Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East
Welch, Minnesota 55089

May 30, 2000

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

LER 2-00-01
Reactor Trip from 22% Power While Shutting Down for Refueling, Caused by Feedwater Heater Hi Hi Level Turbine Trip Signal

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

This event was reported via the Emergency Notification System in accordance with 10 CFR Part 50, Section 50.72, on April 28, 2000. Please contact us if you require additional information related to this event.

Joel P. Sorensen

Joel P. Sorensen
Plant Manager
Prairie Island Nuclear Generating Plant

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

Attachment

IE22

RGN-001

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104
EXPIRES 6/30/01

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT 2	05000 - 306	1 OF 3

TITLE (4)

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
04	28	00	00	01	0	05	30	00	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
POWER LEVEL (10)	22		20.2201(b)		20.2203(a)(2)(v)		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 and in Text, 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 EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

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

ABSTRACT LIMIT TO 1400 SPACES, I.E., APPROXIMATELY 15 SINGLE-SPACED TYPEWRITTEN LINES) (16)
NCR FORM 366 (6-1998)

On April 28, 2000, Unit 2 was undergoing an orderly shutdown in preparation for refueling. During the course of the shutdown, at approximately 2240, the unit tripped from 22% power. The reactor trip was initiated by a turbine trip which was initiated by a Hi Hi Level signal from 23B Feedwater Heater. The unit remains shutdown for refueling.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION	
Prairie Island Nuclear Generating Plant Unit 2	05000 306	00	-- 01 --	00	2 OF 3

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

EVENT DESCRIPTION

On April 28, 2000, Unit 2 was undergoing an orderly shutdown in preparation for refueling. During the course of the shutdown, at approximately 2240, the unit tripped from 21.6% power. The first-out annunciator was 'Turbine Trip-Reactor Trip', indicating that the reactor trip was caused by a turbine[1] trip. Review of the Sequence of Events report from the plant process computer (ERCS) and review of the plant parameters validated that the cause of the reactor trip was a turbine trip. Integrated plant response to the trip was normal. Feedwater isolation valves were closed to maintain steam generator level but dual indication gave ambiguous information regarding their actual positions. Plant response and further evaluation verified that the valves were indeed isolating flow to the steam generators. Operator response to the trip was timely, appropriate, and in accordance with procedures.

CAUSE OF THE EVENT

Cause of the event was a turbine trip due to a momentary 23B Feedwater Heater Hi Hi Level signal. There was a similar trip on November 9, 1998 at 22% power decreasing for a planned refueling shutdown. The cause of that trip was not definitely determined because no method existed to record all turbine trip input signals. An action in response to that trip was to provide inputs from each of the trip signals to the Emergency Response Computer System (ERCS). The trip on April 28, 2000 was definitely confirmed to be from 23B Feedwater Heater Hi Hi Level.

There was not an actual Hi Hi level condition in the heater but rather an anomalous signal from the level instrumentation. The level control system instrumentation and valves for the low pressure feedwater heaters[2] were checked for calibration and proper operation. All setpoints were found to be in tolerance and the instrumentation in good condition. It has been concluded that the cause is related to the piping and valving geometry associated with the level instrumentation[3] for 23B heater, allowing for a momentary Hi Hi level signal to be generated due to flashing in the lower sensing line of the Hi Hi level switch. In this event, the signal was activated for only 0.8 seconds.

ANALYSIS OF THE EVENT

This event is reportable under 10CFR50.73(a)(2)(iv) as an unplanned actuation of the reactor protection system. The health and safety of the public were unaffected since the plant systems responded as designed to the automatic trip.

[1] EIIS Component Identifier: TRB
[2] EIIS Component Identifier: HX
[3] EIIS Component Identifier: LI

Significance Determination

No specific risk assessment has been performed because the reactor trip and plant response were as expected with no equipment failures.

Performance Indicators Assessments

Since no system, structure, or component was inoperable, this event did not involve either a partial or complete loss of a safety system function.

This event affects the number of unplanned scrams per 7,000 critical hours.

However, the event did not involve a scram with a loss of normal heat removal.

CORRECTIVE ACTION

The feedwater isolation valves' torque switch settings have been adjusted to operate unambiguously.

The feedwater heater drain level control system was thoroughly inspected and calibrated.

A globe valve[4] in the instrumentation piping has been replaced with a gate valve and its orientation in the line adjusted. The piping has been checked for proper slope and adjusted.

Instrumentation and procedural changes will be made to avoid trips due to transient level indications while minimizing exposure to turbine damage due to excessive level in the feedwater heaters.

FAILED COMPONENT IDENTIFICATION

None.

PREVIOUS SIMILAR EVENTS

The turbine trip / reactor trip of November 9, 1998 is considered to be similar, LER 2-98-05.

[4] EIIS Component Identifier: V