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 feed.

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 operable 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

2. Primary Containment Integrity

   a. Primary Containment Integrity, ad 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 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 unless an exemption has been granted by the Commission.

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.

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 quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to 75 to 125% of the total measured leakage at Pa. 5. Type B and C tests shall be conducted with gas at $\geq$ Pa at intervals no greater than 24 months except for tests involving the main steam line isolation valves which may be tested at $\geq$25psig. A combined leakage rate of less than or equal to 0.6La shall be demonstrated for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests. A leakage rate of less than or equal to 11.5 scf per hour shall be demonstrated for each main steam 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 six month intervals by conducting an overall airlock leakage test at Pa and demonstrating that overall airlock leakage rate is less than 0.05La. If there have been no airlock openings since the last test at Pa, this test may be conducted at 10 psig with a demonstration that overall airlock leakage is less than 0.007La.

2. After each opening by conducting an overall airlock leakage test at 10 psig and verifying the leakage rate is less than or equal to 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 released during primary system blowdown from 1000 psig.

Since all of the gases in tbe 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 evaluated at the primary containment maximum allowable accident leak rate of 1.2% has been evaluated by the NRC Staff[1]. Computed offsite doses are well below the guidelines of 10 CFR Part 100.

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(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.

(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"