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June 14, 2017

L-MT-17-046
10 CFR 50.90

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

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

LER 2017-003-00 "Main Steam Isolation Valve Leakage Exceeds Technical Specification Requirements"

Enclosed is the Monticello Nuclear Generating Plant (MNGP) Licensee Event Report (LER) 2017-003-00, "Main Steam Isolation Valve Leakage Exceeds Technical Specification Limits." This condition is reportable to the NRC in accordance with 10 CFR 50.73(a)(2)(i)(B), as an operation or condition which was prohibited by the plant's Technical Specifications.

Summary of Commitments

This letter makes 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: Administrator, Region III, USNRC
Project Manager, Monticello Nuclear Generating Plant, USNRC
Resident Inspector, Monticello Nuclear Generating Plant, USNRC

NRC FORM 366 (06-2016) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018
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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Main Steam Isolation Valve Leakage Exceeds Technical Specification Limits

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
04	20	2017	2017	- 003	- 00	06	14	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)
10. POWER LEVEL	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
000	☐ 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 NUMER *(Include Area Code)*
Steve Sollom, Licensing Engineer	763-295-1611

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
B	SB	ISV	A391	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 April 20, 2017 during outage 1R28 Local Leak Rate Testing (Appendix J), AO-2-86C, “13 Outboard Main Steam Isolation Valve,” had an unacceptable as-found leak rate. The measured leakage rate was 187.8 standard cubic feet per hour (scfh) which exceeds the Monticello Nuclear Generating (MNGP) Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.12 limit of 100 scfh. AO-2-86C was declared inoperable and the valve was subsequently disassembled to make repairs. The valve’s stem, discs, upper/lower wedges, disc retainer, and wedge pin were replaced and retested. The as-left leak rate after completion of the work was 2.64 scfh.

This component failure is reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by TS 3.6.1.3, "Primary Containment Isolation Valves," since AO-2-86C likely had been inoperable for greater than the TS 3.6.1.3, Required Action A.1, Completion Time of 8 hours to isolate a main steam line, and the Completion Time for TS 3.6.1.3, Required Action F, to be in Mode 3 in 12 hours and Mode 4 in 36 hours when the completion time of A.1 is not met. There were minimal safety consequences associated with the condition since the primary containment isolation function was maintained by the inboard valve.

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NARRATIVE

EVENT DESCRIPTION

On April 20, 2017, with the plant at 0% power in Mode 5 (Refueling), during refueling outage 1R28, Local Leak Rate Testing (Appendix J) of AO-2-86C, “13 Outboard Main Steam [SB] Isolation Valve [ISV],” had an unacceptable as-found leak rate. The measured leak rate was 187.8 standard cubic feet per hour (scfh) which exceeds the Monticello Nuclear Generating (MNGP) Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.12 limit of 100 scfh. AO-2-86C was declared inoperable and the valve was subsequently disassembled to make repairs. The valve’s stem, discs, upper/lower wedges, disc retainer, and wedge pin were replaced and retested. The as-left leak rate after completion of the work was 2.64 scfh.

This component failure is reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by TS 3.6.1.3, "Primary Containment Isolation Valves," since AO-2-86C likely had been inoperable for greater than the TS 3.6.1.3, Required Action A.1, Completion Time of 8 hours to isolate a main steam line, and the Completion Time for TS 3.6.1.3, Required Action F, to be in Mode 3 in 12 hours and Mode 4 in 36 hours when the completion time of A.1 is not met.

The basis for the reportable condition is the change in wear rate associated with AO-2-86C valve internals. In 2011, AO-2-86C (Anchor Darling model W9324183 18”-900 venturied double) was disassembled and showed unexpected accelerated wear and excessive damage. The stem, upper and lower wedges, disc retainers and discs were replaced. The last as-found leak rate for AO-2-86C was 64.1 scfh in the 2015 refueling outage (1R27). After the actuator was replaced in 1R27 the as-left leakage was 4.4 scfh. Based on these data points it is concluded that the leak rate increased during the cycle and the valve likely had exceeded the TS SR limits during the cycle preceding 1R28.

EVENT ANALYSIS

The event was determined to be reportable in accordance with 10 CFR 50.73 (a)(2)(i)(B), “Any operation or condition which was prohibited by the plant’s Technical Specifications.” Specifically, this component failure is reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by TS 3.6.1.3, "Primary Containment Isolation Valves," since AO-2-86C likely had been inoperable for greater than the TS 3.6.1.3, Required Action A.1, Completion Time of 8 hours to isolate a main steam line, and the Completion Time for TS 3.6.1.3, Required Action F, to be in Mode 3 in 12 hours and Mode 4 in 36 hours when the completion time of A.1 is not met.

This event is not classified as a safety system functional failure as the inboard valve was fully operational.

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SAFETY SIGNIFICANCE

There were minimal safety consequences associated with the condition. The inboard MSIV on Main Steam line “C” (AO-2-80C) was tested for both leak rate and closing time over the past cycle and each test was completed satisfactorily. Therefore, the primary containment isolation capability of the main steam lines remained operable which ensured the required isolation safety function was maintained.

CAUSE

The failure was attributed to oscillating of the disc and wear on the trunnion pin. The oscillation caused wear between the downstream disc trunnion and mating upper wedge hole. As the wear increased, the disc dropped, increasing the gap between the disc retainer and disc groove thereby allowing further rotation of the disc. Eventually, the corner at the end of the disc groove started to contact one of the ends of the retainer plate and wear into it. This resulted in interference between the downstream disc groove area and the bottom corner of the disc retainer. This interference prevented or restricted the ability of the upper part of the downstream disc to move axially towards its corresponding body seat thereby resulting in a gap or reduced seating force in portions of the seat. Based on this, the increased leakage of the valve is attributable to wear which led to reduced seating force or a gap (due to interference) in the valve disc as it contacts the valve body seat.

CORRECTIVE ACTION

The entirety of the internal disc pack was replaced. This includes the stem, discs, upper/lower wedges, disc retainer, and wedge pin. A modification was made to hard face the trunnion outer diameter, upper wedge hole inner diameter, and disc grooves with a Stellite 21 overlay to help reduce the amount of wear. Other improvements were made to the disc retainers as well.

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].