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Monticello Nuclear Generating Plant
2807 W County Road 75
Monticello, MN 55362

January 21, 2016

L-MT-16-006
10 CFR 50.73

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

Monticello Nuclear Generating Plant
Docket 50-263
Renewed Facility Operating License No. DPR-22

LER 2015-007, "Loss of Residual Heat Removal Capability"

Enclosed, is the Monticello Nuclear Generating Plant (MNGP) Licensee Event Report (LER) 2015-007 regarding a loss of Residual Heat Removal capability. This condition is reportable to the NRC in accordance with 10 CFR 50.73(a)(2)(v)(B), as an Event or Condition that Could have Prevented the Fulfillment of the Safety Function of Structures or Systems that are Needed to Remove Residual Heat.

Summary of Commitments

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

Peter A. Gardner
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosure

cc: Regional Administrator, Region III, USNRC
Project Manager, MNGP, USNRC
Resident Inspector, MNGP, USNRC

NRC FORM 366 (02-2014)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 01/31/2017

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE

Loss of Residual Heat Removal Capability

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
11	24	2015	2015 -	007	- 00	01	21	2016	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
3	☐ 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)
10. POWER LEVEL	☐ 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)
0 %	☐ 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

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Andrew Kouba, Licensing Engineer	(763) 271-7251

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
N/A	N/A	N/A	N/A	N/A	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 November 24, 2015 at 0534 hours, the Monticello Nuclear Generating Plant was at 0% power in Mode 3 (Hot Shutdown) for a forced outage. While initially placing Shutdown Cooling (SDC) in service, the 12 Residual Heat Removal (RHR) pump tripped approximately 8-10 seconds after start due to the closure of the RHR SDC suction isolation valves. When placing SDC in service, flow rapidly increased after opening the RHR Division 2 Low Pressure Coolant Injection (LPCI) outboard injection valve causing a localized pressure transient in the reactor recirculation pump suction piping that resulted in an isolation of the SDC suction line. Reactor pressure vessel (RPV) pressure remained stable at approximately 30 psig.

Prior to attempting to place 'B' SDC in service, the Condensate system and the 'F' Safety Relieve Valve were in service providing decay heat removal. Immediate actions were taken to restore 'B' RHR SDC to operable status, thus an alternative method of decay heat removal was already established by the Condensate system and 'F' Safety Relief Valve.

NARRATIVE

EVENT DESCRIPTION

On November 24, 2015 at 0534 hours, the Monticello Nuclear Generating Plant was at 0% power in Mode 3 (Hot Shutdown) for a forced outage. While initially placing Shutdown Cooling (SDC) [KE] in service, the 12 Residual Heat Removal (RHR) [BO] pump [P] tripped approximately 8-10 seconds after start due to the closure of the RHR SDC suction isolation valves [ISV].

When placing SDC in service, flow was rapidly increased after opening the RHR Division 2 Low Pressure Coolant Injection (LPCI) outboard injection valve [INV], causing a localized pressure transient in the reactor recirculation pump suction piping that resulted in an isolation of the SDC suction line. The RHR High Reactor Pressure annunciator [PA] was received and immediately cleared as the pressure switch [PS] upstream of 12 Recirculation Pump [AD] Suction valve in the 'B' Recirculation Loop actuated causing a Group 2 containment isolation signal. However, this was not expected as Reactor Pressure Vessel (RPV) [RPV] steam dome pressure remained stable at approximately 30 psig.

At 0535 hours, immediate actions were taken to restore 'B' RHR SDC to operable status.

At 0545 hours, an alternative method of decay heat removal was established by utilizing the Condensate [SD] system and 'F' Safety Relief valve [RV].

At 1354 hours, the 12 RHR pump and 12 RHR Service Water pump were successfully placed in service on SDC and the plant reached Mode 4 (Cold Shutdown) at 1428 hours.

EVENT ANALYSIS

The event was determined to be reportable in accordance with 10 CFR 50.73(a)(2)(v)(B) as an Event or Condition that Could have Prevented the Fulfillment of the Safety Function of Structures or Systems that are Needed to Remove Residual Heat. This event is considered a Safety System Functional Failure per NEI 99-02, Revision 7.

SAFETY SIGNIFICANCE

Prior to attempting to place 'B' SDC in service, the Condensate system and the 'F' Safety Relieve Valve were in service providing decay heat removal. These systems remained in service and, as demonstrated by steadily lowering RPV pressure and temperature, provided adequate decay heat removal until SDC was placed in service. Additionally, the Reactor Water Cleanup System [CE] was available for decay heat reject if needed. After the closure of the SDC suction valves and subsequent trip of the 12 RHR pump, immediate actions were taken to restore SDC to operable status. Since the reactor remained adequately cooled, there were no actual consequences as a result of the initial failed attempt to place SDC in service. There was no impact to the health and safety of the public.

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NARRATIVE

CAUSE

Both reactor high pressure SDC isolation pressure switches are located on the 'B' Recirculation Suction Piping. When initially placing SDC in service the LPCI outboard injection valve was opened and flow into the 'B' Recirculation system increased to approximately 4000 gpm in several seconds. This rapid flow increase caused a localized pressure transient in the 'B' Recirculation pump piping that resulted in the isolation of the SDC suction valves. Closure of the SDC suction valves subsequently caused a trip of the 12 RHR pump due to loss of pump suction. Written documentation in the operations manual did not adequately address the sensitivity of the pressure switches while placing 'B' SDC in service.

CORRECTIVE ACTION

Since the Condensate system and the 'F' Safety Relieve Valve were already in service providing decay heat removal, an alternate method of decay removal did not need to be established. Immediate actions were taken to restore 'B' SDC to operable status.

The Operations Manual used to place 'B' SDC in service was re-performed in its entirety to verify proper valve alignment, ensure the piping was full of water, and verify acceptable temperatures existed prior to attempting to place the system in service. This included venting the RHR suction and discharge lines prior to placing 'B' SDC in service. Existing procedural guidance allowed the associated LPCI injection valve to be slowly throttled open to achieve required RHR pump flow without introducing a pressure transient that would challenge the reactor high pressure SDC isolation setpoint.

The 12 RHR pump was successfully started and placed in SDC mode to cool down the plant to MODE 4. The Operations Manual has been updated to provide additional guidance for placing SDC in service including the pressure switch sensitivity to injection flow rate changes when changing the position of the LPCI outboard injection valve.

PREVIOUS SIMILAR EVENTS

There were no Licensee Event Reports with similar causes of loss SDC within the 3 last years.

ADDITIONAL INFORMATION

The Institute of Electrical and Electronics Engineer codes for equipment are denoted by [XX].