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. 76
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 October 4, 1990, 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. 76, 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

L B Marsh

L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 19, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 76

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
39	39
127	127
151	151
169	169
189	189
198b	198b

3.1 The IRMs are calibrated by the heat balance method such that 120/125 of full scale on the highest IRM range is below 20% of rated neutron flux (see Specification 2.3.A.2). The requirement that the IRM detectors be inserted in the core assures that the heat balance calibration is not invalidated by the withdrawal of the detector.

Although the operator will set the set points within the trip settings specified on Table 3.1.1, 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, operator setting error, drift of the set point, etc. Therefore, such deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the following amounts:

Trip Function	Deviation	Trip Function	Deviation
3. High Flux IRM	+2/125 of scale	*7. Reactor Low Water Level	-6 inches
5. High Reactor Pressure	+10 psi	8. Scram Discharge Volume High Level	+1 gallon
6. High Drywell Pressure	+1 psi	9. Turbine Condenser Low Vacuum	-1/2 in. Hg

* This indication is reactor coolant temperature sensitive. The calibration is thus made for rated conditions. The level error at low pressures and temperatures is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap, and not the indicated level.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or 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 the actions specified in 3.1.B.2 are not initiated as specified.

If an unsafe failure is detected during surveillance testing, it is desirable to determine as soon as possible if other failures of a similar type have occurred and whether the particular function involved is still operable or capable of meeting the single failure criterion. To meet the requirements of Table 3.1.1, it is necessary that all instrument channels in one trip system be operable

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
E. Safety/Relief Valves	E. Safety/Relief Valves

3.0 LIMITING CONDITIONS FOR OPERATION

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.

   c. The Low-Low Set function for three non-Automatic Pressure Relief Valves shall be operable as required by Specification 3.2.H.

2. If Specification 3.6 E.1.a is not met, initiate an orderly shutdown and have reactor coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

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.

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.

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

I. Deleted

C. Secondary Containment

1. Except as specified in 3.7.C.2 and 3.7.C.3, Secondary Containment Integrity shall be maintained during all modes of plant operation.

2. Secondary Containment Integrity is not required when all of the following conditions are satisfied:

   a. The reactor is subcritical and Specification 3.3.A is met.

   b. The reactor water temperature is below 212°.

   c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A

   d. The fuel cask or irradiated fuel is not being moved within the reactor building.

3. With an inoperable secondary containment isolation damper, restore the inoperable damper to operable status or isolate the affected duct by use of a closed damper or blind flange within eight hours.

4. If Specifications 3.7.C.1 through 3.7.C.3 cannot be met, initiate a normal orderly shutdown and have the reactor in the Cold Shutdown condition within 24 hours. Alterations of the

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

   a. Secondary containment capability to maintain at least a 1/4 inch of water vacuum under calm wind ($2 < u < 5$ mph) conditions with a filter train flow rate of $\leq$4,000 scfm, shall be demonstrated at each refueling outage prior to refueling. Verification that each automatic damper actuates to its isolation position shall be performed at each refueling outage and after maintenance, repair or replacement work is performed on the damper or its associated actuator, control circuit, or power circuit.

Bases Continued:

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system whose failure could result in uncovering the reactor core are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified in USAR Table 5.2-3b are adequate to prevent loss of more coolant from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, this isolation valve closure time is sufficient to prevent uncovering the core.

The primary containment isolation valves are highly reliable, have low service requirement, and most are normally closed. The initiating sensor and associated trip channels are also checked to demonstrate the capability for automatic isolation. Reference Section 5.2.2.5.3 and Table 5-2-3b USAR. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a more reliable system.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
4. Offgas Treatment System	4. Offgas Treatment System
a. The offgas treatment system shall be in operation whenever the main condenser air ejector system is in operation. Components of the system shall be operated to provide the maximum holdup time obtainable except during periods of equipment maintenance.	a. Following each isotopic analysis of a sample of gases from the steam jet air ejector required by 4.8.B.5, verify that the maximum storage tank activity limit specified in 3.8.B.4.e cannot be exceeded using the method in the ODCM.
b. With gaseous waste being discharged for more than 7 days with a holdup time of less than 50 hours, within 30 days submit to the Commission a special report which includes the following information:	
1. Identification of equipment or sub-systems not functional and the reason.	
2 Action(s) taken to restore equipment to functional status.	
3. Summary description of action(s) taken to prevent a recurrence.	