November 25, 1987

Docket No. 50-263

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Mr. D. M. Musolf, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has issued the enclosed Amendment No. 55 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. This amendment is in response to your application dated May 1, 1986, as revised July 15 and October 7, 1987.

The amendment revises the Technical Specifications (TS) to conform to the NRC Standard Technical Specifications for Appendix J testing, including the staff-approved modifications and exemptions. The changes also clarify and eliminate a number of interpretation problems. Specifically, the amendment revises the wording of TS Section 4.7.A.2, "Primary Containment Integrity," and associated bases to conform to the wording of NRC Standard TS (NUREG-0123). The amendment also (1) changes the airlock testing requirements for Type B testing; (2) increases the TS value of Pa, Peak Containment Accident Pressure, from 41 psig to 42 psig; (3) deletes the requirement for inerting system makeup monitoring as specified in Section 4.7.A.2.6; (4) revises the Bases for Sections 3.7 and 4.7 to reflect the above changes; and (5) adds action statements consistent with NUREG-0123 to Section 3.7.A.2 on containment integrity limiting condition for operation.

A copy of our related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

The issuance of this amendment completes our work effort under TAC No. 61448.

Sincerely,

/S/

Dominic C. DiIanni, Project Manager
Project Directorate III-3
Division of Reactor Projects

Enclosures:
1. Amendment No. 55 to License No. DPR-22
2. Safety Evaluation

cc w/enclosures:
See next page

*SEE PREVIOUS CONCURRENCE

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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. 55
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 May 1, 1986, as supplemented July 15 and October 7, 1987 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 Appendix A, as revised through Amendment No.55 , 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

Kenneth E. Perkins

Kenneth E. Perkins, Director
Project Directorate III-3
Division of Reactor Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 25, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 55

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
157	157
158	158
159	159
160	160
161	161
162	--
175	175
183	183
184	184
185	--
186	--
187	--

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

d. During reactor isolation conditions the reactor pressure vessel shall be depressurized to <200 psig at normal cooldown rates if the suppression pool temperature exceeds 120°F.

e. The suppression chamber water volume shall be ≥68,000 and ≤72,910 cubic feet.

f. Two channels of torus water level instrumentation shall be operable. From and after the date that one channel is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 30 days unless such channel is sooner made operable. If both channels are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding six hours unless at least one channel is sooner made operable.

4.0 SURVEILLANCE REQUIREMENTS

d. Whenever there is indication of relief valve operation with a suppression pool temperature of ≥160°F and the primary coolant system pressure >200 psig, an extended visual examination of the suppression chamber shall be conducted before resuming power operation.

e. The suppression chamber water volume shall be checked once per day.

f. The suppression chamber water volume indicators shall be calibrated semi-annually.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. **Primary Containment Integrity** a. 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 when performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t). Without Primary Containment Integrity, restore Primary Containment Integrity within one hour or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.	2. Primary Containment Integrity 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 Requirement 4.7.A.2.b.4 for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.6La.

3.0 LIMITING CONDITIONS FOR OPERATION

b. When Primary Containment Integrity is required, leakage rates shall be limited to:

1. An overall integrated leakage rate of less than or equal to La, 1.2 percent by weight of the containment air per 24 hours at Pa, 42 psig.

2. A combined leakage rate of less than or equal to 0.6La for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests when pressurized to Pa.

3. Less than or equal to 11.5 scf per hour for any one main steam isolation valve when tested at 25 psi.

With the measured overall integrated primary containment leakage rate exceeding 0.75La, or the measured combined leakage rate for all penetrations and valves, except main steam isolation valves, subject to Type B and C testing exceeding 0.6La, or the measured leak rate exceeding 11.5 scf per hour for any one main steam isolation valve, restore leakage rates to less than or equal to these values prior to increasing reactor coolant system temperature above 212°F or, alternatively, restore measured leakage rates to within these limits within one hour or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

b. The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

1. Three Type A overall integrated containment leakage rate tests shall be conducted at 40±10 month intervals* during shutdown at ≥ Pa during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

2. If any periodic Type A test fails to meet 0.75La, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75La, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75La, at which time the above test schedule may be resumed.

3. All Type A test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

*The second test of the second 10-year service period may be conducted during the 1989 refueling outage.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	4. The accuracy of each Type A test shall be verified by a supplemental test which: a. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25La, and b. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test, and c. Requires the rate of gas injected into the containment or bled from the containment during the supplemental test to be limited between 75 to 125% of La. d. Type B and C tests shall be conducted with gas at $\geq$ Pa at each refueling shutdown (maximum interval of 24 months), except for tests involving the main steam line isolation valves. Main steam isolation valve tests shall be conducted with gas at $\geq$ 25 psig each 18 months. A combined leakage rate of $\leq$ 0.6La shall be demonstrated for all penetrations and valves, except for main steam line isolation valves, subject to Type B and C tests. A leakage rate of $\leq$ 11.5 scf per hour shall be demonstrated for each main steam line isolation valve.

3.0 LIMITING CONDITIONS FOR OPERATION

c. When Primary Containment Integrity is required, the primary containment airlock shall be operable with:

1. Both doors closed except when the airlock is being used, then at least one airlock door shall be closed, and

2. An overall airlock leakage rate of less than or equal to 0.05La at Pa or 0.007La at 10 psig.

With the primary containment airlock inoperable, maintain at least one airlock door closed and restore the airlock to Operable status within 24 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

c. The primary containment airlock shall be demonstrated operable:

1. At each refueling shutdown, and at six month intervals thereafter, by conducting an overall airlock leakage test at $\geq$ Pa and demonstrating that overall airlock leakage rate is $\leq$ 0.05La. This test interval may be extended up to the next refueling outage (up to a maximum interval between tests at Pa of 24 months) if there have been no air lock openings since the last successful test at Pa.

2. After each opening by conducting an overall airlock leakage test at $\geq$ 10 psig and verifying the leakage rate is $\leq$ 0.007La. If the airlock is being used for multiple openings, this test is not required after each opening, but shall be performed at least once per 72 hours.

3. At six month intervals by verifying that only one door can be opened at a time. If the airlock has not been used since the last door interlock test, this test is not required.

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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 release 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. 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.

Bases:

4.7 A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident. Daily checks are specified of pool temperature and volume to ensure that these parameters are within their allowable ranges.

The interiors of the drywell and suppression chamber are painted to prevent corrosion. The inspection of the paint during each 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 specified is adequate.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally change 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 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 design basis loss of coolant accident was analyzed at the primary containment maximum allowable accident leak rate of 1.2% and has been evaluated by the NRC Staff(1). Computed offsite doses are well below the guidelines of 10 CFR Part 100.

(1) Safety Evaluation by the Division of Reactor Licensing, US Atomic Energy Commission, in the Matter of Northern States Power Company Monticello Nuclear Generating Plant, Unit 1, Docket No. 50-263, March 18, 1970, Section 4.1.

Bases Continued:

While the design of the Monticello plant predates 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," testing substantially conforms to the requirements of Appendix J. The design of the plant was thoroughly reviewed to determine where compliance with Appendix J was impossible or impractical. In each case where a departure from the requirements of Appendix J was identified, a request for exemption from the requirements of Appendix J or a plant modification was proposed and submitted for NRC Staff review. Exemptions were proposed in those cases where compliance with Appendix J would have provided no meaningful improvement in plant safety.

In their review of Appendix J compliance(1), the NRC Staff approved a number of exemption requests, denied others, and provided necessary interpretation and clarification of the requirements of Appendix J. The Technical Specification surveillance requirements reflect the results of this review.

Exemption from the requirements of Appendix J was provided in the following areas:

a. Testing of valves sealed by water

b. Low pressure testing of main steam line isolation valves

c. Low pressure testing of the primary containment airlock

d. Reduced airlock testing frequency when the airlock is in frequent use

The Monticello airlock is tested by pressurizing the space between the inner and outer doors. Individual door seal leakage tests cannot be performed. Since the inner door is designed to seat with containment pressure forcing the door closed, special bracing must be installed for each leakage test. The outer door must be opened to install and remove this bracing. Because of the complexity of this operation, up to 24 hours may be necessary to perform a leakage test.

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(1) Letter from D G Eisenhut, Director, Division of Licensing, USNRC, dated June 3, 1984, "Safety Evaluation by the Office of NRR, Appendix J Review".

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated May 1, 1986, as supplemented July 15 and October 7, 1987, Northern States Power Company (the licensee) requested an amendment to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment request proposed the following changes to the Technical Specifications (TS) concerned with the containment system.

1. Revise the wording of TS Section 4.7.A.2, "Primary Containment Integrity," and associated bases to conform to the wording of NRC Standard TS (NUREG-0123);

2. Change the airlock testing requirements for Type B testing;

3. Increase the TS value of Pa, Peak Containment Accident Pressure, from 41 psig to 42 psig;

4. Delete the requirement for inerting system makeup monitoring as specified in Section 4.7.A.2.6;

5. Revise the Bases for Sections 3.7 and 4.7 to reflect the above changes; and

6. Add Action Statements consistent with NUREG-0123 to TS Section 3.7.A.2 on containment integrity limiting condition for operation.

The July 15 and October 7, 1987 submittals contained additional information or clarified the changes requested and did not substantially alter the action noticed, or affect the staff's initial determination in the Federal Register on August 16, 1986.

2.0 EVALUATION

1. Revised Wording of TS Section 4.7.A.2 (paragraphs 4.7.A.2.b.4.c and 4.7.A.2.b.4.d)

   The licensee proposes to revise the wording of TS Section 4.7.A.2 to conform to the wording of General Electric Standard Technical Specifications. The staff has reviewed the licensee's revisions and finds them acceptable. Specifically, in the case of paragraph 4.7.A.2.b.4.c dealing with the rate of gas injected into containment, by letter dated October 7, 1987, the licensee clarified the requirement regarding the limitation of the gas to be injected between 75 and 125% of La. By letter dated July 15, 1987, the licensee relocated the contents of paragraph 4.7.A.2.b.5 to 4.7.A.2.b.4.d and specified that the test interval for Type B and C tests and the main steam isolation valves shall be conducted at each refueling outage. The licensee did not specify the test interval for the feedwater isolation valves as previously suggested by the staff. The licensee stated that the feedwater isolation valves are tested in accordance with Type C test requirements as specified in Appendix J to 10 CFR Part 50. The staff finds that the feedwater isolation valves do not have valve leakage collection systems and the test can be performed at 24-month Type C test intervals as required for all the containment isolation valves. Therefore, it is acceptable not to specify leak rate test frequencies for the feedwater isolation valves.

On this basis, the staff finds the changes to paragraphs 4.7.A.2.b.4.c and 4.7.A.2.b.4.d acceptable.

2. <u>Revision to TS 4.7.A.2.c - Airlock Testing Requirements</u>

By letter dated October 7, 1987, the licensee submitted a revision to the Technical Specification, deleting testing of the airlock at reduced pressure which had been previously proposed in paragraph 4.5.C.1. The licensee's revision to paragraph 4.5.C.1 also included the provision to extend the test interval up to the next refueling outage (up to a maximum interval of 24 months) if there have not been airlock openings since the last successful test at Pa. The licensee stated that an exemption from Appendix J requirements for extending the 6-month airlock test interval was previously granted by NRC in a letter dated June 3, 1984. The staff has examined the NRC's Appendix J exemption document and finds that the proposed statement in TS 4.5.C.1 conforms to the exemption as granted. On this basis, the staff finds the licensee's proposed TS changes acceptable. However, any maintenance or repair of the airlock should be considered as an airlock opening and should not be included in the Appendix J test interval exemption.

3. <u>Increase design basis accident pressure Pa from 41 to 42 psig</u>

The licensee increased the TS value of Pa from 41 psig to 42 psig because of the new GE containment response calculations. GE has assumed additional break area of the RHR intertie line installed in 1984, yielding a peak accident pressure of 42 psig when rounded off to the nearest Psi. The licensee has submitted a GE analysis report entitled "Monticello Design Basis Accident Containment Pressure and Temperature Response for FSAR Update." The licensee stated that all future testing will be performed at the higher pressure. The staff has reviewed the GE report and finds the increased Pa acceptable since the new containment response analysis is based on a plant modification and is conservative for use as the leak testing pressure. The licensee should review all leak test acceptance criteria based on the new Pa value.

4. Delete TS Section 4.7.A.2.6 requirements for inerting system makeup monitoring

The licensee proposes to delete the monitoring requirements for the nitrogen makeup line. The licensee states that this specification is based on a false hypothesis that monitoring makeup flow into the containment would detect significant changes in containment leakage. Fourteen years of plant operating experience has proven this requirement to be impractical. The licensee concludes that this is not a requirement of the GE Standard Technical Specifications and proposes to delete it.

The staff concurs with the licensee's decision because the small amount of makeup flow to the containment could not be used to monitor low leakage out of containment at the low differential pressure between the inside and outside of the drywell during normal operation. Therefore, deleting TS 4.7.A.2.6 requirements for inerting system makeup monitoring is acceptable.

5. Revision of Bases for Sections 3.7.A and 4.7.A of the TS

In section 3.7.A, the licensee has changed containment pressure to 42 psig and updated the FSAR reference. In section 4.7.A, the licensee has proposed to change checks of the suppression chamber temperature and volume from weekly to daily to ensure that these parameters are within their allowable ranges, delete a paragraph concerning primary containment preoperational test pressure as described in Section 5.2.3 of the FSAR, and change containment maximum allowable accident leak rate from 1.5 wt.% per day to 1.2% on NRC offsite dose analyses. These revisions are to reflect the above described changes and to update the TS and, therefore, are acceptable.

6. Add Action Statements for Section 3.7.A.2 of the TS

The licensee proposes to add the Action Statements using a format similar to Section 3.6.1.2 of the GE Standard Technical Specifications, which provides the bases for containment leakage rate measurement. The staff has reviewed the bases contained in the Action Statements and finds them acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. S. Guo and D. C. DiIanni

Dated: November 25, 1987