Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East
Welch, Minnesota 55089

September 5, 1995

10 CFR Part 50
Section 50.73

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

Deficiencies discovered in flow testing
of the residual heat removal system

The Licensee Event Report for this occurrence is attached. In the report, we made one new NRC commitment:

Procedure changes will be made which will take recirculation flow into account when determining RHR flow to the reactor vessel.

Please contact us if you require additional information related to this event.

K T Albrecht for

Michael D Wadley
Plant Manager
Prairie Island Nuclear Generating Plant

c: Regional Administrator - Region III, NRC
NRR Project Manager, NRC
Senior Resident Inspector, NRC
Kris Sanda, State of Minnesota

Attachment

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Prairie Island Nuclear Generating Plant U1	05000 282	1 OF 3

TITLE (4) Deficiencies discovered in flow testing of the residual heat removal system

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
8	4	95	95	-- 010 --	00	09	05	95	Prairie Island U2	05000 306
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
			20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
R G Fraser	612-388-1121

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

Engineering review of surveillance procedures used to satisfy Technical Specification 4.5.B.3.h.2, verifying RHR pump minimum flow of 1800 gpm to the reactor vessel after RHR system modifications that alter system flow characteristics, has not always been properly verified. The RHR recirculation line tap, which is located downstream of the flow orifice, bypasses approximately 75 to 150 gpm from the reactor vessel injection line. The surveillance procedures, performed at refueling intervals, call for reactor vessel injection flow to be greater than 1800 gpm, but with the bypass flow subtracted, actual flow could have been as low as 1650 gpm. On August 4, 1995, the findings were reported to the plant Operations Committee, who concluded the event is reportable.

Further review of RHR system maintenance history revealed that there were cases where minor modifications to valves in the reactor vessel injection flowpath were performed without subsequent full flow testing to meet Technical Specification flowpath verification requirements. The cases where full flow was not verified after modifications involved RHR pump discharge check valves and RHR flow control valves. In these cases, alternate testing was performed that verified that the valves were operable.

EVENT DESCRIPTION

Engineering review of surveillance procedures used to satisfy Technical Specification 4.5.B.3.h.2 revealed that Residual Heat Removal Pump (RHR) flow of 1800 gpm through the reactor vessel injection line (EIIS System Identifier BP) has not always been properly verified. The RHR recirculation line tap, which is located downstream of the flow orifice, bypasses approximately 75 to 150 gpm from the reactor vessel injection line. The surveillance procedures, performed at refueling intervals, call for reactor vessel injection flow to be greater than 1800 gpm, but with the bypass flow subtracted, actual flow could have been as low as 1650 gpm. On August 4, 1995, the findings were reported to the plant Operations Committee, who concluded the event is reportable.

Further review of RHR system maintenance history revealed that there were cases where minor modifications to valves in the reactor vessel injection flowpath were performed without subsequent full flow testing to meet Technical Specification flowpath verification requirements. The cases where full flow was not verified after modifications involved RHR pump discharge check valves and RHR flow control valves. In these cases, alternate testing was performed that verified that the valves were operable.

CAUSE OF THE EVENT

This event was the result of a deficient procedure which failed to consider the effect of the recirculation line downstream of the flow orifice.

In the cases where full flow testing was not performed after minor modifications, alternate testing was deemed to satisfy Technical Specification requirements.

ANALYSIS OF THE EVENT

This event is reportable pursuant to 10CFR50.73(a)(2)(i)(B) since the 1800 gpm flow required by Technical Specification 4.5.B.3.h.2 was not verified.

The large break LOCA analysis assumes a flow of 1600 gpm to the reactor vessel. The testing performed has always shown greater than 1600 gpm flow to the reactor vessel.

For the cases where full flow testing was not performed after minor modification, alternate testing provided reasonable assurance that the RHR system would perform as designed. The alternate testing verified valve freedom of movement and also verified the flowpath through the recirculation line.

CORRECTIVE ACTION

Procedure changes will be made which will take recirculation flow into account when determining RHR flow to the reactor vessel.

The event has been discussed with involved personnel.

FAILED COMPONENT IDENTIFICATION

None.

PREVIOUS SIMILAR EVENTS

Deficiencies in surveillance testing performance have been identified as a result of engineering reviews, but this is the first event affecting emergency core cooling systems.