Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East
Welch, Minnesota 55089

February 5, 1996

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

Main Steam Safety Valve
Found Outside the 1% Tolerance

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.

Michael D Wadley

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

130009

IE22

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), 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

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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

TITLE (4)

Main Steam Safety Valve Found Outside the 1% Tolerance

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 05000
1	6	96	96	-- 01 --	00	2	5	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): N

POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jack Leveille	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	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)

On January 6, 1996, Unit 1 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-21-16 was determined to be 1097 psig, 1.9% above the nominal setpoint of 1077 psig. The as-found lift setpoints of the other 9 valves were within the ± 1% Technical Specification acceptance criteria. Valve RS-21-16 was adjusted so that its as-left lift setpoint was within ± 1% of its nominal setpoint.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Prairie Island Nuclear Generating Plant U-1	05000 282	96	-- 01 --	00	2 OF 3

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

EVENT DESCRIPTION

On January 6, 1996, Unit 1 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-21-16 was determined to be 1097 psig, 1.9% above the nominal setpoint of 1077 psig. The as-found lift setpoints of the other 9 valves were within the ± 1% Technical Specification acceptance criteria. Valve RS-21-16 was adjusted so that its as-left lift setpoint was within ± 1% of its nominal setpoint.

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 fact, the current 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.

Test equipment inaccuracies must be included in the measurement to assure the actual setpoints are within the analyzed ± 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

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Prairie Island Nuclear Generating Plant U-1	05000 282	96	-- 01 --	00	3 OF 3

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

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-21-16 was measured at 1.9% above its nominal setpoint. This measurement was 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 ± 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 one valve was found to lift outside that tolerance band, the event is reportable pursuant to 10CFR50.73(a)(2)(i)(B).

CORRECTIVE ACTION

Lift setpoints for the valve measured outside the ± 1% tolerance was 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 ± 3%.

FAILED COMPONENT IDENTIFICATION

None.

PREVIOUS SIMILAR EVENTS

A previous similar event was reported as Unit 1 LER 94-004 and Unit 2 LER 95-001.