



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. 6  
License No. DPR-22

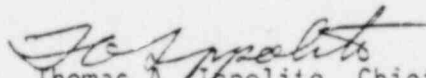
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Northern States Power Company (the licensee) dated August 30, 1977, as supplemented by letter dated March 15, 1978 and revised August 28, 1978, January 5, 1979, February 26, 1979, July 27, 1979, March 5, 1980, and July 16, 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. 6, 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

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

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: June 6, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 6

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.

<u>Remove</u>	<u>Insert</u>
ii	ii
iv	iv
vi	vi
70	70
71	71
128	128
138	138
139	139
140	140
141	141
142	142
143	143
144	144
151	151
152	152
153	153
--	229e
	229f

	<u>Page</u>
3.4 and 4.4 Standby Liquid Control System	93
A. Normal Operation	93
B. Operation with Inoperable Components	94
C. Volume-Concentration Requirements	95
3.4 and 4.4 Bases	99
3.5 and 4.5 Core and Containment Cooling Systems	101
A. Core Spray System	101
B. LPCI Subsystem	103
C. RHR Service Water System	106
D. HPCI System	108
E. Automatic Pressure Relief System	109
F. RCIC System	111
G. Minimum Core and Containment Cooling System Availability	112
H. Deleted	
I. Recirculation System	114
3.5 Bases	115
4.5 Bases	120
3.6 and 4.6 Primary System Boundary	121
A. Reactor Coolant Heatup and Cooldown	121
B. Reactor Vessel Temperature and Pressure	122
C. Coolant Chemistry	123
D. Coolant Leakage	126
E. Safety/Relief Valves	127
F. Deleted	
G. Jet Pumps	128
H. Shock Suppressors (Snubbers)	129
3.6 and 4.6 Bases	144
3.7 and 4.7 Containment Systems	156
A. Primary Containment	156
B. Standby Gas Treatment System	166
C. Secondary Containment	169
D. Primary Containment Isolation Valves	170
3.7 Bases	175
4.7 Bases	183

	<u>Page</u>
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
3.15 and 4.15 Inservice Inspection and Testing	229e
3.15 and 4.15 Bases	229f
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

**POOR ORIGINAL**

LIST OF TABLES

<u>Table No.</u>		<u>Page</u>
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	57
3.2.4	Instrumentation that Initiates Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation	59
3.2.5	Instrumentation that Initiates a Recirculation Pump Trip	60
3.2.6	Instrumentation for Safeguards Bus Degraded Voltage and Loss of Voltage Protection	60a
3.2.7	Trip Functions and Deviations	70
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
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

**POOR ORIGINAL**

Table 3.2.7  
Trip Functions And Deviations

	Trip Function	Deviation
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Specification 3.2.E.3 and Table 3.2.4	Ventilation Plenum Radiation Monitors	+0.2 Mr/Hr
	Refueling Floor Radiation Monitors	+5 Mr/Hr
	Low Reactor Water Level High Drywell Pressure	-6 inches + 1 psi
Primary Containment Isolation Functions Table 3.2.1	Low Low Water Level	-3 inches
	High Flow in Main Steam Line	+2 %
	High Temp. in Main Steam Line Tunnel	+2 <sup>o</sup> F
	Low Pressure in Main Steam Line	-10 psi
	High Drywell Pressure	+1 psi
	Low Reactor Water Level	-6 inches
	HPCI High Steam Flow	+7,500 lb/hr
	HPCI Steam Line Area High Temp.	+2 <sup>o</sup> F
	RCIC High Steam Flow	+2250 lb/hr
RCIC Steam Line Area High Temp	+2 <sup>o</sup> F	

Table 3.2.7 - Continued  
Trip Function and Deviations

	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
	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 2.3
	RBM Downscale	-2/125 of Scale
	RBM Upscale	Same as APRM Upscale
Instrumentation That Initiates Recirculation Pump Trip	High Reactor Pressure	+ 12 psi
	Low Reactor Water Level	- 3 Inches

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip settings, or, when a sufficient number of devices have been affected by any means such that the automatic function is incapable of operating within the allowable deviation while in a reactor mode in which the specified function must be operable or when actions specified are not initiated as specified.



## 3.0 LIMITING CONDITIONS FOR OPERATION

F. Deleted

G. Jet Pumps

Whenever the reactor is in the Startup or Run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the plant shall be placed in a cold shutdown condition within 24 hours.

3.6/4.6

## 4.0 SURVEILLANCE REQUIREMENTS

F. Deleted

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the Startup or Run modes, jet pump operability shall be checked daily by verifying that all the following conditions do not occur simultaneously:

1. The two recirculation loop flows are unbalanced by 15% or more when the recirculation pumps are operating at the same speed.
2. The indicated value of core flow rate is 10% or more less than the value derived from loop flow measurements.

INTENTIONALLY BLANK

INTENTIONALLY BLANK

INTENTIONALLY BLANK

INTENTIONALLY BLANK

INTENTIONALLY BLANK

INTENTIONALLY BLANK

INTENTIONALLY BLANK



Bases Continued 3.6 and 4.6:

The safety/relief valves have two functions; i.e. power relief or self-actuated by high pressure. The solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate opening of the valves. This function is discussed in Specification 3.5.E. In addition, the valves can be operated manually.

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9 of the ASME Pressure Vessel Code Section III Nuclear Vessels requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

It is realized that there is no way to repair or replace the bellows during operation and the plant must be shut down to do this. The thirty-day period to do this allows the operator flexibility to choose his time for shutdown; meanwhile, because of the redundancy present in the design and the continuing monitoring of the integrity of the other valves, the overpressure pressure protection has not been compromised. The auto-relief function would not be impaired by a failure of the bellows. However, the self-actuated overpressure safety function would be impaired by such a failure.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system quarterly provides assurance of bellows integrity.

When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

F. Deleted

POOR ORIGINAL

INTENTIONALLY BLANK

G. Jet pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within + 5%, the flow rates in both recirculation loops will be verified by Control Room monitoring instruments. If the two flow rate values do not differ by more than . . . , riser and nozzle assembly integrity has been verified. If they do differ by 10% or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10% or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

POOR ORIGINAL

## 3.0 LIMITING CONDITIONS FOR OPERATION

3.15 INSERVICE INSPECTION AND TESTINGApplicability:

Applies to components which are part of the reactor coolant pressure boundary and their supports and other safety-related pressure vessels, piping, pumps, and valves.

Objective:

To assure the integrity of the reactor coolant pressure boundary and the operational readiness of safety-related pressure vessels, piping, pumps, and valves.

Specification:A. Inservice Inspection

1. To be considered operable, Quality Group A, B, and C components shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for continued service of ASME Code Class 1, 2, and 3 components, respectively, except where relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(1).

## 4.0 SURVEILLANCE REQUIREMENTS

4.15 INSERVICE INSPECTION AND TESTINGApplicability:

Applies to the periodic inspection and testing of components which are part of the reactor coolant pressure boundary and their supports and other safety-related pressure vessels, piping, pumps, and valves.

Objective:

To verify the integrity of the reactor coolant pressure boundary and the operational readiness of safety-related pressure vessels, piping, pumps, and valves.

Specification:A. Inservice Inspection

1. Inservice inspection of Quality Group A, B, and C components shall be performed in accordance with the requirements for ASME Code Class 1, 2, and 3 components, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50; Section 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(1).

Bases 3.15 and 4.15

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

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

A program of inservice testing of Quality Group A, B, and C pumps and valves is also in effect at the Monticello plant. Technical Specifications related to this program will be issued following NRC review and approval of the pump and valve testing program.