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. 98
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 January 23, 1997, as supplemented January 28, March 4, June 19, July 2, July 16 (2 letters), July 21, and July 25, 1997, 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 adding paragraph 2.C.8 to Facility Operating License No. DPR-22 as follows:

   2.C.8. Additional Conditions

   The Additional Conditions contained in Appendix C, as revised through Amendment No. 98, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of its date of issuance. Implementation is as specified in Appendix C.

FOR THE NUCLEAR REGULATORY COMMISSION

Tae Kim

Tae Kim, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments: 1. Page 5 of the License*
2. Bases pages 112, 113, and 176

Date of Issuance:

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*Page 5 and Appendix C of the license are attached, for convenience, for the composite license to reflect this change.

8. Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 98 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

D. Northern States Power Company shall immediately notify the NRC of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

E. Northern States Power Company shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

F. The licensee shall observe such standards and requirements for the protection of the environment as are validly imposed pursuant to authority established under Federal and State law and as determined by the Commission to be applicable to the facility covered by this facility operating license.

G. This license is effective as of the date of issuance and shall expire at midnight, September 8, 2010.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by: Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Attachments:
1. Appendix A - Technical Specifications
2. Appendix B - (Deleted per Amendment 15, 12/17/82)
3. Appendix C - Additional Conditions

Date of Issuance: January 9, 1981

APPENDIX C

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-22

Northern States Power Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
98	The emergency operating procedures (EOPs) shall be changed to require manual isolation of torus and drywell sprays prior to the point where primary containment pressure would not provide adequate net positive suction head (NPSH) for the emergency core cooling system (ECCS) pumps, change the caution statement regarding NPSH in the Primary Containment Pressure EOP to include the core spray pumps, and add a caution statement regarding NPSH considerations for pressure control while venting to control primary containment pressure.	Prior to starting up from the current maintenance outage, or August 1, 1997, whichever is later.
98	Finalize the additional containment analysis and associated NPSH evaluation which extends the existing long-term case evaluation to the time when the required containment overpressure returns to atmospheric conditions. Changes to the requested long-term containment overpressure, if any, shall be promptly reported to the NRC prior to starting up the unit from the current maintenance outage.	Prior to starting up from the current maintenance outage, or August 1, 1997, whichever is later.
98	Submit the results of the additional containment analysis and associated NPSH evaluation discussed above.	Within 90 days from the date of plant startup from the current maintenance outage, or November 1, 1997, whichever is later.

Amendment Number	Additional Condition	Implementation Date
98	Update Section 5.2 of the Updated Safety Analysis Report by incorporating Figure E.2 of the NSP submittal dated July 16, 1997.	Within 90 days from the date of plant startup from the current maintenance outage, or November 1, 1997, whichever is later.
98	Process a 10 CFR 50.59 evaluation to change the EOP definition of adequate core cooling to 2/3 core height. The corresponding EOP changes and the required operator training shall also be completed. Final implementations shall be completed when all the 10 CFR 50.59 evaluation requirements are satisfied.	Within 180 days from the date of plant startup from the current maintenance outage, or February 1, 1998, whichever is later.

automatically controls three selected safety-relief valves although the safety analysis only takes credit for two valves. It is therefore appropriate to permit one valve to be out-of-service for up to 7 days without materially reducing system reliability.

B. RHR Intertie Line

An intertie line is provided to connect the RHR suction line with the two RHR loop return lines. This four-inch line is equipped with three isolation valves. The purpose of this line is to reduce the potential for water hammer in the recirculation and RHR system when required to cooldown with an isolated or idle recirculation system. The isolation valves are opened during a cooldown to ensure a uniform cooldown of the RHR injection piping. If one recirculation loop is isolated or idle, these valves and associated piping allow the operating loop to cool the isolated or idle loop. The RHR loop return line isolation valves receive a closure signal on LPCI initiation. In the event of an inoperable return line isolation valve, there is a potential for some of the LPCI flow to be diverted to the broken loop during a loss of coolant accident. Surveillance requirements have been established to periodically cycle the RHR intertie line isolation valves. In the event of an inoperable RHR loop return line isolation valve, either the inoperable valve is closed or the other two isolation valves are closed to prevent diversion of LPCI flow. The RHR intertie line flow is not permitted in the Run Mode to eliminate 1) the need to compensate for the small change in jet pump drive flow or 2) a reduction in core flow during a loss of coolant accident.

C. Containment Spray/Cooling Systems

Two containment spray/cooling subsystems of the RHR system are provided to remove heat energy from the containment and control torus and drywell pressure in the event of a loss of coolant accident. A containment spray/cooling subsystem consists of 2 RHR service water pumps, a RHR heat exchanger, 2 RHR pumps, and valves and piping necessary for Torus Cooling and Drywell Spray. Torus Spray is not considered part of a containment spray/cooling subsystem. Placing a containment spray/cooling subsystem into operation following a loss of coolant accident is a manual operation.

The most degraded condition for long term containment heat removal following the design basis loss of coolant accident results from the loss of one diesel generator. Under these conditions, only one RHR pump and one RHR service water pump in the redundant division can be used for containment spray/cooling. The containment temperature and pressure have been analyzed under these conditions assuming service water and initial suppression pool temperature are both 90°F. Acceptable margins to containment design conditions have been demonstrated. Therefore the containment spray/cooling system is more than ample to provide the required heat removal capability. Refer to USAR Sections 5.2.3.3, 6.2.3.2.3, and 8.4.1.3.

During normal plant operation, the containment spray/cooling system provides cooling of the suppression pool water to maintain temperature within the limits specified in Specification 3.7.A.1.

The surveillance requirements provide adequate assurance that the containment spray/cooling system will be operable when required. The head and flow requirements specified for the RHR service water pumps provide assurance that the minimum required service water flow can be supplied to the RHR heat exchangers for the most degraded condition for long-term containment heat removal following the design basis loss of coolant accident.

D. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor isolated from the turbine and when the feedwater system is not available. The pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therefore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

The surveillance requirements provide adequate assurance that the RCIC system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

E. Cold Shutdown and Refueling Requirements

The purpose of Specification 3.5.E is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment spray/cooling subsystems may be out of service. This specification allows all core and containment spray/cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.E.2 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may by necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

Bases Continued:

Vent system downcomer submergence is three feet below the minimum specified suppression pool water level. This length has been shown to result in reduced postulated accident loading of the torus while at the same time assuring the downcomers remain submerged under all seismic and accident conditions and possess adequate condensation effectiveness.[3]

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 conditions which lead to minimum containment pressure, adequate net positive suction head (NPSH) is maintained for the core spray, RHR, and HPCI pumps under loss of coolant accident conditions. Analyses were performed for a broad range of pump combinations and failure modes to define the minimum amount of containment pressure available to provide adequate NPSH in the short and long term. Refer to Section 5.2.3.3 of the USAR for a discussion of these analyses and figures which demonstrate graphically the amount of pressure required and the minimum containment pressure available to supply the required NPSH for the emergency core cooling pumps in the limiting pump combinations evaluated. No pump cavitation will occur over either the short or long term periods under conditions resulting in minimum containment pressure.

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(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.

(3) General Electric NEDE-21885-P, "Mark I Containment Program Downcomer Reduced Submergence Functional Assessment Report," June, 1978.