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)
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TITLE (4)

Combined Bypass 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	DOCKET NUMBER(S)
0 2	1 8	8 8	8 8	— 0 0 6	— 0 0	0 3	1 5	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 0

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

[ ] 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)	[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)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
Stephen L. Stadnick, Engineer X4427	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
B	W K	I S V	M 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On 2/18/88 the plant was in Mode 2, reactor critical after refueling, power level was zero, temperature was 532 degrees fahrenheit and pressure was 2263 psig. Local leakage rate testing determined the combined bypass leakage rate was 0.02907 percent per day. This rate exceeds the Technical Specification 3.6.1.2.C limit of 0.017 La or 0.0085 percent per day.

The primary contributor to the high leakage rate is the inner containment isolation valve, 2-SSP-16.1, on the normal containment sump piping. The cause of the leakage is due to debris lodged in the caged area of the valve that prevented the plug from entering into the seat.

The debris was removed, the seating surfaces were cleaned up, and the subsequent leakage test was satisfactory. The as-left leakage rate for combined bypass leakage is 0.0012 percent per day.

To prevent debris from becoming lodged under the seat the plant will modify the flow characteristics of the valve during the next refueling outage.

Similar Events: 86-012, 85-00^, 84-005, 82-006, 80-032, 79-034.

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

I. Description of Event

On 2/18/88 the plant was in mode 2 with power level at 10(-1) percent, temperature was 532°F and pressure was 2263 psig. The local leakage rates were being tabulated after the refueling outage. The combined bypass leakage rate was determined to be 0.02907 percent per day. This exceeds the Technical Specification limit of 0.017 La or 0.0085 percent per day.

The cause of this event is due to the high leakage rate on the inner containment isolation valve on penetration number 14. This is the normal containment sump penetration (WK).

There were no other systems or secondary functions that actuated as a result of this event.

The method of discovery for this event was the performance of a surveillance test and subsequent completion of the calculation for combined bypass leakage rate.

There were no major operator actions as a result of this event.

There were no automatic or manually initiated safety responses as a result of this event.

II. Cause of Event

The cause of the high leakage is due to debris being lodged in the seat area of the valve. Specifically, a plastic electrician's tie-wrap was lodged in the caged part of the valve preventing full closure of the plug into the seat.

III. Analysis of Event

This event is reported pursuant to paragraph 50.73(a)(2)(i)(B), an operation or condition prohibited by the Plant Technical Specifications.

The combined bypass leakage is determined by the addition of the leakage rates of the penetrations listed in table 3.6-1 of the Technical Specifications. These are leakage paths that could bypass the secondary containment (Enclosure Building). Penetration 14, the normal containment sump (WK), and 67 and 68, the refueling water purification system (DA) comprise this leakage. For penetration 14, the higher of the inner and outer containment isolation valves is considered in the bypass leakage calculation. This assumes the valve with the lower of the two leakage fails. For penetrations 67 and 68 the inner and outer valves on each penetration are tested together and one leakage rate is obtained and considered in the bypass leakage calculation. These rates were 113.5 and 27.1 sccm respectively. The inner valve for penetration 14 had 50,000 sccm and the outer valve had 415 sccm. When the actual leakage thru the penetration is considered along with the leakages thru the refueling water purification penetrations, the combined bypass leakage is 0.00032%/day.

This is well below the Technical Specification limit. Based on the above, there are no safety consequences as a result of this event.

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

IV. Corrective Action

After the initial local leakage test for the inner valve on penetration 14 the valve was disassembled for repair. The repair included removal of the debris, polishing of the plug and seat and reassembly of the valve. The subject retest of the valve indicated an acceptable leakage rate result.

Debris lodged in the seat area of the valve has been identified as the cause of the excessive leakage. The containment isolation valves installed in this piping system are Masoneilan flow control valves. The design of this valve includes a plug and seat held in place by a cylindrical cage. This cage contains holes to allow passage of fluid. This type of valve tends to collect debris and prevents the plug from seating tightly.

Previous action to prevent recurrence included installation of screens in the drain trench to the sump. This has eliminated the majority of the debris that entered the sump but has not been completely successful in debris elimination as evidenced by the tie wrap caught in the cage of the valve. To prevent future instances, the flow characteristics of the inner and outer valves will be modified during the next refueling outage. This will allow any debris that is pumped from the sump to pass thru the containment isolation valves on penetration 14.

V. Additional Information

The normal containment sump piping penetration is 3" diameter pipe. The inside and outside containment isolation valves are Masoneilan (M120) model number 13456-25 air operated flow control valves. The refueling water purification system piping penetrations are 4" diameter pipe. The inside and outside containment isolation valves are Velan (V085) model number B12-054B-13MS 4" diameter bolted bonnet manually operated gate valves.

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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March 15, 1988
MP-11627
Re: 10CFR50.73(a)(2)(i)(B)

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 88-006-00

Gentlemen:

This letter forwards the Licensee Event Report 88-006-00 required to be submitted within thirty days pursuant to paragraph 50.73 (a)(2)(i)(B).

Yours truly,

NORTHEAST NUCLEAR ENERGY COMPANY

Stephen E. Scace
Station Superintendent
Millstone Nuclear Power Station

SES/SLS:mo

Attachment: LER 88-006-00

cc: W. T. Russell, Region 1
W. J. Raymond, Senior Resident Inspector

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