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Prairie Island Nuclear Generating Plant
Northern States Power Company, Minnesota

06/05/2009

L-PI-09-064
10 CFR 50.73

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

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

LER 1-09-04, Residual Heat Removal System Inoperability While in Mode 4 Due to Potential Steam Voiding

Northern States Power Company, a Minnesota corporation (NSPM) herewith encloses Licensee Event Report (LER) 1-09-04. The LER describes multiple occurrences of a condition where, while in Mode 4, both trains of the Residual Heat Removal (RHR) System were inoperable. The conditions were identified in both Prairie Island Nuclear Generating Plant Unit 1 and Unit 2.

The condition was identified as a result of scoping for Generic Letter 2008-01.

Summary of Commitments

This letter contains no new commitments and no changes to existing commitments.

Michael D. Wadley

Michael D. Wadley
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
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ENCLOSURE

LICENSEE EVENT REPORT 1-09-04

4 Pages Follow

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES: 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0066), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Prairie Island Nuclear Generating Plant (PINGP) Unit 1	05000282	1 of 4

4. TITLE
Residual Heat Removal System Inoperability While in Mode 4 Due to Potential Steam Voiding

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 PINGP Unit 2	DOCKET NUMBER 0500306
04	06	2009	2009	– 004 –	00	06	05	2009	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: Mode 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☑ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☑ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☑ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Jorge O'Farrill	651.388.1121

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

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)*

During scoping for Generic Letter 2008-01, a concern with the Residual Heat Removal System (RHR) was identified. High temperature water had the potential to flash to steam during a loss of coolant accident (LOCA) scenario while in Mode 4. Subsequent analysis determined that, while in Mode 4, the temperature of the fluid at the RHR pump suction header exceeded 226 degrees Fahrenheit (F) which, under LOCA conditions, could have resulted in a void being entrained into the common RHR pump suction. This condition rendered both trains of RHR inoperable.

This condition was allowed to occur because operating procedures failed to address voiding due to hot fluid in the RHR piping. An additional contributor to the condition was the failure to adequately address Westinghouse Nuclear Safety Advisory Letter (NSAL) 93-004. Current corrective actions have been issued to modify operating procedures to manage RHR void formation.

EVENT DESCRIPTION

On December 16, 2008 Prairie Island Nuclear Generating Plant (PINGP) performed ultrasonic testing measurements on residual heat removal[1] (RHR) piping susceptible to void formation in order to address gas accumulation concerns mentioned in Generic Letter (GL) 2008-01. A void was found on the common line from the Unit 1 Reactor Coolant System (RCS) hot leg piping to both of the RHR pump suctions. An engineering analysis determined that operability of the RHR system was not affected by this void.

In March 2009, based on the past operability analysis performed for the December 16, 2008 void and an operating experience (OE) report from Wolf Creek dated October 2008, PINGP concluded that the RHR system piping in both units would be vulnerable to void formation under certain plant conditions and must be considered inoperable under those conditions. High temperature water in the RHR system has a potential to flash to steam during a loss of coolant accident (LOCA) scenario during heat-up in Mode 3 (Hot Standby) or during heat-up or cooldown in Mode 4 (Hot Shutdown). When the RHR water temperature is above 226 degrees F in Modes 3 or 4, steam voids could potentially form and transport to the RHR pump suction which would render the pump inoperable. As a result, the RHR system must be considered inoperable any time the RHR system temperature was above 226 degrees F in Modes 3 or 4.

On April 6, 2009 PINGP engineering personnel evaluated the applicability of the potential void formation conditions described by the March 2009 past operability analysis. Engineering personnel determined that within the previous three years the RHR system was sufficiently cool before entering Mode 3 such that inoperability while in Mode 3 was not a concern for either Unit 1 or Unit 2. However, within the previous three years there were multiple identified occurrences where, while in Mode 4, RHR system temperature was greater than 226 degrees F which rendered both trains of the RHR system inoperable.

In Mode 4, a flow path is required to provide recirculation flow via the RHR subsystem from the containment sump into each of the reactor vessel upper plenum nozzles. The inoperability of both trains of the RHR system under the described condition would cause a loss of safety function for the RHR system. Thus, this condition is reportable under 10 CFR 50.73(a)(2)(ii)(B) for being in an unanalyzed condition that significantly degrades plant safety, 10 CFR 50.73(a)(2)(v)(B) as a condition that could have prevented fulfillment of a safety function, and 10 CFR 50.73(a)(2)(vii) due to the common cause inoperability of two independent trains in the same system.

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[1] EIIS System Identifier: BP

EVENT ANALYSIS

In 1993, Westinghouse completed an industry evaluation concerning the potential for water to flash to steam in the RHR pump suction line. Nuclear Safety Advisory Letter (NSAL) 93-004 recommended that plant operating procedures be reviewed to verify that the potential for forming steam voids was precluded. One option presented was to force cool the piping. Since PINGP procedures use forced cooling, it was concluded that the guidance had been met. However, it appears that both the preparer and reviewer of the response to NSAL-93-004 failed to recognize that the forced cooling line returns too close to the RHR pump suction to effectively cool all of the RHR suction piping. Additionally, NSAL-93-004 did not provide recommendations for protecting one train of RHR for emergency core cooling system (ECCS) function while in Mode 4.

The issuance of GL 2008-01 by the NRC and an industry OE report by Wolf Creek prompted PINGP staff to analyze the potential vulnerabilities of the RHR system during an ECCS actuation for a LOCA. If a steam void is entrained into the RHR pump suction, the pump could become inoperable.

A calculation was done to determine the maximum allowable RHR temperature to preclude void formation in the RHR pump suction header. The temperature was determined to be the saturation temperature of the fluid for the minimum pressure that would be present at the RHR pump suction header. This pressure was determined by considering the static head of and ambient pressure on the fluid supplied to the pump, sump strainer losses, and piping losses affecting the system. Due to this minimum pressure, maximum temperature allowable at the RHR hot leg suction was determined to be 226 degrees F.

The April 6, 2009 past operability review by engineering personnel determined that the total time over the previous three years that Unit 1 RHR would have been inoperable due to this condition was 47 hours, 40 minutes, with the longest single inoperability being, 19 hours, 28 minutes. The total time that Unit 2 RHR would have been inoperable due to this condition was 77 hours, 43 minutes, with the longest single inoperability being 23 hours, 8 minutes.

Both trains of RHR being rendered inoperable during the described condition represents a safety system functional failure as described in 10 CFR 50.73(a)(2)(v)(B).

SAFETY SIGNIFICANCE

The potential existed during Mode 4 for either a forced outage or refueling outage that the RHR system would not have functioned as designed during a LOCA. At Mode 4, RCS temperature is maintained between 200 degrees F - 350 degrees F and pressure is maintained below 425 psig before the RHR system is placed in-service. At these temperatures and pressures, which are well below Mode 1 values, the probability of occurrence of a design basis accident is reduced.

In the event that the RHR system became inoperable, abnormal and emergency procedures would provide guidance to restore cooling to the core. When the plant is in Mode 4 and above 218 degrees F, a safety injection pump would be available to provide flow to the RCS in a timely manner and the RHR pumps need not be relied on for short-term LOCA mitigation. With the high head Safety Injection (SI) pump operating, the refueling water storage tank water inventory would eventually deplete causing the need for recirculation. During this time, should steam binding of the RHR pumps occur, the operating SI pump would have no suction source from the RHR pump discharge. However, the charging system would be available to provide highly borated water via emergency boration. Finally, steam generators and auxiliary feedwater would also be available to remove decay heat if needed.

There were no actual consequences to the health and safety of the public and the safety significance of this event is considered minimal.

CAUSE AND CORRECTIVE ACTIONS

The cause for this declared inoperability was the failure of PINGP procedures to adequately support all functions of RHR operation in Modes 3 and 4 without impacting one another. Additional causes include the failure to adequately address Westinghouse NSAL-93-004 in 1993 and the failure to take long term corrective actions to prevent voiding after the water hammer event at PINGP in 1999.

A formal calculation of the allowable RHR water temperature has been completed. Based on the results of the calculation, procedures will be revised to manage RHR void formation.

PREVIOUS SIMILAR EVENTS

On January 9, 1999, preparations were made to place RHR in service for plant cooldown. When the isolation valve between the hot leg and RHR pump suction was opened, the valve stopped after approximately one quarter travel. Operators observed a pressurizer level and pressure decrease. The likely cause of this transient was water hammer created when RCS water collapsed a void in the common RHR hot leg suction piping.

The report for this non-conformance concluded that this void was formed as a result of operating procedures in Modes 3 and 4. Short term corrective actions included an addition to procedures to eliminate the possibility of water hammer by filling the line before placing shutdown cooling in service. However, long term corrective actions were not completed to eliminate the voiding of the hot leg suction line.