Northeast
Utilities System

107 Selden Street, Berlin, CT 06037

Northeast Nuclear Energy Company
P.O. Box 270
Hartford, CT 06141-0270
(203) 665-5000

April 19, 1996

Docket No. 50-423
B15669

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

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

This letter forwards Licensee Event Report 95-020-01, which updates the cause and corrective action for the event. The supplement updates Licensee Event Report 95-020-00, which was submitted within thirty (30) days pursuant to 10CFR50.73(a)(2)(i)(B).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

M. H. Brothers
Unit Director, Millstone Unit No. 3

Attachment: LER 95-020-01

cc: T. T. Martin, Region I Administrator
A. C. Cerne, Senior Resident Inspector, Millstone Unit No. 3
V. L. Rooney, NRC Project Manager, Millstone Unit No. 3

IE22
1/1

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), 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.

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TITLE (4)
Reactor Coolant System Pressure Boundary Leak from a Valve Bonnet Stem Leak-Off Pipe

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
12	02	95	95	020	01	04	19	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
		20.2201(b)	20.2203(a)(2)(v)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)
POWER LEVEL (10)	000	20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
		20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
		20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
		20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
		20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
William J. Temple, Nuclear Licensing Supervisor	(860)437-5904

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	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

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

On December 2, 1995, with the plant in Mode 5, Cold Shutdown, a leak was discovered on the valve stem leak-off pipe for the Residual Heat Removal System supply from a Reactor Coolant System loop inboard isolation valve (3RHS*MV8701C). This pipe had been modified to vent the valve bonnet to the RCS, to prevent pressure locking / thermal binding of the valve disc.

This condition is conservatively being reported in accordance with 10CFR50.73(a)(2)(i)(B) as a condition prohibited by Technical Specifications. Technical Specification 3.4.6.2 requires "no pressure boundary leakage" in Modes 1-4. This report is being made due to the possibility of having unisolable leakage past the disc of 3RHS*MV8701C, even with the valve in the closed position during Modes 1, 2, and 3. Some leakage is possible since the disc is a flex wedge design, which could allow pressure upstream of the valve to leak past the upstream seat of the disc and into the bonnet chamber. This pressure then assists in seating the downstream seat of the disc. (See Sketch 1 for leakage path through the valve). In Mode 4, one train of RHR is required for plant cooldown, which could require opening this valve. 3RHS*MV8701C could then be placed on its backseat to isolate the bonnet, and the downstream isolation valve(s) 3RCS*V969, V2002, V2003 could be closed to isolate the leak. (See Sketch 2 for piping layout). However, if the opposite train of RHR is being used to cool the RCS, 3RHS*MV8701C would remain closed and the leak path through the valve still exists.

An initial inspection indicated a crack near the toe of the fillet weld between the pipe and the valve body. The leaking section of pipe was removed for further evaluation which resulted in the discovery of six linear indications at a 45-degree angle to the pipe. (See Sketch 3 for inspection results). None of the indications were in the weld material, all were located in the pipe.

Although this incident involved reactor coolant leakage, it had low safety significance. The pipe connection to the RCS has a 0.375-inch diameter orifice to limit RCS leakage. Also, the hole in the valve bonnet between the stuffing box and the stem leak-off pipe is only 0.25-inch in diameter. These restrictions limit potential leakage to within the capacity of the normal makeup system.

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

I. Description of Event

On December 2, 1995, with the plant in Mode 5, a leak was discovered on the valve stem leak-off pipe for the Residual Heat Removal System (RHS) supply from Reactor Coolant System (RCS) Loop A inboard isolation valve 3RHS*MV8701C. The leak was discovered during a visual inspection for RCS leakage. The leak was apparently due to a cracked pipe near the weld joining the stem leak-off pipe to the valve bonnet. The inboard isolation valve 3RHS*MV8702C, for the RHS supply from RCS Loop D, was also inspected for leakage and none was noted. This valve is identical to 3RHS*MV8701C, and has had the same modifications performed on the stem leak-off piping.

The leak was considered RCS pressure boundary leakage since the stem leak-off pipe is routed back to the RCS. The valve stem leak-off had been routed to the RCS to address pressure locking / thermal binding concerns for this valve. This results in the potential for leakage past the disc of 3RHS*MV8701C, even with the valve in the closed position during Modes 1, 2, and 3. Some leakage is possible since the disc is a flex wedge design, which can allow pressure upstream of the valve to leak past the upstream seat of the disc and into the bonnet chamber. This pressure then assists in seating the downstream seat of the disc. See attached Sketch 1 for the leakage path through the valve. In Mode 4, one train of RHR is required for plant cool down, which could require opening this valve. Valve 3RHS*MV8701C could then be placed on its backseat to isolate the bonnet, and downstream isolation valve(s) 3RCS*V969, V2002, V2003 could be closed to isolate the leak. See attached Sketch 2 for clarification of the piping layout. However, if the opposite train of RHR is being used to cool the RCS, 3RHS*MV8701C would remain closed and the leak path through the valve still exists. The existence of pressure boundary leakage in Modes 1-4 is a condition prohibited by Technical Specifications.

At the time of discovery, the plant was in Cold Shutdown due to a previously identified RCS pressure boundary leak through an instrument tap that had been immediately reported on November 30, 1995 (LER 95-019-00). Although the condition was discovered while the plant was shut down, it was determined that the condition may have existed while the plant was previously operating.

Due to the small size of the leak, no operator actions were required, no automatic or manually initiated safety response was required, and no other system or functions were affected.

II. Cause of Event

Initial inspection indicated a small crack in the pipe near the toe of the fillet weld between the pipe and the valve body. A further examination of the removed section of piping revealed six linear indications in the pipe, and none in the weld. Five of the indications were at a 45-degree angle to the pipe. The most likely cause of the indications is from mechanically induced tensile overload consisting of a bending or twisting motion. The failures were not the result of a fatigue mechanism, material incompatibilities, the welding process, or service related conditions. The attached Sketch 3 shows the locations of the linear indications.

III. Analysis of Event

The condition is being conservatively reported in accordance with 10CFR 50.73(a)(2)(i)(B) an operation or condition prohibited by Technical Specifications. Technical Specification 3.4.6.2 requires "no pressure boundary leakage."

Although this incident involved a reactor coolant leak, it had low safety significance. The piping connecting the stem leak-off pipe to the RCS was designed with a flow restriction. There is a 0.375-inch diameter orifice where the line connects to the RCS. The valve stuffing box to the valve stem leak-off pipe is restricted by a 0.25-inch diameter hole.

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

Also, the valve is normally closed during plant operation, greatly restricting flow into the bonnet area. This serves to limit potential RCS leakage to less than the capacity of the normal make-up system. The RCS unidentified leakage rate remained low during the time frame up to discovery of the leak. Unidentified reactor coolant leakage is collected in the containment drain system and is monitored to fractions of a gallon per minute, with a maximum allowed leakage of one gallon per minute. The Containment Atmosphere Gaseous and Particulate Radioactivity Monitoring System also is used for identification of reactor coolant leaks. The leak reported here was not distinguishable by either of these systems, and was ultimately identified by a visual inspection.

This event did not prevent any safety system or equipment from performing its safety function. However, a U-bolt located below the leak was damaged due to boric acid corrosion and was subsequently replaced. An evaluation of the system with the degraded U-bolt determined the system was operable. No system or other equipment was out of service as a result of this event.

IV. Corrective Action

The failed pipe and valve bonnet weld were removed. A new pipe was installed and welded into the valve bonnet. The fit-up, visual, and liquid penetrant examinations of the weld were satisfactory. A visual examination of the weld was conducted at normal operating pressure and there was no leakage.

The location and configuration of the linear indications are consistent with mechanically induced tensile overload consisting of a bending or twisting motion. It cannot be determined exactly when or how the leak-off pipe was mishandled. This condition may have been present since construction/startup. The problem then manifested itself after the pressure locking / thermal binding modification. During this modification, packing which normally isolates the leak-off piping from system pressure, was removed. This allows the upper bonnet area to vent to avoid the pressure locking / thermal binding. This would also be the first time that this pipe saw system pressures.

The now modified configuration is less susceptible to future tensile overloads. The longer piping configuration is readily visible and would distribute any loads over a longer length from the original six-inch pipe nipple. Additionally, flexible hose and pipe supports added as a result of the modification assist in protecting the pipe from future recurrences.

The inboard isolation valve 3RHS*MV8702C, for the RHS supply from RCS Loop D, was also inspected for leakage and none was noted. This valve is identical to 3RHS*MV8701C, and has had the same modifications performed on the stem leak-off piping.

V. Additional Information

Stem leak-off is 1/2" sch. 80 pipe, SA316 type F304 material.

There has been no previous LER similar to this failure of a stem leak-off pipe.

At the time of discovery, the plant was in Cold Shutdown due to a previously identified RCS pressure boundary leak through an instrument tap that had been immediately reported on November 30, 1995. That plant condition was previously reported in LER 95-019-00.

Similar Events

There has been no previous LER similar to this failure of a stem leak-off pipe.

Manufacturer Data

EIIS System Codes
Reactor Coolant System - AB
Residual Heat Removal System - BP

EIIS Component Codes
Pipe

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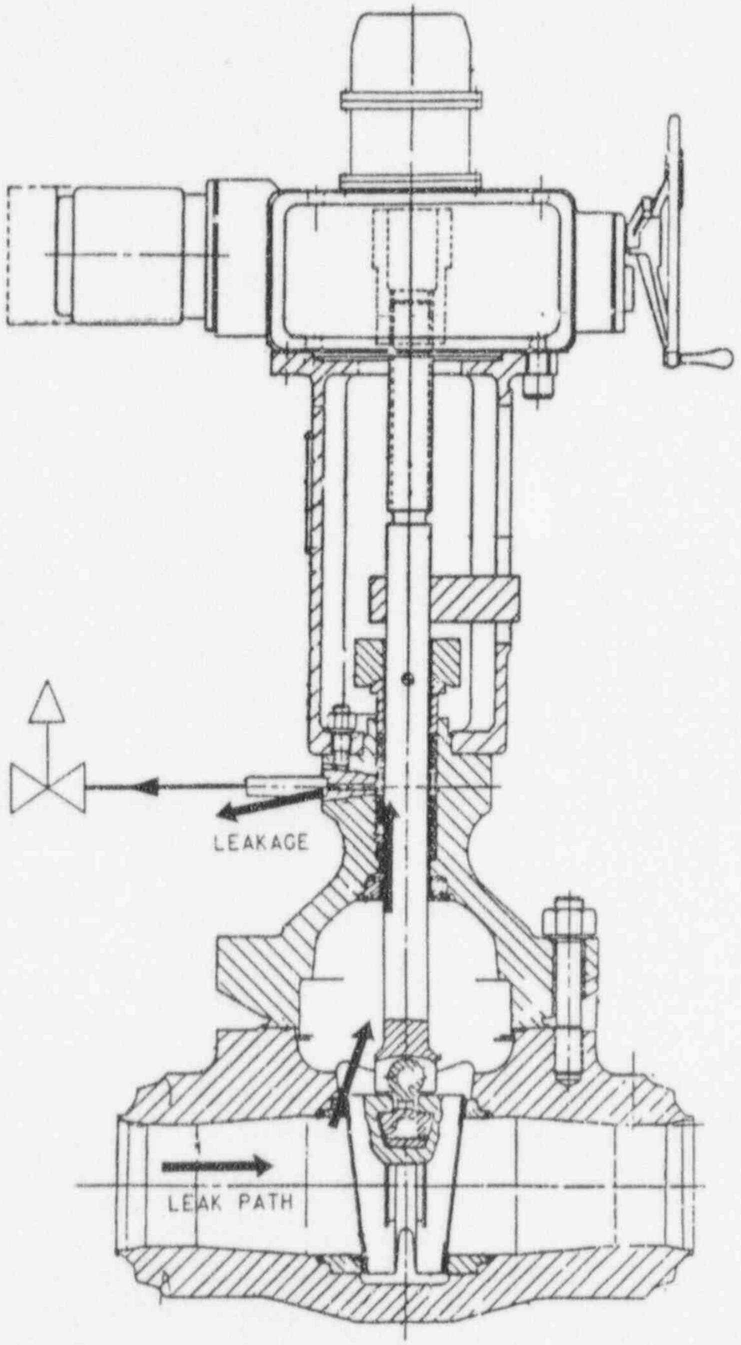

SKETCH 1
3RHS*MV8701C

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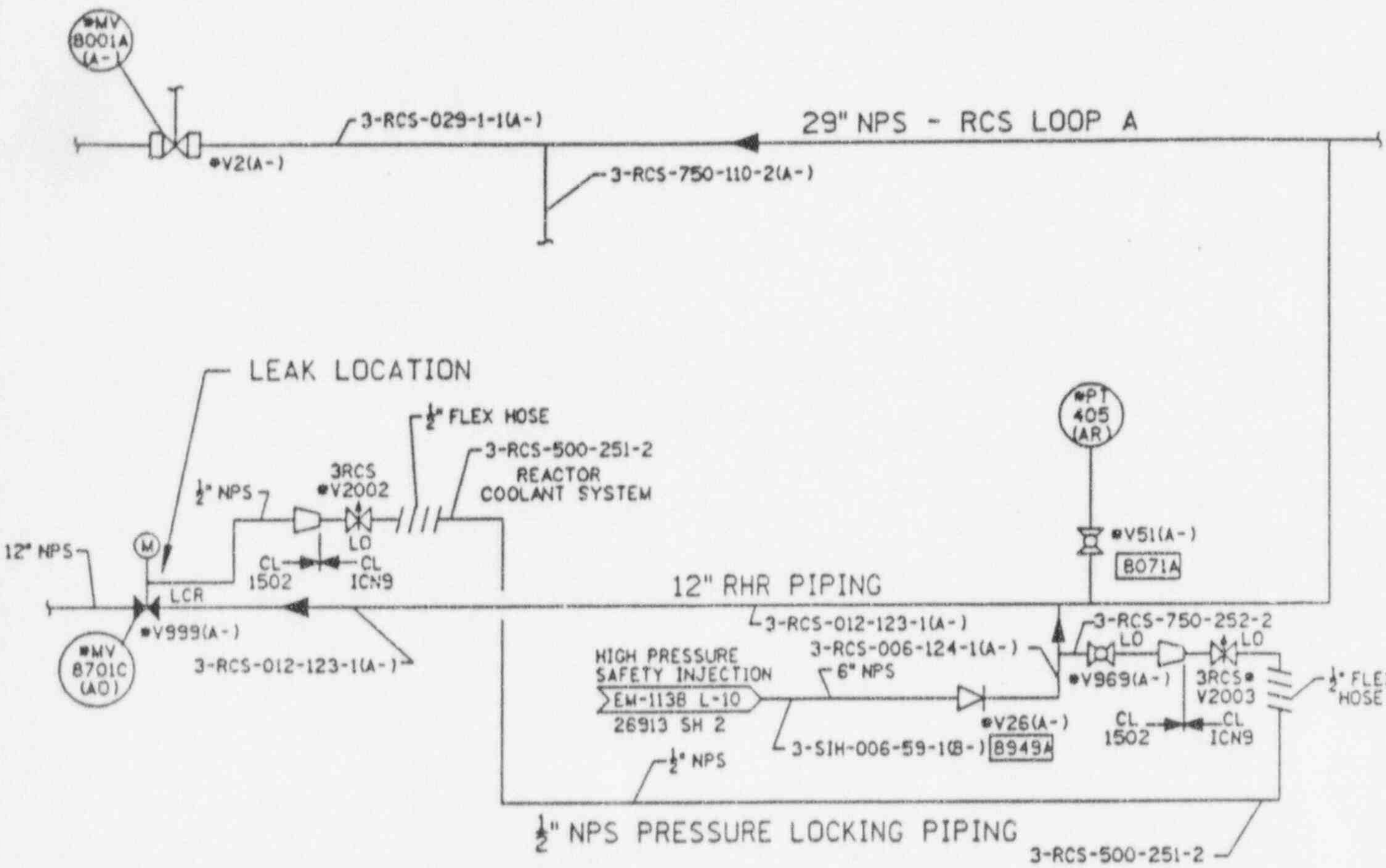

SKETCH 2

3RHS*MV8701C PRESSURE LOCKING / THERMAL BINDING PIPING

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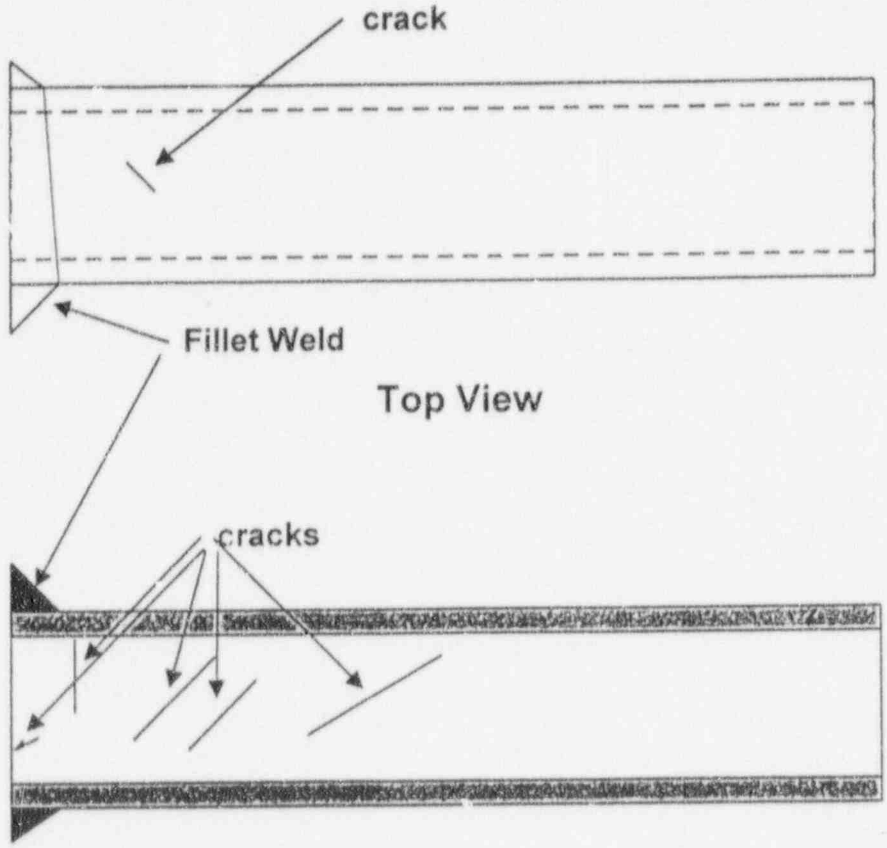

SKETCH 3
Locations of Linear Indications