



Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East  
Welch, Minnesota 55089

October 25, 1996

10 CFR Part 50  
Section 50.90

U S Nuclear Regulatory Commission  
Attn: Document Control Desk  
Washington, DC 20555

**PRAIRIE ISLAND NUCLEAR GENERATING PLANT**  
Docket Nos. 50-282 License Nos. DPR-42  
50-306 DPR-60

**License Amendment Request Dated October 25, 1996**  
**Incorporation of Appendix J, Option B Containment Leak Testing Requirements**

Attached is a request for a change to the Technical Specifications, Appendix A of the Operating Licenses, for the Prairie Island Nuclear Generating Plant. This request is submitted in accordance with the provisions of 10 CFR Part 50, Section 50.90.

This amendment request proposes a change to Technical Specification (TS) 4.4.A, "Containment Leakage Tests," the addition of TS.6.5.J, "Containment Leakage Rate Testing Program," and a change to the Bases to the Technical Specifications section B.4.4, "Containment System Tests."

Exhibit A contains a description of the proposed changes, the reasons for requesting the changes, the supporting safety evaluation and significant hazards determination. Exhibit B contains current Prairie Island Technical Specification pages marked up to show the proposed changes. Exhibit C contains the revised Technical Specification pages.

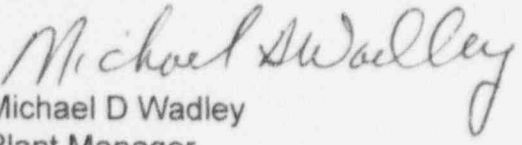
We intend to implement Option B during the next Unit 2 refueling outage, scheduled to start January 25, 1997.

Note that we intend to test the containment vacuum breakers with test intervals allowed up to 60 months based on continued satisfactory performance. This contrasts with our treatment of containment purge and vent valves which will be tested with intervals not to exceed 30 months in accordance with Regulatory Guide 1.163. Attachment 1 provides a discussion of our position regarding the test intervals.

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Please contact Jack Leveille (612-388-1121, Ext. 4662) if you have any questions related to this License Amendment Request.

  
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Plant Manager  
Prairie Island Nuclear Generating Plant

c: Regional Administrator-III, NRC  
NRR Project Manager, NRC  
Senior Resident Inspector, NRC  
State of Minnesota, Attn: Kris Sanda  
J E Silberg

Attachments:

Attachment 1 - Containment Vacuum Breaker Information  
Affidavit  
Exhibit A - Evaluation of Proposed Changes to the Technical Specifications  
Exhibit B - Proposed Changes Marked Up on Existing Technical Specification Pages  
Exhibit C - Revised Technical Specification Pages

## CONTAINMENT VACUUM BREAKER INFORMATION

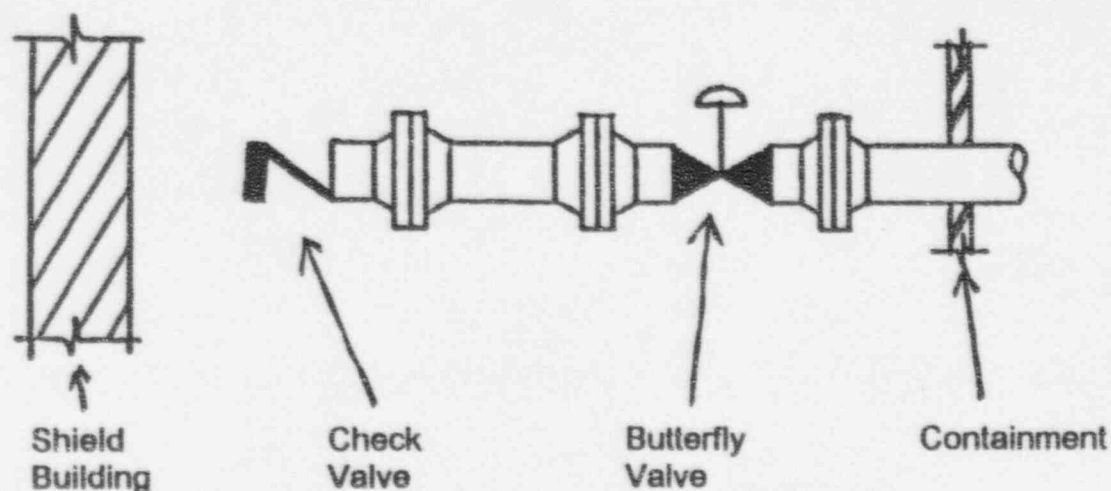
NRC Regulatory Guide 1.163 stipulates that test intervals for containment purge and vent valves should be limited to 30 months with consideration given to operating experience and safety significance. This test interval is shorter than that allowed for other containment penetration valves; the shorter test interval is based on the seat material (resilient seal), the large size of the valves, and the potential unfiltered leak path. Similar to the purge and vent valves, the containment vacuum breaker assembly valves at the Prairie Island Nuclear Generating Plant have resilient seal seat material and are relatively large valves. The following discussion justifies testing the containment vacuum breaker valve assemblies at intervals allowed for other containment isolation valves; i.e., up to 60 months based on continued satisfactory performance.

The vacuum breaker assemblies have an excellent performance history; the measured leakage over the past 10 years is shown on the next page. The administrative leakage limit for the vacuum breaker assembly is 6000 cc/min; as shown, most of the results are less than 10% of this desired value. The seals are designed to last 40 years. The results shown on the next page indicate no definitive trend of seal degradation. Maintenance on the seals is based on testing results. No maintenance has been required on the seals in either unit since plant startup.

The acceptance criteria for seal leakage is based on the overall leakage ( $L_a$ ) to the shield building annulus. The  $L_a$ , assumed in the off-site dose analysis, to the annulus is 144,480 cc/min at a maximum design differential pressure of 46 psi. During an accident, leakage to the annulus is recirculated and filtered by high efficiency HEPA and charcoal filters prior to release. In addition to the vacuum breaker assemblies, the other penetrations which could leak to the annulus are electrical penetrations, hydrogen control valves, and hot piping bellows. These penetrations have, historically, had small leak rates. A review of the last ten years of testing results shows that the average total leakage to the annulus was less than 5,000 cc/min at maximum design differential pressure. This provides substantial margin for an unexpected increase in leakage through a vacuum breaker assembly.

The vacuum breaker assemblies are capable of performing in the postulated post-accident environmental conditions. That is, there are no potential environmental effects which could cause the valve leak tight performance to degrade during the course of the event.

Therefore, based on the performance history of these valve assemblies and the minimal safety significance associated with the assemblies, it is considered acceptable to test these assemblies at increased intervals allowed by Regulatory Guide 1.163 (up to 60 months).



### Unit 1

(leak rates are in cc/minute)

Penetration Number	Description	1996	1994	1992	1991	1990	1988	1987	1986
41A	11 Vacuum Breaker Assembly	265	335	320	500	460	14	190	1000
41B	12 Vacuum Breaker Assembly	560	210	1	200	660	590	180	8

### Unit 2

(leak rates are in cc/minute)

Penetration Number	Description	1995	1993	1992	1990	1989	1988	1986
41A	21 Vacuum Breaker Assembly	400	400	340	330	500	620	35
41B	22 Vacuum Breaker Assembly	11	30	460	65	150	740	380

Each Vacuum Breaker Assembly consists of a Check Valve (24" diameter) and a Butterfly Valve (18" diameter). The two valves in each assembly are tested at the same time; thus, only one leakage value is shown for each assembly.