NSP

Northern States Power Company
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East
Welch, Minnesota 55089

December 27, 1995

10 CFR 50.55a

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

Request for Relief from Code Requirements,
Block Valves in Series with Overpressure Protection Devices

Pursuant to 10 CFR 50.55a(a)(3), Northern States Power (NSP) is hereby requesting relief for Prairie Island Nuclear Generating Plant (PINGP), Units 1 & 2 from the requirements of USAS B31.1-1967 - specifically, Article 122.6.1 and Article 122.6.2(a). This relief would allow NSP to retain the installed configuration of block valves located in series with overpressure protection devices for the regenerative heat exchangers and one train per unit of residual heat removal discharge piping. The block valves are a part of the original standard design provided by Westinghouse, facilitate testing and maintenance of the affected systems, are under direct administrative control, and have been evaluated as having no safety significance. A complete description of the request for relief is as follows.

1. Components for Which Relief Is Requested

A) RHR System

The components for which relief is being requested are RH-7-1 and 2RH-7-1 for Units 1 and 2 respectively. These block valves are installed in identical configurations in each unit. The block valves are locked open in the discharge path of spring loaded check valves RH-6-1 and 2RH-6-1 respectively. RH-6-1

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and 2RH-6-1 provide two functions. They provide letdown from the RHR system to the CVCS letdown line when the RHR system is in operation and provide overpressure protection to the discharge portion of one train of RHR when the system is split. When the system is cross-c nnected, relief valves SI-26-1 and 2SI-26-1 (which are not isolatable) protect both trains. The system configuration is shown as Figure 1.

B) CVCS System

The components for which relief is being requested are VC-16-3 and 2VC-16-3 for Units 1 and 2 respectively. These block valves are installed in identical configurations in each unit. The valves are located in the supply path to spring loaded check valves VC-17-1 and 2VC-17-1 respectively. These valves are not locked as they are required to be manipulated for the addition of Hydrazine to the RCS. VC-17-1 and 2VC-17-1 provide two functions. They provide thermal relief protection for the charging pump line between the Regenerative Heat Exchanger discharge flow control valve and the upstream check valve should somehow the line become isolated with normal letdown still in operation. Additionally, they provide bypass flow around the flow control valve should the valve shut with the pumps running. The system configuration is shown as Figure 2.

2. Code Requirements

These portions of the RHR and CVCS systems were constructed to the requirements of the 1968 edition of ASME Section III and where supplied as original NSSS design by Westinghouse. Section N-910.8 of Article 9 allows the installation of stop valves in the supply or discharge path of relief devices provided there is a system of "positive interlocks or controls". Subsequent ASME Code interpretations determined that manual valves even under direct administrative control did not satisfy the intent of ASME Section III. Prairie Island system design is to the requirements of USAS B31.1-1967. This is incorporated into the facility operating license through inclusion into the FSAR. USAS B31.1-1967 specifically prohibits the use of intervening stop valves in the supply or discharge path of relief devices. Therefore, these configurations are technically outside of the facility licensing basis.

3. Code Requirements from Which Relief Is Requested

Prairie Island requests relief from USAS B31.1-1967 Articles 122.6.1 and 122.6.2.(a) which state the following:

"There shall be no intervening stop valves between piping being protected and its protective device or devices""[122.6.1]

"There shall be no intervening stop valves between the protective device or devices and the point of discharge""[122.6.2.(a)]

4. Proposed Alternative to Code Requirements

As an alternate to the USAS B31.1-1967 requirements, NSP proposes to maintain the current configuration as shown in the attached figures based upon the following considerations;

a) The existing valves are under direct administrative controls which include physical verification of position.

b) The existing configurations are desirable in that they facilitate testing and maintenance.

c) A safety evaluation conducted in accordance with 10 CFR 50.59 to review the effect of operation with these valves closed concluded that there was no safety significance should these valves become isolated.

5. Basis for Relief

Relief from the requirements of USAS B31.1-1967 Articles 122.6.1 and 122.6.2.(a) is requested as follows. This relief request is justified in accordance with 10 CFR 50.55a(a)(i), 50.55a(a)(ii), and 50.55a(f)(6)(i).

a) Administrative controls are in place to ensure that the block valves remain open. These administrative controls include the physical verification of valve position. These administrative controls provide an acceptable alternative for the applications in question.

b) Compliance with code requirements would diminish the ability to test and maintain the systems without a corresponding increase in the level of quality and safety. The block valve and relief valve configuration were a part of the original design configuration specified by the vendor.

c) The current configuration provides an acceptable level of quality and safety. A safety evaluation conducted in accordance with 10 CFR 50.59 to review operation with these valves shut concluded that, while not a desirable operating configuration, no safety concern would originate as a result of it.

In this letter we have made no new Nuclear Regulatory Commission commitments.

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

Michael D Wadley

Michael D Wadley
Plant Manager
Prairie Island Nuclear Generating Plant

c: Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
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J E Silberg

Attachments: Figure 1
Figure 2

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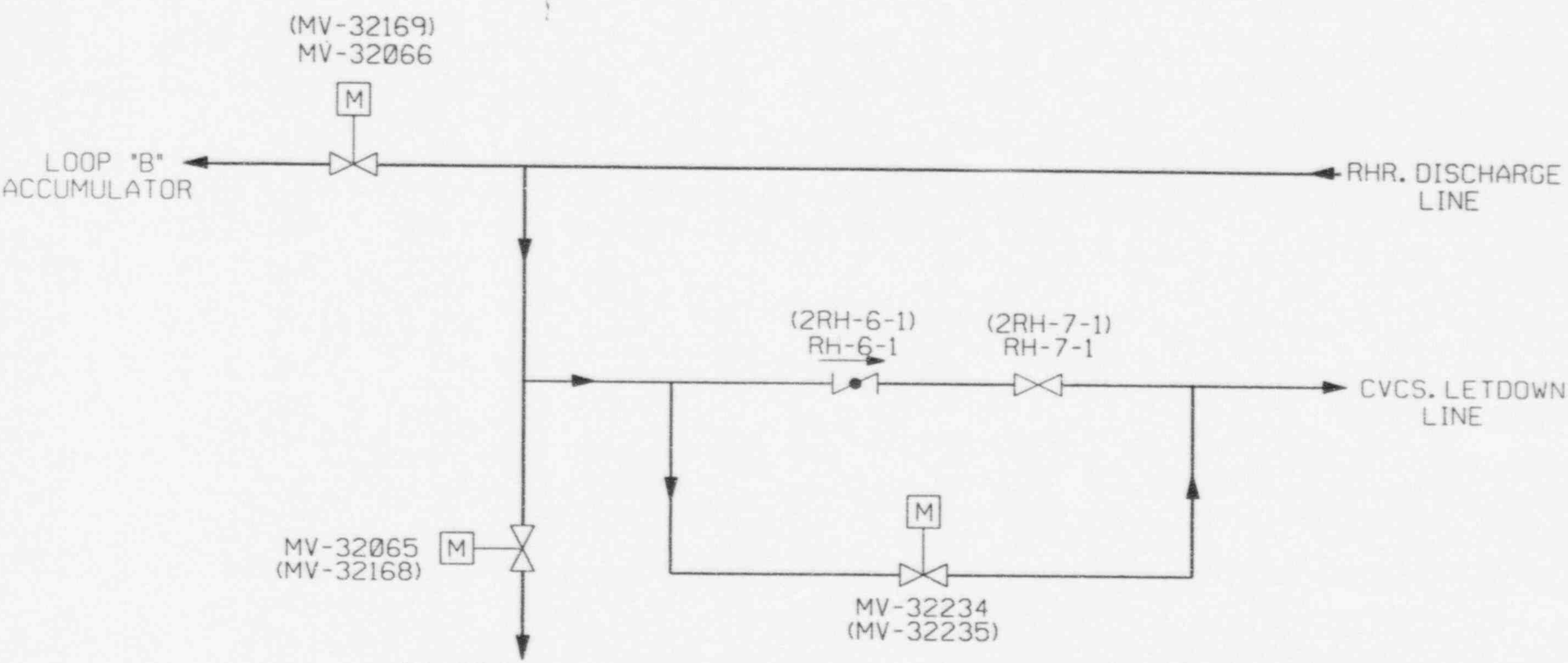

NOTE

1) (   ) DENOTES UNIT 2 VALVE NUMBERS.

DWN CB DATE 2-26-94
CHECKED
PROJECT NO. N/A
APP'D & CERT.
FILMED N/A

SIGNIFICANT NO.
GROUP 1 2 3 4 5 6

FIGURE 1
NRC SUBMITTAL SKETCH

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GEN. PLANT

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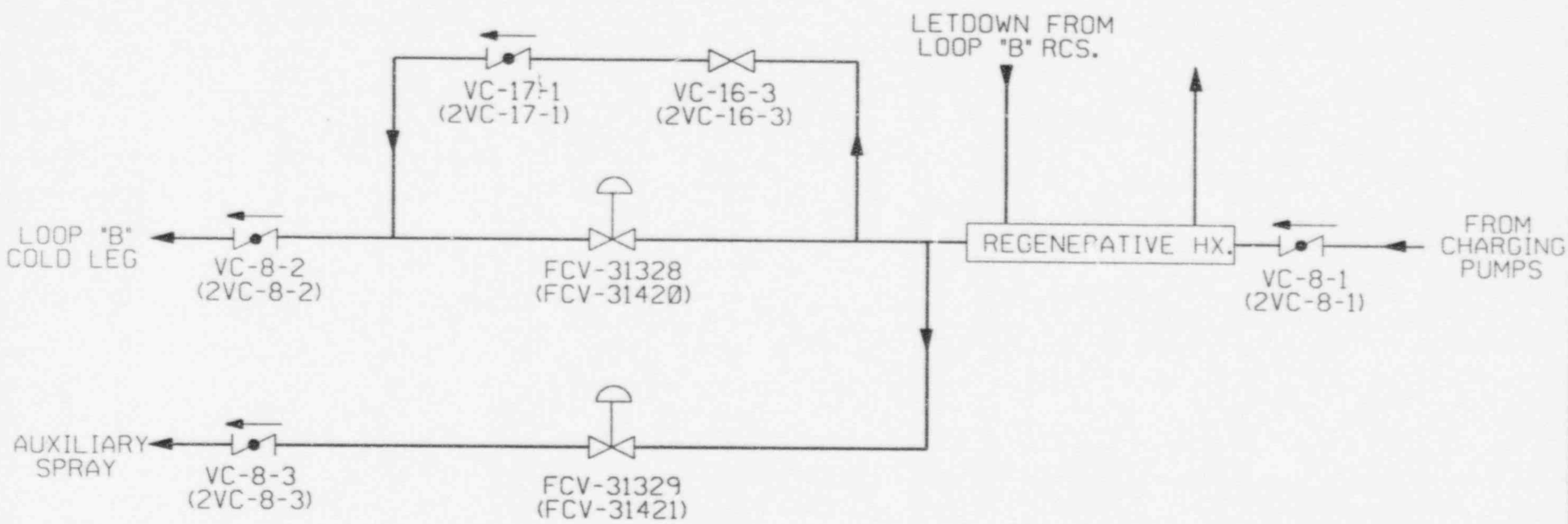

NOTE

1) ( ) DENOTES UNIT 2 VALVE NUMBERS.

DWN DB	DATE 2-26-94	SIGNIFICANT NO.						
CHECKED		GROUP	1	2	3	4	5	6
PROJECT NO.		FIGURE 2 NRC SUBMITTAL SKETCH.						
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