

NORTHEAST UTILITIES



The Connecticut Light And Power Company
Western Massachusetts Electric Company
Holyoke Water Power Company
Northeast Utilities Service Company
Northeast Nuclear Energy Company

General Offices - Selden Street, Berlin Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06414-0270
(203) 665-5000

October 30, 1992
MP-92-1160

Re: 10CFR50.73(a)(2)(i)

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

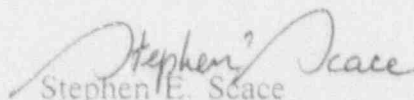
Reference: Facility Operating License No. DPR-65
Docket No. 50-336
Licensee Event Report 90-019-01

Gentlemen:

This letter forwards Licensee Event Report 90-019-00 required to be submitted within thirty (30) days pursuant to 10CFR50.73(a)(2)(i)(B).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY


Stephen E. Scace

Vice President - Millstone Station

SES/SLS:dlr

Attachment: LER 90-019-01

cc: T. T. Martin, Region I Administrator
P. D. Swetland, Senior Resident Inspector, Millstone Unit Nos. 1, 2 and 3
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2

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LICENSEE EVENT REPORT (LER)

Estimated burden per response to comply with this information collection request: 50-6 hrs. Forward comments regarding burden estimate to the Records and Reports Management Branch (p-530), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503.

FACILITY NAME (1) Millstone Nuclear Power Station Unit 2

DOCKET NUMBER (2)

PAGE (3)

0 5 0 0 0 3 3 6 1 OF 0 3

TITLE (4)

Combined Leakage Rate Exceeded

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										
1	0	0	5	9	0	9	0	0	1	9	0	1	1	0	3	0	9	2	

OPERATING MODE (9)	THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more of the following) (11)																								
6	<table border="1"><tr><td>20.402(b)</td><td>20.402(c)</td><td>50.73(a)(2)(iv)</td><td>73.71(b)</td></tr><tr><td>20.405(a)(1)(i)</td><td>50.36(a)(1)</td><td>50.73(a)(2)(v)</td><td>73.71(c)</td></tr><tr><td>20.405(a)(1)(ii)</td><td>50.36(a)(2)</td><td>50.73(a)(2)(vii)</td><td>OTHER (Specify in Abstract below and in Text, NRC Form 366A)</td></tr><tr><td>20.405(a)(1)(iii)</td><td>X 50.73(a)(2)(i)</td><td>50.73(a)(2)(viii)(A)</td><td></td></tr><tr><td>20.405(a)(1)(iv)</td><td>50.73(a)(2)(ii)</td><td>50.73(a)(2)(viii)(B)</td><td></td></tr><tr><td>20.405(a)(1)(v)</td><td>50.73(a)(2)(iii)</td><td>50.73(a)(2)(ix)</td><td></td></tr></table>	20.402(b)	20.402(c)	50.73(a)(2)(iv)	73.71(b)	20.405(a)(1)(i)	50.36(a)(1)	50.73(a)(2)(v)	73.71(c)	20.405(a)(1)(ii)	50.36(a)(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)(ix)	
20.402(b)	20.402(c)	50.73(a)(2)(iv)	73.71(b)																						
20.405(a)(1)(i)	50.36(a)(1)	50.73(a)(2)(v)	73.71(c)																						
20.405(a)(1)(ii)	50.36(a)(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)(ix)																							

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Peter D. Lessard, Engineer, Ext. 5066	AREA CODE 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 NRC	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC
B	C	B	I S V F	1 3 5					

SUPPLEMENTAL REPORT EXPECTED (14)

EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE) NO	1	2	3 1 9 3

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

The plant was in mode 6, power level was zero percent, temperature was 92 degrees Fahrenheit and RCS pressure was 0 psig. Type B and C leakage rate testing was being performed in accordance with Technical Specification Surveillance requirement 4.6.1.2.d. On 10/05/90 a leakage rate test of the Steam Generator Blowdown piping isolation valve caused the combined leakage rate limit of specification 3.6.1.2.b to be exceeded. The leakage rate was such that the test volume could not be pressurized to accident pressure for testing.

The valve was disassembled. Inspection revealed wear in the seat area and plug. The seat and plug were replaced and the test was completed satisfactorily.

The root cause of the high leakage is due to wear in the plug and seat area of the valve due to steam cutting.

There were no safety implications resulting from this event as the valves are on the secondary side of the steam generators.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

Estimated burden per response to comply with 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.

FACILITY NAME (1) Millstone Nuclear Power Station Unit 2	DOCKET NUMBER (2) 0 5 0 0 0 3 3 6 9 0	LER NUMBER (6)			PAGE (3) 0 2 OF 0 3
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
		9 0	0 1 9	0 1	

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

I. Description of Event

The plant was in Mode 6, shutdown for refueling operations. Reactor Coolant System (RCS) temperature was 52 degrees Fahrenheit and pressure was 0 PSIG. Type B and C leakage rate testing of containment penetrations was in progress. On 10/05/90 the #2 Steam Generator Blowdown Penetration Isolation valves were being leakage rate tested. (See the diagram attached for the valve line-up). An attempt was made to pressurize the penetration volume to accident pressure (Pa-56 PSIG) thru the flange at valve 2-MS-17B. Test pressure could not be attained and the source of the leakage was investigated. Inspection of the vent paths thru 2-MS-348 and 2-MS-415 indicated leakage thru 2-MS-220B. When the vent path valves 2-MS-412 and 2-MS-415 were closed the test pressure was almost immediately achieved and the leakage rate was essentially zero. This indicates the source of the leakage to be 2-MS-220B.

II. Cause of Event

The root cause of the event is due to steam cutting of the valve seating surfaces.

III. Analysis of Event

This valve is located on the steam generator blowdown piping. The valve is in a piping system on the secondary side of the plant and has the steam generator tubing as the first containment barrier. Since there was no tube rupture event there are no safety implications as a result of this event.

IV. Corrective Action

The valve was disassembled and inspected. The inspection revealed a worn seat and plug. These items along with the valve stem and various gaskets were replaced. Since this valve is located on the secondary side of the plant, the necessity for Appendix J Leakage testing requirements was reviewed. Based on this review, it has been determined that the Steam Generator Blowdown valves may not be required to be included in the Appendix J, Type C Leakage Testing Program. Current plans are to submit a request to remove these valves from the Appendix J, Type C test requirements.

V. Additional Information

2 inch 600 lb. socket weld Masonellan Globe Valve. Trim material is 316ss, seat ring is 416ss.

Similar Events: 89-003, 88-006, 86-012, 85-003, 84-005, 82-006, 80-032, 79-034.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

Estimated burden per response to comply with this information collection is just 50.0 hrs. Forward comments regarding burden estimate to the Records and Reports Management Branch (2-530), U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503.

FACILITY NAME (1)

DOCKET NUMBER (2)

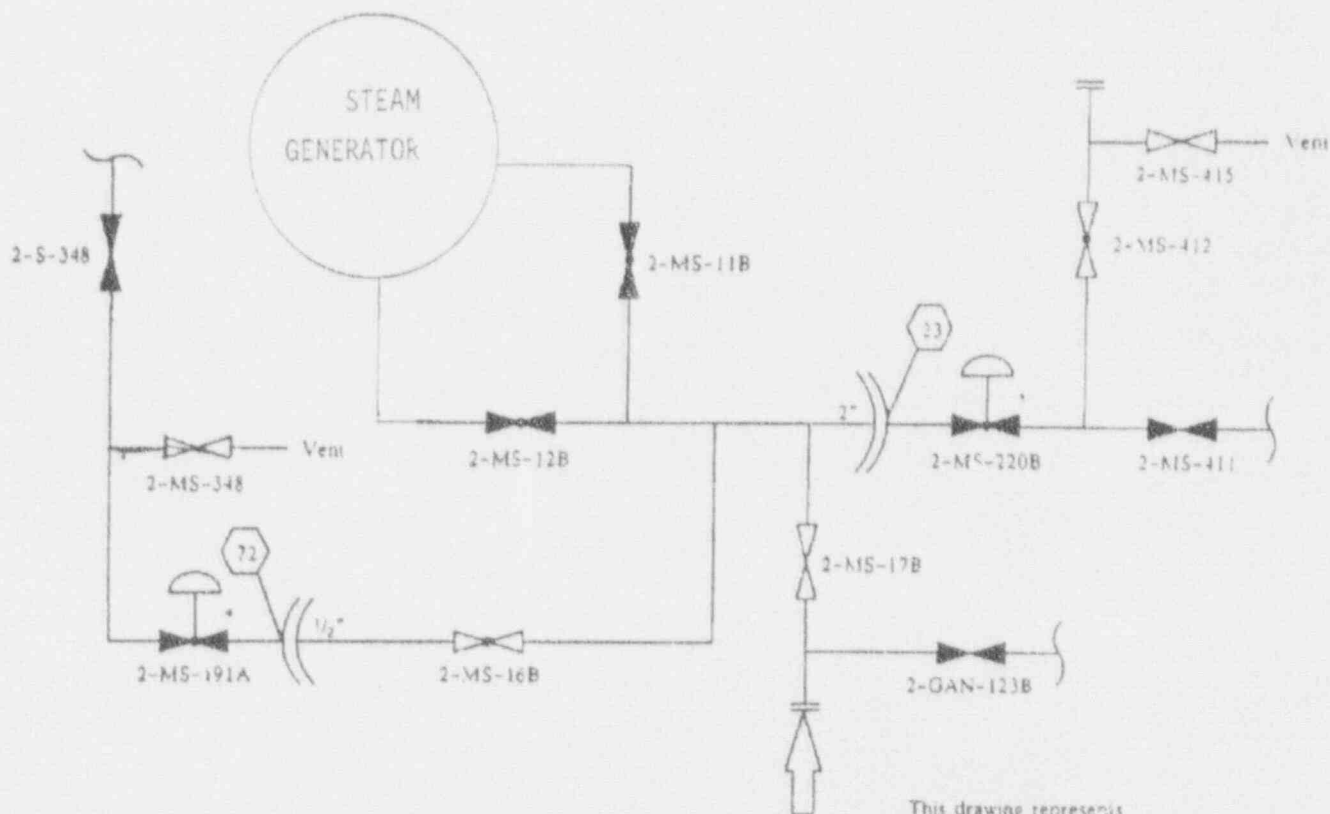
LER NUMBER (6)



PAGE (3)

Millstone Nuclear Power Station
Unit 2

0 5 0 0 0 3 3 6 9 0 0 1 9 0 1 0 3 OF 0 3

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



- 4 Verify NO pressure in S/G 2 before test.
- 3 Penetration must be drained prior to testing.
- 2  Signifies OPEN valve,  Signifies CLOSED valve.
- 1 Valves shown (*) must be stroked full open, then full closed in the normal operating manner (eg. MOV with control switch) prior to testing.

NOTES: Record AS FOUND (AF) and AS LEFT (AL) valve positions on this sheet

This drawing represents
actual test conditions

Signed _____ Date _____

LOCAL LEAK RATE TEST
VALVE LINEUPPenetration 23/72
2-MS-220B & 191B

P&ID 26002

Figure 9.6