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. 30
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 September 7, 1984, 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 Appendix A as revised through Amendment No. 30 , 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

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 16, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 30

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
i	i
vi	vi
23	23
24	24
48	48
--	60b
--	60c
63	63
--	63a
69	69
--	69a
70	70
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--	71a
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Bases Continued:

2.2 The normal operating pressure of the reactor coolant system is approximately 1010 psig. The turbine trip with failure of the bypass system represents the most severe primary system pressure increase resulting from an abnormal operational transient. The safety/relief valves (S/RV's) are sized assuming no direct scram during MSIV closure. The only scram assumed is from an indirect means (high flux). The analysis assumes that only seven of the eight S/RV's are operable and that they open at 1% over their setpoint with a 0.4 second dalay. Reactor pressure remains below the 1375 psig ASME Code limit for the reactor vessel.

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Bases:

2.4 The settings on the reactor high pressure scram, reactor coolant system safety/relief valves, turbine control valve fast closure scram, and turbine stop valve closure scram have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. The APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits. In addition to preventing power operation above 1075 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients.

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1670 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure remains below the 1375 psig ASME Code limit. Only seven of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 1% above their setpoint with a 0.4 second delay.

The operator will set the reactor coolant high pressure scram trip setting at 1075 psig or lower. However, the actual setpoint can be as much as 10 psi above the 1075 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18. In a like manner, the operator will set the reactor coolant system safety/relief valve initiation trip setting at 1108 psig or lower. However, the actual set point can be as much as 11.1 psi above the 1108 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or when a sufficient number of devices have been affected by any means

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F. Recirculation Pump Trip and Alternate Rod Injection Initiation

1. Whenever the reactor is in the RUN mode, the Limiting Conditions for Operation for the instrumentation listed in Table 3.2.5 shall be met.

G. Safeguards Bus Voltage Protection

1. Whenever the safeguards auxiliary electrical power system is required to be operable by Specification 3.9, the Limiting Conditions for Operation for the instrumentation listed in Table 3.2.6 shall be met.

H. Instrumentation for Safety/Relief Valve Low-Low Set Logic

1. Whenver the safety/relief valves are required to be operable by Specification 3.6.E, the Limiting Conditions for Operation for the instrumentation listed in Table 3.2.7 shall be met.

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 ConditionsA
Reactor Scram Detection	-	2(2)	2	2	A or B or C
Reactor Coolant System Pressure for Opening/Closing (1)	1060±3/980±3 psig 1050±3/970±3 psig 1040±3/960±3 psig	2(2)	2	2	A or B or C
Discharge Pipe Pressure Inhibit	50±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

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Table 3.2.7 (continued)

Instrumentation for Safety/Relief Valve Low-Low Set Logic

Notes:

(1) Low-low set and inhibit logic is provided for three non-Automatic Pressure Relief System Valves. The three valves have staggered setpoints as indicated.

(2) Each valve is provided with two trip, or actuation, systems.

(3) Differential pressure with respect to drywell atmosphere.

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

A) One trip system may be inoperable for testing or maintenance for up to 72 hours. If two trip systems cannot be made operable at the end of the 72 hour period, within 24 hours reduce reactor pressure to less than 110 psig and reactor water temperature to less than 345 °F.

B) With two trip systems inoperable, within 24 hours reduce reactor pressure to less than 110 psig and reactor water temperature to less than 345 °F.

C) One low-low set valve may be inoperable for testing or maintenance for up to seven days. If the valve cannot be made operable at the end of the seven day period, within 24 hours reduce reactor pressure to less than 110 psig and reactor water temperature to less than 345 °F.

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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

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
SAFETY/RELIEF VALVE LOW-LOW SET LOGIC			
1. Reactor Scram Sensing	Once/Shutdown (8)	-	-
2. Reactor Pressure - Opening	Once/3 months	Once/Operating Cycle	Once/day
3. Reactor Pressure - Closing	Once/3 months	Once/Operating Cycle	Once/day
4. Discharge Pipe Pressure	Once/3 months	Once/Operating Cycle	-
5. Inhibit Timer	Once/3 months	Once/Operating Cycle	-

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TABLE 4.2.1 - Continued
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

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.

(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.

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

Bases Continued:

increases core voiding, a negative reactivity feedback. High pressure sensors initiate the pump trip in the event of an isolation transient. Low level sensors initiate the trip on loss of feedwater (and the resulting MSIV closure). The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated after reactor isolation in this unlikely postulated event, requiring the trip to be operable only when in the RUN mode is therefore conservative.

Safety/relief valve low-low set logic is provided to prevent any safety/relief valve from opening when there is a elevated water leg in the respective discharge line. A high water leg is formed immediately following valve closure due to the vacuum formed when steam condenses in the line. If the valve reopens before the discharge line vacuum breakers act to return water level to normal, water clearing thrust loads on the discharge line may exceed their design limit. The logic reduces the opening setpoint and increases the blowdown range of three non-APRS valves following a scram. A 15-second interval between subsequent valve actuations is provided assuming one valve fails to

Bases Continued:

open and instrumentation drift has caused the nominal 80-psi blowdown range to be reduced to 60 psi. Maximum water leg clearing time has been calculated to be less than 6 seconds for the Monticello design. Inhibit timers are provided for each valve to prevent the valve from being manually opened less than 10 seconds following valve closure. Valve opening is sensed by pressure switches in the valve discharge line. Each valve is provided with two trip, or actuation, systems. Each system is provided with two channels of instrumentation for each of the above described functions. A two-out-of-two-once logic scheme ensures that no single failure will defeat the low-low set function and no single failure will cause spurious operation of a safety/relief valve. Allowable deviations are provided for each specified instrument setpoint. Setpoints within the specified allowable deviations provide assurance that subsequent safety/relief valve actuations are sufficiently spaced to allow for discharge line water leg clearing.

Although the operator will set the set points within the trip settings specified in Tables 3.2.1 through 3.2.7, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. The deviations could be caused by inherent instrument error, drift of the set point, ect. Therefore, these deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the following amounts:

	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	-6 inches
	High Drywell Pressure	+ 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	+10°F
	Low pressure in Main Steam Line	-10 psi
	High Drywall Pressure	+1 psi
	Low Reactor Water Level	-6 inches
	HPCI High Steam Flow	+7,500 lb/hr
	HPCI Steam Line Area High Temp.	+2°F
	RCIC High Steam Flow	+2250 lb/hr
	RCIC Steam Line Area High Temp	+2°F
	Shutdown Cooling Supply Iso	+7 psi

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	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
	Scram Discharge Volume-High Level	+ 1 gallon
Instrumentation That Initiates Recirculation Pump Trip	High Reactor Pressure	+ 12 psi
	Low Reactor Water Level	-3 Inches

	Trip Function	Deviation
Instrumentation for Safety/Relief Valve Low Set Logic	Reactor Coolant System Pressure for Opening/Closing	$\pm$ 20 psig
	Opening - Closing Pressure	$\geq$60 psi
	Discharge Pipe Pressure Inhibit	$\pm$ 10 psid
	Timer Inhibit	- 3 sec + 10 sec

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.

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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.

   a. The safety valve function (self-actuation) of seven safety/relief valves shall be operable.

   b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.

2. The Low-Low Set function for three non-Automatic Pressure Relief valves shall be Operable as specified in Section 3.2.1.

E. Safety/Relief Valves

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.

   b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.

   c. The integrity of the safety/relief valve bellows shall be continuously monitored.

   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.

D. Coolant Leakage

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached. Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. Systems connected to the reactor coolant system boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period. Other leakage detection methods provide warning of abnormal leakage and are not directly calibrated to provide leak rate measurements.

E. Safety/Relief Valves

Testing of all safety/relief valves each refueling outage ensures that any valve deterioration is detected. A tolerance value of 1% for safety/relief valve setpoints is specified in Section III of the ASME Boiler and Pressure Vessel Code. Analyses have been performed with all valves assumed set 1% high. As discussed in the Section 2.2 Bases, the 1375 psig Code limit is not exceeded in any case.

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.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

1. Deleted

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d. During reactor isolation conditions the reactor pressure vessel shall be depressurized to $<$ 200 psig at normal cooldown rates if the suppression pool temperature exceeds 120°F.

e. The suppression chamber water volume shall be $\geq$ 68,000 and $\leq$ 72,910 cubic feet.

f. Two channels of torus water level instrumentation shall be operable. From and after the date that one channel is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 30 days unless such channel is sooner made operable. If both channels are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding six hours unless at least one channel is sooner made operable.

2. Primary Containment Integrity

Primary containment integrity, as defined in Section 1, shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 Mw(t).

d. Whenever there is indication of relief valve operation with a suppression pool temperature $\geq$ 160°F and the primary coolant system pressure > 200 psig, an extended visual examination of the suppression chamber shall be conducted before resuming power operation.

e. The suppression chamber water volume shall be checked once per day.

f. The suppression chamber water volume indicators shall be calibrated semi-annually.

2. Primary Containment Integrity

a. Integrated Primary Containment Leak Test (IPCLT)

The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972:

1. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 $\pm$ 10 month intervals during shutdown at $\overline{P}_a$ (41 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.*

*The third test of the first 10-year service period shall be conducted during the 1980 refueling shutdown. The first test of the second 10-year period shall be conducted during the 1984 refueling shutdown.