

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 $\geq 68,000$ and $\leq 72,910$ 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.

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4.0 SURVEILLANCE REQUIREMENTS

- d. Whenever there is indication of relief valve operation with a suppression pool temperature of $\geq 160^{\circ}\text{F}$ 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.

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3.0 LIMITING CONDITIONS FOR OPERATION

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.

4.0 SURVEILLANCE REQUIREMENTS

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 \geq 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

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 $0.25L_a$, 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 quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to 75 to 125% of L_a .
 - d. Type B and C tests shall be conducted with gas at $\geq P_a$ 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 ≥ 25 psig at each refueling shutdown (maximum interval of 24 months). A combined leakage rate of $\leq 0.6L_a$ 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 ≤ 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.

3.7/4.7

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. For tests conducted at other than refueling shutdown with no airlock openings since the last test at Pa, this test may be conducted at \geq 10 psig with a demonstration that overall airlock leakage is \leq 0.007La.
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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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)
NORTHERN STATES POWER COMPANY) Docket No. 50-263
(Monticello Nuclear Generating Plant))

EXEMPTION

I.

The Northern States Power Company (NSP/the licensee) is the holder of Facility Operating License No. DPR-22 (the license) which authorizes operation of the Monticello Nuclear Generating Plant, located in Wright County, Minnesota, at steady state reactor core power level not in excess of 1670 megawatts thermal. The license provides, among other things, that it is subject to all rules, regulations and Orders of the Commission now or hereafter in effect.

II.

Section 50.54(o) of 10 CFR Part 50 requires that primary reactor containments for water-cooled power reactors be subject to the requirements of Appendix J of 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," published on February 14, 1973. Appendix J contains the leakage test requirements, schedules, and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment; and on August 1975, each licensee was requested to review the extent to which its facility met these requirements.

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On September 19, 1975, NSP submitted its evaluation of the Monticello Nuclear Generating Plant, and assessed its compliance with the rule. Subsequently, the licensee submitted a combination of proposed Technical Specification changes, exemption requests, and proposed design modifications as NSP's overall plan for achieving compliance with the requirements of Appendix J. The licensee requested certain exemptions from the requirements of Appendix J in a letter dated May 5, 1976. On October 28, 1976 a meeting was held with the licensee to discuss certain aspects of the exemption request. At this meeting, the licensee provided additional information to support various positions. The minutes of this meeting along with the submitted information and interchanged agreements were documented in a summary dated November 12, 1976. In the May 5, 1976 and in the November 12, 1976 documents, NSP requested exemption from Type B testing of certain instrument lines, Type C testing of certain valves, and from Type B testing of pressure and frequency of the drywell air locks, as required by Appendix J.

The Franklin Research Center, as a consultant to NRC, has reviewed all of the licensee's submittals and prepared a Technical Evaluation Report (TER) of its findings. The NRC staff has reviewed the TER and has noted its findings in the Safety Evaluation dated April 1984. The staff has concurred in the TER's bases and findings concerning the conclusions on the exemption request. The conclusion on proposed modifications to the piping and changes to the Technical Specifications are also noted in the Safety Evaluation along with the exception taken by the staff to one position assumed by our consultant.

III.

We have found acceptable the following requests for exemption.

1. Section III.C.1 of Appendix J requires, in part, Type C testing of containment isolation valves which are required to operate intermittently under post-accident conditions. The licensee has requested an exemption from Type C testing requirements for the following valves:

- a) MO-2006, 2007 Torus Spray Line
- b) MO-2008, 2009 Torus Recirculation Line
- c) MO-2020, 2021 Drywell Spray Line.

We have reviewed the licensee's submittals and drawings and have determined that:

- a) Valves, MO-2006 and 2007, in the torus spray line may be exempt from Type C testing because they are sealed by water from the residual heat removal (RHR) pumps under post-accident conditions;
 - b) Valves, MO-2008 and 2009, in the torus recirculation line may be exempt from Type C testing because they are sealed by water from the suppression pool, provided the packing of these valves is not exposed to leakage coming from the torus spray line (MO-2010 and 2011); and
 - c) Valves, MO-2020 and 2021, in the drywell spray line may be exempt from Type C testing because they are sealed by water from the RHR pumps under post-accident conditions.
2. Section III.C.2 of Appendix J requires, in part, that Type C testing be performed at the peak calculated accident pressure (Pa), which for Monticello is 41 psig. NSP requested an exemption from this requirement

for the Main Steam Isolation Valves (MSIVs) to continue testing at 25 psig in accordance with current Technical Specifications rather than at (Pa) as required by Appendix J.

The MSIVs are leak tested by pressurizing between the valves. The MSIVs are angled in the main steam lines in the direction of flow to afford better sealing upon closure. On this basis, we conclude that testing at a reduced pressure of 25 psig is acceptable. A test pressure of Pa acting under the inboard disc is sufficient to lift the disc off its seat, and result in excessive leakage into the reactor vessel. This would result in a meaningless test. The proposed test calls for a test pressure of 25 psig to avoid lifting the disc at the inboard valve. The total observed leakage through both valves (inboard and outboard) is then conservatively assigned to the penetration.

3. NSP requested an exemption from the frequency of Type B testing of the air lock. Specifically, they requested an exemption to do a three-day test of the air lock when it is in use rather than after each use. The revised rule required testing of the air locks as follows:
 - a. Every six months at a pressure of not less than Pa (and after periods when the air lock is opened and containment integrity is not required).
 - b. Within three days of opening (or every three days during periods of frequent opening) when containment integrity is required, at a pressure of Pa or at a reduced pressure as stated in the Technical Specifications.

Our consultant, the Franklin Research Center (FRC), has reviewed the licensee's proposal. Whenever the air lock was opened during the operating cycle, and containment integrity was required, the air lock gasket would be tested following closure if it had been greater than three days since the last leakage test.

FRC concluded that the licensee's proposal to test air lock gaskets within three days of an air lock opening is acceptable.

We agree with the FRC's conclusion that the air lock gasket leakage be tested within three days from an air lock opening. We further agree with the FRC's conclusion that the air lock testing frequency should make adequate allowances to detect potential deterioration of air locks through normal use. However, when the air lock remains closed, that is, there is no opening or closing of the doors to cause degradation of seals or damage to door mechanisms, we find that the reduced pressure testing frequency proposed by the licensee would be adequate to assure that the air lock door seal integrity is maintained.

The staff has reevaluated the six-month test requirement and has developed a revised position which meets the objectives of Appendix J requirements for containment air lock door tests. This revised position still requires the containment air lock to be tested at six-month intervals at a pressure of Pa in accordance with Appendix J, except that this test interval may be extended up to the next refueling outage (up to a maximum interval between Pa tests of 24 months) if there have been no air lock openings since the last successful test at Pa. The

intent of the Appendix J requirement is to assure that the air lock door seal integrity is maintained and that no degradation has occurred as a result of opening of the air lock doors between testing intervals at Pa. This position satisfies the objectives of the requirement. The licensee will be required to propose appropriate modifications to the Technical Specifications.

IV.

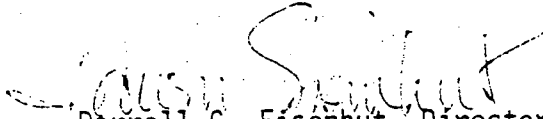
Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12, an exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. Therefore, the Commission hereby approves the following exemption requests:

1. Exemption is granted from the requirements of III.C.1 of Appendix J pertaining to the Type C testing of the torus spray line (MO-2006, 2007) valves and the drywell spray line (MO-2020 and 2021) valves. Similarly, exemption is granted from the requirements of III.C.1 of Appendix J pertaining to the Type C testing of the torus recirculation line (MO-2008 and 20009) valves provided the packing of these valves is not exposed to leakage coming from the torus spray line (MO-2010 and 2011).
2. Exemption is granted from the requirements of Section III.C.2 of Appendix J pertaining to the Type C testing of the Main Steam Isolation Valves, at a test pressure of Pa (peak calculated accident pressure). Testing at a reduced pressure of 25 psig is acceptable because of the unique design of the valves.

3. Exemption is granted to test air lock gaskets within 3 days of an air lock opening.

The NRC staff has determined that the granting of these exemptions will not result in any significant environmental impact and that pursuant to 10 CFR 50.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with this action.

FOR THE NUCLEAR REGULATORY COMMISSION



Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 3rd day of June, 1984.