NSP

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June 7, 1994

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 one new NRC commitment:

> A License Amendment Request will be prepared which will propose a change to Technical Specification 3.4.A.1.a to allow as-found valve measurement tolerances of +/- 3%.

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

E J Watt 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

IE22
1/1

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

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.

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 U1	05000 282	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: Prairie Island U2	DOCKET NUMBER: 05000 306
05	08	94	94	-- 04 --	00	06	07	94	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)

	20.402(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 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)
Arne A Hunstad	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)

On May 8, 1994, 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 5 safety valves on 1 of the 2 main steam lines are tested each refueling outage. 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-21-1 was determined to be 1098 psig, 2% above the nominal setpoint of 1077 psig. The as-found lift setpoint for valve RS-21-5 was determined to be 1111 psig, 1.8% below the nominal lift setpoint of 1131 psig. The as-found lift setpoints of the other 3 valves were within the +/- 1% Technical Specification acceptance criteria. Valves RS-21-1 and RS-21-5 were adjusted so that their as-left lift setpoints were within +/- 1% of their nominal setpoints.

- 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 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-21-1 was measured at 2.0% above its nominal setpoint, while the as-found lift setpoint of valve RS-21-5 was measured at 1.8% 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 +/- 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 +/- 1% tolerance were returned to 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 will be prepared which will propose a change to Technical Specification 3.4.A.1.a to allow as-found valve measurement tolerances of +/- 3%.

FAILED COMPONENT IDENTIFICATION

None.

PREVIOUS SIMILAR EVENTS

Previous similar events were reported as Unit 1 LER's 81-015 and 92-014 in which a main steam safety valve lift setpoint was found outside the +/- 1% tolerance, but the cause of those events was different.

EVENT DESCRIPTION

On May 8, 1994, Unit 1 was in the process of 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 5 safety valves on 1 of the 2 main steam lines are tested each refueling outage. 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-21-1 was determined to be 1098 psig, 2% above the nominal setpoint of 1077 psig. The as-found lift setpoint for valve RS-21-5 was determined to be 1111 psig, 1.8% below the nominal lift setpoint of 1131 psig. The as-found lift setpoints of the other 3 valves were within the +/- 1% Technical Specification acceptance criteria. Valves RS-21-1 and RS-21-5 were adjusted so that their as-left lift setpoints were within +/- 1% of their nominal setpoints.

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.