UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 107
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 February 12, 1999, 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. 107 , 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 its date of issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 24, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 107

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
123	123
124	124
147	147
148	148
158	158
175	175
	175a

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
4. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head are ≥70°F.	4. When the reactor vessel head studs are under tension and the reactor is in the Cold Shutdown Condition, the reactor vessel shell flange temperature shall be permanently recorded.
C. Coolant Chemistry	C. Coolant Chemistry
1. (a) The steady state radioiodine concentration in the reactor coolant shall not exceed 0.25 microcuries of I-131 dose equivalent per gram of water.	1. (a) A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactive iodines of I-131 through I-135 during power operation.
(b) The steady state radioiodine concentration in the reactor coolant shall not exceed 0.02 microcuries of I-131 dose equivalent per gram of water when the reactor coolant temperature is > 212°F, the reactor is not critical, and primary containment integrity has not been established.	(b) A sample of reactor coolant shall be taken and analyzed for radioactive iodines of I-131 through I-135 within 24 hours prior to raising the reactor coolant temperature > 212°F, with the reactor not critical, and with primary containment integrity not established.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	(c) Where steam jet air ejector monitors indicate an increase in radioactive gaseous effluents of 25 percent or 5000 μCi/sec, whichever is greater, during steady state reactor operation a reactor coolant sample shall be taken and analyzed for radioactive iodines.
	(d) Isotopic analysis of reactor coolant samples shall be made at least once per month.
	(e) Whenever the steady state radioiodine concentration of prior operation is greater than 1 percent but less than 10 percent of Specification 3.6.C.1.(a) a sample of reactor coolant shall be taken within 24 hours of any reactor startup and analyzed for radioactive iodines of I-131 through I-135.
	(f) Whenever the steady state radioiodine concentration of prior operation is greater than 10 percent of Section 3.6.C.1.(a) a sample of reactor coolant shall be taken daily and prior to any reactor startup and analyzed for radioactive iodines of I-131 through I-135 as well as the coolant sample and analyses required by Specification 4.6.C.1.(e) above.

Bases 3.6/4.6 (Continued):

The requirements for cold bolt-up of the reactor vessel closure are based on the NDT temperature plus 60°F which is derived from the requirements of the ASME Boiler and Pressure Vessel Code to which the vessel was built. The NDT temperature of the closure flanges, adjacent head and shell material, and stud material is a maximum of 10°F. The minimum temperature for bolt-up is therefore 10° + 60° = 70°F. The neutron radiation fluence at the closure flanges is well below 10^{17} n/cm^2 (E>1 MEV) and therefore radiation effects will be minor and will not influence this temperature.

10 CFR 50 Appendix G requires that pressure tests and leak tests of the reactor vessel required by Section XI of the ASME Code must be completed before the core is critical. This requirement does not apply to component specific leakage tests that do not coincide with reactor vessel hydrostatic or leakage tests and would be required for repair/replacement activities such as change out of a safety relief valve topworks or replacement of a control rod drive assembly. Component specific leakage tests are only used after reactor vessel integrity has been previously demonstrated by a reactor vessel hydrostatic or leakage test. Since component specific leakage tests are not tests required for the reactor vessel, they may be performed during or after reactor startup at nominal operating pressures and temperatures.

Bases 3.6/4.6 (Continued):

C. Coolant Chemistry

In the event of a main steam line break outside primary containment, calculations show the resultant radiological dose at the exclusion area boundary to be less than 10% of the dose guidelines of 10 CFR 100. This dose was calculated on the basis of the radioiodine concentration limit of 2 μCi of I-131 dose equivalent per gram of water. In the event of a postulated high energy line break in the RWCU system outside the drywell, calculations show the resultant radiological dose at the exclusion area boundary to be less than 10% of the dose guidelines of 10 CFR 100. This dose was calculated on the basis of the radioiodine concentration limit of 0.25 μCi of I-131 dose equivalent per gram of water. In the event of a large primary system break in primary containment during a reactor vessel hydrostatic or leakage test with the reactor coolant temperature >212°F, the reactor not critical, and primary containment integrity not established, calculations show the resultant radiological dose at the exclusion area boundary to be conservatively bounded by the dose calculated for a main steam line break outside primary containment. This dose was calculated on the basis of the radioiodine concentration limit of 0.02 μCi of I-131 dose equivalent per gram of water

The reactor coolant sample will be used to assure that the limit of Specification 3.6.C.1(a) is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

Whenever an isotopic analysis is performed, a reasonable effort will be made to determine a significant percentage of those contributors representing the total radioactivity in the reactor coolant sample. Usually at least 80 percent of the total gamma radioactivity can be identified by the isotopic analysis.

It has been observed that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations such as reactor shutdown, reactor power changes, and reactor startup if failed fuel is present. As specified, additional reactor coolant samples shall be taken and analyzed for reactor operations in which steady state radioiodine concentrations in the reactor coolant indicate various levels of iodine releases from the fuel. Since the radioiodine concentration in the reactor coolant is not continuously measured, reactor coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam line.

Materials in the primary system are primarily 304 stainless steel and zircaloy. The reactor water chemistry limits are established to prevent damage to these materials. The limit placed on chloride concentration is to prevent stress corrosion cracking of the stainless steel.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. Primary Containment Integrity	2. Primary Containment Integrity
a. (1) 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 as specified in 3.7.A.2.a.(2) or 3.7.A.2.a.(3).	a. Primary Containment Integrity shall be demonstrated after each closing of each penetration subject to Type B testing, if opened following a Type A or Type B test, by leak rate testing the seal with gas at ≥ Pa, 42 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirements 4.7.A.2.b for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.6La.
(2) Primary Containment Integrity is not required when performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).	
(3) Primary Containment Integrity is not required when performing reactor vessel hydrostatic or leakage tests with the reactor not critical.	
(4) If requirements of 3.7.A.2.a.(1) cannot be met, restore Primary Containment Integrity within one hour or be in at least Hot Shutdown within the next 12 hours and Cold Shutdown within the following 24 hours.	

Bases 3.7:

A. Primary Containment

The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the offsite doses to values less than 10 CFR 100 guideline values in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. Two exceptions are made to this requirement. The first exception is during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit incremental control worth to less than 1.3% delta k. A drop of a 1.3% delta k increment of a rod does not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offers a sufficient barrier to keep offsite doses well within 10 CFR 100 guide line values. The second exception is during the performance of reactor vessel hydrostatic and leakage tests per Section XI of the ASME Code where establishing primary containment integrity would restrict access to the reactor vessel head for performance of required inspections. The reactor vessel hydrostatic and leakage tests are performed with the reactor vessel nearly water solid, at nominal operating pressure, not critical and at low decay heat values. However, the minimum reactor coolant temperature required for these tests as identified in Section 3.6.B can be greater than 212°F which provides the potential for steam, rather than water, leaks. In the unlikely event of a large primary system break with primary containment open to secondary containment the positive pressure created in secondary containment could result in a ground level radiological release to the environment. A limit on reactor coolant activity ensures that the potential resultant radiological dose at the exclusion area boundary will be conservatively bounded by the dose calculated for a main steam line break outside primary containment. In addition, low pressure emergency core cooling systems would be required to be operable during reactor vessel hydrostatic and leakage tests with reactor coolant temperature greater than 212°F providing assurance that adequate core cooling could be achieved to preclude fuel failures and subsequent increase in reactor coolant activity.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat release during primary system blowdown from 1000 psig.

Bases 3.7/4.7 (Continued):

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the maximum allowable primary containment pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. See USAR Section 5.2.3.2.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 42 psig which is below the allowable pressure of 62 psig.