Northeast
Nuclear Energy

Rope Ferry Rd. (Route 156), Waterford, CT 06385

Millstone Nuclear Power Station
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385-0128
(203) 444-4300
Fax (203) 444-4277

The Northeast Utilities System

Donald B. Miller Jr.,
Senior Vice President – Millstone

Re: 10CFR50.73(a)(2)(iv)
September 7, 1995
MP-95-277

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 95-032-00

This letter forwards Licensee Event Report 95-032-00 required to be submitted within thirty (30) days pursuant to 10CFR50.73(a)(2)(iv).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Donald B. Miller, Jr.
Senior Vice President – Millstone Station

DBM/RT:ljs

Attachment: LER 95-032-00

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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NRC Form 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 2	05000336	1 OF 4

TITLE (4)
Manual Reactor Trip Due to Unisolable Secondary Steam Leakage

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 NAME	DOCKET NUMBER
08	08	95	95	032	00	09	07	95	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
		20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	60	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
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	(Specify in Abstract below and in Text, NRC Form 366A)
		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 (Include Area Code)
Philip J. Lutzi, Nuclear Licensing	(203) 440-2072

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

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		09	01	96

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

On August 8, 1995, at 1459 hours with the plant in mode 1 at 60% power, the RCS temperature at 556° and RCS pressure at 2265 PSI, the reactor was manually tripped from the control room due to a steam leak in a secondary system within the turbine building. Plant response was within norm and no automatic actuation of safety systems occurred. Examination of the area where the steam leakage occurred revealed a 14" vertical rupture in the 8" diameter recirculation line from the discharge of the B Heater Drains Pump (HDP) to the Heater Drain Tank.

Immediate corrective action consisted of manually tripping the reactor and placing the plant in mode 3, and stopping the steam leak and restricting access to the local area for personnel protection.

An event review team (ERT) was formed to investigate the cause of the pipe rupture. The ERT corrective actions were provided on a short term and long term basis (see Section IV of this LER). All short term corrective actions will be completed by January 1, 1996. All long term corrective actions will be completed by September 1, 1996.

This event is being reported pursuant to the requirements of 10CFR50.73(a)(2)(iv), any event or condition that resulted in a manual or automatic action of any Engineered Safety Feature (ESF), including the Reactor Protection System RPS.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	95	— 032 —	00	02 OF 04

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

I. Description of Event

On August 8, 1995, at 1459 hours with the plant in mode 1 at 60% power, the Reactor Coolant System temperature at 556° and RCS pressure at 2265 PSI, the reactor was manually tripped from the Control Room due to a secondary system steam leak within the turbine building. The steam leak could not be isolated locally with the turbine on line, therefore, the decision was made to manually trip the reactor and isolate main steam to the turbine building.

All actions taken by plant personnel following the trip were appropriate and equipment responses were as expected, except for one of the Main Steam Safety valves lifting immediately following the unit trip. Subsequent testing revealed the valve's setpoint to be 22 psi below the required minimum.

Examination of the area where the steam leakage occurred revealed a 14" vertical rupture in the 8" diameter recirculation line from the discharge of the B Heater Drains Pump (HDP) to the Heater Drain Tank.

II. Cause of Event

The event was caused by a secondary system steam leak within the turbine building that could not be isolated with the turbine on line. The decision was therefore made to manually trip the reactor and isolate steam to the turbine building.

The cause of the pipe rupture was investigated by an Event Review Team (ERT). Two causal factors were established as follows:

Causal Factor 1

The section of piping where the rupture occurred was degraded and water hammer caused its burst pressure to be exceeded.

The root cause was the introduction of subcooled water into the recirc line and subsequent draining of said line when the "B" Heater Drains Pump was stopped. There were two factors involved:

1. The procedure directed a step sequence which resulted in both the heater drains system subcooling valve and the recirc isolation valve being open at the same time. This allowed the recirc line to partially fill with subcooled liquid. The procedure then directed the stopping of the HDP which allowed the recirc line to drain.

2. The heater drains system design includes a horizontal recirc line which discharges through an open ended pipe to the heater drains tank. This design allows the recirc line to drain when both pumps are secured. The system also includes a flow path of condensate which can be aligned to the HDP suctions. This provides a potential source of subcooled liquid to the recirc line.

The combination of a partially filled (draining) recirc line and a subcooled fluid causes a water hammer-susceptible condition to exist.

Causal Factor 2

The affected pipe wall had thinned to the point that a system transient caused tensile overload (rupture).

The root cause was accelerated flow induced erosion/corrosion resulting from the recirc isolation valve being slightly open and/or leaking by. This condition could have been detected by the Erosion/Corrosion pipe inspection program but was not since the pipe is downstream of a normally closed valve. As such, it was not expected to experience high velocity flow for extended periods of time and hence should not have degraded.

Historically, the HDP recirc isolation valves have been found slightly open after plant start-up. This was predominantly the case on the "A" HDP based on the operational practice of starting the "A" HDP first.

This fact was not captured by the E/C program, so the element of the E/C program which reviews normally closed valves for off-normal operation did not identify the recirc isolation valve's downstream piping as susceptible to E/C. Thus, this section of piping was not inspected.

III. Analysis of Event

This event is being reported pursuant to the requirements of 10CFR50.73(a)(2)(iv), any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). The reactor was manually tripped due to an unisolable secondary system steam leak within the turbine building. There was no safety consequence as a result of this event. All equipment responded as expected and no safety systems were compromised by this secondary system event.

IV. Corrective Action

Reactor Trip

Appropriate measures were taken to stop the steam leak and to restrict access to the local area for personnel protection. The reactor was manually tripped and the plant was placed in Mode 3.

Main Steam Safety Valve Lifting

After the trip, testing of the main steam safety valve that lifted revealed its setpoint to be 22 psi below the required minimum. Given this setpoint the valve was noted to reseat as expected. The valve closed tightly, once pressure decayed below the reseat level. Simmer testing was completed, confirming the low setpoint, and the valve was adjusted to the required higher relief setpoint. A review of valve history and the fact that the remaining relief valves did not lift following this trip indicates that this is an isolated problem, albeit another case of relief setpoint drift.

Pipe Rupture Causal Factor 1

An event review team provided short and long term corrective actions to the steam pipe ruptures. Among them:

(a) Critical piping and components in the Heater Drains System were inspected to identify damage caused by the system transient. The following piping and components were inspected: suction side expansion bellows, heater drains tank internals, subcooling supply piping, recirculation piping, high level dump piping, piping supports and hangers, and component supports. No unacceptable conditions were identified.

(b) A design review of the Heater Drains System to verify adequacy for all operational conditions, update design documents, and propose design changes as applicable will be performed before revising OP 2320, "Feedwater Heater Drains and Vents," to allow securing two heater drains pumps at greater than 50% power. After completing the design review, Operations and Engineering should review the Alarm Response and Abnormal Operating Sections of this procedure. (Long-term)

Pipe Rupture Causal Factor 2

(a) Heater Drain Recirculation isolation valves 2-HD-45A and 2-HD-45B were visually inspected for erosion/corrosion and signs of seat leakage.

(b) Other piping systems on Unit 2 in similar systems or downstream of normally closed or throttled valves were inspected for erosion/corrosion. All UT inspections were found satisfactory.

(c) The Erosion/Corrosion Manual should be revised to require an enhanced SRO and multi-discipline review of non-standard operations before each Refueling Outage in order to update the Erosion/Corrosion Program. (Short-term)

(d) The Erosion/Corrosion Program Manual should include a standard questionnaire designed to elicit historical operations data from Operations, I&C, Engineering, and Maintenance Departments. Following completion of the questionnaires, the responding personnel should be interviewed as a _group_ to build on this data base. (Long-term)

(e) Other NU nuclear units should include normally closed valves on recirculation lines in their E/C Program and inspect the downstream pipe during the next Refuel Outage. Evaluation of other lines with normally closed valves where the potential exists for mispositioning or leak-by will be addressed during the enhanced multi-discipline review process identified earlier. (Long-term)

All short term Corrective Actions will be completed by January 1, 1996 and long term Corrective Actions will be completed by September 1, 1996. A supplement to this LER will be submitted to report the completion of corrective actions.

V. Additional Information

Similar LERs: 91-012-01: Manual Reactor Trip Due to Plant Conditions Resulting From a Rupture in the Reheater Drain Tank to High Pressure Feedwater Heater Pipe

EIIS Code: None