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TO: Mr. V. Stello	ORIG 3 signed	CC 37	OTHER	SENT NRC PDR XX SENT LOCAL PDR XX		
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DESCRIPTION: Ltr trans the following:

ENCLOSURES: Request for Amdt to OL/DPR-22/Tech Specs notarized 1-30-76 consists of Exhibit A containing proposed changes to Tech Specs of App. A & Exhibit B containing a set of Tech Specs pages incorporating the proposed changes.....

(3 Orig & 37 CC rec'd)

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ACKNOWLEDGED

PLANT NAME: Monticello

FOR ACTION/INFORMATION

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

Regulatory Docket File

DOCKETED USNRC FEB 3 1976 Mail Section Docket Clerk

RECEIVED FEB 3 1976 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

January 30, 1976

Mr. Victor Stello, Director
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Stello:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

License Amendment Request Dated January 30, 1976

Attached are 3 originals and 37 conformed copies of a request for change of the Technical Specifications, Appendix A of the Provisional Operating License for the Monticello Nuclear Generating Plant, dated January 30, 1976. This request has been reviewed by the Monticello Operations Committee and the Safety Audit Committee.

This License Amendment Request contains proposed changes to the Technical Specifications governing containment leakage testing. These changes are being submitted in accordance with the request contained in a letter dated August 13, 1975 from Mr. K. R. Goller, Division of Reactor Licensing, USNRC.

We have determined that these changes do not involve an unreviewed safety question.

Yours very truly,

L. O. Mayer

L. O. Mayer, PE
Manager, Nuclear Support Services

LOM/DMM/deb

cc: J. G. Keppler
G. Charnoff
MPCA
Attn: J. W. Ferman
MECCA
Attn: H. J. Vogel
City of St. Paul
Attn: D. L. Ficker
S. J. Gadler

1015

Regulatory Docket File

UNITED STATES NUCLEAR REGULATORY COMMISSION

Received w/Ltr Dated 1-30-76

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT

Docket No. 50-263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE NO. DPR- 22

(License Amendment Request Dated January 30, 1976)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A and Exhibit B. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B is a set of Technical Specification pages incorporating the proposed changes.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By L J Wachter
L J Wachter
Vice President, Power Production &
System Operation

On this 30th day of January, 1976, before me a notary public in and for said County, personally appeared L J Wachter, Vice President, Power Production & System Operation, and first being duly sworn acknowledged that he is authorized to execute this document in behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

Denise E. Branau

DENISE E. BRANAU
NOTARY PUBLIC – MINNESOTA
HENNEPIN COUNTY
My Commission Expires Oct. 10, 1981

EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

LICENSE AMENDMENT REQUEST DATED JANUARY 30, 1976

PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS, APPENDIX A OF PROVISIONAL OPERATING LICENSE DPR-22

Pursuant to 10 CFR 50.59, the holders of Provisional Operating License DPR-22 hereby propose the following changes to the Appendix A Technical Specifications:

Specification and Bases 3.7/4.7.A - Primary Containment

PROPOSED CHANGES

1. Add new Limiting Conditions for Operation and Surveillance Requirements for primary containment leakage testing. The proposed new specifications appear on pages 140 through 143 of Exhibit B.

2. Redesignate existing specifications 3/4.7.A.3 through 3/4.7.A.6 as specifications 3.7/4.7.A.5 through 3.7/4.7.A.8. These changes appear on pages 144 through 148 of Exhibit B. Page 147A will be deleted by these changes.

3. Revise Table 3.7.1 to include automatic isolation valves smaller than 2 inches in diameter which were omitted from the original table. The proposed new table appears on pages 153, 154, and 154A of Exhibit B.

4. Add new Table 4.7.1, "Monticello Containment Penetrations," as shown on pages 154B through 154 O of Exhibit B. This table lists all Type B and Type C testing requirements.

5. Revise the "List of Tables" in the front of the Monticello Technical Specifications to reflect the change in the title of Table 3.7.1 and the inclusion of the new Table 4.7.1.

REASONS FOR CHANGES

In a letter from Mr. K. R. Goller, Division of Reactor Licensing, USNRC, to Mr. L. O. Mayer, NSP, dated August 13, 1975, NSP was requested to determine if containment leakage testing at the Monticello Nuclear Generating Plant conforms to 10 CFR 50, Appendix J. NSP was specifically asked to identify any design features that do not permit conformance with the requirements of Appendix J or any existing Technical Specifications that are less restrictive than Appendix J. A preliminary response to this request was contained in a letter from Mr. L. O. Mayer, NSP, to Mr. K. R. Goller, USNRC, dated September 19, 1975. This letter outlined the following actions and schedule

to attain conformance to Appendix J:

a. A License Amendment Request to be submitted by December 31, 1975 (later rescheduled for January 30, 1976) to revise the Monticello Technical Specifications to conform to Appendix J in those areas where plant design permits.

b. An analysis, to be submitted by March 31, 1976, of systems containing isolation valves which require Type C tests in accordance with the definition contained in Section II.H of Appendix J, but which are not testable. Appropriate design changes will be proposed or a request for exemption from the requirements of Appendix J will be included in conformance with 10 CFR 50.12.

The Technical Specification changes proposed in this License Amendment Request satisfy the first of the two commitments made in our letter of September 19, 1975.

The proposed changes contained in Exhibit B revise the Technical Specifications to remove conflicts with 10 CFR 50, Appendix J. The proposed wording is similar to the wording used in the Technical Specifications of recently licensed BWR's conforming to Appendix J. With the exception of the following clarifying remarks, no further justification for these changes is necessary.

a. Proposed specification 4.7.A.3.a permits NSP to determine an L_t value for the Monticello containment at the next refueling outage and to conduct future leakage tests at a reduced pressure of $P_t=0.5P_a$. This L_t determination is normally completed during the initial integrated leakage test conducted as part of a plant's preoperational testing program. Existing Technical Specifications have required testing at P_a; therefore an L_t value was not determined during the initial test at Monticello. For purposes of scheduling future integrated leakage tests, the test conducted during the next refueling outage would constitute the preoperational Type A leakage rate test specified in Appendix J.

b. Proposed specifications 3.7.A.3.C and 4.7.A.3.f would permit testing of main steam isolation valves at 25 psig. As discussed in the proposed 4.7 Bases on page 163 of Exhibit B, testing of these valves at P_a is not feasible and there is no substantial benefit to be gained from testing at P_a.

c. Proposed specification 4.7.A.4.a permits overall pressure testing of the air lock every three days when the air lock is in use. This is a reasonable surveillance requirement to verify correct door sealing when the air lock is actually in use. Gasket leakage tests are not possible since Monticello air lock doors are not equipped with double gaskets.

d. Table 3.7.1 has been revised to include automatic isolation valves in containment penetrations smaller than two inches in diameter. These penetrations were omitted from the original

table. Table 3.7.1 has also been revised to list isolation valve identification numbers.

e. Table 4.7.1 is a new table that has been included in the proposed Technical Specification changes as a guide in performing Type B and Type C tests. All containment penetrations are listed along with the sealing device or isolation valves in each penetration and the testing required for each.

In cases where conflicts in the table exist and a Type C test is specified for a valve which is not testable in place, the valve is identified as not testable. These conflicts will be resolved in the NSP analysis to be submitted by March 31, 1976 with either a request for waiver from the requirements of Appendix J or proposed modifications to permit testing.

SAFETY EVALUATION

This License Amendment Request is being submitted at the request of the Commission to remove the current conflicts between the surveillance requirements for containment leakage testing in the Monticello Technical Specifications and in 10 CFR 50, Appendix J. The proposed changes revise the Monticello Technical Specifications to conform to Appendix J in all areas where plant design permits. Where possible, the proposed changes follow the wording used in Technical Specifications issued for BWR's currently being licensed and whose testing programs conform to Appendix J.

LICENSE AMENDMENT REQUEST DATED JANUARY 31, 1976

Received w/Ltr Dated 1-30-76

EXHIBIT B

This exhibit consists of the following pages revised or added to incorporate all of the proposed Technical Specification changes:

vii
140 through 148
153
154
154 A through 154 O (new pages)
162 through 164

Existing page 147 A is to be deleted by these changes.

LIST OF TABLES

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

(d) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the torus water temperature exceeds 120°F.

(e) Minimum Water Volume 68,000 cubic feet.

(f) Maximum Water Volume 77,970 cubic feet.

2. Primary containment integrity as defined in the 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 while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 Mw(t).

3. Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. $\leq L_a$ (1.2 percent by weight of the containment air per 24 hours) at Pa (41 psig), or

2. $\leq L_t$ at a reduced pressure of P_t (20.5 psig).

b. A combined leakage rate of $0.6L_a$ for all penetrations and valves (except main steam isolation valves) subject to Type B and C tests when pressurized to P_a.

2. Prior to establishing conditions requiring primary containment integrity verify:

a. All penetrations not capable of being closed by operable containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position.

b. All equipment hatches are closed.

c. Both containment air lock doors are closed.

3. The 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 10CFR50 using the methods recommended in ANSI N45.4 - 1972:

3.0 LIMITING CONDITIONS FOR OPERATION

c. 11.5 scf per hour for any one main steam isolation valve when tested at 25 psig.

With either (a) the overall integrated containment leakage rate exceeding $0.75L_a$ or $0.75L_t$, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding $0.6L_a$, or (c) **a** main steam isolation valve leak rate exceeding 11.5 scf per hour, restore the leakage rate(s) to within acceptable limit(s) prior to increasing the reactor coolant temperature above 212°F.

4.0 SURVEILLANCE REQUIREMENTS

a. During the first refueling outage following the adoption of this specification, a Type A test shall be performed at a pressure P_t of 20.5 psig and a second test performed at a pressure P_a of 41 psig. The maximum allowable test leakage rate, L_t, shall be determined in accordance with section III.A..4(a)(iii) of Appendix J.

b. Following the test specified in 4.7.A.3.a above, three Type A tests shall be conducted at 40 ± 10 month intervals during shutdown at either P_a or P_t during each 10-year service period. One of these tests shall be conducted during the shutdown for each 10-year plant inservice inspection.

c. If any Type A test fails to meet the acceptance criteria of $0.75L_t$ for reduced pressure tests at P_t or $0.75L_a$ for peak pressure tests at P_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two such consecutive tests fail to meet the acceptance criteria, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet the acceptance criteria at which time the schedule specified in 4.7.A.3.b may be resumed.

d. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within $0.25L_t$ for reduced pressure tests at P_t or within $0.25L_a$ for peak pressure tests at P_a.

2. Is of sufficient duration to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the rate that gas is injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the leakage rate measured during the Type A test.

e. Type B and Type C tests shall be conducted at intervals no greater than 24 months, except for tests of the containment air lock, and shall include all testable components listed in Table 4.7.1.

f. Type B and Type C tests shall be conducted at P_a, except for main steam isolation valves. Main steam isolation valves shall be tested at 25 psig.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
4. The containment air lock shall be operable with: a. Both doors closed except when the air lock is being used for normal entry and exit, then at least one air lock door shall be closed. b. An overall air lock leakage rate of $\leq 0.05 L_a$ at at P_a. c. All interlocks function as designed.	4. The containment air lock shall be demonstrated operable by: a. At least once per 6 months by conducting an overall air lock leakage test at Pa and by verifying that the overall leakage rate is within its limit. If the air lock is in use, and containment integrity is required, the air lock shall be tested every three days or after each use, whichever interval is greater. b. During each refueling outage and following repairs to the air lock, by verifying all interlocks function as designed.

3.0 LIMITING CONDITIONS FOR OPERATION

5. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

   a. Except as specified in 3.7.A.5.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when the primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be 0.5 psi.

   b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the suceeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

4.0 SURVEILLANCE REQUIREMENTS

5. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

   a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentation including set point shall be checked for proper operation every three months.

3.0 LIMITING CONDITIONS FOR OPERATION

6. Pressure Suppression Chamber-Drywell Vacuum Breakers

   a. When primary containment is required, all 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.6.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.

4.0 SURVEILLANCE REQUIREMENTS

6. 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 fuel 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)

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

d. One position alarm circuit can be inoperable providing that the redundant position alarm circuit is operable. Both position alarm circuits may be inoperable for a period not to exceed seven days provided that all vacuum breakers are operable.

7. Oxygen Concentration

a. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 5% oxygen with nitrogen gas whenever the reactor is in the run mode, except as specified in 3.7.A.7.b.

b. Within the 24-hour period subsequent to placing the reactor in the run mode following shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 5% by weight, and maintained in this condition. Deinerting may commence 24 hours prior to leaving the run mode for a reactor shutdown.

b. When the position of any drywell-suppression chamber vacuum breaker valve is indicated to be not fully closed at a time when such closure is required, the drywell to suppression chamber differential pressure decay shall be demonstrated to be less than that shown on Figure 3.7.1 immediately and following any evidence of subsequent operation of the inoperable valve until the inoperable valve is restored to a normal condition.

c. When both position alarm circuits are made or found to be inoperable, the control panel indicator light status shall be recorded daily to detect changes in the vacuum breaker position.

7. Oxygen Concentration

Whenever inerting is required, the primary containment oxygen concentration shall be measured and recorded on a weekly basis.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
8. If any of the specifications of 3.7.A cannot be met, the reactor shall be placed in the cold shutdown condition within 24 hours.	

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
B. Standby Gas Treatment System 1. Except as specified in 3.7.B.3 below, both circuits of the standby gas treatment system shall be operable at all times when secondary containment integrity is required.	B. Standby Gas Treatment System 1. Standby gas treatment system surveillance shall be performed as indicated below: a. At least once per operating cycle it shall be demonstrated that: (1) Pressure drop across the combined high-efficiency and charcoal filters is less than 7.0 inches of water, and (2) Inlet heater output is at least 15 kw. b. Within 30 days of the beginning of each refueling outage, whenever a filter is changed whenever work is performed that could affect filter systems efficiency, and at intervals not to exceed six months between refueling outages, it shall be demonstrated that: (1) The removal efficiency of the installed particulate filters for particles having a mean diameter of 0.7 microns shall be

Table 3.7.1 Primary Containment Automatic Isolation Valves

Isolation Group (Note 1)	Application	Isolation Valves Inboard	Isolation Valves Outboard	Permissible Operating Time (sec)	Normal Position
1	Main Steam Isolation	AO-2-80A AO-2-80B AO-2-80C AO-2-80D	AO-2-86A AO-2-86B AO-2-86C AO-2-86D	$3 \leq t \leq 5$	Open
	Main Steam Line Drain	MO-2373	MO-2374	$t \leq 60$	Closed
	Reactor Water Sample	CV-2790	CV-2791	$t < 60$	Closed
2	Drywell Equipment Sump	AO-2561A	AO-2561B	$t \leq 60$	Open
	Drywell Floor Sump	AO-2541A	AO-2541B	$t \leq 60$	Open
	Torus Vent Bypass	-	CV-2384	$t \leq 60$	Closed
	Torus Vent	-	AO-2383	$t \leq 60$	Closed
	Torus Vent	-	AO-2896	$t \leq 60$	Closed
	Drywell Vent Bypass	-	CV-2385	$t \leq 60$	Closed
	Drywell Vent	-	AO-2386	$t \leq 60$	Closed
	Drywell Vent	-	AO-2387	$t \leq 60$	Closed
	Torus Air Purge Air Supply	-	AO-2378	$t \leq 60$	Closed
	Drywell Air Purge Supply	-	AO-2381	$t \leq 60$	Closed
	Containment Air Purge Supply	-	AO-2377	$t \leq 60$	Closed
	TIP Ball Valves (3)	-	-	Note 2	Closed
	Nitrogen Pumpback Suction		CV-7436	$t \leq 60$	Open
	Nitrogen Pumpback Suction		CV-7437	$t \leq 60$	Open

Table 3.7.1 Primary Containment Automatic Isolation Valves (continued)

Isolation Group (Note 1)	Application	Isolation Valves		Permissible Operating Time (sec)	Normal Position
		Inboard	Outboard		
2	RHR Supply	MO-2029	MO-2030	t $\leq$120	Closed
	RHR Head Cooling	MO-2027	MO-2026	t $\leq$120	Closed
	RHR Return to A Loop	-	MO-2014	t $\leq$120	Closed
	RHR Return to B Loop	-	MO-2015	t $\leq$120	Closed
	Containment Nitrogen Supply	-	CV-3269	t $\leq$60	Closed
	Torus Nitrogen Supply	-	CV-3267	t $\leq$60	Closed
	Drywell Nitrogen Supply	-	CV-3268	t $\leq$60	Closed
	Oxygen Analyzer Sample Point	-	CV-3305	t $\leq$60	Open
	Oxygen Analyzer Sample Point	-	CV-3306	t $\leq$60	Open
	Oxygen Analyzer Sample Point	-	CV-3307	t $\leq$60	Open
	Oxygen Analyzer Sample Point	-	CV-3308	t $\leq$60	Open
	Oxygen Analyzer Sample Point	-	CV-3309	t $\leq$60	Open
	Oxygen Analyzer Sample Point	-	CV-3310	t $\leq$60	Open
	Oxygen Analyzer Sample Point	-	CV-3311	t $\leq$60	Open
	Oxygen Analyzer Sample Point	-	CV-3312	t $\leq$60	Open
	Oxygen Analyzer Return	-	CV-3313	t $\leq$60	Open
	Oxygen Analyzer Return	-	CV-3314	t $\leq$60	Open

Table 3.7.1 Primary Containment Automatic Isolation Valves (continued)

Isolation Group (Note 1)	Application	Isolation Valves		Permissible Operating Time (sec)	Normal Position
		Inboard	Outboard		
3	Reactor Water Cleanup Supply	MO-2397	MO-2398	t $\leq$40	Open
	Reactor Water Cleanup Return	--	MO-2399	t $\leq$40	Open
4	HPCI Steam Supply	MO-2034	MO-2035	t $\leq$40	Open
5	RCIC Steam Supply	MO-2075	MO-2076	t $\leq$30	Open

Note 1: Containment isolation groupings are as follows:

Group 1 The valves in Group 1 are closed upon any one of the following conditions:
1. Reactor low low water level
2. Main steam line high radiation
3. Main steam line high flow
4. Main steam line tunnel high temperature
5. Main steam line low pressure (RUN mode only)

Group 2 The valves in Group 2 are closed upon any one of the following conditions:
1. Reactor low water level
2. High Drywell Pressure

Group 3 The actions in Group 3 are initiated by reactor low water level.

Group 4 Isolation valves in the HPCI System are closed upon any one of the following conditions:
1. HPCI steam line high flow
2. HPCI steam line low pressure
3. High temperature in the vicinity of the HPCI steam line

Group 5 Isolation valves in the RCIC System are closed upon any one of the following conditions:
1. RCIC steam line high flow
2. RCIC steam line low pressure
3. High temperature in the vicinity of the RCIC steam line

Note 2: Testing consists of verifying TIP probe automatic withdrawal and ball valve closure on a simulated Group 2 isolation signal.

Table 4.7.1 Monticello Containment Penetrations

Penetration Designation	Description	Applicable Appendix J Type Test	INNER BARRIER			OUTER BARRIER		
			Designation	Type	Testable	Designation	Type	Testable
-	Seismic Restraint Port A	B	-	1	Yes	-	-	-
-	Seismic Restraint Port B	B	-	1	Yes	-	-	-
-	Seismic Restraint Port C	B	-	1	Yes	-	-	-
-	Seismic Restraint Port D	B	-	1	Yes	-	-	-
-	Seismic Restraint Port E	B	-	1	Yes	-	-	-
-	Seismic Restraint Port F	B	-	1	Yes	-	-	-
-	Seismic Restraint Port G	B	-	1	Yes	-	-	-
-	Seismic Restraint Port H	B	-	1	Yes	-	-	-
-	Drywell Head	B	-	1	Yes	-	-	-

Table 4.7.1 Monticello Containment Penetrations (continued)

Penetration Designation	Description	Applicable Appendix J Test Type	INNER BARRIER			OUTER BARRIER		
			Designation	Type	Testable	Designation	Type	Testable
X-1	Equipment Hatch	B	-	-	-	-	1	Yes
X-2	Air Lock (Note 6)	B	-	-	-	-	-	Yes
X-3	Not Assigned	NONE	-	-	-	-	-	-
X-4	Head Access Hatch	B	-	-	-	-	1	Yes
X-5A - 5H	Drywell-Torus Vent Pipes	NONE (Note 7)	-	-	-	-	-	-
X-6	CRD Access Hatch	B	-	-	-	-	1	Yes
X-7A	Bellows	B	-	-	-	-	2	Yes
	Primary Steam Line A	C (Note 1)	AO-2-80A	3	Yes	AO-2-86A		Yes
X-7B	Bellows	B	-	-	-	-	2	Yes
	Primary Steam Line B	C (Note 1)	AO-2-80B	3	Yes	AO-2-86B	3	Yes
X-7C	Bellows	B	-	-	-	-	2	Yes
	Primary Steam Line C	C (Note 1)	AO-2-80C	3	Yes	AO-2-86C	3	Yes
X-7D	Bellows	B	-	-	-	-	2	Yes
	Primary Steam Line D	C (Note 1)	AO-2-80D	3	Yes	AO-2-86D	3	Yes
X-8	Bellows	B	-	-	-	-	2	Yes
	Primary Steam Drain	C	MO-2373	4	Yes	MO-2374	4	Yes

Table 4.7.1 Monticello Containment Penetrations (continued)

Penetration Designation	Description	Applicable Appendix J Type Test	INNER BARRIER			OUTER BARRIER		
			Designation	Type	Testable	Designation	Type	Testable
X-9A	Bellows	B	-	-	-	-	2	Yes
	Feedwater Line	C	FW-97-2	5	Yes	FW-94-2	5	Yes
X-9B	Bellows	B	-	-	-	-	2	Yes
	Feedwater Line	C	FW-97-1	5	Yes	FW-94-1	5	Yes
X-10	Bellows	B	-	-	-	-	2	Yes
	Steam to RCIC	C	MO-2075	4	Yes	MO-2076	4	Yes
X-11	Bellows	B	-	-	-	-	2	Yes
	Steam to HPCI	C	MO-2034	4	Yes	MO-2035	4	Yes
X-12	Bellows	B	-	-	-	-	2	Yes
	RHR Supply	C	MO-2029	4	Yes	MO-2030	4	Yes
X-13A	Bellows	B	-	-	-	-	2	Yes
	RHR Return to B Loop	C	AO-10-46B	6	Yes	MO-2015	4	Yes
X-13B	Bellows	B	-	-	-	-	2	Yes
	RHR Return to A Loop	C	AO-10-46A	6	Yes	MO-2014	4	Yes
X-14	Bellows	B	-	-	-	-	2	Yes
	RWCU Supply	C	MO-2397	4	Yes	MO-2398	4	Yes
X-15	Spare Penetration	NONE	-	-	-	-	17	-

Table 4.7.1 Monticello Containment Penetrations (continued)

Penetration Designation	Description	Applicable Appendix J Type Test	INNER BARRIER			OUTER BARRIER		
			Designation	Type	Testable	Designation	Type	Testable
X-16A	Bellows	B	-	-	-	-	2	Yes
	Core Spray B	C	AO-14-13B	6	Yes	MO-1754	4	Yes
X-16B	Bellows	B	-	-	-	-	2	Yes
	Core Spray A	C	AO-14-13A	6	Yes	MO-1753	4	Yes
X-17	Bellows	B	-	-	-	-	2	Yes
	Head Cooling	C	MO-2027	4	Yes	MO-2026	4	Yes
X-18	Floor Sump Discharge	C	-	-	-	AO-2541A	7	Yes
						AO-2541B	7	Yes
X-19	Equip Sump Discharge	C	-	-	-	AO-2561A	7	Yes
						AO-2561B	7	Yes
X-20	Demin Water Supply	NONE (Note 2)	-	-	-	DM-57	8	-
						DM-58	8	-
X-21	Service Air Supply	NONE (Note 2)	-	-	-	AS-39	8	-
						AS-40	8	-
X-22	Instrument Air	C	-	-	-	CV-1478	9	No
X-23	RBCCW to Drywell	C	-	-	-	RBCC-15	5	No
X-24	RBCCW from Drywell	C	-	-	-	MO-1426	4	No

Table 4.7.1 Monticello Containment Penetrations (continued)

Penetration Designation	Description	Applicable Appendix J Type Test	INNER BARRIER			OUTER BARRIER		
			Designation	Type	Testable	Designation	Type	Testable
X-25	Drywell Ventilation Exhaust	C	-	-	-	AO-2386	10	Yes
						AO-2387	10	Yes
						CV-2385	9	Yes
X-26	Drywell Ventilation Supply	C	-	-	-	AO-2377	10	Yes
						AO-2381	10	Yes
						CV-3268	9	Yes
						CV-3269	9	Yes
X-27A - 27C	Instrumentation	NONE (Note 3)	-	-	-	-	18	-
X-27D	Oxygen Analyzer Sample Point	C	-	-	-	CV-3305	9	Yes
						CV-3306	9	Yes
X-27E	Oxygen Analyzer Sample Point	C	-	-	-	CV-3307	9	Yes
						CV-3308	9	Yes
X-27F	Oxygen Analyzer Sample Point	C	-	-	-	CV-3309	9	Yes
						CV-3310	9	Yes

Table 4.7.1 Monticello Containment Penetrations (continued)

Penetration Designation	Description	Applicable Appendix J Type Test	INNER BARRIER			OUTER BARRIER		
			Designation	Type	Testable	Designation	Type	Testable
X-28A - 28F	Instrumentation	NONE (Note 3)	-	-	-	-	18	-
X-29A - 29 D	Instrumentation	NONE (Note 3)	-	-	-	-	18	-
X-24E - 24F	Instrumentation	NONE (Note 8)	-	-	-	-	19	-
X-30A - 30F	Spare Penetrations	NONE	-		-	-	17	-
X-31A,B,D,E,F	Instrumentation	NONE (Note 3)	-	-	-	-	18	-
X-31C	Spare	NONE	-	-	-	-	17	-
X-32A,B,D,E,F	Instrumentation	NONE (Note 3)	-	-	-	-	18	-
X-32C	Drywell Flood Level Level Switch	NONE (Note 8)	-	-	-	-	19	-
X-33A - 33F	Instrumentation	NONE (Note 3)	-	-	-	-	18	-
X-34A - 34F	Spare Penetrations	NONE	-	-	-	-	17	-
X-35A,B,C	TIP Probes (Note 4)	C	-	-	-	-	15	No
X-35D	Spare Penetration	NONE	-	-	-	-	17	-
X-35E	TIP Purge Supply	C	-	-	-	-	5	No
						-	16	No

Table 4.7.1 Monticello Containment Penetrations (continued)

Penetration Designation	Description	Applicable Appendix J Type Test	INNER BARRIER			OUTER BARRIER		
			Designation	Type	Testable	Designation	Type	Testable
X-36	CRD Hydraulic Return	C	CRD-34	5	No	CRD-31	5	Yes
X-37A - 37D	CRD Insert Lines (121)	NONE (Note 5)	-	5	-	-	-	-
X-38A - 38D	CRD Withdraw (121)	NONE (Note 5)	-	5	-	-	-	-
X-39A	Drywell Spray B	C	-	-	-	MO-2021	4	Yes
						MO-2023	4	Yes
X-39B	Drywell Spray A	C	-	-	-	MO-2020	4	Yes
						MO-2022	4	Yes
X-40A-A-40D-F	Instrumentation	NONE (Note 3)	-	-	-	-	18	-
X-41	Recirc Loop B Sample	C	CV-2790	9	Yes	CV-2791	9	Yes
X-42	Standby Liquid Control	C	XP-7	5	No	XP-6	5	Yes
X-43 - 47	Spare Penetrations	NONE	-	-	-	-	17	-
X-48	Nitrogen Pumpback Suction	C	-	-	-	CV-7436	11	Yes
						CV-7437	11	Yes
X-49A - 49F	Instrumentation	NONE (Note 3)	-	-	-	-	18	-
X-50A - 50D	Instrumentation	NONE (Note 3)	-	-	-	-	18	-
X-50E - 50F	Instrumentation	NONE (Note 8)	-	-	-	-	19	-
X-51A - 51F	Instrumentation	NONE (Note 3)	-	-	-	-	18	-
X-52A - 52F	Instrumentation	NONE (Note 3)	-	-	-	-	18	-
X-53 - X-99	Not Assigned	NONE	-	-	-	-	-	-

Table 4.7.1 Monticello Containment Penetrations (continued)

Penetration Designation	Description	Applicable Appendix J Type Test	INNER BARRIER			OUTER BARRIER		
			Designation	Type	Testable	Designation	Type	Testable
X-100A - 100D	Electrical Penetration	B	-	-	-	-	11	Yes
X-100E	Spare Penetration	NONE	-	-	-	-	17	-
X-101A,101C	Spare Penetrations	NONE	-	-	-	-	17	-
X-101B,101D	Electrical Penetration	B	-	-	-	-	11	Yes
X-102	Spare Penetration	NONE	-	-	-	-	17	-
X-103	Electrical Penetration	B	-	-	-	-	11	Yes
X-104A -- 104D	Electrical Penetration	B	-	-	-	-	11	Yes
X-104E	Spare Penetration	NONE	-	-	-	-	17	-
X-105A,105C, 105D	Electrical Penetration	B	-	-	-	-	11	Yes
X-105B	Spare Penetration	NONE	-	-	-	-	17	-
X-106	Spare Penetration	NONE	-	-	-	-	17	-
X-107	Spare Penetration	NONE	-	-	-	-	17	-
X-108 - X-199	Not Assigned	NONE	-	-	-	-	-	-
X-200A	Torus Hatch (45°)	B	-	-	-	-	1	Yes
X-200B	Torus Hatch (225°)	B	-	-	-	-	1	Yes
X-201A - 201H	Torus Vent Pipes	NONE (Note 7)	-	-	-	-	-	-
X-202A,B,C,D, E,F,G,H, J,K	Drywell-Torus Vacuum Breakers	NONE (Note 7)	-	-	-	-	-	-

Table 4.7.1 Monticello Containment Penetrations (continued)

Penetration Designation	Description	Applicable Appendix J Type Test	INNER BARRIER			OUTER BARRIER		
			Designation	Type	Testable	Designation	Type	Testable
X-202I	Not Assigned	NONE	-	-	-	-	-	-
X-203	Not Assigned	NONE	-	-	-	-	-	-
X-204A - 204D	Torus Ring Header	NONE (Note 7)	-	-	-	-	-	-
X-205	Torus Ventilation Exhaust	C	-	-	-	AO-2383	10	Yes
						AO-2384	10	Yes
						AO-2896	10	Yes
X-206A - 206D	Torus Instrumentation	NONE (Note 8)	-	-	-	-	19	-
X-207A - 207H	Torus Vent Pipe Drains	NONE (Note 7)	-	-	-	-	-	-
X-208A - 208H	Relief Valve Discharge Pipes	NONE (Note 7)	-	-	-	-	-	-
X-209A - 209D	Torus Instrumentation	NONE (Note 8)	-	-	-	-	19	-
X-210A	RHR and Core Spray B Test Line to Torus	NONE (Note 9)	-	-	-	RHR-8-2	5	-
						MO-2007	4	-
						MO-2009	12	-
						MO-1750	12	-
						CS-10-2	13	-

Table 4.7.1 Monticello Containment Penetrations (continued)

Penetration Designation	Description	Applicable Appendix J Type Test	INNER BARRIER			OUTER BARRIER		
			Designation	Type	Testable	Designation	Type	Testable
X-210B	RHR and Core Spray A Test Line to Torus	NONE (Note 9)	-	-	-	RHR-8-1	5	-
						MO-2006	4	-
						MO-2008	12	-
						MO-1749	12	-
						CS-10-1	13	-
X-211A	RHR B Torus Spray	C	-	-	-	MO-2007	4	No
						MO-2009	4	No
						MO-2011	4	No
X-211B	RHR A Torus Spray	C	-	-	-	MO-2006	4	No
						MO-2008	4	No
						MO-2010	4	No
X-212	RCIC Turbine Exhaust	C	-	-	-	RCIC-9	5	Yes
						RCIC-10	5	No
X-213A,213B	Flanged Bottom Torus Drains	NONE (Note 10)	-	-	-	-	1	-
X-214	Oxygen Analyzer Return	C	-	-	-	CV-3313	9	Yes
						CV-3314	9	Yes
X-215 - 217	Spare Penetrations	NONE	-	-	-	-	17	-

Table 4.7.1 Monticello Containment Penetrations (continued)

Penetration Designation	Description	Applicable Appendix J Type Test	INNER BARRIER Designation	Type	Testable	OUTER BARRIER Designation	Type	Testable
X-218	Torus-Reactor Building Vacuum Breakers and Ventilation Supply	C	-	-	-	AO-2379	10	Yes
						DWV-8-2	14	Yes
						AO-2380	10	Yes
						DWV-8-1	14	Yes
						AO-2378	10	Yes
						CV-3267	9	Yes
X-219	Spare Penetration	NONE	-	-	-	-	17	-
X-220	Oxygen Analyzer Sample Point	C	-	-	-	CV-3311	11	Yes
						CV-3312	11	Yes
X-221	HPCI Turbine Exhaust	C	-	-	-	HPCI-9	5	Yes
						HPCI-10	5	No
X-222	HPCI Steam Line Drains	NONE (Note 9)	-	-	-	HPCI-14	5	-
						HPCI-15	5	-
X-223	RCIC Steam Line Drains	NONE (Note 9)	-	-	-	RCIC-16	5	-
						RCIC-17	5	-
X-224A	RHR B Suction	NONE (Note 9)	-	-	-	MO-1987	4	-
X-224B	RHR A Suction	NONE (Note 9)	-	-	-	MO-1986	4	-
X-225	HPCI Suction	NONE (Note 9)	-	-	-	MO-2061	4	-
						MO-2062	4	-

Table 4.7.1 Monticello Containment Penetrations (continued)

Penetration Designation	Description	Applicable Appendix J Type Test	INNER BARRIER			OUTER BARRIER		
			Designation	Type	Testable	Designation	Type	Testable
X-226A	Core Spray B Suction	NONE (Note 9)	-	-	-	MO-1742	4	-
X-226B	Core Spray A Suction	NONE (Note 9)	-	-	-	MO-1741	4	-
X-227	RCIC Suction	NONE (Note 9)	-	-	-	MO-2100	4	-
X-228	Not Assigned		-	-	-	-	-	-
X-229	Capped Penetrations (Formerly Control Air to Vacuum Breakers)	NONE	-	-	-	-	17	-
X-230	Electrical Penetration	B	-	-	-	-	11	Yes

Explanation of Notes:

1. Main steam isolation valves are normally tested by pressuring the connecting volume between inboard and outboard valves. Since test pressure tends to unseat the inboard valve, a lower test pressure than P_a is specified.

2. Isolation is accomplished using manual valves in the containment supply line. These valves are opened only when containment integrity is not required. The valves are closed in accordance with valve lineup checklists which are completed prior to plant heatup.

3. One-inch instrumentation lines equipped with excess flow check valves. Subject to leakage testing in accordance with Technical Specification 4.7.D.1.b. Leakage can occur only through rupture of the line or its associated instrument outside of containment.

4. TIP probes are withdrawn on a containment isolation signal and the line is isolated by automatic closure of a ball valve. A shear valve can be manually actuated from the Control Room in the event a probe fails to retract. A solenoid valve in the purge supply line automatically closes on a containment isolation signal.

5. Containment isolation of the CRD hydraulic control lines is accomplished with a ball check valve internal to each drive mechanism and the normally closed hydraulic system control valve.

6. The drywell air lock is constructed with both doors opening inward so that containment pressure will tend to seat the door seals. During overall air lock pressure tests, a support member is installed on the inner door to prevent the door from being forced open.

7. These are internal penetrations between the drywell and torus.

8. Instrumentation lines not equipped with excess flow check valves. Leakage can occur only through rupture of the line or its associated instrument outside of containment.

9. This penetration terminates at the bottom of the suppression pool. It is not exposed to the containment atmosphere.

10. These drains are installed at the bottom of the suppression pool.

Table 4.7.1 Monticello Containment Penetrations (continued)

Barrier Type Codes

1	Double gasketed seal
2	Hot pipe expansion bellows
3	Air operated globe valve
4	Motor operated gate valve
5	Check valve
6	Testable check valve
7	Air operated gate valve
8	Manual gate valve
9	Diaphragm air operated control valve
10	Air operated butterfly valve
11	Electrical penetration
12	Motor operated globe valve
13	Manually operated globe valve
14	Self-actuating vacuum breaker
15	Ball Valve
16	Solenoid Valve
17	Spare Penetration - welded cap
18	Instrument Line with excess flow check valve
19	Instrument Line without excess flow check valve

Bases Continued:

rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID 14844, the calculated maximum total whole body passing cloud dose is about 3 rem and the calculated maximum total thyroid dose is approximately 150 rem at the low population zone distance of one mile for the duration of the accident. The resultant doses that were calculated for the two-hour accident dose at the exclusion zone boundary of 1600 feet are lower than the above stated doses. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss of coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between off-site doses and 10 CFR 100 guidelines. The fission product source term defined in was also used in the design of the facility engineered safety features, including filter sizing.

The maximum allowable leakage rate, L_a, at the calculated peak accident pressure P_a of 41 psig at test conditions was derived from the maximum allowable accident leak rate of about 1.5% per day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air depleted of oxygen, whereas under test conditions the test medium would be air or nitrogen at ambient conditions. Considering the differences in mixture composition and temperatures, the appropriate correction factor applied was 0.8 as determined from the guide on containment testing and results in an L_a of 1.2 wt%/day. (4)

(4) TID 20583, Leakage Characteristics of Steel Containment Vessels and the Analysis of Leakage Rate Determinations

Bases Continued:

Although the dose calculations suggest that the accident leak rate could be allowed to increase to about 2.4% per day before the guideline thyroid dose value given in 10 CFR 100 would be exceeded, establishing the test limit of 1.2% per day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The operational limit is derived by multiplying the allowable test leak rate by 0.75, thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The surveillance testing requirements for overall integrated contaiment leakage tests (Type A tests) and local penetration leakage tests (Type B and Type C tests) are consistent with the requirements of 10CFR50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

The leakage limitations and surveillance testing requirements placed on the containment air lock are designed to provide assurance that this component is leak tight while at the same time providing a reasonable degree of freedom of access to the drywell. The special testing provisions specified for the air lock are consistent with the requirements of 10CFR50, Appendix J.

Main steam isolation valves are normally tested by pressurizing the connecting volume between the inboard and outboard valves in one steam line. The inboard main steam isolation valve is subjected to test pressure in a direction opposite to that existing following an accident. During testing, test pressure tends to unseat the valve and can lead to an invalid measurement. For this reason, the test pressure has been limited to 25 psig for main steam isolation valve testing and the allowable leakage has been stated in terms of a 25 psig test pressure.

Bases Continued:

Results of loss of coolant accident analyses indicate that fission products would not be released directly to the environs because of leakage through the main line isolation valves due to holdup in the steam system complex. Although this effect shows that an adequate margin exists with regard to release of fission products, the results of leak tests on the main steam line isolation valves will be closely followed in order to determine the adequacy of these valves to perform their intended function.