Docket No. 50-263 November 5, 1981

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Mayer:

The Commission has issued the enclosed Amendment No. 8 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. This amendment consists of changes to the Technical Specifications in response to your application dated June 4, 1981.

These changes to the Technical Specifications reflect plant modifications being made as part of the Mark I Containment Long-Term Program.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Kenneth T. Eccleston, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 8 to DPR-22
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosure
See next page

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Mr. L. O. Mayer
Northern States Power Company

cc:

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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. 8
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 June 4, 1981 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 Appendices A and B as revised through Amendment No. 8 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

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 5, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 8

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
164	164
175	175
176	176
178	178

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

a. When primary containment is required, all eight drywell-suppression chamber vacuum breakers shall be operable and positioned in the closed position as indicated by the position indication system, except during testing and except as specified in 3.7.A. 4.b and c below.

b. Any drywell-suppression chamber vacuum breaker may be nonfully closed as indicated by the position indication and alarm systems provided that drywell to suppression chamber differential pressure decay does not exceed that shown on Figure 3.7.1.

c. Up to two drywell-suppression chamber vacuum breakers may be inoperable provided that: (1) the vacuum breakers are determined to be fully closed and at least one position alarm circuit is operable or (2) the vacuum breaker is secured in the closed position or replaced by a blank flange.

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

a. Operability and full closure of the drywell-suppression chamber vacuum breakers shall be verified by performance of the following:

(1) Monthly each operable drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle.

(2) Once each operating cycle, drywell to suppression chamber leakage shall be demonstrated to be less than that equivalent to a one-inch diameter orifice and each vacuum breaker shall be visually inspected. (Containment access required)

(3) Once each operating cycle, vacuum breaker position indication and alarm systems shall be calibrated and functionally tested. (Containment access required)

(4) Once each operating cycle, the vacuum breakers shall be tested to determine that the force required to open each valve from fully closed to fully open does not exceed that equivalent to 0.5 psi acting on the suppression chamber face of the valve disc. (Containment access required)

Bases:

3.7 A. Primary Containment

The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the off-site 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. An exception is made to this requirement 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 off-site doses well within 10 CFR 100 guide line values.

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 released during primary system blowdown from 1000 psig.

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. Reference 5.2.3 FSAR.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 41 psig which is below the allowable pressure of 62 psig.

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 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.

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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.

Bases Continued:

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and between the suppression chamber and reactor building during a loss of coolant accident so that structural integrity of the containment is maintained.

The vacuum relief system between the pressure suppression chamber and reactor building consist of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 1 psi. The external design pressure is 2 psig. One valve may be out of service for repairs for a period of seven days. This period is based on the low probability that system redundancy would be required during this time. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

The capacity of the drywell vacuum relief valves is sufficient to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to less than the design limit of 2 psi. Capacity of the vacuum relief valves has been confirmed using a sizing model developed in conjunction with the Mark I Containment Long Term Program.(2) With six of the eight valves operable, the pressure differential is limited to less than 2 psi and containment integrity is assured.

In addition to the above considerations, postulated leakage through the vacuum breaker to the suppression chamber air space could result in a partial bypass of pressure suppression in the event of a LOCA or a small or intermediate steam leak. This effect could potentially result in exceeding containment design pressure. As a result of the leakage potential, the containment response has been analyzed for a number of postulated conditions. It was found that the maximum allowable bypass area for any postulated break size was equivalent to a six-inch diameter opening.(1) This bypass corresponds to a

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(1) Report on Torus to Drywell Vacuum Breaker Tests and Modifications for Monticello Nuclear Generating Plant, dated March 12, 1973, submitted to Mr D J Skovholt, AEC-DL, from Mr L O Mayer, NSP.

(2) "Monticello Torus-to-Drywell Vacuum Breaker Requirements," Nutech, Inc, December, 1980, submited as Exhibit C, Northern States Power Company License Amendment Request dated June 4, 1981.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 8 TO LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

1.0 Introduction

By letter dated June 4, 1981 Northern States Power Company (the licensee) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The requested changes reflect plant modifications being made as part of the Mark I Containment Long-Term Program.

2.0 Background Information

Monticello currently has ten 18-inch Atwood and Morrill vacuum breakers on the end of the eight vent lines in the pressure suppression chamber (torus). Six of these vent lines have one vacuum breaker and two of the vent lines have two vacuum breakers. As part of the Mark I Containment Long-Term Program, the licensee has reevaluated vacuum breaker sizing requirements. The licensee determined that the limiting transient occurs with both drywell sprays initiated simultaneously in a steam filled drywell (following onset of a LOCA). Northern States Power Company also determined that six vacuum breakers would keep the torus to drywell differential pressure well below the two psid design pressure. Based on this finding, the licensee proposed changing the Technical Specifications to require that eight vacuum breakers be operable under normal conditions in lieu of the current requirement of ten operable vacuum breakers.

3.0 Evaluation

1. Torus - Drywell Vacuum Breakers

The licensee has evaluated the following cases: (1) inadvertent spray operation; (2) drywell spray following onset of a LOCA; and (3) vessel reflood through the postulated break. The licensee determined that the limiting transient occurs for drywell spray following onset of a LOCA and that six vacuum breakers would keep torus-to-drywell differential pressure well below the 2 psid design limit.

We have performed independent confirmatory calculations and agree with the licensee's conclusion that six vacuum breakers are sufficient to keep the torus to drywell differential pressure below the design value.

Based on this finding, we conclude it would be preferable for the licensee to mount only eight vacuum breakers on the vent lines in the torus instead of the ten vacuum breakers called for in the existing Technical Specifications. Reducing the total number of vacuum breakers from ten to eight would also reduce the potential for drywell-torus bypass leakage. Accordingly, we find the licensee's proposed Technical Specification changes acceptable.

4.0 Environmental Considerations

We have determined that the amendment does not involve a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

5.0 Conclusions

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a singificant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 5, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 8 to Facility Operating License No. DPR-22, issued to Northern States Power Company, which revised Technical Specifications for operation of the Monticello Nuclear Generating Plant (the facility) located in Wright County, Minnesota. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to reflect plant modifications being made as part of the Mark I Containment Long-Term Program.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated June 4, 1981 and (2) Amendment No. 8 to License No. DPR-22. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Environmental Conservation Library, Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota. A copy of item (2) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 5th day of November 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T.A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing