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December 11, 2017

L-PI-17-049
10 CFR 50.73

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant, Unit 2
Docket No. 50-306
Renewed Facility Operating License No. DPR-60

LER 50-306/2017-002-00, Reactor Coolant System Shutdown Communication Line Vent Through Wall Defect

Northern States Power Company, a Minnesota Corporation (NSPM), doing business as Xcel Energy, hereby submits Licensee Event Report (LER) 50-306/2017-002-00, Reactor Coolant System Shutdown Communication Line Vent Through Wall Defect.

If there is any question or if any additional information is needed, please contact Leonard Sueper, at 612-330-6917.

Summary of Commitments

This letter makes no new commitments and no revisions to existing commitments.

Scott Northard
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC

ENCLOSURE

Licensee Event Report 50-306/2017-002-00

3 pages follow

NRC FORM 366 (04-2017)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104	EXPIRES: 03/31/2020

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Prairie Island Nuclear Generating Plant	05000306	1 OF 3

4. TITLE
Reactor Coolant System Shutdown Communication Live Vent Through Wall Defect

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
10	16	2017	2017	- 002	- 00	12	11	2017	FACILITY NAME	DOCKET NUMBER
										05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
5	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☒ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
0	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Leonard Sueper, Senior Regulatory Engineer	(612) 330-6917

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	AB	VTV	V085	Y	N/A	N/A	N/A	N/A	N/A

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On October 16, 2017, with Unit 2 shutdown for a refueling outage, investigation into a boric acid indication identified a through wall leak at the socket weld that joins a 3/4 inch line to Loop A Reactor Coolant System (RCS)[AB] shutdown communication line valve 2RC-8-37)[VTV]. The leak was isolated by closed valves that would have limited primary coolant leakage to within the capacity of the charging system when the reactor coolant system was pressurized. The quantity of dry boric acid at the location was small (estimated at 1/2 teaspoon in volume). This failure constituted a welding or material defect in the primary coolant system that was not found acceptable under ASME Section XI and an event or condition prohibited by Technical Specifications.

The cause of the leakage was determined to be stress corrosion cracking. Valve 2RC-8-37 was replaced. In addition, Prairie Island Nuclear Generating Plant intends to perform phased array ultrasonic inspections of socket welds on similar Class 1 piping containing stagnant water during future refueling outages.

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NARRATIVE

DESCRIPTION OF EVENT

On October 16, 2017, with Prairie Island Nuclear Generating Plant (PINGP) Unit 2 shutdown for refueling outage 2R30, an indication of leakage was found on the pipe socket weld upstream of valve 2RC-8-37 [VTV] during boric acid corrosion examinations. Valve 2RC-8-37 functions as a vent path on the normally isolated 3/4 inch Loop A shutdown communication line [AB]. Subsequent non-destructive examination confirmed a pressure boundary leak existed, which was not found acceptable under ASME Section XI.

EVENT ANALYSIS

This event is being reported in accordance with 10 CFR 50.73(a)(2)(ii)(A), “Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded” because the weld defect in the primary coolant system was not found acceptable under ASME Section XI. In addition, the pressure boundary leakage exceeded the zero reactor coolant system operational leakage limit specified in Tech Spec 3.4.14 and therefore was reportable per 50.73(a)(2)(i)(B) as a condition prohibited by plant technical specifications.

CAUSE

Laboratory analysis determined the cause of the leak to be stress corrosion cracking. Contributing causes included weld material sensitization and the presence of sulfur and oxygen on internal surfaces. Oxygen is introduced from containment air when the system is in use during outages and is not removed by the normal primary system oxygen scavenging mechanisms because the line is normally isolated / stagnant.

SAFETY SIGNIFICANCE

There was no actual safety consequence associated with this event. The quantity of dry boric acid at the location was small (estimated at 1/2 teaspoon in volume). Even if the weld had experienced a complete circumferential failure, the leak was isolated by closed valves that would have limited primary coolant leakage to within the capacity of the charging system when the reactor coolant system was pressurized (see figure next page).

CORRECTIVE ACTIONS

Valve 2RC-8-37 and the associated weld were replaced and the reactor coolant pressure boundary was restored.

PINGP intends to perform phased array ultrasonic inspections of socket welds on similar Class 1 piping containing stagnant water during future PINGP Unit 1 and 2 refueling outages.

PREVIOUS SIMILAR OCCURRENCES

A similar boric acid deposit was previously identified during refueling outage 2R29 in 2015 on valve 2RC-8-37 at the same location. However, visual and dye penetrant testing performed in accordance with plant procedures showed no indication of flaws or an active leak. The origin of the boric acid residue on the valve at the time was incorrectly attributed to a prior leak (e.g. reactor coolant pump seal leak) or maintenance activity.

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