NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Containment Integrated Leakage in Excess of Tech Spec Limit

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 5 0 0 0
0 4	0 9	8 7	8 7	— 0 0 2 —	0 0	0 5	0 6	8 7	NONE	0 5 0 0 0

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

OPERATING MODE (9):
POWER LEVEL (10): 0

[ ] 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
C. W. Fay, Vice President - Nuclear Power	4 1 4 2 2 1 - 2 8 1 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
B	J B	I S V								

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[X] NO				

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

On April 9, 1987 the Unit 1 "as found" containment integrated leak rate was determined to be in excess of the allowable Technical Specification limit (0.75 Lt). A local leak rate test of four steam generator level transmitter isolation valves resulted in a collective leak rate of 70% of the allowable leak rate. The containment "as found" leak rate, taking into account the local leak rate of the four isolation valves, was 133% of the allowable leark rate (0.75 Lt). The cause of the valve failure was failed/leaking packing in three of the four valves. The affected valves were replaced with new valves and successfully tested during the ILRT.

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EVENT DESCRIPTION

On April 4, 1987, Unit 1 was shut down for its annual refueling outage. On April 9, 1987, following the integrated leak rate test of the Unit 1 containment, it was determined that the "as found" leak rate was in excess of 0.75 Lt. Technical Specifications 15.4.4.I.B.3 requires a limit of 0.75 Lt for reduced pressure tests.

Preceding the Unit 1 outage, leakage from the secondary side of both steam generators to the containment was estimated to be approximately 1740 gallons of water per day. Nearly all of this leakage was past the packing of three steam generator level tap isolation valves and was accommodated by periodic drainage of the containment sump. These valves were:

Valve No.	Valve Description
MS473C	Second off isolation valve to level transmitter ILT 473 ("B" S/G)
MS471C	Second off isolation valve to level transmitter ILT 471 ("B" S/G)
MS462A	Second off isolation valve to level transmitter ILT 462 ("A" S/G)

To determine the leak rate that would exist past the packing of the suspect valves at containment test pressure, with a vented steam generator, a local leak rate test of each suspect valve was performed on April 8, 1987. The 3 valves and their attached piping were removed from the steam generators without disturbing the valves. A fourth level tap isolation valve MS462B was also removed to accommodate the removal of MS462A. These valves were then placed in a test chamber that could be pressurized to containment test pressure (30 psig). The test chamber was configured such that air pressure could be applied to the outside of each valve and its associated piping. Any leakage thru the valve packing would go through the valve and associated piping which were open to atmosphere. The total leakage was determined by measuring the flow of air to the test chamber at 30 psig. The combined leak rate for all four level tap isolation valves was 89000 sccm at 30 psig. This is equivalent to 0.1484% of containment volume per day or 70% of the allowable containment integrated leak rate of 0.75 LT (0.212% per day).

The containment integrated "as found" leak rate, calculated at the 95% confidence level was 0.1338% of containment volume per day (63% of the allowable leak rate) exclusive of the leakage attributable to the steam generator level tap isolation valves. Adding the leakage for the isolation valves resulted in an "as found" containment integrated leak rate of 0.282% of containment volume per day or 133% of the allowable leak rate (0.212% per day).

The subject valves are Velan Valve Corp. model 274W, 3/4-inch, 600 psig, carbon steel, globe valves. These valves are located inside containment and are manual isolation valves for associated narrow range steam generator level transmitters.

All four valves, MS462A and B and MS471C and MS473C were replaced prior to the integrated leak rate test of the containment. In this manner the leak rate past the newly installed valves was included in the overall integrated leak rate results so as to permit determination of the "as left" condition of the containment.

As previously mentioned, the containment integrated leak rate was 63% of the allowable leak rate which demonstrates that the containment integrated leakage, with the newly installed valves, is well within acceptable limits.

GENERIC IMPLICATIONS

No generic implications related to this incident have been discovered.

REPORTABILITY

This report is filed pursuant to 10 CFR 50.73(a)(2)(i), "Any operation or condition prohibited by the plant's Technical Specifications." The Energy Industry Identification System component function identifier is ISV and the system code for the valves is JB.

CAUSE

The cause of the failure to meet the specification for the integrated leak rate was the packing failure of one valve (MS462A). Only minor packing leaks were evident on the other two valves, MS473C and MS471C.

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

SAFETY ASSESSMENT

Operation of Unit 1 during the last fuel cycle posed no safety hazard to the employees of Point Beach Nuclear Plant or the general public. During the local leak rate test of the four valves, a test pressure of 30 psig was applied to the exterior of the valves with no internal pressure applied. During a hypothetical loss of coolant accident, pressure in the steam generator would remain higher than containment pressure during the accident and any leakage through the subject valves would be from the steam generator into the containment. Therefore, although the leakage past the packing on these valves contributed significantly to the overall leak rate during the containment integrated leak rate test, it would not contribute any significant leakage out of the containment during an actual loss of coolant accident while the steam generator was pressurized.

Following a hypothetical steam line break accident, the steam generator may depressurize. If the break is inside the containment the leakage from these valves is of no concern. If it is outside the containment, there is no containment pressure to force the containment atmosphere through the valve, through the steam generator, to the environment.

The leakage became evident early in 1987 and was monitored. The leakage was quantified by the amount of water drained from the sump. The water was periodically analyzed to assure the secondary side leak was not masking a primary system or service water system leak. The radiation monitors which would respond to a primary coolant leak were operable. Biweekly containment inspections were made to check for other leaks and monitor the valve leakage. Containment humidity was also trended to provide assurance that other equipment would not be affected. The operation of the associated level indicators was not affected.

CORRECTIVE ACTIONS

The subject valves were replaced and successfully tested as part of the integrated leak rate test.

SIMILAR OCCURRENCES

No similar occurrences of this type are known to have occurred at Point Beach Nuclear Plant.

Wisconsin Electric POWER COMPANY
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VPNPD-87-194
NRC-87-049

May 6, 1987

U. S. NUCLEAR REGULATORY COMMISSION
Document Control Desk
Washington, D. C. 20555

Gentlemen:

DOCKET 50-266
LICENSEE EVENT REPORT 87-002-00
CONTAINMENT INTEGRATED LEAK RATE IN EXCESS OF
TECHNICAL SPECIFICATION LIMITS
POINT BEACH NUCLEAR PLANT, UNIT 1

Enclosed is Licensee Event Report 87-002-00 for Point Beach Nuclear Plant Unit 1 detailing the failure of the Unit 1 containment to pass its integrated leak rate test in the "as found" condition.

LER-87-002-00 is filed pursuant to 10 CFR 50.73(a)(2)(i), "Any operation or condition prohibited by the plant's Technical Specifications."

If you have any questions, please do not hesitate to contact us.

Very truly yours,

C. W. Fay
Vice President
Nuclear Power

Enclosure

Copies to NRC Resident Inspector
NRC Regional Administrator, Region III

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