Northern States Power Company
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East
Welch, Minnesota 55089

April 18, 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

Reactor Trip Caused by Failure of Feedwater Regulating Valve

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 March 19, 1996. 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

220066

JE22

NRC FORM 366 (4-95) 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 04/30/98

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Prairie Island Nuclear Generating Plant Unit 2	05000 306	1 OF 3

TITLE (4)
Reactor Trip Caused by Failure of Feedwater Regulating Valve

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
03	19	96	96	-- 01 --	00	4	18	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): N
POWER LEVEL (10): 100

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

	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)	X	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
B	SJ	FCV	C635	Yes						

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)

At approximately 0011 hours on March 19, 1996, while operating at 100 percent reactor power, Unit 2 experienced a reactor trip generated by a protective signal received from Low Low Steam Generator Water Level. Significant observations by Control Room personnel included indication of 100 percent demand signal on CV-31135 and indications of decreasing water level in 21 Steam Generator with no feedwater flow in Loop A. Additionally, a lock-in trouble annunciation for the feedwater control system was received. Also heard and felt by Operations personnel immediately preceding this event was a loud thud which was probably water hammer.

After the plant was in a stable condition, an investigation was begun and the cause of the trip was determined. CV-31135, Loop A Feedwater Regulating Valve, was isolated and stroked with no externally visible impairment of operation. But because of observed indications and plant response, it was decided to disassemble CV-31135 as it was the most likely candidate to initiate this event. During disassembly of CV-31135 it was observed that the valve plug had separated from the valve stem.

The valve stem/valve plug separation resulted in the valve plug dropping completely into the valve cage obstructing all Loop A feedwater flow to 21 Steam Generator. This ultimately resulted in the loss of the unit on a Low Low Steam Generator Water Level reactor trip.

EVENT DESCRIPTION

At approximately 0011 hours on March 19, 1996, while operating at 100 percent reactor power, Unit 2 experienced a reactor trip generated by a protective signal received from Low Low Steam Generator Water Level. Significant observations by Control Room personnel included indication of 100 percent demand signal on CV-31135 and indications of decreasing water level in 21 Steam Generator (EIIS Component ID - SG) with no feedwater (EIIS System ID - SJ) flow in Loop A. Additionally, a lock-in trouble annunciation for the feedwater control system (EIIS System ID - JB) was received. Also heard and felt by Operations personnel immediately preceding this event was a loud thud which was probably water hammer.

The plant response to the reactor trip was as expected.

After the plant was in a stable condition, an investigation was begun and the cause of the trip was determined. CV-31135, Loop A Feedwater Regulating Valve (EIIS Component ID - FCV), was isolated and stroked with no externally visible impairment of operation. But because of observed indications and plant response, it was decided to disassemble CV-31135 as it was the most likely candidate to initiate this event.

CAUSE OF THE EVENT

During disassembly of CV-31135 it was observed that the valve plug had separated from the valve stem. Visual examination revealed that the valve stem had fractured leaving a small piece remaining in the valve plug. Examination of the remnant piece suggested a ductile tensile separation with no visual indication of torsional stress. However, the through pin between the valve plug and the valve stem showed obvious torsional shear. The assembly of the valve stem into the valve plug is a threaded connection beneath a taper joint. The valve stem is torqued into the taper joint on assembly, creating a friction interface between the mated surfaces which assists in sustaining the joint. The joint is then cross drilled with a through pin inserted to maintain the friction interface. It is unlikely at this point that any tensile preload is developed in the threaded area. It is known that a large torsional stress is exerted on the valve stem/plug during low power/low flow operations and has been a contributing factor in similar failures. It is surmised, based upon the visual evidence, that this torsional stress created the shear failure in the through pin. The failure of the through pin allowed the valve stem to begin to unthread from the valve plug undermining the friction interface. This condition now exposed the valve stem to tensile loading and would exacerbate any pre-existing material flaw conditions. This resulted at some point in the ultimate failure of the valve stem with the attendant separation of the valve plug.

Visual examination of CV-31135 indicates that the most likely cause of the component failure appears to be a pre-existing material flaw that when exposed to tensile stress resulted in the fracture of the valve stem. The torsional stress that low power/low flow conditions create provided the opportunity for the stem defect(s) to become exposed to the tensile stress which ultimately led to the valve stem/valve plug separation. The valve stem/valve plug separation resulted in the valve plug dropping completely into the valve cage obstructing all Loop A feedwater flow to 21 Steam Generator. This ultimately resulted in the loss of the unit on a Low Low Steam Generator Water Level reactor trip.

ANALYSIS OF THE EVENT

This event is reportable pursuant to 10CFR50.73(a)(2)(iv) since this was an unplanned actuation of the reactor protection system. Response of the primary system was as expected. Health and safety of the public were unaffected.

CORRECTIVE ACTION

Prior to return to power after rework of CV-31135, CV-31136 (Loop B Feedwater Regulating Valve) was stroked to confirm that the existing condition matched the original calibration data and that there was no indication of a similar separation on loop B.

It is planned to modify the joint configuration to increase the joint resistance to any applied torsional moment and thereby reduce exposure to similar future failures.

It is planned to review the operating procedures and the digital feedwater control system to determine if it is possible to transit through the period of low flow conditions on reactor startup more expeditiously. This would reduce exposure to the window in which the feedwater control valves are exposed to the large torsional force. Before any operational changes are considered, however, the valve stem torsional stresses will be measured for varying plant conditions in order to evaluate the need for any operational changes.

FAILED COMPONENT IDENTIFICATION

Copes-Vulcan 12 inch, 900 #, air-operated control valve, Model No. D-100

PREVIOUS SIMILAR EVENTS

There have been no other similar events reported.