



Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East
Weich, Minnesota 55089

June 6, 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

Two Main Steam Safety Valves Found to Lift Outside of
Their Technical Specification Tolerance of +/- 1%

The Licensee Event Report for this occurrence is attached. In the report, we made no new NRC commitments.

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

MS Wadley for

Roger O Anderson
Director
Licensing and Management Issues

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

Attachment

9506140307 950606
PDR ADDCK 05000306
S PDR

IF22
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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)

Prairie Island Nuclear Generating Plant U2

DOCKET NUMBER (2)

05000 306

PAGE (3)

1 OF 3

TITLE (4) Two Main Steam Safety Valves Found to Lift Outside of Their Technical Specification Tolerance of +/- 1%

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
05	12	95	95	-- 01 --	00	06	06	95	Prairie Island U1	05000 282
									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)							
POWER LEVEL (10)		0	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)	
			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	
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		Abstract below	
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		and in Text, NRC Form 366A)	

LICENSEE CONTACT FOR THIS LER (12)

NAME

Arne A Hunstad

TELEPHONE NUMBER (Include Area Code)

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)

YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
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ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On May 12, 1995, Unit 2 was being shut down for a scheduled refueling outage. The Reactor Coolant System had been borated to the Cold Shutdown boron concentration and the secondary steam system was being maintained at 915 psig in order to test the main steam safety valves. The test involves stabilizing the valve body temperature, then using a hydraulic lift assist device to simulate the higher steamline pressure needed to lift the safety valve. The valves are tested one at a time to determine the actual lift setpoint via a correlation with the hydraulic oil pressure in the test equipment. The as-found lift setpoint for valve RS-21-19 was determined to be 1138 psig, 1.6% above the nominal setpoint of 1120 psig. The as-found lift setpoint for valve RS-21-13 was determined to be 1097 psig, 1.2% below the nominal lift setpoint of 1110 psig. The as-found lift setpoints of the other 8 valves were within the +/- 1% Technical Specification acceptance criteria. Valve RS-21-13 was adjusted so that its as-left lift setpoint was within +/- 1% of their nominal setpoints. Subsequent tests of valve RS-21-19 produced lift setpoints with the +/- 1% tolerance; therefore, no adjustments were made.

NRC FORM 366A (5-92)		U.S. NUCLEAR REGULATORY COMMISSION		APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95	
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION				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)		LER NUMBER (6)	
Prairie Island Unit 2		05000 306		YEAR 95	SEQUENTIAL NUMBER -- 01 --
				REVISION NUMBER 00	PAGE (3) 2 OF 3

TEXT (if more space is required, use additional copies of NRC Form 366A) (17)

EVENT DESCRIPTION

On May 12, 1995, Unit 2 was being shut down for a scheduled refueling outage. The Reactor Coolant System had been borated to the Cold Shutdown boron concentration and the secondary steam system was being maintained at 915 psig in order to test the main steam safety valves. The test involves stabilizing the valve body temperature, then using a hydraulic lift assist device to simulate the higher steamline pressure needed to lift the safety valve. The valves are tested one at a time to determine the actual lift setpoint via a correlation with the hydraulic oil pressure in the test equipment. The as-found lift setpoint for valve RS-21-19 was determined to be 1138 psig, 1.6% above the nominal setpoint of 1120 psig. The as-found lift setpoint for valve RS-21-13 was determined to be 1097 psig, 1.2% below the nominal lift setpoint of 1110 psig. The as-found lift setpoints of the other 8 valves were within the +/- 1% Technical Specification acceptance criteria. Valve RS-21-13 was adjusted so that its as-left lift setpoint was within +/- 1% of their nominal setpoints. Subsequent tests of valve RS-21-19 produced lift setpoints within the +/- 1% tolerance; therefore, no adjustments were made.

CAUSE OF THE EVENT

The instrumentation typically used for safety valve testing has a total combined inaccuracy of slightly less than +/- 1%. For this test, use of new high accuracy (less than +/- 0.5%) digital equipment was planned. Therefore, it was expected that some as-found lift setpoints would likely be outside the +/- 1% specified in Technical Specifications.

In summary, the Technical Specifications limit of +/- 1% does not provide adequate margin to accommodate test instrumentation inaccuracies.

ANALYSIS OF THE EVENT

In anticipation of finding lift setpoints outside the +/- 1% limits, Safety Evaluation #379 had been prepared to justify new acceptance criteria of +/- 3% of the nominal setpoint. Safety Evaluation #379 considered USAR accident analyses and normal operating scenarios. The safety evaluation shows, with a +/- 3% setpoint tolerance, that:

- auxiliary feedwater pumps flow capacity remain within accident design assumptions.
- the Technical Specification safety limit curves and USAR transient analyses are still valid.
- the design function of motor operated valves in the main steam system remains adequate with the increased setpoint tolerance.

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FACILITY NAME (1)		DOCKET NUMBER (2)		LER NUMBER (6)							
Prairie Island Unit 2		05000 306		<table border="1"> <tr> <td>YEAR</td> <td>SEQUENTIAL NUMBER</td> <td>REVISION NUMBER</td> </tr> <tr> <td>95</td> <td>-- 01 --</td> <td>00</td> </tr> </table>		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	95	-- 01 --	00
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				PAGE (3)							
				3 OF 3							

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Test equipment inaccuracies must be included in the measurement to assure the actual setpoints are within the analyzed $\pm 3\%$. The total combined inaccuracy of the test equipment used during this test has been calculated to be less than 0.5% of the nominal setpoints. Therefore, any measured setpoints less than 2.5% of the nominal setpoint are assured to be within the analyzed 3% tolerance. The as-found lift setpoint of valve RS-21-19 was measured at 1.6% above its nominal setpoint, while the as-found lift setpoint of valve RS-21-13 was measured at 1.2% below the nominal setpoint. Both of these measurements were within 2.5% of the nominal setpoints, and thus were bounded by the conclusions of Safety Evaluation #379.

The ASME Section XI allows a 3% measurement tolerance on code safety valve lift setpoints to assure vessel stresses are within design. All measurements were within the $\pm 3\%$ tolerance.

Therefore, based on the Safety Evaluation conclusions and the ASME Code acceptance criteria, there are no safety concerns with these measurements, and the health and safety of the public were unaffected.

Technical Specification 3.4.A.1.a requires that the steam generator safety valves lift settings be within 1% of their nominal setpoints. Since 2 valves were found to lift outside that tolerance band, the event is reportable pursuant to 10CFR50.73(a)(2)(i)(B).

CORRECTIVE ACTION

Lift setpoints for both valves measured outside the $\pm 1\%$ tolerance were left within the tolerance specified in the Technical Specifications.

Safety Evaluation #379 had been prepared in anticipation of this event. The Safety Evaluation provides a basis for a License Amendment Request. A License Amendment Request has been submitted which proposes a change to Technical Specification 3.4.A.1.a to allow as-found valve setpoint tolerances of $\pm 3\%$.

FAILED COMPONENT IDENTIFICATION

None.

PREVIOUS SIMILAR EVENTS

A previous similar event was reported as Unit 1 LER 94-004.