NRC FORM 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
MONTICELLO NUCLEAR GENERATING PLANT	0 5 0 0 0 2 6 3	1 OF 0 5

TITLE (4): Main Steam Isolation Valves Local Leak Rate Exceeded - Root Cause To Be Determined

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 1	3 1	9 3	9 3	— 0 0 3 —	0 0	0 3	0 2	9 3		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
Dave Pennington, System Engineer	6 1 2	2 9 5 - 1 3 5 4

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	S B	I S V	A 5 8 5	Yes					

SUPPLEMENTAL REPORT EXPECTED (14)

[X] YES (If yes, complete EXPECTED SUBMISSION DATE) [ ] NO

EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	0 3	2 3	9 3

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

Abstract

Main Steam Line Isolation Valve leak rate tests conducted during a refueling outage determined that the Technical Specification leakage limit for individual valves was exceeded. The root cause of the event is still being evaluated. The valves were inspected by a plant engineer and a valve manufacturer's engineering representative and it appears that the cause of failure may differ for each valve. Further evaluation is necessary to determine the appropriate long term corrective actions. The Main Steam Isolation Valves will be repaired and re-tested prior to startup. A supplemental report will be submitted to discuss the root cause or causes of this event and long term corrective actions.

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ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

Description

On January 31, 1993, with the plant in a refueling outage, it was determined that Technical Specification 3.7.A.2.b.3 leak rate criteria had been exceeded. The Technical Specification states, "When Primary Containment Integrity is required, leakage rates shall be limited to...Less than or equal to 11.5 scf per hour for any one main steam isolation valve when tested at 25 psi". When tested, 7 of the 8 Main Steam Isolation Valve (EIIS Component: ISV) demonstrated leakage in excess of the limit specified.

Table 1 below is a summary of "as-found" Main Steam Isolation Valve leakage (without air pressure on the valve actuator) when tested at 25 psi:

Table 1
(Test Results Without Air Pressure on Actuator)

MAIN STEAM LINE	INBOARD ISOLATION VALVE LEAKAGE	OUTBOARD ISOLATION VALVE LEAKAGE
A	50.1 scfh	6388 scfh
B	32.7 scfh	202.3 scfh
C	3.3 scfh	12.5 scfh
D	24 scfh	146 scfh

This was the first time Main Steam Isolation Valves have been leak rate tested with the air vented from the actuators, and this change resulted in a more challenging test than had been performed previously. To determine the effect of this change in test methodology, three of the inboard Main Steam Isolation Valves were re-tested with air pressure on the actuator. Table 2 below is a summary of "as-found" leakage, with air on the valve actuator, when tested at 25 psi:

Table 2
(Test Results With Air Pressure on Actuator)

MAIN STEAM LINE	INBOARD ISOLATION VALVE LEAKAGE
A	8.9 scfh
B	28 scfh
D	13.5 scfh

A plant engineer and an engineering representative from the valve manufacturer inspected the valves. The results of this inspection are still under review.

This event is a condition prohibited by Technical Specification and is reportable per 10 CFR 50.73.

Cause

The cause of this event is still under evaluation, but it is suspected that the leakage may be due to a differing combination of factors for each valve. Further evaluation and confirmation of the root cause (or causes) of this problem is necessary before long term corrective actions can be developed.

Analysis

The minimum pathway leakage without air pressure on the Main Steam Isolation Valve actuators was 168 scfh corrected for 42 psig containment pressure. With air on the actuators, the minimum path way leakage was 81.8 scfh corrected for 42 psig containment pressure. During design basis loss of coolant accident conditions the air supplied to the Main Steam Isolation Valve accumulators is assumed to be lost. However, the accumulators would still be available to close the valves and to maintain them closed for a period of at least 10 minutes. Also, during the loss of coolant accident event containment pressure

peaks at 42 psig for two seconds and then begins to rapidly decay. Prior to and after that peak, pressure at the Main Steam Isolation Valves would be below 42 psig and main steam isolation valve leakage would be correspondingly lower. Therefore, when averaged over the course of the event, the leak rate through the valves would be less than the minimum pathway leak rate (81.8 scfh) that was calculated above based on local leak rate test results.

The maximum allowable containment leakage rate (La) is 1.2 weight % per day at 42 psig (Pa), which equates to 458 scfh at 42 psig. Plant Technical Specification 3.7.A.2.b.2 requires that the combined leakage rate for all penetrations and valves, except main steam isolation valves, subject to Type B and Type C local leak rate testing shall be limited to 0.6La, or 275 scfh, when pressurized to Pa. A preliminary review of "as-found" local leak rate test data indicates that this limit, which is based on maximum pathway leakage, was not exceeded.

When the "as-found" Main Steam Isolation Valve leakage (adjusted to 42 psi) is added to the preliminary total "as-found" minimum pathway leakage for all valves and penetrations that undergo type B and type C testing, the total is 205 scfh. An Engineering review of the minimum pathway leakage and current assumptions for the Control Room Dose Analysis concluded that the additional Main Steam Isolation Valve leakage would not have caused the allowable dose limit of GDC-19 to be exceeded. Similarly, the health and safety of the public was not affected because even with the as-found leakage of the Main Steam Isolation Valves factored in, this leak rate is well below the 1.2 weight % per day leak rate used in the plant safety analysis.

<u>Corrective Actions</u>

The following actions have been completed:

> An engineering representative from the valve manufacturer was called in to inspect the valves. The results of this inspection are still under review.

The following actions will be completed:

1. Further evaluation of this event will be performed to confirm the root cause or causes. The results of this evaluation will be used to develop appropriate long term corrective actions to prevent recurrence.

2. The Main Steam Isolation Valves will be repaired and re-tested satisfactorily prior to startup.

3. A supplement to this report will be submitted prior to start-up to discuss the results of our root cause evaluation and to describe our long term corrective actions.

<u>Failed Component Identification</u>

Main Steam Isolation Valves (7)
Manufacturer: A585, Atwood and Morrill Company Incorporated
Type: 18" Y Globe Valve

<u>Previous Similar Events</u>

Licensee Event Report 91-005 reported the failure of three Main Steam Isolation Valves to pass the local leak rate test. The cause of the failure was an oxide film on the valve seats and normal wear. At that time it was concluded that the oxide film was a normal build up and that further corrective actions were not needed. The valves were repaired and tested satisfactorily.