DCS MS-016

FEB 23 1983

DISTRIBUTION:

Docket File	OPA
NRC PDR	RDiggs
L PDR	PMKreutzer (3)
NSIC	RAClark
ORB#3 Rdg	Gray File (+4)
DBrinkman	ASLAB
DEisenhut	DDiIanni
OELD	SECY
ELJordan (2)	ACRS-10
TBarnhardt (8)	LSchneider

Docket Nos. 50-282
and 50-306

Mr. D. M. Musolf
Nuclear Support Services Department
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has issued the enclosed Amendment Nos. 62 and 56 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, respectively in response to your application dated August 7, 1975 as revised by letters dated December 3, and December 22, 1982.

These amendments revise the common Technical Specifications (TS) for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 as a result of our review of the TS for compliance with the requirements of Appendix J to 10 CFR Part 50. Other changes include provisions for reducing the 24 hour containment leak rate test period, deletions of obsolete leak testing requirements of the shield and auxiliary buildings (TS 4.4.A.3) and increasing the allowable leakage rate for the overall airlock door tests (TS 4.4.A.5.c).

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

Sincerely,

Original signed by

Dominic C. DiIanni, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 62 to DPR-42
2. Amendment No. 56 to DPR-60
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

8303140077 830223
PDR ADOCK 05000282
P PDR

OFFICE	ORB#3:DL	ORB#3:DL	CSB	ORB#5:DL	ORB#3:DL	AD:OR:DL	OELD
SURNAME	PMKreutzer	DCDiIanni/pn	WButler	WPaulson	RAClark	GCLainas	
DATE	1/21/83	1/24/83	2/4/83	2/1/83	2/7/83	2/7/83	2/10/83

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

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

DISTRIBUTION:
Docket File
ORB#3 Rdg
PMKreutzer

Docket No. 50-282/50-306

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: NORTHERN STATES POWER COMPANY, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Amendment Nos. 62 and 56
Referenced documents have been provided PDR.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE→	ORB#3:DL					
SURNAME→	PMKreutzer/pn					
DATE→	2/23/83					

NRC FORM 102 7—79

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 62
License No. DPR-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated August 27, 1975, as revised by letters dated December 3 and December 22, 1982, 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-42 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 62 , 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

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 23, 1983

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 56
License No. DPR-60

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated August 27, 1975, as revised by letters dated December 3 and December 22, 1982, 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-60 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 56 , 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

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 23, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 62 TO FACILITY OPERATING LICENSE NO. DPR-42

AMENDMENT NO. 56 TO FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
4.4-1	4.4-1
4.4-2	4.4-2
4.4-3	4.4-3
4.4-4	4.4-4
4.4-5	4.4-5
4.4-6	4.4-6
4.4-7	4.4-7
4.4-8	4.4-8
4.4-9	-
Table 4.4-1 pg 1 of 5	Table 4.4-1 pg 1 of 5
Table 4.4-1 pg 2 of 5	Table 4.4-1 pg 2 of 5
Table 4.4-1 pg 3 of 5	Table 4.4-1 pg 3 of 5
Table 4.4-1 pg 4 of 5	Table 4.4-1 pg 4 of 5
Table 4.4-1 pg 5 of 5	Table 4.4-1 pg 5 of 5

4.4 CONTAINMENT SYSTEM TESTS

Applicability

Applies to integrity testing of the steel containments, shield buildings, auxiliary building special ventilation zone, and the associated systems including isolation valves and emergency ventilation systems.

Objective

To assure that potential leakage from containment of either unit to the environs following a hypothetical loss of coolant accident in that unit is held within values assumed in the accident analysis.

Specification

A. Containment Leakage Tests

Periodic and post-operational integrated leakage rate tests of each containment shall be performed in accordance with the requirements of 10CFR50, Appendix J, "Reactor Containment Leakage Testing for Water Cooled Power Reactors," as published in the Federal Register, Volume 38, February 14, 1973.

1. Type A tests shall initially be performed in accordance with the reduced pressure test program as defined in paragraph III A4(a)(1) of Appendix J. Periodic tests shall be in accord with either the reduced or peak pressure test program defined in Paragraph III A5. Tests shall include the following conditions:

    a. The absolute method of leakage rate testing will be used as the method for performing the test. The controlled leak-off method of leakage rate testing will be used for verification. Test will be conducted in accordance with the provisions of ANSI N45.4-1972.

    b. A Type A test may be terminated in less than 24 hours if the procedures of Bechtel Topical Report BN-TOP-1 Revision 1 are followed completely.

    c. An initial leakage rate test will be performed at a pressure of 23 psig (P_t) and a second test at 46 psig (P_a).

    d. The design basis accident leakage rate (L_a) shall be 0.25 weight percent per 24 hours at pressure P_a.

2. Initial and periodic type B (except airlocks) and type C tests of penetrations (Table TS.4.4-1) shall be performed at a pressure of 46 psig (P_a) in accordance with the provisions of Appendix J, Section III.B and Section III.C, and Specification 4.4.A.5. The airlocks shall be tested initially and at six-month intervals at 46 psig by pressurizing the inner volume. In addition, when containment system integrity is required, each airlock shall be tested every 3 days if it is in use by pressurizing the intergasket space to 10 psig.

3. Type A tests will be considered to be satisfactory if the acceptance criteria delineated in Appendix J, Section III.A are met.

4. Type B and C tests will be considered to be satisfactory if the combined leakage rate of all components subjected to Type B and C tests does not exceed 60% of the L_a and if the following conditions are met.

   a. For pipes connected to systems that are in the ABSVZ (Designated ABSVZ in Table TS.4.4-1) the total leakage past isolation valves shall be less than 0.1 weight percent per 24 hours at pressure P_a.

   b. For pipes connected to systems that are exterior to both the shield building and the ABSVZ (designated EXTERIOR in Table TS.4.4-1) the total leakage past isolation valves shall be less than 0.01 weight percent per 24 hours at pressure P_a.

   c. For airlocks, the leakage shall be less than 1% of the L_a at 10 psig for door intergasket tests and 5% of the L_a at 46 psig for overall airlock tests.

5. The retest schedules for Type A, B, and C tests will be in accordance with Section III.D of Appendix J. Each shield building shall be retested in accordance with the Type A test schedule for its containment. The auxiliary building special ventilation zone shall be retested in accordance with the Type A test schedule for Unit 1 containment.

6. Type A, B and C tests will be in accordance with Section V of Appendix J. Inspection and reporting requirements of each shield building test shall be the same as for Type A tests. The auxiliary building special ventilation zone shall have the same inspection and reporting requirements as for the Type A tests of Unit 1.

B. Emergency Charcoal Filter Systems

1. Periodic tests of the shield building ventilation system shall be performed at quarterly intervals to demonstrate operability. Each redundant train shall be initiated from the control room and determined to be operable at the time of its periodic test if it meets drawdown performance computed for the test conditions with 75% of the shield building inleakage specified in Figure TS 4.4-1 after initiation and achieve a pressure $\leq$-2.0 inches of water gage.

2. Periodic tests of the auxiliary building special ventilation system shall be performed at approximately quarterly intervals to demonstrate its operability. Each redundant train shall be initiated from the control room and determined to be operable at the time of periodic test if it isolates the normal ventilation system and produces a measureable negative pressure in the ABSVZ within 6 minutes after initiation.

3. At least once per operating cycle, or once each 18 months, whichever comes first, tests of the filter units in the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System shall be performed as indicated below:

   a. The pressure drop across the combined HEPA filters and the charcoal adsorbers shall be demonstrated to be less than 6 inches of water at system design flow rate ($\pm$10%).

   b. The inlet heaters and associated controls for each train shall be determined to be operable.

   c. Verify that each train of each ventilation system automatically starts on a simulated signal of safety injection and high radiation (Auxiliary Building Special Ventilation only).

4. a. The tests of Specification 3.6.E.2 shall be performed at least once per operating cycle, or once every 18 months whichever occurs first, or after every 720 hours of system operation or following painting, fire or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

b. Cold DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing that could affect the HEPA bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing that could affect the charcoal adsorber bank bypass leakage.

d. Each circuit shall be operated with the heaters on at least 10 hours every month.

5. Perform an air distribution test on the HEPA filter bank after any maintenance or testing that could affect the air distribution within the systems. The test shall be performed at rated flow rate (±10%). The results of the test shall show the air distribution is uniform within ±20%.

C. Containment Vacuum Breakers

The air-operated valve in each vent line shall be tested at quarterly intervals to demonstrate that a simulated containment vacuum of 0.5 psi will open the valve and a simulated accident signal will close the valve. The check valves as well as the butterfly valves will be leak-tested during each refueling shutdown in accordance with the requirements of Specification 4.4.A.2.

D. Residual Heat Removal System

1. Those portions of the residual heat removal system external to the isolation valves at the containment, shall be hydrostatically tested for leakage at 12-month intervals.

2. Visual inspection shall be made for excessive leakage from components of the system. Any visual leakage that cannot be stopped at test conditions shall be measured by collection and weighing or by another equivalent method.

3. The acceptance criterion is that maximum allowable leakage from either train of the recirculation heat removal system components (which includes valve stems, flanges and pump seals) shall not exceed two gallons per hour when the system is at 350 psig.

4. Repairs shall be made as required to maintain leakage within the acceptance criterion in Specification 4.4.D.3

5. If repairs are not completed within 7 days, the reactor shall be shut down and depressurized until repairs are effected and the acceptance criterion in 3. above is satisfied.

E. Containment Isolation Valves

During each refueling shutdown, the containment isolation valves, shield building ventilation valves, and the auxiliary building normal ventilation system isolation valves shall be tested for operability by applying a simulated accident signal to them.

F. Post Accident Containment Ventilation System

During each refueling shutdown, the operability of system recirculating fans and valves, including actuation and indication, shall be demonstrated.

G. Containment and Shield Building Air Temperature

Prior to establishing reactor conditions requiring containment integrity, the average air temperature difference between the containment and its associated Shield Building shall be verified to be within acceptable limits.

H. Containment Shell Temperature

Prior to establishing reactor conditions requiring containment integrity, the temperature of the containment vessel wall shall be verified to be within acceptable limits.

Basis

The containment system consists of a steel containment vessel, a concrete shield building, the auxiliary building special ventilation zone (ABSVZ), a shield building ventilation system, and an auxiliary building special ventilation system. In the event of a loss-of-coolant accident, a vacuum in the shield building annulus will cause most leakage from the containment vessel to be mixed in the annulus volume and recirculated through a filter system before its deferred release to the environment through the exhaust fan that maintains vacuum. Some of the leakage goes to the ABSVZ from which it is exhausted through a filter. A small fraction bypasses both filter systems.

The freestanding containment vessel is designed to accommodate the maximum internal pressure that would result from the Design Basis Accident.(1) For initial conditions typical of normal operation, 120°F and 15 psia, an instantaneous double-ended break with minimum safeguards results in a peak pressure of less than 46 psig at 268°F.

The containment will be strength-tested at 51.8 psig and leak-tested at 46.0 psig to meet acceptance specifications.

The safety analysis(2)(3) is based on a conservatively chosen reference set of assumptions regarding the sequence of events relating to activity release and attainment and maintenance of vacuum in the shield building annulus and the auxiliary building special ventilation zone, the effectiveness of filtering, and the leak rate of the containment vessel as a function of time. The effects of variation in these assumptions, including that for leak rate, has been investigated thoroughly. A summary of the items of conservatism involved in the reference calculation and the magnitude of their effect upon off-site dose demonstrates the collective effectiveness of conservatism in these assumptions.

Several penetrations of the containment vessel and the shield building could, in the event of leakage past their isolation valves, result in leakage being conveyed across the annulus by the penetrations themselves, thus bypassing the function of the shield building vent system.[5] Such leakage is estimated not to exceed .025% per day. A special zone of the auxiliary building has minimum-leakage construction and controlled access, and is designated as a special ventilation zone where such leakage would be collected by either of two redundant trains of the auxiliary building special vent system. This system, when activated, will supplant the normal ventilation and draw a vacuum throughout the zone such that all outleakage will be through particulate and charcoal filters which exhaust to the shield building exhaust stack.

The design basis loss-of coolant accident was initially evaluated by the AEC staff[3] assuming primary containment leak rate of 0.5% per day at the peak accident pressure. Another conservative assumption in the calculation is that primary containment leakage directly to the ABSVZ is 0.1% per day and leakage directly to the environs is 0.01% per day. The resulting two-hour doses at the nearest site boundary and 30-day doses at the low population zone radius of 1½ miles are less than guidelines presented in 10CFR100.

Initial leakage testing of the shield building and the ABSV resulted in a greater inleakage than the design basis. The staff has reevaluated doses for these higher inleakage rates and found that for a primary containment leak rate of 0.25% per day at peak accident pressure, the offsite doses are about the same as those initially calculated for higher primary containment leakage and lower secondary containment inleakage[6].

The residual heat removal systems functionally become a part of the containment volume during the post-accident period when their operation is changed over from the injection phase to the recirculation phase. Redundancy and independence of the systems permit a leaking system to be isolated from the containment during this period, and the possible consequences of leakage are minor relative to those of the Design Basis Accident [4]; however, their partial role in containment warrants surveillance of their leak-tightness.

The limiting leakage rates from the recirculation heat removal system are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a design basis accident. The test pressure, 350 psig, achieved either by normal system operation or hydrostatically testing gives an adequate margin over the highest pressure within the system after a design basis accident. A recirculation heat removal system leakage of 2 gal/hr will limit off-site exposure due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident.

The shield building ventilation system consists of two independent systems that have only a discharge point in common, the shield building vent. Both systems are normally activated and one alone must be capable of accomplishing the design function of the system. During the first operating cycle, tests were performed to demonstrate the capability of the separate and combined systems under different wind conditions. During quarterly operability tests, the drawdown transient of shield building pressure is compared to the computed predicted drawdown transient for non-accident conditions and leakage equal to 75% of Figure TS.4.4-1 (840 cfm at -2.0 INWG). The -2.0 INWG setpoint of the recirculation damper must be reached and the equilibrium pressure in the annulus must be less than -1.82 INWG to demonstrate adequate shield building leak tightness.

Pressure drop across the combined HEPA filters and charcoal absorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to verify operability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. A charcoal adsorber tray which can accommodate a sufficient number of representative adsorber sample modules for estimating the amount of penetration of the system adsorbent through its life is currently under development. When this tray is available, sample modules will be installed with the same batch characteristics as the system adsorbent and will be withdrawn for the methyl iodide removal efficiency tests. Each module withdrawn will be replaced or blocked off. Until these trays can be installed, to guarantee a representative adsorbent sample, procedures should allow for the removal of a tray containing the oldest batch of adsorbent in each train, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. One sample will be submitted for laboratory analysis and the other held as a backup. If test results are unacceptable, all adsorbent in the train will be replaced. Adsorbent in the tray removed for sampling will be renewed. Any HEPA filters found defective will be replaced. Replacement charcoal adsorber and HEPA filters will be qualified in accordance with the intent of Regulatory Guide 1.52 - Rev. 1 June 1976.

If significant painting, fire, or chemical release occurs such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals, or foreign material, the same tests and sample analysis will be performed as required for operational use.

Operation of each train of the system for 10 hours every month will demonstrate operability of the system and remove excessive moisture which may build up on the adsorber.

Periodic checking of the inlet heaters and associated controls for each train will provide assurance that the system has the capability of reducing inlet air humidity so that charcoal adsorber efficiency is enhanced.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

A minimum containment shell temperature of 30°F has been specified to provide assurance that an adequate margin above NDTT exists. Evaluation of data collected during the first fuel cycle of Unit No. 1 shows that this limit can be approached only when the plant is in cold shutdown. Requiring containment shell temperature to be verified to be above 30°F prior to plant heatup from cold shutdown provides assurance that this temperature is above NDTT prior to establishing conditions requiring containment integrity[(7)].

A maximum temperature differential between the average containment and annulus air temperatures of 44°F has been specified to provide assurance that offsite doses in the event of an accident remain below those calculated in the FSAR. Evaluation of data collected during the first fuel cycle of Unit No. 1 shows that this limit can be approached only when the plant is in cold shutdown. Requiring this temperature differential to be verified to be less than 44°F prior to plant heatup from cold shutdown provides assurance that this parameter is within acceptable limits prior to establishing conditions requiring containment integrity[(7)].

References

(1) FSAR, Section 5, and Appendix 14-C
(2) FSAR, Section 14, and Appendix G
(3) Safety Evaluation Report, Sections 6.2 and 15.0
(4) FSAR, Section 14
(5) FSAR, Section 14.3.6
(6) Letter to NSP from AEC dated November 29, 1973
(7) NSP Report, "Prairie Island Containment Systems Special Analyses," dated April 9, 1976.

UNIT 1 AND UNIT 2 PENETRATION DESIGNATION FOR LEAKAGE TESTS

Penetration Number	Penetration Description	Penetration Designation (Note 3)	Type of Test
1	Pressure Relief Tank to Gas Analyzer	ABSVZ	C
2	Pressure Relief Tank Nitrogen Supply	Exterior	C
3A	Dead Weight Tester	Note (1)	-
3B	Pressure Instrument	Note (1)	-
4	Primary Vent Header	ABSVZ	C
5	RC Drain Tank Pump Discharge	ABSVZ	C
6A, 6B	Steam lines	Note (2)	-
(6C, 6D in Unit 2)	Bellows	Annulus	B
7A, 7B	Feedwater lines	Note (2)	-
(7C, 7D in Unit 2)	Bellows	Annulus	B
8A, 8B	Steam Gen Blowdown	Note (2)	-
(8C, 8D in Unit 2)	Bellows	Annulus	B
9	RHR Loop Out	Note (5)	-
9	Bellows	Annulus	B
10	RHR Loop Out	Note (5)	-
10	Bellows	Annulus	B
11	Letdown line	ABSVZ	C
11	Bellows	Annulus	B
12	Charging line	ABSVZ	C
13A, 13B	RC Pump Seal Supply	ABSVZ	C
14	RC Pump Seal Return	ABSVZ	C
15	Pressurizer Steam Sample	ABSVZ	C
16	Pressurizer Liquid Sample	ABSVZ	C

UNIT 1 AND UNIT 2 PENETRATION DESIGNATION FOR LEAKAGE TESTS

Penetration Number	Penetration Description	Penetration Designation (Note 3)	Type of Test
17	Loop B Hot Leg Sample	ABSVZ	C
18 18	Fuel Transfer Tube (4) Bellows	ABSVZ Annulus	B B
19	Service Air (4)	ABSVZ	B
20	Instrument	Exterior	C
21	RC Drain Tank to Gas Analyzer	ABSVZ	C
22	Containment Air Sample In	ABSVZ	C
23	Containment Air Sample Out	ABSVZ	C
24	Spare		-
25A	Containment Purge Exhaust (4)	ABSVZ	B
25B	Containment Purge Supply (4)	ABSVZ	B
26	Containment Sump "A" Discharge	ABSVZ	C
27A-1, 27A-2	Steam Generator Blowdown Sample	Note (2)	-
27B (51 in Unit 2)	Fire Protection (4)	ABSVZ	B
27-1, 27-2 (27C-1 and 27C-2 in Unit 2)	OILT Instruments	ABSVZ	B
27D	Spare		-
28A, 28B	Safety Injection	Note (5)	-
29A, 29B	Containment Spray	ABSVZ	C
30A, 30B	Low Head SI Suction from Sump B	ABSVZ	C

UNIT 1 AND UNIT 2 PENETRATION DESIGNATION FOR LEAKAGE TESTS

Penetration Number	Penetration Description	Penetration Designation (Note 3)	Type of Test
31	Accumulator Nitrogen	Exterior	C
32A, 32B	CC to RC Pumps	Note (5)	-
33A, 33B	CC from RC Pumps	Note (5)	-
34	Electrical Penetration	Annulus	B
35	SI and Accumulator	Note (5)	-
36A,B,C,E	Spares		-
36D (50 in Unit 2)	Instrumentation	Note (1)	-
37A,B,C D	Cooling Water to Fan Coil Units	Note (5)	-
38A,B,C,D	Cooling Water from CC to Excess Letdown Heat Exchanger	Note (5)	-
40	CC from Excess Let-down Heat Exchanger	Note (5)	-
41A, 41B	Containment Vacuum Breaker	Annulus	C
41C	Spare		-
42A-1	Post-LOCA Hydrogen Control Air Supply	Annulus	C
42-2	Post-LOCA Hydrogen Control Vent	Annulus	C
42-3	Sample to Gas Analyzer	Exterior	C

UNIT 1 AND UNIT 2 PENETRATION DESIGNATION FOR LEAKAGE TESTS

Penetration Number	Penetration Description	Penetration Designation (Note 3)	Type of Test
42B (53 in Unit 2)	Inservice Purge Supply Valves (6)	ABSVZ	C
42B (53 in Unit 2)	*Inservice Purge Supply Blind Flange(4)	Annulus	B
42C (54 in Unit 2)	Containment Heating Steam (4)	ABSVZ	B
42D, 42E	Spare		-
42F-1 (42E-1 in Unit 2)	Heating Steam Condensate Return(4)	ABSVZ	B
42F-2 (42E-2 in Unit 2)	Heating Steam Return Vent(4)	ABSVZ	B
42G	Spare		
43A (52 in Unit 2)	Inservice Purge Exhaust Valves(6)	ABSVZ	C
43A (52 in Unit 2)	*Inservice Purge Exhaust Blind Flange(4)	Annulus	B
43B,C,D	Spares		
44	Containment Vessel Pressurization (4)	ABSVZ	B
45	Reactor Makeup to Pressurizer Relief Tank	ABSVZ	C
46A, 46B (46C, 46D in Unit 2)	Auxiliary Feedwater	Note (2)	-
47	Electrical Penetration	Annulus	B
48	Low Head SI	Note (5)	-
49A	Instrumentation	Note (1)	-
49B (55 in Unit 2)	Demineralized Water (4)	ABSV	B

*Testing required following modification to inservice purge system of each unit during 1983 refuleing outages.

UNIT 1 AND UNIT 2 PENETRATION DESIGNATION FOR LEAKAGE TESTS

Penetration Number	Penetration Description	Penetration Designation (Note 3)	Type of Test
50-1	Post-LOCA Hydrogen Control Air Supply	Annulus	C
50-2	Post-LOCA Hydrogen Control Vent	Annulus	C
50-3	Sample to Gas Analyzer	Exterior	C
	Equipment Door	Annulus	B
	Personnel Airlock	Annulus	B
	Maintenance Airlock	Annulus	B

Notes:

1. Instrumentation lines. No Type B or C testing required.

2. Steam and feedwater lines. Type C testing not required since valves are not relied upon to prevent containment leakage.

3. Penetration Designations
   - ABSVZ - pipes connected to systems that are located in the Auxiliary Building Special Ventilation Zone
   - Exterior - pipes connected to systems that are exterior to the Shield Building and ABSVZ
   - Sealed - pipes that will be sealed by water in space between isolation barriers following LOCA
   - Annulus - penetration that would leak to the Shield Building annulus following LOCA

4. These penetrations have blank flanges. Penetrations 18, 25A, 25B, 27-1, 27-2, 27C-1, and 27C-2 have blind flanges on the inside only. Penetrations 42B, 43A, 52, and 53 have a blind flange in the annulus only.

5. Safety injection, RHR, cooling water, and closed cooling water system valves not relied upon to prevent containment leakage.

6. The leakage test for this penetration is only required prior to use of the inservice purge system.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 62 TO FACILITY OPERATING LICENSE NO. DPR-42 AND AMENDMENT NO. 56 TO FACILITY OPERATING LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

On June 25, 1976 [1], the NRC requested Northern States Power Company (NSP) to review its containment testing program for Prairie Island, Unit Nos. 1 and 2 and the associated Technical Specifications (TS), for compliance with the requirements of Appendix J to 10 CFR Part 50.

Appendix J to 10 CFR Part 50 was published on February 14, 1973. Since by this date there were already many operating nuclear plants and a number more in advanced stages of design or construction, the NRC decided to have these plants re-evaluated against the requirements of this new regulation. Therefore, beginning in August 1975, requests for review of the extent of compliance with the requirements of Appendix J were made of each licensee. Following the initial responses to these requests, NRC staff positions were developed which would assure that the objectives of the testing requirements of the above cited regulation were satisfied. These staff positions have since been applied in our review of the submittals filed by the licensee for Prairie Island Unit Nos. 1 and 2. The results of our evaluation are provided below.

2.0 EVALUATION

Our consultant, the Franklin Research Center (FRC), has reviewed the licensee's submittals [2, 3, 4 and 6] and prepared the attached evaluation of containment leakage tests for the Prairie Island Nuclear Generating Plant. We have reviewed this evaluation and concur in its bases and findings. These findings by our consultant indicate that several issues did not fully comply with the requirements of Appendix J and therefore were considered unresolved. In order to resolve these issues the licensee submitted by letters dated December 3 and 22, 1982 [7 and 8] a revised amendment request modifying the affected areas of the original submittal [6]. The licensee's revised submittals address the unresolved issues raised in our consultant's report as follows.

A. The licensee's initial submittal proposed to test airlock door seals at 10 psig every 3 days when the airlock is in use and during each 6 month interval. The 10 psig test pressure for the 6 month test interval is unacceptable. Appendix J requires a Type B test of the entire airlock door at the anticipated accident presssure at least once every 6 months. The licensee has revised the proposed TS request (TS 4.4A.2) to meet this requirement in that the test pressure was raised from 10 to 46 psig (the anticipated accident pressure) for the 6 month test interval. This does meet the requirements of Appendix J. On this basis, we find the change acceptable.

B. The licensee's initial submittal [6] proposed to test certain valves hydraulically and convert the results to an air equivalent leakage rate using a scaling factor of 60. As indicated in our consultant's report, this proposed test method is unacceptable. The licensee revised the proposed TS change request (TS 4.4A.4) [7 and 8] by replacing the leakage rate determination by hydraulic measurements with the leakage rate determination meeting the requirement of Appendix J. On this basis, we find the revised proposed TS change acceptable.

C. The licensee's revised TS shows that Table 4.4-1 has a number of isolation valves that are not subjected to Type C testing of Appendix J. These valves are in the residual heat removal systems, safety injection, component cooling system and the test line for the accumulators. All of these systems are designed as Class I seismic, free to sustain a single failure and are missile protected. In all cases, the system operating pressure during an accident exceeds the postulated containment accident pressure. In addition, the design of the component cooling water system furnishing coolant to the reactor cooling water pumps and the excess letdown heat exchanger is the same design as the system that furnishes cooling water to the fan coils. Our consultant has found that the containment isolation valves associated with the fan coil may be excluded from Type C testing in which we also agree.

The licensee has stated that the same justification that was given for the isolation valves for the fan coils also applies for the isolation valve in the cooling lines of excess letdown heat exchanger. This justification is that: (1) the system is a closed system inside containment designed to perform post accident safeguard functions; (2) the piping is Seismic Class 1, missile protected free from a single active failure; and (3) operates at pressures greater than the containment accident pressure. Furthermore, in the event of a pipe rupture, the leakage past isolation valve would be into containment and not out. On this basis, we agree with the licensee that these isolation valves associated with excess letdown heat exchanger need not be relied upon to prevent the escape of containment air to the outside atmosphere and need not be subjected to a periodic type C testing.

Other containment penetration isolation valves that the licensee proposes not to perform a periodic Type C testing are the valves in the component cooling water lines supplying coolant to the reactor cooling water pumps and the valves in the safety injection system. The licensee has stated that these valves are in an open position to perform safeguard functions during accident conditions and therefore are not relied upon to perform a containment isolation function. In addition, the basis used for not testing the isolation valves associated with the fan coils also apply for the case of the isolation valves associated with the reactor cooling water pumps since both cooling systems are of the same design. If the isolation valves in the safety injection system are required to close during post accident condition and the valves were to leak, the leakage would be returned to the containment sump where it could be fed back into the safety injection system and in no way would containment air leak out to the atmosphere. On this basis we agree with the licensee that the isolation valves in the safety injection system and the valves in the component cooling water lines servicing the reactor cooling water pumps need not be relied upon and therefore Type C testing is not necessary.

In regard to the isolation valves in the residual heat removal system, the licensee indicated in Table 4.4-1 that these valves would not be subjected to a Type C testing. The justification for this proposed position is that the residual heat removal system external to the isolation valves is required to be leak tested annually at a test pressure of 350 psig by TS 4.4.D and during accident conditions the system pressure is maintained at 200 psig which is well in excess of the postulated containment accident pressure of 46 psig. In addition, if the isolation valves were to leak, the leakage would be into the containment. The residual heat removal system is also connected to the containment sump enabling a sufficient fluid inventory to maintain a fluid seal at the valves during the accident period. Furthermore, these isolation valves would be opened to remove the reactor decay heat approximately 36 hours after accident when the reactor is brought to cold shutdown. On this basis we agree with the licensee that these valves need not be relied upon to seal the containment atmosphere during an accident. Therefore, Type C testing of the isolation valve associated with the residual heat removal system is not necessary.

The licensee is also proposing not to test the isolation valves that are closed during an accident condition which are in the test line of the accumulator. Any leakage past these valves will return the fluid back to the containment sump which would contain a sufficient fluid inventory to assure a liquid seal during the accident period. The fluid filled line external to these valves will be pressurized to the safety injection pressure (i.e. > 1000 psi) during the accident period which is well in excess of accident pressure inside containment. On this basis we agree with the licensee that leakage past these valves need not be relied upon for containment and therefore Type C testing is not necessary.

D. Our consultant's report shows that the extrapolation of the leakage rate at the test pressure to the accident pressure for the airlock door proposed by the licensee is unacceptable. This extrapolation was proposed by the licensee's letter dated May 30, 1980 [4] to justify the 10 psig airlock door test during each six month interval. Our consultant's position on this matter is no longer applicable since the licensee has committed by the modified proposed amendment request [7] to test the airlock doors during each 6 month interval test at the accident pressure (46 psig) instead of 10 psig as proposed earlier. On this basis there is no need for any extrapolation as proposed by the licensee or recommended by our consultant since the licensee would be required by the proposed amendment to measure the leakage rate at the accident pressure.

E. The licensee proposed changes to TS Table 4.4-1 which included updating the penetration numerical notation and designation, deleting column labeled "test methods" and revising the column labeled "type of test." The column "test method" identifies the penetrations that are hydrostatically and pneumatically tested. By the proposed change and in order to meet the requirements of Appendix J, the hydrostatic test is unacceptable. Since the pneumatic test method is the only method used, the column "test method" does not serve a useful purpose and we agree with the licensee that it should be deleted. The column labeled "Type of Test" is revised to show the proposed Type "C" test for the various penetrations that are now hydrostatically tested. We have reviewed all of the licensee's proposed changes to Table 4.4-1 and have determined that these changes are necessary to meet the requirements of Appendix J. In addition, these changes in no way relax the requirements of the containment system tests nor reduce the level of plant safety. On this basis, we find the proposed changes to Table 4.4-1 acceptable.

Based on the above, we conclude that the licensee, by the implementation of the proposed revised amendment [7 and 8], is in compliance with the requirements of Appendix J. Therefore, we find the proposed changes to the TS for the containment system tests resulting from our Appendix J review for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 acceptable.

The licensee, by letters dated December 3 and 22, 1982 [7 and 8], has requested other changes to Section 4.4 "Containment System Tests" which are not related to the Appendix J review. Our evaluation of these changes is as follows:

I. <u>Reduction of the 24 hour Type A Containment Leak Rate Test</u>

By letter dated November 2, 1977 [3], the licensee requested a reduction in the 24 hour test duration period for the primary containment integrated leak rate test (Type A tests of Appendix J). Section 7.6 of ANSI N45.4-1972 requires a minimum test duration of 24 hours. An exception to this requirement is permitted provided that the leakage rate can be accurately determined during a shorter test time period. The Bechtel Corporation prepared a topical report (Bechtel BN-TOP-1 Revision 1) titled "Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants", that includes a criterion that, if applied completely during the containment leak rate testing period, the time period of 24 hours may be reduced.

By our letter dated February 1, 1973 [9], the NRC staff issued a safety evaluation which finds the Bechtel report acceptable. On this basis, we advised the licensee of our acceptance of this document and, if followed in its entirety, the leak rate testing period may be reduced to less than the 24 hours as specified in Section 7.6 of ANSI N45.4-1972. The licensee, by his letter dated December 3, 1982 [7] has committed to follow the Bechtel criteria by requesting a TS requirement. On this basis we find the inclusion of the Bechtel BN-TOP-1 Revision 1 into the TS acceptable.

II. Shield Building and Auxiliary Building Leak Tests (TS 4.4.A.3)

The licensee has proposed by letter dated December 3, 1982 [7] to delete TS 4.4.A.3 related to leak testing the shield and auxiliary buildings. Specifically, the TS that would be deleted reads as follows.

> "Tests of each shield building and auxiliary building special ventilation zone (ABSVZ) will be in accordance with the provisions of Appendix J, Section IV B and the following conditions:
>
> a. Each shield building shall be functionally tested initially and periodically with the same frequency as the primary containment. Tests will be performed at pressures $\leq$ -2 inches of water gage. Tests results are acceptable if in-leakage is less than 75% of that in Figure TS.4.4-1.
>
> b. The auxiliary building special ventilation zone shall be functionally tested initially and periodically with the same frequency as the primary containment. Test results are acceptable if one fan will maintain the zone at a negative pressure with an opening in the ABSVZ boundary of at least 10 square feet.

c. For initial tests of the shield building for each unit and the ABSVZ, the negative pressures will be determined from measurements of several locations in each region, and will include an allowance for anticipated plant operating and environmental conditions. For subsequent tests, the pressures may be determined from other instrumentation, determined to be representative of the negative pressure in each region, as established and reported in accordance with Table TS.6.7-1, Item 2."

The Specifications 4.4.B.1 and 4.4.B.2 do meet the provision of Appendix J Section IV B in that Type A tests are conducted on shield and auxiliary buildings assuring that leakage rate meets the requirements of Appendix J. This assurance is achieved when operability is demonstrated by meeting the drawdown performance specified in the TS 4.4.B.1 for the shield building and when a measurable negative pressure is achieved within 6 minutes specified for the auxiliary building in TS 4.4.B.2.

TS 4.4.A.3a is equivalent to TS 4.4.B.1 except for the maximum negative pressure which is not specified in TS 4.4.B.1 for functionally testing the shield building. In order to resolve this difference we requested that the phrase "and achieve a pressure $\leq$ -2.0 inches of water gage" be added to the last sentence of TS 4.4.B.1. This addition was discussed with and agreed to by the licensee. TS 4.4.A.3a is less restrictive than TS 4.4.B.1 in that TS 4.4.A.3a requires the test be performed at the same frequency as the primary containment tests while TS 4.4.B.1 requires the test to be performed quarterly.

TS 4.4.A.3b achieves the equivalent results as TS 4.4.B.2 in regard to verifying the leak tightness of the auxiliary building. This equivalance is based on our judgement that the required results of achieving a negative pressure with an opening of the pressure boundary of at least 10 ft^2, in the case of TS 4.4.A.3b, is equivalent to the required results of achieving a measurable negative pressure in the auxiliary building special ventilation system within 6 minutes after the initiation of each redundant train in the case of TS 4.4.B.2. However, TS 4.4.B.2 is more restrictive than TS 4.4.A.3b from the standpoint of test frequency.

TS 4.4.A.3c requires the licensee to perform special functional test of the shield building and the auxiliary building special ventilation system during the plant initial start period. The licensee has performed these tests and reported their results in the report dated April 9, 1976. We have reviewed these results and find them acceptable.

Based on the above evaluation, we conclude that the requirement of TS 4.4.A.3, 4.4.A.3a, 4.4.A.3b and 4.4.A.3c do not serve a meaningful purpose and therefore are no longer necesssary since their functions are adequately addressed in the existing TS 4.4.B.1 and 4.4.B.2. On this basis we agree with the licensee that TS 4.4.A.3, 4.4.A.3a, 4.4.A.3b and 4.4.A.3c may be deleted.

III. Airlock Door Allowable Seal Leakage

Introduction

By letter dated December 22, 1982 [9], the licensee requested a change to TS 4.4.A.5c to permit an increase in the allowable leakage rate for the overall airlock door tests. Specifically, the proposed change permits an increase in the allowable leakage rate for the overall airlock door tests of the containment buildings for Prairie Island Unit Nos. 1 and 2 from 1500 scc/m at 46 psig and 256 scc/m at 10 psig to .05 La (12,900 scc/m) at 46 psig and .01 La (2,580 scc/m) at 10 psig.

Discussion and Evaluation

The existing TS 4.4.A.5c was amended by letter dated March 5, 1982 that issued amendment Nos. 55 and 49 to the Facility Operating Licenses Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Plant, Unit Nos. 1 and 2 respectively. The amendment changed TS 4.4.A.5c to read as follows:

> "For airlocks, the leakage shall be less than 256 scc/m at 10 psig for door intergasket tests and 1500 scc/m at 46 psig for overall airlock tests."

The licensee proposes to change TS 4.4.A.5c to allow a further increase in the leakage rate from the existing 256 scc/m at 10 psig and 1500 scc/m at 46 psig for the overall airlock tests to .01 La at 10 psig and .05 La at 46 psig, the postulated accident pressure. The difference between the proposed and the existing TS is an increase in leakage rate from

256 to 2,580 scc/m at 10 psig and from 1500 to 12,900 scc/m at 46 psig. The licensee requested this change based on the latest door test results which showed that lower leakage rate values were nearly impossible to meet with new Grafoil seals. The original silicon rubber seals have been replaced with the Grafoil seals which are qualified to meet the potential radiation dose of the design bases accident condition recommended in NUREG-0578 and NUREG-0737. Although Grafoil seals can withstand a higher radiation dose for accident conditions, they do not have a low leakage sealing capability equivalent to silicon rubber seals. In addition, the proposed leakage rate values are the same as those recommended in the standard TS which we find acceptable. Therefore, due to the near impossibility of meeting the existing leakage rate requirements, the difference in sealing capability and the higher leakage rate values found acceptable by the staff, resulted in the request by the licensee for this license amendment.

We agree with the licensee regarding this change, since the allowable leakage rate is the same as those we find acceptable for plants having the standard specifications in place. In addition, the leakage rate proposed by the licensee still contains a comfortable margin since this is a small fraction of the total leakage rate allowed for Type B and C tests of 10 CFR Part 50 Appendix J. Appendix J of 10 CFR Part 50 allows a total leakage at a rate not to exceed 154,800 scc/m for the total Type B and C leakage tests which includes the leakage of the airlock door. This proposed TS does not in any way change this total leakage rate requirement which the licensee must meet. In addition, by permitting this change the licensee will be required to reduce the leakage rate at other containment building leak paths that are subjected to the Type B and C tests so that the total leakage rate from the containment building does not exceed the allowable value (i.e., 154,800 scc/m at 46 psig) of 10 CFR Part 50 Appendix J.

Based on the above evaluation we find that the proposed TS which changes the allowable leak rate for the overall airlock tests will not alter the existing level of protection to the public. On this basis we find the licensee's proposed change to TS 4.4.A.5c acceptable.

IV. Containment Purge

By letter dated December 3, 1982 [7], the licensee included changes to TS Table 4.4-1 related to containment purge generic issue B-24 and NUREG-0737 item II.E.4.2. Specifically the licensee shows changes from Type C to Type B testing in the column titled "type of test" as related to those penetrations (i.e., Penetration Numbers 25A, 25B, 42B (53 in Unit 2) and 43A (52 in Unit 2)) for containment purge. These changes stem from the licensee's commitments to place blind flanges at these penetrations as discussed in our safety evaluation issued by letter dated September 9, 1982. We have reviewed these proposed changes and conclude that the licensee meets those portions of the commitments which are addressed in our safety evaluation of September 9, 1982 related to the testing of the blind flanges and the testing of the purge valves (18 inch only) as they affect TS Table 4.4-1. On this basis we find these changes acceptable.

Other commitments regarding this containment purge issue are addressed in the licensee's proposed amendment issued by his letter dated October 29, 1982 and are under review by the staff.

V. <u>Proposed Change in the Allowable Leakage Rate of Auxiliary Building Special Ventilation Zone (ABSVZ) and Exterior Pipes of the Shield Building (ABSVZ) and TS 4.4.A.5a and 4.4.A.5b</u>

By our letter dated March 27, 1975 [10] we requested the licensee to revise the TS related to the ABSVZ (TS 4.4.A.5a) and to the exterior of the shield building and the ABSVZ (TS 4.4.A.5b). These specifications dealing with ABSVZ and the exterior of the shield building that includes the ABSVZ presently appear as specifications 4.4.A.4a and 4.4.A.4b respectively. Specifically we requested the licensee to propose a TS change increasing the allowable leakage rate for the ABSVZ (TS 4.4.A.5a) and to proportionally decrease the allowable leakage rate for the exterior of the shield building and ABSVZ (TS 4.4.A.5b). The licensee by letter dated December 3, 1982 [7] indicated a change in the allowable leakage ratio is not necessary based on the subsequent leakage tests results.

<u>Discussion and Evaluation</u>

Our requested change by letter dated March 27, 1975 [10] is based on the initial local leakage rate test data of the Unit No. 1 containment building issued by the licensee June 6, 1974 and our staff's assumption related to water filled lines. The initial local leakage rate test data and these assumptions indicated 130% of that allowed by the TS for the case of ABSVZ, and only 4% of that allowed by the TS for the case of the exterior of the shield building that includes the ABSVZ. Based on these results at the time, we requested that a higher allowable leakage rate should be specified for the ABSVZ and a proportional lower leakage rate should be specified for the exterior of the shield building that includes the ABSVZ. The 130% and the 4% were based on total calculated leakage rate values of 67,000 and 210 scc/m respectively, where leakage rate was assumed through pipes that are water filled during plant operation and accident conditions. These total leakage rates were determined from a volumetric scaling factor for converting water leakage to air. However, based on our review of the licensee's containment test program for compliance with the requirements of Appendix J, we find that to test these valves hydraulically and convert the results to an air equivalent leakage rate is unacceptable (note Item 2B of this SE). The penetrations that were used in our assumptions for determining the total leakage rates are associated with the chemical and volume control system, the sampling system, the decay heat removal system, the containment spray system the safety injection system and the component cooling system. All of these penetrations associated with the systems will undergo either a Type C test as called for in Appendix J (using air as test fluid) or testing is not required based on our review of the licensee's compliance with the requirements of Appendix J (note Item 2C of this SE).

The other term used in determining the 130% and 4% values was based on the allowable leakage rate values per the TS as 51,600 and 5,160 scc/m which are equivalent to 0.05 and 0.005 weight percent per 24 hours at pressure P_a respectively. These values are in error since the allowable leakage rates in the TS has always been 0.10 and 0.01 weight percent per 24 hours at pressure P_a which are equivalent to 103,200 and 10,320 scc/m respectively. In summary, the allowable leakage rates used in determining the 130% and 4% were one half of what appears in the TS.

Changing leakage ratio of allowable leakage of ABSVZ and the allowable in the interior of the shield building that includes ABSVZ will not improve the level of plant safety since the total allowable leakage rate from which dose levels at the exclusion boundary are calculated is unchanged.

In addition, there was fragmented contaiment leakage rate test data available when we made our request in 1975. Since then, additional test data has become available from containment leakage tests conducted on both units. We have reviewed this data and agree with the licensee that our suggested change in the allowable leakage rate ratio is unnecessary.

On this basis we conclude that the existing requirements in the TS as shown in 4.4.A.4a and 4.4.A.4b pertaining to the allowable leakage rate for the ABSVZ and the exterior of the shield building that includes ABSVZ are adequate.

Environmental Consideration

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

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: February 23, 1983

Attachment: FRC Technical
Evaluation Report

Principal Contributors:
D. C. DiIanni
J. Pulsipher

REFERENCES

1. NRC
   Letter to Mr. L. O. Mayer (NSP)
   Subject: Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2
   June 25, 1976

2. Mr. L. O. Mayer (NSP)
   Letter to Mr. K. R. Goller (NRC)
   Subject: Prairie Island Compliance with the Requirements of 10 CFR Part 50 Appendix J
   August 9, 1977

3. Mr. L. O. Mayer (NSP)
   Letter to Mr. V. Stello, (NRC)
   Subject: Request to Terminate Containment Integrated Leak Rate Test in Less than 24 Hours
   November 2, 1977

4. Mr. L. O. Mayer (NSP)
   Letter to Director NRR
   Subject: Information on Implementation of 10 CFR Part 50 Appendix J
   May 30, 1980

5. Mr. A. Schwencer (NRC)
   Letter to Mr. L. O. Mayer (NSP)
   Subject: Information on Implementation of 10 CFR Part 50 Appendix J
   April 11, 1980

6. Mr. L. O. Mayer (NSP)
   Letter to Mr. A. Giambusso (NRC)
   Subject: License Amendment Request dated August 7, 1975
   August 7, 1975

7. Mr. D. Musolf (NSP)
   Letter to Director NRR
   Subject: Revision No. 1 to License Amendment Request (August 7, 1975)
   December 3, 1982

8. Mr. D. Musolf (NSP)
   Letter to Director NRR
   Subject:
   December 22, 1982

9. R. C. DeYoung
   Letter to Mr. R. D. Allen, Vice President Bechtel Corporation
   Subject: Review of Bechtel Topical Report (BN-TOP-1 Revision 1)
   February 1, 1973

10. Karl Neil (NRC)
    Letter to Mr. L. O. Mayer (NSP)
    Subject: NRC Review of Initial Leak Rate Test
    March 27, 1975

TECHNICAL EVALUATION REPORT

CONTAINMENT LEAKAGE RATE TESTING

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND UNITS 1 AND 2

NRC DOCKET NO. 50-282, 50-306

NRC TAC NO. 06250, 08729

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC ASSIGNMENT 1

FRC TASKS 43, 44

Prepared by

Franklin Research Center
20th and Race Street
Philadelphia, PA 19103

Author: T. J. DelGaizo

FRC Group Leader: T. J. DelGaizo

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: Y. S. Huang

March 25, 1982

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

CONTENTS

FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

Mr. T. J. DelGaizo contributed to the technical preparation of this report through a subcontract with WESTEC Services, Inc.

1. BACKGROUND

On June 25, 1976 [1], the NRC requested Northern States Power Company (NSP) to review the containment leakage testing program at Prairie Island Units 1 and 2 and to provide a plan for achieving full compliance with 10CFR50, Appendix J, Containment Leakage Testing. This plan was to include appropriate design modifications, changes to technical specifications, or requests for exemption from the requirements pursuant to 10CFR50.12, where necessary.

On August 9, 1976 [2], NSP responded to the NRC's request, identifying the following departures from the requirements of 10CFR50, Appendix J:

- o Containment fan coil unit isolation valves not Type C tested.
- o Airlock door seals tested at 10 psig rather than at 46 psig.
- o Some isolation valves tested in a direction opposite to that existing under accident conditions.
- o Some Type C tests of containment isolation valves performed hydraulically rather than pneumatically, and an air/water leakage correlation factor applied.
- o Airlocks tested every 3 days when in use, rather than after each use, by pressurizing the door seals.

On November 2, 1977 [3], NSP requested authorization to substitute a statistical containment leak rate test completion criterion for the 24-hour test duration requirement of Section 7.6 of ANSI N45.4-1972. The statistical procedure was designed to verify that the measured leakage rate, at the 95% confidence level, is less than the leakage rate acceptance criterion. The issue of Type A testing in less than 24 hours, however, is being reviewed by the NRC staff on a generic basis and therefore is not a part of this report.

On May 30, 1980 [4], NSP responded to a request for additional information from the NRC dated April 11, 1980 [5]. In this submittal, NSP provided additional justification for previously submitted exemption requests.

The purpose of this report is to provide technical evaluations of outstanding issues regarding the implementation of 10CFR50, Appendix J, at

Prairie Island. Consequently, technical evaluations of requests for exemption from the requirements of Appendix J, as submitted in Reference 2 and amplified in Reference 4, are included. In addition, Reference 2 indicated that a previously submitted License Amendment Request dated August 7, 1975 [6] is significant to the implementation of Appendix J at Prairie Island. NSP stated that the technical specification changes of this License Amendment Request along with the exemptions from certain requirements regarding the above-mentioned departures are necessary to provide conformance with Appendix J. Therefore, technical evaluations of the proposed technical specification changes of Reference 6 are also included in this report.

2. EVALUATION CRITERIA

Code of Federal Regulations, Title 10, Part 50 (10CFR50), Appendix J, Containment Leakage Testing, contains the criteria used for the evaluation of exemption requests. Where applied to the evaluations, the criteria are either referenced or briefly stated, where necessary, to support the results. Furthermore, in recognition of plant-specific conditions which could lead to requests for exemption not explicitly covered by the regulations, the NRC directed that the technical review constantly emphasize the basic intent of Appendix J, i.e., that potential containment atmospheric leakage paths be identified, monitored, and maintained below established limits.

3. TECHNICAL EVALUATION

3.1 REQUEST FOR EXEMPTION FROM THE REQUIREMENTS OF APPENDIX J

In Reference 2, NSP stated:

> It has been our understanding that exemption from certain requirements of Appendix J has been granted as a result of the Commission's review of the Prairie Island testing program and issuance of appropriate Technical Specifications prior to licensing.

For the purpose of a generic review of the status of implementation of 10CFR50, Appendix J, at all operating reactors, licensee responses to the NRC's generic letter (Reference 1 in the case of Prairie Island) are evaluated on their own merits or on subsequently provided information. Consequently, all reported deviations from the requirements of Appendix J which require exemptions are considered to be requests for exemption regardless of possible prior reviews or agreements. The items evaluated in the following subparagraphs are treated as requests for exemption from the requirements of Appendix J even if the correspondence from NSP never formally requested that exemptions be granted.

3.1.1 Fan Coil Unit Isolation Valves

In Reference 2, NSP stated:

> Containment Fan Coil Unit isolation valves are not subjected to Type C tests as required by Section III.A.1.(d) of Appendix J. The Technical Specifications specifically exclude these valves from local leakage tests since they are considered to be installed in systems which are "sealed" to containment leakage.

In Reference 5, NSP stated:

> Fan coil units inside containment are provided with water from the plant cooling water system when they are operating in their safeguards mode. Portions of the cooling water system serving the fan coil units are free from single failures, designed as Class I seismic, and are missile protected. Cooling water system pressure exceeds maximum postulated containment accident pressure. There is no potential for leakage of radioactive material out of the containment via the cooling water system.

In the event of accident, the cooling water supply and return isolation valves remain open to satisfy their safeguards function. In the event of a fan coil unit or associated piping rupture the containment manual isolation valves would be closed to prevent the entry of non-borated water into containment. Pressure against the closed isolation valves is maintained by 1/2-inch equalizing lines. The water supply for this "seal" is provided by the cooling water system pumps (3 motor driven and 2 diesel driven) which take suction from the Mississippi River.

Evaluation

Section III.A.1.(d) of Appendix J requires Type C testing of containment isolation valves in systems that are normally filled with water and operating under post-accident conditions. Section II.B., however, defines containment isolation valves as those valves relied upon to perform a containment isolation function. Section II.D defines leakage as the escape of containment air to outside atmosphere. Therefore, although the fan coil units are normally filled with water and operating under post-accident conditions, Type C testing of the containment isolation valves is not required if the valves are not relied upon to prevent the escape of containment air to the outside atmosphere.

NSP has stated that there is no potential for leakage of radioactive material out of the containment via the cooling water system. NSP's justification for this statement is that the system is a closed system inside containment designed to perform a post-accident safeguard function, designed Seismic I, missile protected, free from single active failure which would prevent operation, and operates at pressures in excess of postulated maximum containment atmosphere. Furthermore, in the event of a piping rupture of this system, leakage past the isolation valves would be into the containment and not out. Consequently, FRC concurs with NSP that the isolation valves of this system are not relied upon to prevent the escape of containment air to the outside atmosphere. The fan coil unit isolation valves may be excluded from Type C testing and no exemption from the requirements of Appendix J is necessary because Appendix J does not require the testing of these valves.

3.1.2 Testing of Containment Airlocks

In Reference 2, NSP stated:

Airlocks door seals are not tested at Pa as required by Section II.B.2. The Technical Specifications permit this test to be performed at 10 psig. We believe this is an appropriate pressure to use since a higher pressure will produce erroneous results.

Airlocks are not tested after each use as required by Section III.D.2. The Technical Specification permits testing to be conducted every three days if an airlock is in use by pressurizing the door seals. We believe that the requirement to test an airlock after each opening is not practical.

In Reference 5, NSP stated:

The Prairie Island Technical Specifications require airlock door leakage to be less than the design leakage of the door seals reported by NPS in Supplement No. 1 to the Initial Unit No. 1 Reactor Containment Building Leak Rate Test Report. The value reported was an arbitrary 1 cc/min/lineal inch of resilent seal at test pressure (Pt = 10 psig). There is no need to correct leakage from test pressure to peak accident pressure (Pa) since the leakage acceptance criterion is not stated in terms of full pressure.

An attempt to clarify this issue was made in NSP's Prairie Island License Amendment Request dated August 7, 1975. Refer to Exhibit A, Item 1(c). No action has been taken on this request by the NRC Staff.

If extrapolation were necessary, the following method could be used. If Pt is the gauge test pressure used and Pa is the gauge pressure that the results are to be corrected to, a conservative factor to apply to the leakage measured at Pt would be Pa/Pt.

Evaluation

Sections III.B.2 and III.D.2 of Appendix J require that containment airlocks be tested at peak calculated accident pressure (Pa) at 6-month intervals and after each opening when opened in the interim between 6-month tests. These requirements were imposed because airlocks represent potentially large leakage paths which are more prone to human error than other containment penetrations. Type B penetrations (other than airlocks) require testing in accordance with Appendix J at intervals not to exceed 2 years.

Appendix J was published in 1973. A compilation of airlock events from Licensee Event Reports submitted since 1969 shows that airlock testing in accordance with Appendix J has been effective in prompt identification of airlock leakage but that rigid adherence to the after-each-opening requirement may not be necessary.

Since 1969, there have been approximately 70 reported instances in which airlock testing results have exceeded allowable leakage limits. Of these events, 25% were the result of leakage other than that resulting from improper seating of airlock door seals. These failures were generally caused by leakage past door-operating mechanism handwheel packing, door-operating cylinder shaft seals, equalizer valves, or test lines. These penetrations are not unlike other Type B or Type C containment penetrations except that they may be operated more frequently. Since airlocks are tested at a pressure of Pa every 6 months, these penetrations are tested, at a minimum, four times more frequently than typical Type B or C penetrations. The 6-month test is therefore considered to be both justified and adequate for the prompt identification of this leakage.

Improper seating of the airlock door seals, however, is not only the most frequent cause of airlock failures (the remaining 75%), but also represents the large potential leakage path. While testing at a pressure of Pa after each opening will identify seal leakage, seal leakage can also be identified by alternative methods such as pressurizing between double-gasketed door seals (for airlocks designed with this type of seal) or pressurizing the airlock to pressures other than Pa. Furthermore, experience gained in testing airlocks since the issuance of Appendix J indicates that the use of one of these alternative methods may be preferable to the full-pressure test of the entire airlock.

Reactor plants designed prior to the issuance of Appendix J often do not have the capability to test airlocks at Pa without the installation of strongbacks or the performance of mechanical adjustments to the operating mechanisms of the inner doors. This is because the inner doors are designed to seat with accident pressure on the containment side of the door, and therefore, the operating mechanisms were not designed to withstand accident pressure in the opposite direction. When the airlock is pressurized for a local airlock test (i.e., pressurized between the doors), pressure is exerted on the airlock side of the inner door causing the door to unseat and preventing the conduct of a meaningful test. The strongback or mechanical adjustments prevent the unseating of the inner door, allowing the test to proceed. The installation of strongbacks or performance of mechanical adjustments is time consuming (often taking several hours), may result in additional radiation exposure to operating personnel, and may also cause degradation to the operating mechanism of the inner door with consequential loss of reliability of the airlock. In addition, when conditions require frequent openings over a short period of time, testing at Pa after each opening becomes both impractical (tests often take from 8 hours to several days) and accelerates the rate of exposure to personnel and degradation of mechanical equipment.

For these reasons, it is concluded that the intent of Appendix J is satisfied and the undesirable effects of testing after each opening are reduced if a satisfactory test of the airlock door seals is performed within 3 days of each opening or every 72 hours during periods of frequent openings whenever containment integrity is required. The test of the airlock door seals may be performed by pressurizing the space between the double-gasketed seals (if so equipped) or by pressurizing the entire airlock to a pressure less than Pa that does not require the installation of strongbacks or performance of other mechanical adjustments. If the reduced pressure airlock test is employed, the results of this test must be conservatively extrapolated to the results of the Pa air test. Further, a 1980 revision to Section III.D.2 of Appendix J incorporated the above provisions into the regulation.

In view of the foregoing discussion, NSP's proposal to test airlock door seals at 10 psig every 3 days when the airlock is in use is acceptable in meeting the after-each-opening requirement of Appendix J, but unacceptable in meeting the requirement for the semiannual test. A Type B test of the entire airlock assembly every 6 months at peak calculated accident pressure (Pa) is essential to the verification of airlock integrity and must be performed in accordance with Appendix J.

Furthermore, both the acceptance criteria of 1 cc/min/lineal inch of resilent seal and the extrapolation factor of Pa/Pt, discussed in Reference 5, are unacceptable. Section III.B.3 of Appendix J requires that the total of all Type B and Type C tests (local leakage rate tests) be less than 0.6 La (maximum allowable containment integrated leakage). Therefore, Appendix J requires that the airlock leakage at Pa, when combined with leakage from local testing of penetrations and isolation valves in accordance with Appendix J, does not exceed 0.6 La. Since this leakage rate is in terms of Pa, the results of testing at Pt must be conservatively extrapolated to Pa.

The extrapolation that consists of multiplying the leakage rate measured at Pt by Pa/Pt to determine the leakage rate at Pa, as proposed by NSP, is not considered acceptable because it is not necessarily conservative. In the absence of knowledge of the leakage path geometry, it is possible that the leakage path consists of the space between two very closely spaced surfaces. Since air is compressible, the mass flow rate measured at Pt should be multiplied by:

$$\frac{[(Pa + Patm)^2 - (Patm)^2]\ (\mu t)}{[(Pt + Patm)^2 - (Patm)^2]\ (\mu a)}$$

where Pa and Pt are in psig. Patm is discharge pressure for the leakage path in psia, μa is the visosity of air at the temperature at which a test at Pa would be performed, and μt is the viscosity of air at the temperature of the test. For example, if Pa = 60 psig, Pt = 10 psig, Patm = 14.7 psia, and μt = μa, then the extrapolation factor is 13.6 rather than 6 as obtained from the formula Pa/Pt.

3.1.3 Direction of Test Pressure

In Reference 2, NSP stated:

> In a small number of cases isolation valves are tested in a direction opposite to the existing under accident conditions. There is no provision

for testing these valves in the correct direction. There is no assurance that testing these valves in the reverse direction results in a conservative leakage measurement.

In Reference 5, NSP stated:

Testing of blind flanges involves pressurizing the piping between the inboard and outboard flanges. The inboard flange is pressurized in the reverse direction. This is a conservative test since test pressure acts to unseat the inboard flange. Testing of airlock overall leakage is, for the same reason, a conservative test since the inner door is pressurized in the reverse direction which tends to open the door.

Testing globe valves in the reverse direction is acceptable if this results in applying pressure under the seat. Testing butterfly valves in the reverse direction is also acceptable if they are constructed for sealing in either direction. Reverse direction testing should also pressurize the valve stem seal (if any) or the integrity of the stem seal should be verified in some other manner. This position is consistent with the requirements of IWV-3423 of Section XI of the ASME Boiler and Pressure Code, 1977 Edition, Summer 1978 Addenda, which is used in conducting the inservice inspection program valve leakage tests. Testing gate valves in the reverse direction is generally unacceptable.

Each of the penetrations tested in the reverse direction will be reviewed to determine if it meets the above criteria. Procedure changes or modifications will be made to allow testing in the direction of post-accident pressure if they are found to be feasible. This review will be completed prior to the conduct of the 1981 refueling outage Type B and Type C tests.

Evaluation

Section III.C.1 of Appendix J requires that Type C tests be performed by local pressurization applied in the same direction as that for which the valve would be required to perform its safety function, unless it can be determined that the results from tests in which the pressure is applied in a different direction will provide equivalent or more conservative results. In Reference 5, NSP has provided criteria by which it will determine whether or not reverse direction testing in the case of the valves at Prairie Island will provide equivalent or conservative results.

FRC concurs that the criteria specified by NSP for this determination are sufficient. Reverse direction testing of valves which satisfy the criteria is acceptable and no exemption is required. Reverse direction testing of valves which do not satisfy the criteria is unacceptable and exemptions are not appropriate.

3.1.4 Water Testing of Isolation Valves

In Reference 2, NSP stated:

All Type C tests are not conducted using air or nitrogen pressure as required by Section III.C.2 The Technical Specifications permit a large number of isolation valves to be hydrostatically tested at Pa. The test results are corrected to equivalent gas leakage by applying an appropriate air/water leakage correlation factor. It is not practical to drain and vent these penetrations to conduct an air or nitrogen test.

In Reference 5, NSP stated:

The Prairie Island Technical Specification currently allows water tests of the following penetrations:

RHR Supply and Return
Charging Line
Reactor Coolant Pump Seal Supply and Return
Safety Injection
Containment Spray
Containment Sump ECCS Suction
Low Head Safety Injection

All of these lines will remain water filled and intact outside containment following a loss of coolant or steam line break accident. Present practice is to apply an air/water leakage scaling factor of approximately 60. This scaling factor effectively limits the permitted water leakage to a total of a few liters/min to permit the overall containment penetration leakage rate criterion to be satisfied. Leakage rates of this magnitude do not raise serious questions concerning available makeup inventory of any of the systems involved.

Evaluation

Section III.C.2 of Appendix J requires that valves, unless pressurized with fluid from a seal system, be pressurized with air or nitrogen at a pressure of Pa. This is because Appendix J is concerned with measuring the rate of escape of containment air to the outside atmosphere, and therefore, the test medium must closely approximate post-accident containment air. Where it is not convenient to test certain valves with air or nitrogen, water testing may be acceptable where the measured leakage rate can be conservatively correlated to equivalent air leakage. However, to date, no acceptable correlation factor has been demonstrated by any licensee nor is NSP's proposed scaling factor of 60 considered acceptable. Nevertheless, where it can be demonstrated that, because of the design of the system (i.e., safety-related, missile-protected, designed to remain intact and water-filled or operational post-accident, etc.), the isolation valves will be water sealed throughout the post-accident period, Appendix J does not require testing with air or nitrogen because these valves are not relied upon to prevent the escape of containment air to the outside atmosphere.

NSP's justification provided in Reference 5 that leakage rates of a few liters/min do not raise serious questions concerning available makeup inventory of any of the systems involved does not provide ample assurance that these valves will remain water sealed throughout the post-accident period. A clear demonstration that valves will remain water sealed is one that meets the Appendix J criteria for seal systems given in Section III.C.3.b.

3.1.5 Draining of Systems for Type A Testing

In Reference 2, Table 1, Item 4, NSP stated:

> The primary system is vented to the containment atmosphere, but coolant is not drained to expose systems communicating with the primary system to the air test pressure. Each system is, however, subjected to Type C test if practicable.

Further explaining this statement in Reference 5, NSP provided the following:

> Section III.A.1.(d) of Appendix J requires "...portions of the fluid systems that are part of the reactor coolant pressure boundary and are open directly to the containment atmosphere under post-accident conditions and become an extension of the boundary of the containment shall be opened or vented to the containment atmosphere prior to and during the test. Portions of closed systems inside containment that penetrate containment and rupture as a result of a loss of coolant accident shall be vented to the containment atmosphere". We believe this requires that draining and venting of those systems inside containment which may communicate with the post-accident atmosphere either through design or due to failure of non-seismic or non-missile protected piping. Systems designed for the accident environment (such as the seismic,

missile protected fan coil units and the secondary side of the steam generators) need not be drained and vented.

Section III.A.1.(d) does not require systems which are normally operating and filled with water following an accident to be vented to the containment. Type C tests of isolation valves in these systems are required, however.

Prairie Island does not conform to the requirements of Section III.A.1.(d) since many of the isolation valve tests are performed using water and not air as required by Section III.C of Appendix J. As discussed in Item 2.4, however, we believe water tests are more appropriate for these isolation valves. Also, as discussed eariler, tests are performed in some cases with pressure applied in a direction opposite to post-accident pressure. The validity of this testing will be reviewed as noted in our response to Item 2.3.

Evaluation

FRC concurs with NSP's interpretation of the venting and draining requirements of Section III.A.1.(d). As to testing of isolation valves with water in lieu of air or nitrogen and testing of valves in the reverse direction, these items have been evaluated in Sections 3.1.4 and 3.1.3, respectively.

3.2 PROPOSED TECHNICAL SPECIFICATION CHANGES

In Reference 6, NSP submitted a License Amendment Request which included proposed revisions to Technical Specification 4.4.A, Containment Leakage Tests. Although submitted prior to the NRC's generic letter of June 1976, this License Amendment Request had been submitted to provide conformance with the requirements of Appendix J. Technical evaluations of the proposed changes are provided in the following subparagraphs.

3.2.1 Proposed Specification 4.4.A.2, Type B and C Testing

This proposed specification requires that Type B and C tests (except for airlocks) be performed at 46 psig (Pa) in accordance with Sections III.B and III.C of Appendix J.

Evaluation

This proposed specification is in accordance with Appendix J and is acceptable.

3.2.2 Proposed Specification 4.4.A.3, Airlock Testing

This proposed specification requires airlock testing at 6-month intervals and airlock seal testing, by pressurizing the intergasket space, every 3 days when in use at a pressure of 10 psig.

Evaluation

This proposed specification does not conform to the requirements of Sections III.B.2 and III.D.2 of Appendix J with regard to testing of containment airlocks and is, therefore, unacceptable. For a detailed evaluation of NSP's airlock testing proposal, see Section 3.1.2 of this report.

3.2.3 Proposed Specification 4.4.A.4, Hydrostatic Testing of Isolation Valves

The proposed specification provides that penetrations which are hydrostatically tested at 46 psig with the measured leakage converted to equivalent gas leakage use a volumetric scaling factor of 280 scc/min air leakage to 1 cc (at 46 psig)/min of water leakage.

Evaluation

Section III.C.2 of Appendix J requires that Type C testing be performed with air or nitrogen as a test medium. NSP's proposal to perform this testing hydrostatically and convert the results to equivalent air leakage is unacceptable for the reasons given in Section 3.1.4 of this report.

3.2.4 Proposed Specification 4.4.A.7, Type B and C Testing Acceptance Criteria

The proposed specification requires that Type B and C test be considered satisfactory if the combined leakage rate of all components subjected to Type B and C testing does not exceed 60% of La and if additional conditions are met.

Evaluation

Sections III.B.3 and III.C.3 of Appendix J require that the combined total of the Type B and Type C tests are not exceed 0.6 La. Consequently, this proposed specification is acceptable. The additional requirements imposed by NSP in 4.4.A.7.a and 4.4.A.7.b are not material to the requirements of Appendix J and are not evaluated as part of this report, but are left to the discretion of the Licensee.

3.2.5 Proposed Specification 4.4.A.8, Retest Requirements

The proposed specification requires retest schedules for Type A, B, and C tests be in accordance with Section III.D of Appendix J.

Evaluation

This requirement conforms to Appendix J and is acceptable.

3.2.6 Proposed Specification 4.4.A.9, Inspection and Reporting

The proposed specification provides various requirements regarding the inspection and reporting requirements for the Type A, B, and C tests.

Evaluation

This proposed specification is in accordance with Section V of Appendix J and in accordance with information previously reviewed by the NRC, and is acceptable.

3.2.7 Proposed Table 4.4-1, Penetration Designation for Leakage Tests

The proposed table provides a listing of containment penetrations and the type testing performed.

Evaluation

Subject to the findings of Sections 3.1.3 and 3.1.4 of this report regarding the testing of penetrations, this proposed table is acceptable.

4. CONCLUSIONS

Technical evaluations of all outstanding issues regarding the implementation of 10CFR50, Appendix J, at Prairie Island (requests for exemption and proposed technical specification changes) were provided. The conclusion of these evaluations are provided below:

- o Fan coil isolation valves may be excluded from Type C testing, and no exemption is needed because Appendix J does not require testing of these valves.
- o NSP's proposal to test airlock door seals at 10 psig every 3 days when the airlock is in use is acceptable, but testing airlocks at 10 psig is not sufficient to satisfy the requirement to test at Pa every 6 months.
- o NSP's proposal to determine the adequacy of reverse direction testing for certain valves in accordance with NSP's stated criteria is acceptable. Reverse direction testing of valves which satisfy the criteria is acceptable, and no exemptions are necessary. Exemptions are not appropriate for valves which do not satisfy the criteria.
- o NSP's proposal to test certain valves hydraulically and to convert the results to equivalent air leakage using a scaling factor of 60 is unacceptable.
- o NSP's interpretation of Section III.A.1.(d) regarding the draining and venting of systems during Type A testing is agreed with by FRC.
- o The proposed changes to the Technical Specifications at Prairie Island submitted by NSP in August 1975 were found to be acceptable with the exception of airlock testing and hydraulic testing of isolation valves for the reasons enumerated above.

5. REFERENCES

1. NRC
   Letter to Mr. L. O. Mayer (NSP)
   Subject: Prairie Island Nuclear Generating Plants Units 1 and 2
   June 25, 1976

2. Mr. L. O. Mayer (NSP)
   Letter to Mr. K. R. Goller (NRC)
   Subject: Prairie Island Compliance with the Requirements of 10CFR50 Appendix J
   August 9, 1977

3. Mr. L. O. Mayer (NSP)
   Letter to Mr. V. Stello (NRC)
   Subject: Request to Terminate Containment Integrated Leak Rate Test in Less than 24 Hours
   November 2, 1977

4. Mr. L. O. Mayer (NSP)
   Letter to Director NRR
   Subject: Information on Implementation of 10CFR50 Appendix J
   May 30, 1980

5. Mr. A. Schwencer (NRC)
   Letter to Mr. L. O. Mayer (NSP)
   Subject: Information on Implementation of 10CFR50 Appendix J
   April 11, 1980

6. Mr. L. O. Mayer (NSP)
   Letter to Mr. A. Giambusso (NRC)
   Subject: License Amendment Request dated August 7, 1975
   August 7, 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 62 to Facility Operating License No. DPR-42, and Amendment No. 56 to Facility Operating License No. DPR-60 issued to Northern States Power Company (the licensee), which revised Technical Specifications for operation of Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

These amendments revise the common Technical Specifications (TS) for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 as a result of our review of the TS for compliance with the requirements of Appendix J to 10 CFR Part 50. Other changes include provisions for reducing the 24 hour containment leak rate test period, deletions of obsolete leak testing requirements of the shield and auxiliary buildings (TS 4.4.A.3) and increasing the allowable leakage rate for the overall airlock door tests (TS 4.4.A.5.c).

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

Ted that the issuance of these amendments
1 nnt environmental impact and that pursuant
10 nental impact statement or negative declar-
on ppraisal need not be prepared in connection
is .

Fopect to this action, see (1) the application
ame75 as revised by letters dated December 3
Decent Nos. 62 and 56 to License Nos. DPR-42
aDPR's related Safety Evaluation. All of
the lic inspection at the Commission's Public
Documer , Washington, D.C. and at the Environmental
Conserv Mall, Minneapolis, Minnesota 55401.
A copy o obtained upon request addressed to the
U. S. Nuc n, Washington, D.C. 20555, Attention:
Director,

Dated this 23rd day of February, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing