

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 06114-0270
(203)665-5000

May 1, 1991

MP-91-365

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

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

Reference: Facility Operating License No. NPF-49
Docket No. 50-423
Licensee Event Report 90-030-01

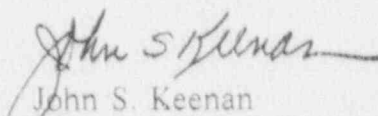
Gentlemen:

This letter forwards Licensee Event Report 90-030-01 which is being submitted as an update report by May 1, 1991, in accordance with LER 90-030-00. Licensee Event Report 90-030-00 was submitted pursuant to 10CFR50.73(a)(2)(iv), any event or condition that resulted in manual actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: Stephen E. Scace
Director, Millstone Station

BY: 
John S. Keenan
Millstone Unit 2 Director

SES/TGM:ljs

Attachment: LER 90-030-01

cc: T. T. Martin, Region I Administrator
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2 and 3
D. H. Jaffe, NRC Project Manager, Millstone Unit Nos. 1 and 3

9105080322 910510
PDR ADOCK 05000423
S PDR

(ent No 266578649)
IF22
11

LICENSEE EVENT REPORT (LER)

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 (B-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)										PAGE (3)									
Millstone Nuclear Power Station Unit 3										0 5 0 0 0 4 2 3										1 OF 0 6									
TITLE (4)																													
Manual Reactor Trip Due to Moisture Separator Reheater Piping Line Breaks																													
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																	
												0 5 0 0 0 0																	
1	2	3	1	9	0	9	0	0	0	3	0	0	1	0	5	0	1	9	1	0 5 0 0 0 0									
OPERATING MODE (9)					THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more of the following) (11)																								
1					20.402(b)					20.402(c)					<input checked="" type="checkbox"/> 50.73(a)(2)(iv)					73.71(b)									
POWER LEVEL (10)					20.405(a)(1)(i)					50.36(c)(1)					50.73(a)(2)(v)					73.71(c)									
8 6					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)					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														
Terry McNatt, Engineer, Ext. 5592															AREA CODE														
															2 0 3 4 4 7 - 1 7 9														
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																				
SUPPLEMENTAL REPORT EXPECTED (14)										EXPECTED SUBMISSION DATE (15)					MONTH DAY YEAR														
<input checked="" type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE)										NO					0 5 0 7 9														

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (15)

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

Estimated burden per response to comply with this information collection request: 60.0 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 3	DOCKET NUMBER (2) 0 5 0 0 0 4 2 3 9 0	LER NUMBER (6)			PAGE (3) 0 2 OF 0 6
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
		0 3 0	0 1		

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

I. Description of Event

On December 31, 1990, at 1636 hours with the plant in Mode 1 at 86% power, 580 degrees Fahrenheit and 2250 psia, a manual reactor trip was initiated due to two six inch Moisture Separator Reheater drain line (DSM) piping breaks in the Turbine Building. Following the trip a Main Steam Line Isolation was initiated to minimize the release of steam into the Turbine Building. Breaks occurred in lines downstream of the respective level control valves 3DSM-LV20A1 and B1, due to severe wall thinning caused by erosion/corrosion (refer to Figure 2). The piping failure resulted in the release of approximately 127,000 gallons of steam/water from the condensate piping and hotwell and 65,000 gallons of water from the condensate surge tank. The thermal energy of the fluid released from the ruptured piping activated the fire protection sprinkler system releasing an additional 25,000 gallons of water into the Turbine Building. In addition to mechanical and electrical damage in the Turbine Building, a power loss caused the isolation of instrument air to the containment, resulting in the loss of normal pressurizer spray flow and the isolation of normal letdown flow. Reactor Coolant Pump Seal Injection was reduced to the minimum allowable. Pressurizer level and pressure increased. One Power Operated Relief Valve (PORV) lifted three times to control reactor coolant system pressure.

At the time of the trip, operators verified that the reactor trip and bypass breakers were open, that all control rods were fully inserted, and that neutron flux was decreasing. An Auxiliary Feedwater actuation occurred as a result of steam generator low-low level signals. This is a normal plant response following a trip from 86% power. A Main Steam Line Isolation was manually initiated due to the line break in the Turbine Building. No additional engineered safety features were required or initiated. There was no operational, maintenance, or construction activities in progress at the time which affected the event. The plant stabilized at approximately 1841 hours based on stable reactor coolant system temperatures, the restoration of normal letdown flow, and restoration of normal pressurizer pressure control.

II. Cause of Event

The cause of the failure of the DSM lines was severe wall thinning downstream of the control valves. The wall thickness of the carbon steel, six inch, schedule 40 pipe had decreased from a nominal thickness of 0.280 inches to 0.020 inches at the failure sites. The cause of the thinning is attributed to single phase erosion/corrosion. Piping design, fluid temperature, high fluid velocity, and oxygen content of the fluid contributed to the accelerated wear rate. The original piping design in the early 70's had not considered the significant reductions in fluid oxygen levels that would be achieved by the time the unit was ready for operation. The minimum thickness occurred adjacent to the control valve(s), and increased by 0.011 inches per inch downstream from the valve(s).

The configuration of the failed piping was a horizontal run of 10" piping which reduced to 6" piping upstream of a 6" control valve. The 6" piping extended up to a 6" manual isolation valve, then expanded to 10" piping, before tying into the Condensate System. While it is normal to reduce the line size of piping directly upstream of a control valve, in typical piping designs, the piping is increased to the original line size immediately downstream of the associated control valve (see Figure 2). The fluid velocity in the piping downstream of the control valves has been calculated to be in the range of 17 ft/sec, which when coupled with the low oxygen levels produces a high erosion rate.

An erosion/corrosion monitoring and inspection program for these non-safety related piping systems was developed and has been in use following the failure of non-safety related piping at Surry in 1986. NRC Notice 86-106, Generic Letter 89-08 and other documents aided in this development by providing information regarding computer programs such as the EPRI CHEC program to predict piping areas susceptible to single phase erosion/corrosion and the developmental MIT program. A data entry error resulted in the DSM system piping not being entered into the computer program. A comprehensive review has determined that no other piping was omitted.

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 3	DOCKET NUMBER (2) 0 5 0 0 0 4 2 3 9 0	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
		0 3 0	0 1	0 3	OF	0 6	

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

III. Analysis of Event

This event is being reported in accordance with 10CFR50.73(a)(2)(iv) as an event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System. An immediate notification was made in accordance with 10CFR50.72(b)(2)(ii).

There were no significant safety consequences due to this event. The loss of instrument air to Containment resulted in a loss of normal non-safety related pressurizer pressure and level control. As a result, pressurizer pressure increased to the PORV setpoint. The PORV's functioned as designed to prevent pressure increases beyond the PORV setpoint. The safety grade letdown flowpath was available, but it was not used because restoration of the normal flowpath was expected. After the PORV cycled the second time, both the valve open and valve closed lights on the main control boards remained illuminated. PORV discharge tailpiece temperature was used to verify the PORV had reseated. The loss of power in the Turbine Building led to a loss of power to the plant process computer, rendering the Safety Parameter Display System (SPDS) unavailable. Control Room operators used the backup status trees to verify critical status functions. Monitoring functions normally performed by the computer were performed manually until power was restored to the computer.

IV. Corrective Action

The damaged sections of pipe were removed. Temporary pipe caps of a material consistent with the original system design were installed. The plant returned to power utilizing the Moisture Separator drain system alternate drain lines. (See Figure 1). During the third refueling outage (RFO3), which began on February 2, 1991, the six-inch diameter piping unstream and downstream of the level control valves was replaced with six-inch piping having greater erosion/corrosion resistant properties (i.e., 5% chrome-moly, schedule 40 pipe (A-335, grade P5)). The existing piping configuration was determined to be acceptable with the change in piping material.

The erosion/corrosion program was reviewed to verify that no other systems or portions of systems were omitted from the erosion/corrosion analysis. The affected DSM lines were added to the inspection program. These lines will remain in the inspection program even though they would be exempted because of the high chrome content of the replacement piping. Prior to the restart of the plant following the pipe break, nondestructive examinations were performed on the remaining sections of piping, on welds upstream and downstream of the failed piping, on a representative sample of piping with similar configurations, and downstream of the alternate drain line control valves. The DSM piping downstream of the alternate drain line control valves was inspected again during the refueling outage. The remaining piping systems with similar configurations were inspected during the refueling outage.

The complete erosion/corrosion program was remodeled using the CHECMATE program. The CHECMATE inspection output was analyzed to identify any piping that required inspection prior to returning to service following the refueling outage. A total of 164 erosion/corrosion examinations were performed prior to and during RFO3. Three piping lines were identified as requiring repairs in addition to the original DSM piping. All repairs have been completed. In addition, three piping lines were identified as requiring monitoring during the operating cycle (i.e., Cycle 4).

The instrument air containment isolation valve control circuitry was modified to prevent valve isolation on a loss of power to the current to pneumatic (I/P) controller.

An engineering task force was formed to investigate the root cause and to provide corrective action recommendations. The task force report has been issued and the report is currently being reviewed. An update report will be submitted by June 7, 1991, to discuss implementation of the corrective action recommendations.

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-630), 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 3	DOCKET NUMBER (2) 0 5 0 0 0 4 2 3 9 0	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
		0 3 0	0 1	0 4	OF	0 6	

TEXT (If more space is required, use additional NRC Form 365A, s). (17)

V. Additional Information

There have been no similar events that have occurred at Millstone 3.

EHS CodesSystem

Main Steam Reheat System-SB

Component

Level Control Valves-LCV

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 3

DOCKET NUMBER (2)

0 5 0 0 0 4 2 3 9 0

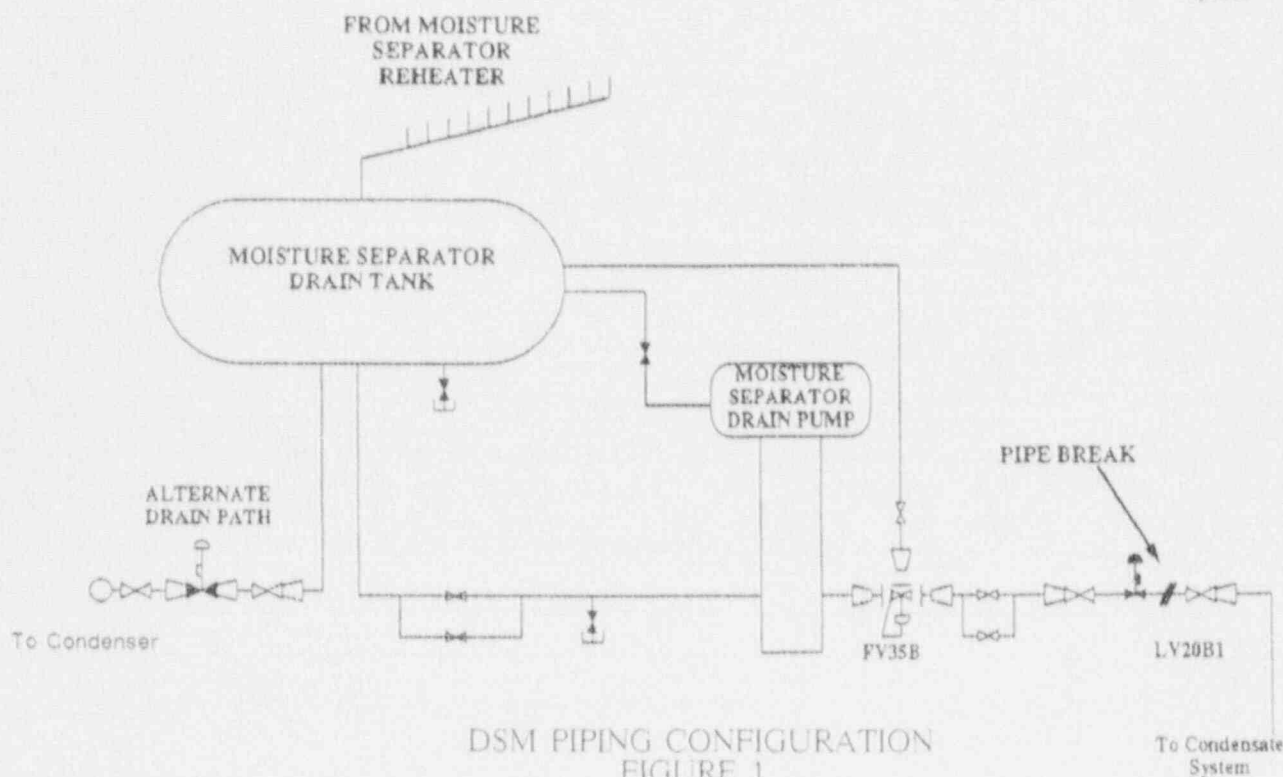
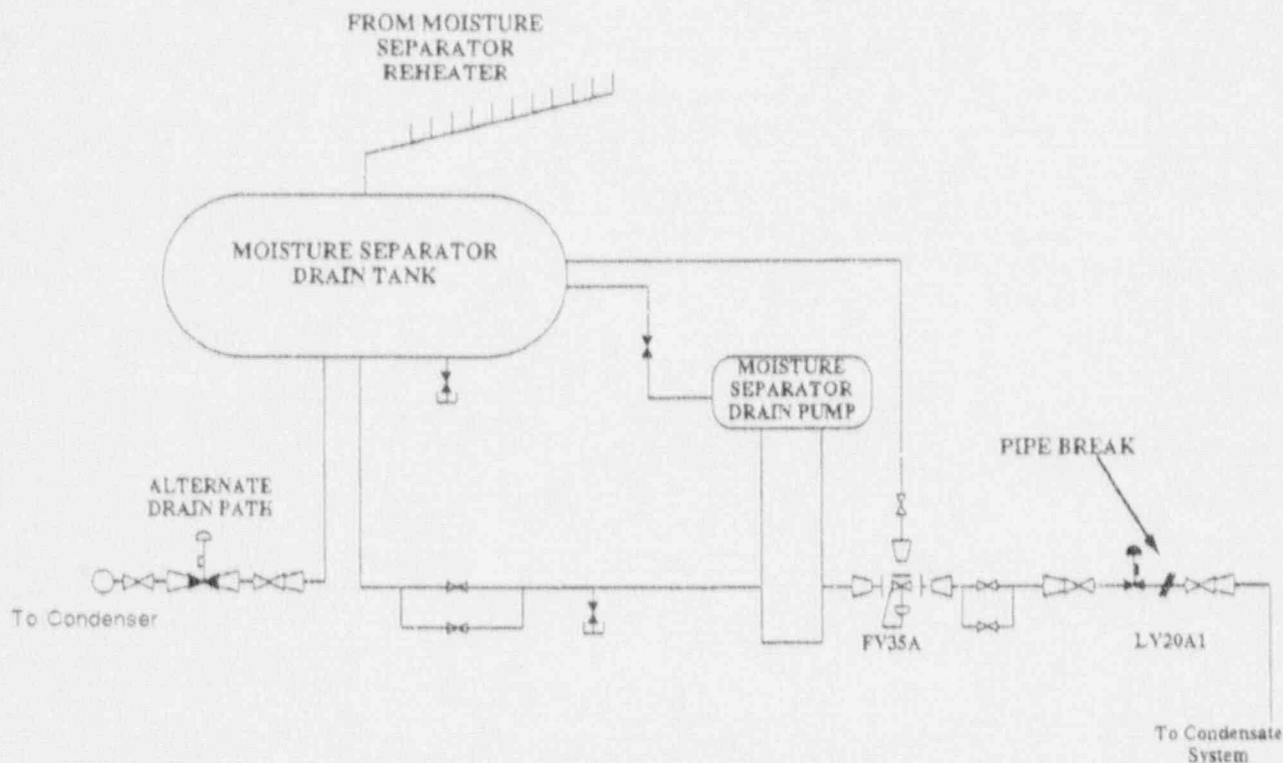
LER NUMBER (6)

YEAR SEQUENTIAL NUMBER REVISION NUMBER

0 3 0 0 1 0 5 OF 0 6

PAGE (3)

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

DSM PIPING CONFIGURATION
FIGURE 1

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)

DOCKET NUMBER (2)

LER NUMBER (6)

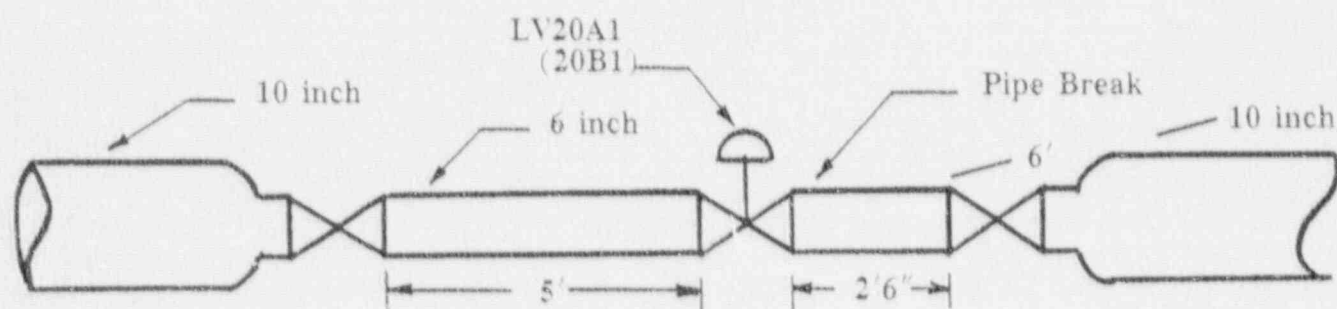
PAGE (3)

Millstone Nuclear Power Station
Unit 3

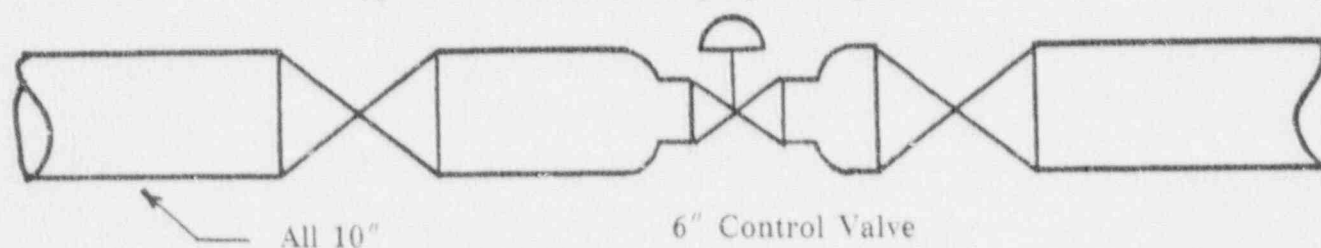
0 5 0 0 0 4 2 3 9 0 - 0 3 0 - 0 1 6 OF 0 6

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

DSM Control Valve Piping Configuration



Typical Control Valve Piping Configuration

DSM VS. TYPICAL PIPING CONFIGURATION
FIGURE 2