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

OCT 6 1975

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 13
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 March 24, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-22 is hereby amended to read as follows:

"B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 21."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Karl R. Goller / for

Roger S. Boyd, Acting Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 21 to the
Technical Specifications

Date of Issuance: OCT 6 1975

ATTACHMENT TO AMENDMENT NO. 13

CHANGE NO. 21 TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

The Technical Specifications contained in Appendix A, attached to Provisional Operating License No. DPR-22, are hereby changed by replacing pages 139, 140, 157 and 161 with revised pages bearing the same numbers and additional pages 157A and 161A. Changed areas on the revised pages are reflected by marginal lines.

LIMITING CONDITIONS FOR OPERATION

CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment.

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, except as permitted by specification 3.5.G.4, the suppression pool water volume and temperature shall be maintained within the following limits.

    (a) Maximum Water Temperature during normal operation 90°F.

    (b) Maximum Water Temperature during any test operation which adds heat to the suppression pool - 100°F and shall not be above 90°F for more than 24 hours.

    (c) If Torus Water Temperature exceeds 110°F, initiate an immediate scram of the reactor. Power operation shall not be resumed until the pool temperature is

4.0 SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment.

1. The suppression chamber water level and temperature shall be checked once per day. A visual inspection of the suppression
21 chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage. Whenever there is indication of relief valve operation which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an extended visual examination of the suppression chamber shall be conducted before resuming power operation.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

21 (d) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the torus water temperature exceeds 120°F.

(e) Minimum Water Volume 68,000 cubic feet.

(f) Maximum Water Volume 77,970 cubic feet.

2. Primary containment integrity as defined in the 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).

2. The primary containment integrity shall be demonstrated as follows:

(a) Integrated Primary Containment Leak Test (IPCLT)

(1) An integrated leak rate test shall be performed prior to initial unit operation at an initial test pressure (Pt) of 41 psig.

(2) Subsequent leak rate tests shall be performed without preliminary leak detection surveys or leak repairs immediately prior to or during the test, at an initial pressure of approximately 41 psig.

(3) Leak repairs, if necessary to permit integrated leak rate testing, shall be preceded by local leak rate measurements where possible. The leak rate differ-

Bases Continued:

3.7 A. Primary Containment

length of four feet, which resulted in complete condensation. Thus with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humboldt Bay [1] and Bodega Bay [2] tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

For an initial maximum suppression chamber water temperature of 90°F and assuming the normal complement of containment cooling pumps (2 LPCI pumps and 2 containment cooling service water pumps), containment pressure is not required to maintain adequate net positive suction head (NPSH) for the core spray, LPCI and HPCI pumps. However, during an approximately one-day period starting a few hours after a loss-of-coolant accident, should one RHR loop be inoperable and should the containment pressure be reduced to atmospheric pressure through any means, adequate NPSH would not be available. Since an extremely degraded condition must exist, the period of vulnerability to this event is restricted by Specification 3.7.A.1.b by limiting the suppression pool initial temperature and the period of operation with one inoperable RHR loop.

(1) Robbins, C. H., "Tests of Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

Bases Continued:

3.7 A. Primary Containment

If a loss of coolant accident were to occur when the reactor water temperature is below 330°F, the containment pressure will not exceed the 62 psig design pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperatures above 212°F provides additional margin above that available at 330°F.

Bases:

4.7 A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a weekly check of the temperature and volume is adequate to assure that adequate heat removal capability is present. For additional margin, these will be checked once per day.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact and is not deteriorating. Experience with this type of paint indicates that the inspection interval is adequate.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature
21 normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress. Visual inspection of the suppression chamber including water line regions each refueling outage is adequate to detect any changes in the suppression chamber structures.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss of coolant accident. The peak drywell pressure would be about 41 psig, which would rapidly reduce to 25 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 25 psig within 10 seconds, equalizes with drywell pressure and thereafter rapidly decays with the drywell pressure decay. See Section 5.2.3 FSAR.

The design pressure of the drywell and absorption chamber is 56 psig. See Section 5.2.3 FSAR. The design leak rate is 0.5%/day at a pressure of 56 psig. As indicated above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

Bases Continued:

4.7 A. Primary Containment

The design basis loss of coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5% day at 41 psig. The analysis showed that with this leak