Xcel Energy®

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

January 21, 2016

L-MT-16-005
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 2015-006-00 "Reactor Scram due to Group 1 Isolation from Foreign Material in the Main Steam Flow Instrument Line"

Enclosed is the Monticello Nuclear Generating Plant (MNGP) Licensee Event Report (LER) 2015-006-00, Reactor Scram due to Group 1 Isolation from Foreign Material in the Main Steam Flow Instrument Line. This condition is reportable to the NRC in accordance with 10 CFR 50.73(a)(2)(iv)(A), as an event that resulted in automatic actuation of a reactor protection system.

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)	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 FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), 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.

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4. TITLE
Reactor Scram due to Group 1 Isolation from Foreign Material in the Main Steam Flow Instrument Line

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	23	2015	2015	- 006	- 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)*			
1	☐ 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)
100 %	☐ 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)*
Stephen Sollom, Licensing Engineer	(763) 271-1611

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
E	AD	MG	G080	Y	X	SB	SOL	N/A	Y

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 23, 2015, a trip of the # 11 Reactor Recirculation Pump occurred, followed by a Group 1 isolation which resulted in a reactor scram. A post scram troubleshooting investigation determined a large spike in differential pressure occurred in the 'C' main steam flow instrumentation line at the time of the Group 1 initiation event.

The root cause of this event was determined to be legacy foreign material present in the 'C' main steam flow instrumentation line. This foreign material obstructed the instrumentation line and resulted in the momentary sensed high steam flow. The sensed high steam flow was not due to an actual high steam flow condition in the 'C' main steam line.

Since the presence of foreign material in the instrument line is a legacy issue, the corrective action for the root cause was to remove the foreign material. The corrective action for the trip of the reactor recirculation pump, will be to revise the fleet procedure to require verification of torque on accessible electrical connections for critical components which are bench tested and also to ensure that accessible soldered and crimped electrical terminations are inspected for signs of degradation during bench testing.

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NARRATIVE

EVENT DESCRIPTION

On November 23, 2015, at approximately 10:40, with the plant in Mode 1 and 100 percent power, a trip of the # 11 Reactor Recirculation Pump [AD] occurred. Following the # 11 Reactor Recirculation Pump trip, at approximately 10:41, a large spike in differential pressure on the 'C' main steam line flow instrumentation line occurred. Due to this differential pressure spike, all four of the Main Steam Line 'C' High Flow Isolation Pressure Differential Indicating Switches [PDIS] actuated, causing a Group 1 isolation. The Group 1 isolation caused the main steam isolation valves (MSIV) [ISV] to close, resulting in a reactor scram due to the MSIVs being less than 90% open at approximately 10:41. The 'D' outboard MSIV did not close until approximately 130 seconds after the Group 1 signal was sent. The other seven MSIVs closed as expected. All control rods [AA] inserted, and all other safety systems responded as expected. Division II secondary containment isolation instrumentation, secondary containment isolation valves, and the Standby Gas Treatment System were inoperable for planned maintenance prior to the event.

Also prior to the event, the # 4 Turbine Stop Valve 90% open limit switch, which inputs to the reactor protection system (RPS), was inoperable and the channels had previously been placed in trip to meet Technical Specification Required Actions 3.3.1.1.A and 3.3.1.1.B and remained in trip throughout the event. The inoperable equipment did not impact the scram response.

With the MSIVs closed, normal Reactor Pressure Vessel (RPV) [RPV] pressure control [PC] was not available. The Low-Low Set System operated as expected to control reactor pressure by cycling the 'H' Safety Relief Valve (SRV) [RV]. The 'H' SRV first opened at approximately 10:49. At approximately 10:58, the High Pressure Coolant Injection (HPCI) [BJ] System was placed in service in the pressure control mode. The 'H' SRV cycled five (5) times prior to placing HPCI in service and cycled one additional time after HPCI was placed in pressure control mode.

Following the scram, RPV water level immediately decreased to +3 inches, causing a Group 2 Primary Containment Isolation signal when RPV level decreased below +9 inches. The Group 2 signal remained in effect until it was reset at approximately 13:05. In response to the RPV water level reduction, the Feedwater Regulating Valves (FRVs) opened to restore RPV level. With the MSIVs closed, the only steam exiting the RPV was through the 'H' SRV and the steam supplied to operate the HPCI system, however pressure was below the 'H' SRV Low-Low Set setpoint and HPCI had not yet been placed in service. Under these conditions, RPV water level rose. Operations responded by closing the FRVs. This action lessened the RPV level increase, but RPV water level continued to rise, tripping the Reactor Feedwater Pumps at approximately 10:42.

At approximately 11:02, with HPCI in the pressure control mode and RPV water level at approximately +22 inches the final actuation of the 'H' SRV occurred. Indicated RPV level immediately increased above the high RPV level setpoint of +48 inches, causing a trip of the HPCI System. HPCI could not be placed in service until the 'H' SRV closed and RPV water level was below +48 inches. At approximately 11:03, the 'H' SRV closed and indicated RPV water level lowered to +6 inches. HPCI was once again placed in service in the pressure control mode and the # 11 Reactor Feedwater Pump (RFP) was started.

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The HPCI System was able to control RPV pressure without additional actuations of the 'H' SRV and the Condensate and Feedwater System was able to control RPV water level for the remainder of the event.

Following plant stabilization, Mode 4 was achieved on November 24, 2015, at approximately 14:28.

EVENT ANALYSIS

Pursuant to 10 CFR 50.72 paragraph (b)(2)(iv)(B) for the RPS actuation and paragraph (b)(3)(iv)(A) as an emergency safety feature (ESF) actuation, a four hour event notification was made to the NRC. The event was determined to be reportable in accordance with 10 CFR 50.73 (a)(2)(iv)(A), "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph 10 CFR 50.73 (a)(2)(iv)(B)".

This event is not classified as a safety system functional failure.

SAFETY SIGNIFICANCE

The occurrence of the Group 1 isolation signal caused by the 'C' main steam line flow instrument did not compromise safety systems or components required to mitigate the consequences of an accident or transient, with the exception of temporarily resulting in losing the Main Condenser heat sink as an alternative means for decay heat removal. The excess flow check valve (EFCV) on the upstream 'C' main steam line flow instrument was degraded by the pressure wave that caused the Group 1 isolation. The EFCV is designed to isolate the downstream instrument line in the event of an instrument line break. The integrity of the line was not compromised during the event. Therefore, there were no consequences resulting from the failure of this valve. The Group 1 isolation and subsequent reactor scram is within the analyzed and anticipated frequency of a Group 1 initiation and is bounded by the Main Steam Line Break accident analysis documented in the USAR.

The 'D' outboard MSIV was delayed in closing following the Group 1 isolation signal. However, the 'D' inboard MSIV closed within the required time frame, isolating the 'D' main steam line. Furthermore, subsequent evaluation has determined a condition requiring a Group 1 isolation did not exist. All other safety systems and components responded as required to mitigate the transient and ensure adequate core cooling. Operators performed as expected and in accordance with procedures.

In conclusion, this transient that occurred on November 23, 2015 did not impact the health and safety of the public.

CAUSE

The Group 1 isolation and subsequent reactor scram was caused by legacy foreign material in the common high side instrument line tap of the 'C' main steam flow instrument line which obstructed proper steam line flow instrument operation. This foreign material resulted in a high sensed flow condition by all four flow switches on the 'C' main steam flow instrument line during the down-power event (loss of a recirculation pump).

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The foreign material in the instrument line is considered a legacy issue (believed to be from initial construction). Foreign Material Exclusion (FME) programs and controls have been improved to preclude similar conditions since that time. Therefore, this legacy issue is not indicative of current plant programs, processes, or behaviors.

The cause of the #11 Reactor Recirculation Pump trip that precipitated the down power was less than adequate maintenance strategy that did not provide sufficient process controls to ensure the bench testing of high critical components identified and then corrected equipment deficiencies prior to installation. As a result, a loose wire on the #11 Recirculation Pump Motor-Generator set voltage regulator was not detected and corrected prior to the event. This loose wire led to a trip of the #11 Reactor Recirculation Pump, a core flow reduction, and a reactor power reduction.

CORRECTIVE ACTION

Immediate corrective actions included removing the foreign material (nut) from the instrumentation line and properly attaching the loose wire. The EFCV and the MSIV issues were fixed prior to plant start up. Long term corrective actions are to revise a fleet procedure to require verification of proper torque on accessible electrical connections for critical components which are bench tested, and also to ensure that accessible soldered and crimped electrical terminations are inspected for signs of degradation during bench testing.

Since the original plant construction, the FME programs and controls have been improved to preclude similar conditions. Therefore, no additional FME corrective actions are required.

PREVIOUS SIMILAR EVENTS

There were no previous similar licensee event reports in the past three years.

ADDITIONAL INFORMATION

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