NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 1	0 5 0 0 0 2 4 5	1 OF 0 6

TITLE (4)
Local Leak Rate Test Failures

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 6	0 6	8 7	8 7	0 1 5	0 1	0 9	2 9	8 7		0 5 0 0 0
										0 5 0 0 0

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

OPERATING MODE (9)					
POWER LEVEL (10)	0 0 0	[ ] 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)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
		[ ] 20.405(a)(1)(iv)	[X] 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)
Lou Georgian, Engineer - Ext. 4198	2 0 3 4 4 7 - 1 7 9 1

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
X	J M	I S V	C 6 6 5	Y	X	B M	I S V	C 2 5 5	Y
X	C E	I S V	M 1 2 0	Y	X	A A	I S V	V 0 8 5	Y

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[ ] NO				

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

On June 6, 1987 at 1000 hours, while performing Local Leak Rate Testing (LLRT) during the 1987 refuel outage, it was identified that the "B" main steam isolation valves could not meet the required leak rate as specified in Technical Specification 4.7.f.2.c. Testing of all primary containment isolation valves, cable penetrations and manways as required by 10CFR50 Appendix J revealed additional isolation valves that did not meet the local leak rate test requirement.

All valves that failed to meet the local leakage rate test requirements were satisfactorily retested subsequent to repairs.

There were no consequences.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

I. Description of Event

During the 1987 refuel outage, all testable primary containment isolation valves, cable penetrations and manways were local leak rate tested. Of the total number of valves, penetrations and manways tested, seven (7) primary containment isolation valves required maintenance in order to meet the leakage requirement of Technical Specification 4.7.A.3.e. Technical Specifications require that no single penetration or isolation valve(s), except the main steam isolation valve(s) (MSIV), exceed a leakage rate of 18.8 SCFH at 43 psig. No single MSIV may exceed a leak rate of 11.5 SCFH at 25 psig. (This correlates to a leakage rate of approximately 15 SCFH at 43 psig.)

Technical Specifications further require that a total combined leakage rate for all testable isolation valves and penetrations be less than 300.3 SCFH at 43 psig.

The total "as found" combined leakage rate for all testable isolation valves and penetrations exceeded the allowable limit of 300.3 SCFH ($0.6L_a$). Following repairs to valves that exceeded the allowable leakage rate, the total "as left" leakage rate was 153.2 SCFH.

The following is a list of valves and their associated containment penetration that failed to pass the local leak rate test and their "as found" and "as left" leakage rates.

	Penetration	As Found (At 43 psig)	As Left (At 43 psig)
A. Main Steam Isolation Valves			
1-MS-2A, 1-MS-2B (tested concurrently)	X-7B	16.18 SCFH	10.23 SCFH
B. Clean-up Valves			
1-CU-2, 1-CU-2A, 1-CU-3, 1-CU-5 (tested concurrently)	X-14	>265.0 SCFH*	8.67 SCFH
C. Cove Spray			
1-CS-5A	X-16A	19.92 SCFH	2.16 SCFH
D. Head Spray			
1-HS-4, 1-HS-5	X-17	>265.0 SCFH*	4.05 SCFH
E. Post Accident Sampling			
1-PAS-17, 1-PAS-20	X-12	60.1 SCFH	0.21 SCFH
F. Atmospheric Control Valves			
1-AC-7, 1-AC-8, 1-AC-9 1-AC-10, 1-AC-11, 1-AC-12 (tested concurrently)	X-25/ X-202D	>265.0 SCFH*	5.45 SCFH

*Maximum reading within the capability of the local leakage rate test equipment. Actual leakage rates for these valves (not quantified) are in excess of those recorded.

II. Cause of Event

A. Main Steam Isolation Valves
1-MS-2B Damaged pilot valve disc.

B. Clean-up System
1-CU-2A Valve seat required lapping.

C. Core Spray
1-CS-5A Minor wear on wedge.

D. Head Spray
1-HS-5 Broken spring.

E. Post Accident Sampling
1-PAS-17 Plunger tube scored.

F. Atmospheric Control Valves
1-AC-8 Slightly worn seat ring.
1-AC-10 Worn seat ring.

NOTE: All valves were satisfactorily retested subsequent to repairs.

III. Analysis of Event

° In the case of the main steam isolation valves, only the outboard isolation valve (1-MS-2B) exhibited leakage that exceeded the allowable limits. This is evident from the "as found" and "as left" leakages for 1-MS-2A and 1-MS-2B. As a result, the first isolation valve (1-MS-2A) would have performed the required isolation function for this penetration.

° The clean-up system valves 1-CU-2, 1-CU-2A, 1-CU-3, and 1-CU-5, which are tested concurrently failed to meet the acceptance criteria. Valve 1-CU-2A (½" globe valve) was the only valve that required re-work which consisted of lapping the seat and re-setting the stroke. Since 1-CU-2A is a bypass around the inboard isolation valve 1-CU-2, the outboard isolation valves 1-CU-3 and 1-CU-5 would have performed the required isolation function for this penetration.

° The core spray system is a closed loop system with valve 1-CS-5A providing the containment isolation function. Any leakage through valve 1-CS-5A would be contained by the pressure boundary of the system. The pressure boundary of the system would as a result perform the required isolation function.

The integrity of the pressure boundary is demonstrated on a monthly basis during surveillance testing of the core spray system.

° In the case of the head spray isolation valves, only the outboard isolation valve 1-HS-5 exhibited leakage that exceeded the allowable limits. The recorded leakage rate for the inboard isolation valve 1-HS-4, was zero (0) SCFH. As a result, the inboard isolation valve 1-HS-4 would have performed the required isolation function for penetration X-17.

II. Cause of Event (Continued)

° Post Accident Sampling System valves 1-PAS-17 and 1-PAS-20 (3/4" solenoid valves) provide the sample point for the Shutdown Cooling (SDC) loop and are located between valves 1-SD-1, 1-SD-2 and 1-SD-2A (SDC isolation valves). Valves 1-PAS-17 and 1-PAS-20 are in series with respect to each other with 1-PAS-20 being the first isolation valve off the SDC loop. Although 1-PAS-17 exhibited leakage that exceeded the allowable limits, 1-PAS-20 leakage was recorded at zero (0) SCFH and as a result would have performed the required isolation function.

° Due to the configuration of the Atmospheric Control System, valves 1-AC-7, 1-AC-8, 1-AC-9, 1-AC-10, 1-AC-11 and 1-AC-12 are tested concurrently and provide the isolation function for penetrations X-25 and X-202D. The excessive leakage rate for these penetrations was attributed primarily to valve 1-AC-10 (12" butterfly valve). Both valves 1-AC-8 and 1-AC-10 are the outboard containment isolation valves. To determine the integrity of penetrations X-25 and X-202D, valve 1-AC-8 was removed and a blank flange was installed in its place. Retesting of these penetrations showed no appreciable change in the excessive leakage rate. Valve 1-AC-10 was then removed and a blank flange was installed in its place. With both 1-AC-8 and 1-AC-10 removed and blank flanges installed, the inboard isolation valves 1-AC-7, 1-AC-9, 1-AC-11 and 1-AC-12 were tested and the recorded leakage rate for these valves was 11.44 SCFH which is well within the allowable limit of 18.8 SCFH. As a result it was determined that the inboard isolation valves 1-AC-7, 1-AC-9, 1-AC-11 and 1-AC-12 would have performed the required isolation function for penetrations X-25 and X-202D.

IV. Corrective Action

Main Steam Isolation Valves

1-MS-2B Replaced pilot valve disc and ground pilot valve seat.

Clean-up System

1-CU-2A Lapped seat and plug and adjusted stroke.

Core Spray System Valve

1-CS-5A Ground seat and wedge.

Head Spray System Valve

1-HS-5 Replaced broken spring. Reground seat and machined plug.

Post Accident Sampling System Valve

1-PAS-17 Replaced plunger tube.

Atmospheric Control Valves

1-AC-8 Replaced seat ring and "O" rings.
1-AC-10 Replaced seat ring and "O" rings.

V. Additional Information

A. Failed Component Identification

The description of the valves that failed the local leak rate test is as follows:

Valve No.	Size/Type	Manufacturer
1-MS-2B	20"/"Y" globe	Crane Company
1-CU-2A	1/2"/globe	Masoneilan
1-CS-5A	10"/gate	Chapman
1-HS-5	2"/lift check	Velan
1-PAS-17	3/4"/solenoid	Target Rock
1-AC-8	18"/butterfly	Allis Chalmers
1-AC-10	12"/butterfly	Allis Chalmers

B. Previous Similar Events

LER 85-023
LER 84-005
LER 82-23/3L
LER 82-23/3X-1
LER 80-14/1D
LER 80/14/1T

★U.S.GPO:1986-0-624-538/455

NRC Form 366B
(9-83)

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EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Millstone Nuclear Power Station Unit 1	0 5 0 0 0 2 4 5	8 7	— 0 1 5 —	0 1	0 6 OF 0 6

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	I P	I S V	T 0 2 0	Y					
X	V P	I S M	A 1 8 0	Y					
X	V B	I S M	A 1 8 0	Y					

NRC FORM 366B
(9-83)

NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

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September 29, 1987
MP-10917
Re: 10CFR50.73(a)(2)(ii)

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Reference: Facility Operating License No. DPR-21
Docket No. 50-245
Licensee Event Report 87-015-01

Gentlemen:

This letter forwards the updated Licensee Event Report 87-015-01. This update provides additional information with regard to local leak testing performed during the 1987 refueling outage.

Yours truly,

NORTHEAST NUCLEAR ENERGY COMPANY

Stephen E. Scace
Station Superintendent
Millstone Nuclear Power Station

SES/EJG:mo

Attachment: LER 87-015-01

cc: W. T. Russell, Region I
T. Rebelowski, Senior Resident Inspector

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