



Monticello Nuclear Generating Plant  
2807 West County Road 75  
Monticello, MN 55362-9637

Operated by Nuclear Management  
Company LLC

January 7, 2002

US Nuclear Regulatory Commission  
Attn: Document Control Desk  
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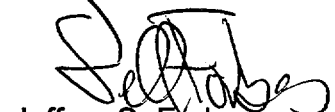
**MONTICELLO NUCLEAR GENERATING PLANT**  
Docket No. 50-263 License No. DPR-22

**LER 2001- 012**

**Refueling Testing Identifies Containment Isolation Valve Leakage Greater than  
Allowed by the Technical Specifications**

A Licensee Event Report for this occurrence is attached. This report contains no new NRC commitments.

Contact Thomas Parker, at (763) 295-1014 if you require further information.



Jeffrey S. Forbes  
Site Vice President  
Monticello Nuclear Generating Plant

**Enclosure**

c: Regional Administrator - III NRC  
NRR Project Manager, NRC  
Sr. Resident Inspector, NRC  
Minnesota Department of Commerce

**LICENSEE EVENT REPORT (LER)**(See reverse for required number of  
digits/characters for each block)

Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@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 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**

Monticello Nuclear Generating Plant

**2. DOCKET NUMBER**

05000263

**3. PAGE**

1 OF 7

**4. TITLE**

Refueling Testing Identifies Containment Isolation Valve Leakage Greater than Allowed by the Technical Specifications

**5. EVENT DATE**MO DAY YEAR  
11 08 2001**6. LER NUMBER**YEAR SEQUENTIAL  
NUMBER REV.  
NO  
2001 - 012 -**7. REPORT DATE**MO DAY YEAR  
01 07 2002**8. OTHER FACILITIES INVOLVED**FACILITY NAME DOCKET NUMBER  
05000FACILITY NAME DOCKET NUMBER  
05000**9. OPERATING  
MODE**

N

**10. POWER  
LEVEL**

0

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

20.2201(b)

20.2203(a)(3)(ii)

50.73(a)(2)(ii)(B)

50.73(a)(2)(ix)(A)

20.2201(d)

20.2203(a)(4)

50.73(a)(2)(iii)

50.73(a)(2)(x)

20.2203(a)(1)

50.36(c)(1)(i)(A)

50.73(a)(2)(iv)(A)

73.71(a)(4)

20.2203(a)(2)(i)

50.36(c)(1)(ii)(A)

50.73(a)(2)(v)(A)

73.71(a)(5)

20.2203(a)(2)(ii)

50.36(c)(2)

50.73(a)(2)(v)(B)

OTHER

20.2203(a)(2)(iii)

50.46(a)(3)(ii)

50.73(a)(2)(v)(C)

Specify in Abstract below or

20.2203(a)(2)(iv)

50.73(a)(2)(i)(A)

50.73(a)(2)(v)(D)

in NRC Form 366A

20.2203(a)(2)(v)

X

50.73(a)(2)(i)(B)

50.73(a)(2)(vii)

20.2203(a)(2)(vi)

50.73(a)(2)(i)(C)

50.73(a)(2)(viii)(A)

20.2203(a)(3)(i)

50.73(a)(2)(ii)(A)

50.73(a)(2)(viii)(B)

**12. LICENSEE CONTACT FOR THIS LER****NAME**

Tom Parker

**TELEPHONE NUMBER (Include Area Code)**

763-295-1014

**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	B	CE	ISV	A391	Y

**14. SUPPLEMENTAL REPORT EXPECTED**

YES (If yes, complete EXPECTED SUBMISSION DATE).

X

NO

**15. EXPECTED  
SUBMISSION  
DATE**

MONTH

DAY

YEAR

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

Local leak rate testing during a refueling outage indicated that the following limits established in the Technical Specifications for Containment Isolation Valve leakage were exceeded: 1) the main steam isolation valve leakage limit and 2) the combined maximum flow path leakage rate for all containment penetrations and valves subject to Type B and C tests. Five valves with excessive leakage were identified, inspected and repaired as necessary. The valves were retested satisfactorily. The total minimum flow path leakage was not exceeded, therefore, this event did not affect the health and safety of the public.

# LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
Monticello Nuclear Generating Plant	05000263	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	2 of 7
		2001	- 012	- 00	

NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

## Description

During the November 2001 refueling outage, local leak rate testing indicated that limits established in the Technical Specification for Containment Isolation Valve Leakage were exceeded.

Technical Specification 3.7.A.2.b.3 limits main steam isolation valve leakage to "Less than or equal to 46 scf per hour combined maximum flow path leakage for all main steam isolation valves when tested at 25 psig." The leakage associated with main steam isolation valve AO-2-86 C and D<sup>1</sup> caused this limit to be exceeded. See table below for more details.

Technical Specification 3.7.A.2.b.2 states: "A combined maximum flow path leakage rate of less than or equal to 0.6La for all penetrations and valves, subject to Type B and C tests when pressurized to Pa, 42 psig." "La" is the maximum allowable leakage rate in percent by weight of the containment air volume per day at Pa. "Pa" is the calculated peak containment pressure related to the design bases accident. The leakage associated with three containment isolation valves, MO-2397, MO-2398<sup>2</sup> (Reactor Water Cleanup System) and CV-2790<sup>3</sup> (Recirculation System) caused this limit to be exceeded. See table below for more details.

The combined maximum flow path leakage is the total containment maximum pathway leakage that would exist if the valve with the lowest leakage in each penetration failed to close.

Valves with Technical Specification Excessive Leakage			
Test Date	Valve Name	Valve Description	"As Found" Leakage Rate <sup>4</sup>
11/8/01	AO-2-86C Outboard Main Steam Isolation Valve on the "C" Main Steam Line	18" Venturi, Spring Actuated double disc gate Valve (Installed in 1994)	33 scfh @ 25 psig
11/8/01	AO-2-86D Outboard Main Steam Isolation Valve on the "D" Main Steam Line	18" Venturi, Spring Actuated double disc gate Valve (Installed in 1994)	27 scfh @ 25 psig
11/11/01	MO-2397 Reactor Water Cleanup Inlet Inboard Isolation Valve	Motor Operated 4" 600# Double Disc Gate Valve (Installed in 2000).	95 scfh @ 42 psig
11/11/01	MO-2398 Reactor Water Cleanup Inlet Outboard Isolation Valve	Motor Operated 4" 600# Double Disc Gate Valve (Installed in 2000).	839 scfh @ 42 psig

<sup>1</sup> EIIS System code = SB, EIIS Component Code = ISV

<sup>2</sup> EIIS System code = CE

<sup>3</sup> EIIS System code = AD

<sup>4</sup> Standard Cubic Feet Per Hour = scfh or scf per hour

# LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
Monticello Nuclear Generating Plant	05000263	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	3 of 7
		2001	012	00	

NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

11/10/01	CV-2790 RECIRC Sample Line Inboard Isolation Valve	Air Operated 3/4" - Double Disc Gate Valve (Installed in 1991).	138 scfh @ 42 psig
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## Event Analysis

### Analysis of Reportability

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(i)(B): "Any operation or condition prohibited by the plant's Technical Specifications . . ."

No safety system functional failure occurred, since the combined minimum flow path leakage rate was less than 0.6La.

### Safety Significance

The measured total containment minimum pathway leakage was less than 0.6La. If a condition requiring containment isolation had occurred, any releases would have been less than those assumed in the plant safety analysis. The health and safety of the public was not affected by this event.

## Cause

AO-2-86C, Outboard Main Steam Isolation Valve on the "C" Main Steam Line  
AO-2-86D, Outboard Main Steam Isolation Valve on the "D" Main Steam Line

The outboard Main Steam Isolation Valves are double disc gate valves with independent upstream and downstream discs. The upstream disc is equipped with a small hole to prevent pressure locking. The downstream disc therefore accomplishes the valve isolation function.

All four outboard Main Steam Isolation Valves were disassembled. Packing leakage, seating surface scratches (AO-2-86C only) and radial cracks in the disc seating surfaces were the cause of the excessive leakage in AO-2-86C and D.

Minor radial cracks were also found in AO-2-86B, Outboard Main Steam Isolation Valve on the "B" Main Steam Line. These radial cracks did not compromise the leak tightness of the valve.

**LICENSEE EVENT REPORT (LER)**

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Monticello Nuclear Generating Plant	05000263	2001	012	00	4 of 7

**NARRATIVE** (If more space is required, use additional copies of NRC Form 366A) (17)

Valve body wear was also discovered on all four valves, although it did not affect the isolation function. The upper portion of the downstream body seat and the valve body guides were worn by the disc rotating during service. The cause of the radial cracks is due to a combination of residual stresses from manufacturing and mechanical stress associated with the disc rotating during service and interacting with the stationary seats.

Reference – CR 20017070

MO-2397 Reactor Water Cleanup System Inlet Inboard Isolation Valve  
MO-2398 Reactor Water Cleanup System Inlet Outboard Isolation Valve

The seating surface of MO-2397 was found to be gouged. The seating surface of MO-2398 was found to be severely scored. No source of the foreign material could be identified. These valves were replaced during the last refueling outage.

Reference – CR 20017090 and 20017091

CV-2790 RECIRC Sample Line Inboard Isolation Valve

The valve actuator bench setting was found to be lower than the manufacturer's recommendation. The leakage was caused by a small amount of corrosion products on the seats and the air operator had a spring setting below specifications.

Reference – CR 20017092

**Actions**

AO-2-86C, Outboard Main Steam Isolation Valve on the "C" Main Steam Line  
AO-2-86D, Outboard Main Steam Isolation Valve on the "D" Main Steam Line

The downstream discs of all four outboard Main Steam Isolation Valves were replaced, the body seats were lapped and the valves were repacked. A galled stem on AO-2-86D was replaced.

All four valves were modified to prevent disc rotation.

The outboard Main Steam Isolation Valves were successfully retested.

**LICENSEE EVENT REPORT (LER)**

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
Monticello Nuclear Generating Plant	05000263	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	5 of 7
		2001	- 012	- 00	

NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

MO-2397 Reactor Water Cleanup System Inlet Inboard Isolation Valve  
MO-2398 Reactor Water Cleanup System Inlet Outboard Isolation Valve

The valve seats were repaired and were successfully retested. No source of the foreign material could be identified nor was any foreign material recovered..

CV-2790 RECIRC Sample Line Inboard Isolation Valve

The valve seats were cleaned and the air operator adjusted. The valve was successfully retested.

Failed Component Identification

AO-2-86C and D - Outboard Main Steam Isolation Valve on the "C" and "D" Main Steam Line  
Valve Manufacturer: Flowserve (formerly Anchor/Darling Valve Company)

Size of Valve: 18" venturi

Pressure Class: 900 psig

Model Number: W9324183 (NX-56482)

MO-2397 and MO-2398

Valve Manufacturer: Flowserve (formerly Anchor-Darling)

Size of Valve: 4" double-disc gate valve

Pressure Class: 900 psig

Model Number: W9925312 (NX-9235-45), Serial Number E-758A-2-1 (MO-2397), E-758A-2-2 (MO-2398)

CV-2790

Valve Manufacturer: Flowserve (formerly Anchor/Darling Valve Company)

Size of Valve: 3/4" - Double Disc Gate Valve

Pressure Class: 1878 psig

Model Number: Valve Serial Number EB069-1-1 (Tech Manual NX-17293)

**LICENSEE EVENT REPORT (LER)**

FACILITY NAME (1)		DOCKET (2)	LER NUMBER (6)			PAGE (3)
Monticello Nuclear Generating Plant		05000263	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	6 of 7
			2001	- 012	- 00	

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Similar Events In the Last Ten Years

There have been past LERs on failed Main Steam Isolation Valves of the inboard Main Steam Isolation Valve design, but there have been no failures of Main Steam Isolation Valves of the outboard valve design. Testing verified that the inboard Main Steam Isolation Valves performed the function properly. The inboard Main Steam Isolation Valves were original plant equipment and have a different design than the outboard valve design.

No other similar failures have occurred.

(1-2001)

**LICENSEE EVENT REPORT (LER)****FAILURE CONTINUATION**

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
Monticello Nuclear Generating Plant	05000263	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	7 of 7
		2001	- 012	- 00	

**COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)**

CAUSE	SYSTEM	COMPONENT	MANU- FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU- FACTURER	REPORTABLE TO EPIX
X	AD	ISV	A391	Y					