
RV27-H1	SAFETY-RELIEF	DRYWELL	950	320			I
RV27-H1A	SAFETY-RELIEF	DRYWELL	950	230			I
RV27-H5	SAFETY-RELIEF	DRYWELL	945	270			I
RV27-H6	SAFETY-RELIEF	DRYWELL	945	270			I
RV27-N1	SAFETY-RELIEF	DRYWELL	956	270	X	X	I
RV27A-H2A	SAFETY-RELIEF	DRYWELL	953	290			I
RV27A-H5	SAFETY-RELIEF	DRYWELL	953	290			I
RV27A-H9	SAFETY-RELIEF	DRYWELL	938	290			I
RV27A-N1	SAFETY-RELIEF	DRYWELL	956	270	X	X	I
SS-1	MAIN STEAM	DRYWELL	953	279	X		I
SS-1AR	RECIRCULATION	DRYWELL	922	315	X	X	I
SS-1BR	RECIRCULATION	DRYWELL	922	135	X	X	I
SS-11	FEEDWATER	DRYWELL	952	302			I
SS-12	FEEDWATER	DRYWELL	952	058			I
SS-13	FEEDWATER	DRYWELL	952	258			I
SS-14	FEEDWATER	DRYWELL	952	096			I

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	HIGH RADIATION AREA	DIFFICULT TO REMOVE	ACCESSIBLE -A INACCESSIBLE-I
SS-17A	RHR	DRYWELL	964	072	X		I
SS-17B	RHR	DRYWELL	964	072	X		I
SS-18A	RHR	DRYWELL	964	288			I
SS-18B	RHR	DRYWELL	964	286			I
SS-19	RHR	DRYWELL	964	341			I
SS-2	MAIN STEAM	DRYWELL	953	061	X		I
SS-2AR	RECIRCULATION	DRYWELL	927	302	X	X	I
SS-2BR	RECIRCULATION	DRYWELL	927	122		X	I
SS-2C	RHR	DRYWELL	964	019	X		I
SS-3	MAIN STEAM	DRYWELL	950	212			I
SS-3AR	RECIRCULATION	DRYWELL	927	328		X	I
SS-3BR	RECIRCULATION	DRYWELL	927	148		X	I
SS-4	MAIN STEAM	DRYWELL	950	148			I
SS-4AR(A)	RECIRCULATION	DRYWELL	934	302			I
SS-4AR(B)	RECIRCULATION	DRYWELL	934	323			I
SS-4BR(A)	RECIRCULATION	DRYWELL	934	120			I
SS-4BR(B)	RECIRCULATION	DRYWELL	934	149			I
SS-40	HPCI	MAIN STEAM CHASE					I
SS-5AR	RECIRCULATION	DRYWELL	941	315		X	I
SS-5BR	RECIRCULATION	DRYWELL	941	135		X	I
SS-6AR	RECIRCULATION	DRYWELL	953	261	X		I
SS-6BR	RECIRCULATION	DRYWELL	953	099	X		I
SS-7	MAIN STEAM	DRYWELL	953	240	X		I
SS-7AR	RECIRCULATION	DRYWELL	953	323			I
SS-7BR	RECIRCULATION	DRYWELL	953	032			I
SS-8	MAIN STEAM	DRYWELL	953	120	X		I
SS-8AR	RECIRCULATION	DRYWELL	927	270		X	I
SS-8BR	RECIRCULATION	DRYWELL	927	090		X	I
SS-21	RHR	TORUS FL LV - S WALL					A
SS-22	RHR	TORUS FL LV - S WALL					A
SS-23	RHR	B RHR ROOM FL LV					A
SS-24	RHR	A RHR ROOM FL LV					A
SS-25	RHR	TORUS CATWK-SE WALL					A
SS-26	CORE SPRAY	B RHR ROOM FL LVL					A
SS-27	CORE SPRAY	B RHR ROOM FL LVL					A
SS-28A	CORE SPRAY	A RHR ROOM FL LVL					A
SS-28B	CORE SPRAY	A RHR ROOM FL LVL					A
SS-29	RHR	OVER N2 ANALYZER	954			X	A
SS-30	RHR	OVER N2 ANALYZER	954			X	A
SS-31	RHR	TORUS CATWK					A
SS-32A	RHR	A RHR ROOM - BY HX	916			X	A
SS-32B	RHR	A RHR ROOM - BY HX	916			X	A
SS-33	RHR	ABOVE TORUS					A
SS-34	RHR	ABOVE TORUS					A
SS-35	HPCI	HPCI ROOM - N WALL	912			X	A
SS-36A	HPCI	HPCI ROOM - FL LVL					A
SS-36B	HPCI	HPCI ROOM - FL LVL					A

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	HIGH RADIATION AREA	DIFFICULT TO REMOVE	ACCESSIBLE -A INACCESSIBLE-I
SS-37	HPCI	HPCI ROOM - W WALL	905			X	A
SS-38A	RCIC	RCIC ROOM - W WALL	906			X	A
SS-38B	RCIC	RCIC ROOM - W WALL	906			X	A
SS-41	CORE SPRAY	ABOVE TORUS CATWK	927				A
SS-42	HPCI	ABOVE TORUS RING HDR	906				A

TABLE 3.7.1, continued

PRIMARY CONTAINMENT ISOLATION

Group 3: The actions in Group 3 are initiated by reactor low water level or high drywell pressure.

Group 4: Isolation valves in the high pressure coolant injection system (HPCI) are closed upon any one of the following signals:
1. HPCI steam line high flow
2. HPCI steam line low pressure
3. High temperature in the vicinity of the HPCI steam line

Group 5: Isolation valves in the reactor core isolation cooling system (RCIC) are closed upon any one of the following signals:
1. RCIC steam line high flow
2. RCIC steam line low pressure
3. High temperature in the vicinity of the RCIC steam line

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

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 ensure 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 the reactor is in the startup or run mode, 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 Water Level (Fuel Zone)	2	1	B
Safety/Relief Valve Position (One Channel Pressure Switch and One Channel Thermocouple Position Indication per Valve)	2/Valve	1/Valve	A

*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 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 inoperable channel(s) shall be restored to operable status within 30 days or be in at least Hot Shutdown within the next 12 hours.

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 inoperable channel(s) shall be restored to operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours.

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 Water Level (Fuel Zone)	-	Once/Operating Cycle	Once/month (Note 3)
Safety/Relief Valve Position (Pressure Switches)	-	Once/Operating Cycle	Once/month (Note 2)
Safety/Relief Valve Position (Thermocouples)	-	Once/Operating Cycle	Once/month (Note 2)

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) Proper instrument response shall be verified during each safety/relief valve actuation.

(3) These instruments are off-scale high during normal plant operation.

Bases:

3.14/4.14 The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Learned Task Force Status Report and Short Term Recommendations".

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

A. The Plant Manager has the overall full-time onsite responsibility for safe operation of the facility. During periods when the Plant Manager is unavailable, he may delegate this responsibility to other qualified supervisory personnel.

B. The Northern States Power corporate organizational structure relating to the operation of the plant is shown in Figure 6.1.1.

C. The minimum functional organization for operation of the plant shall be as shown in Figure 6.1.2 and:

1. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.1.1.

2. At least one licensed operator shall be in the control room when fuel is in the reactor.

3. At least two licensed operators shall be present in the control room during cold startup, scheduled reactor shutdown, and during recovery from reactor trips.

4. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

5. All alterations of the reactor core shall be directly supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

6. A fire brigade of at least three members shall be maintained on site at all times. The fire brigade shall not include the four members of the shift organization required for safe shutdown of the reactor or more than one member of the site security force.

D. Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Superintendent Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents. The training program shall be under the direction of a designated member of Northern States Power management.

TABLE 6.1.1

MINIMUM SHIFT CREW COMOSITION (Note 1)

CATEGORY	APPLICABLE PLANT CONDITIONS	
	COLD SHUTDOWN OR REFUELING OPERATION	ABOVE COLD SHUTDOWN
No. Licensed Senior Operators (LSO)	1 (Note 2)	1
Total No. Licensed Operators (LSO & LO)	2	3
Total No. Licensed and Unlicensed Operators	3	5
Shift Technical Advisor	0	1

NOTES:

1. Shift crew composition may be one less than the minimum requirements for a period of time not to exceed two hours in order to accomodate an unexpected absence of one duty shift crew member provided immediate action is taken to restore the shift crew composition to within the minimum requirements specified.

2. Does not include the licensed Senior Reactor Operator, or Senior Reactor Operator Limited to Fuel Handling, supervising alterations of the reactor core.

1. a. Paragraph 20.203 "Caution signs, lables, signals and controls." In lieu of the "Control device" or alarm signal required by paragraph 20.203(c)(2), each high radiation area in which the intensity of radiation is 1000 mRem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. The above procedure shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mRem/hr, except that locked doors shall be provided to prevent unauthorized entry into these areas and the keys to these locked doors shall be maintained under the administrative control of the Plant Manager.

2. A program shall be implemented to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

   a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

   b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

   A program acceptable to the Commission was described in a letter dated December 31, 1979, from L O Mayer, NSP, to Director of Nuclear Reactor Regulation, "Lessons Learned Implementation".

3. A program shall be implemented which will ensure the capability to accurately determine the airborne iodine concentration in essential plant areas under accident conditions. This program shall include the following:

   a. Training of personnel,

   b. Procedures for monitoring, and

   c. Provisions for maintenance of sampling and analysis equipment.

   A program acceptable to the Commission was described in a letter dated December 31, 1979, from L O Mayer, NSP, to Director of Nuclear Reactor Regulation, "Lessons Learned Implementation".

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 2 TO LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

1.0 Introduction

By letter dated December 12, 1980, Northern States Power Company (the licensee) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The requested changes would incorporate certain of the TMI-2 Lessons Learned Category "A" requirements into the Monticello Technical Specifications. The licensee's request is in direct response to the NRC staff's letter of July 2, 1980.

2.0 Background Information

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the Three Mile Island Unit 2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter to Northern States Power Company dated March 21, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their Technical Specifications to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. The licensee's application is in direct response to our request. Finally, additions to the table of hydraulic snubbers were made as part of the Mark I Containment Long Term Program. Each of the issues identified by the NRC staff and the licensee's response is discussed in the Evaluation below.

3.0 Evaluation

1. Emergency Power Supply/Inadequate Core Cooling

As applicable to Boiling Water Reactors (BWRs), we indicated that water level instrumentation is important to post accident monitoring and that surveillance of this instrumentation should be performed. The licensee

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has provided (Tables 3.14.1 and 4.14.1) Technical Specifications to include reactor vessel water level instrumentation. These specifications provide ACTION statements for inoperable channels. Surveillance requirements for sensor checks and calibration are also included. The frequency of surveillance meets our guidelines. Based on this review, we find the licensee's proposal acceptable.

2. Valve Position Indication

Our requirement for installation of a reliable position indicating system for relief and safety valves was based on the need to provide the operator with a diagnostic aid to reduce the ambiguity between indications that might indicate either an open relief/safety valve or a small line break. Such a system need not be safety grade provided that backup methods of determining valve position are available.

The licensee's request would add the primary indicating system (pressure switches) and secondary indicating system (thermocouples) for the safety relief valves to the specifications. The Monticello primary relief system employs only safety relief valves. Actions have been specified for the condition of inoperability of both primary and secondary detector channels and to provide for torus temperature monitoring in event of the inoperability of both the primary and secondary detectors. Additionally, surveillance requirements have been included which meet our guidelines. Based on our review, we find the licensee's proposed changes satisfy our guidelines and are acceptable.

3. Containment Isolation

Our request indicated that the specifications should include a Table of Containment Isolation Valves which reflect the diverse isolation signal requirement of this Lessons Learned issue. The licensee has modified the containment isolation system so that diverse parameters will be sensed to ensure automatic isolation of non-essential systems under postulated accident conditions. We have reviewed this system in our Lessons Learned Category "A" Safety Evaluation dated March 21, 1980.

The modification is such that it does not result in the automatic loss of containment isolation after the containment isolation signal is reset. Reopening of containment isolation valves would require deliberate operator action. The existing Technical Specifications and the Technical Specifications submitted by the licensee (Tables 3.2-1 and 3.7-1) list the affected valves by isolation group and the diverse signals sensed to initiate containment isolation of each valve group. Table 4.2-1 of the Monticello Appendix "A" Technical Specifications has been modified to include surveillance requirements.

We have reviewed the existing Technical Specifications and the modifications to the Monticello Technical Specifications agreed to by the licensee and have determined that these specifications satisfy our requirements and are, therefore, acceptable.

4. Shift Technical Advisor (STA)

Our request indicated that the Technical Specifications related to minimum shift manning should be revised to reflect the addition of an STA. The STA function includes both accident and operating experience assessment. The licensee proposed the addition of an STA to the minimum shift composition and the specific qualifications of the individual.

These qualifications state that the STA shall have a bachelors degree or equivalent in a scientific or engineering descipline with specific training in plant design, and response and analysis of the plant for transients and accidents. Since our position does not require degreed STAs until January 1, 1981, the licensee requested that the effective date for the STA requirement be delayed until that time.

Based on our review, we find that the licensee's submittal satisfies our requirements and is acceptable.

5. Integrity of Systems Outside Containment

Our letter dated July 2, 1980, indicated that the license should be amended by adding license conditions related to the Systems Integrity Measurements Program. This condition would require the licensee to implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels.

The licensee instead incorporated such requirements into Section 6.5 of the Monticello Technical Specifications. This program for leakage reduction includes provisions establishing preventive maintenance and periodic visual inspection requirements and for periodic systems leak requirements and thus satisfies our requirements.

6. Iodine Monitoring

Our letter dated July 2, 1980 indicated that the license should be amended by adding license conditions related to Improved Iodine Measurements capability. This condition would require the licensee to implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 2 to Facility Operating License No. DPR-22, issued to Northern States Power Company, which revised Technical Specifications for operation of the Monticello Nuclear Generating Plant (the facility) located in Wright County, Minnesota. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to incorporate certain of the TMI-2 Lessons Learned Category "A" requirements and to add additional snubbers to the table of safety related hydraulic snubbers.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

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For further details with respect to this action, see (1) the application for amendment dated December 12, 1980, and (2) Amendment No. 2 to License No. DPR-22. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Environmental Conservation Library, Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota. A copy of item (2) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing

Dated at Bethesda, Maryland, this 2nd day of March 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing