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)
Point Beach Nuclear Plant, Unit 1	0 5 0 0 0 3 0 1	1 OF 0 3

TITLE (4) Containment Isolation Valve Leak Rate in Excess of Technical Specification Limits

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
1 0	0 3	8 6	8 6	0 0 5	0 0	1 0	3 1	8 6	None	0 5 0 0 0
									None	0 5 0 0 0

OPERATING MODE (9): 0

POWER LEVEL (10): 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	
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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	B D	I S V	V 0 8 5	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 October 3, 1986, during a type "C" leak rate test, the "A" reactor coolant pump component cooling water supply line check valve (755A) leak rate exceeded the containment leakage rate allowed by Technical Specifications 15.4.4.II.B and III.B.

Valve 755A, a four-inch, 150-pound, carbon steel, Velan swing check, appeared to stick partially open during the initial Type "C" test. The required test pressure could not be achieved and, therefore, the leak rate at the rated pressure could not be quantified.

Because of past leakage problems with this valve, it will be removed and replaced with a new valve prior to startup of Unit 2. The replaced valve will be inspected to determine if the cause of the leakage can be identified.

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

EVENT DESCRIPTION

On September 27, 1986, Unit 2 was shut down for its annual refueling. On October 3, 1986, the total "as-found" leakage for Type "B" and "C" local leak rate testing exceeded 0.6 la when excessive leakage through the "A" reactor coolant pump component cooling water supply containment isolation valve (755A0 was found. The limit 0.6 La is required by Technical Specifications 15.4.4.II.B and 15.4.4.III.B.

Valve 755A was tested according to procedure; however, the required test pressure of 60 psig could not be achieved in the test volume. An indicated test pressure of 18.3 psig was achieved while an indicated leak rate of greater than 162,000 scfm existed. Note that this leak rate is not the actual leak rate which would exist if the test volume could be pressurized to 60 psig. Therefore, in view of the test findings, it is assumed that the leak rate of 755A by itself would violate the total leakage limits set forth in the Technical Specifications.

The subject valve is a four-inch, 150-pound, carbon steel, check valve model 73908 manufactured by Velan Corporation. This check valve is located inside containment in an incoming component cooling water line to the "A" reactor coolant pump. Additional isolation capability is provided by Valve 754A located outside containment, which can be remotely operated from the control room. Operator action would be required to shut this valve in the event it was needed to establish containment isolation. This valve was subjected to a Type "C" test with satisfactory results.

When Valve 755A was last tested after its repair prior to the start-up of Unit 2 in the fall of 1985, it had a leak rate of 1680 sccm. (See LER 85-002-01.) In 1985 it was anticipated that this repair was successful, since a similar Unit 1 valve was repaired in 1985 in the same manner and had passed its leak test after a full year of operation.

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."

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(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

SAFETY ASSESSMENT

Operation of Unit 2 during the last fuel cycle posed no safety hazard to the employees of Point Beach Nuclear Plant or the general public for several reasons.

First, during the test procedure, no actual backflow condition is created through the check valve as would be expected under accident conditions. Therefore, there is no shutting force created by reverse flow against the disc during the test. It is very likely that during a design basis accident, should the containment atmosphere be exposed to a broken CCW line creating reverse flow in the line, liquid remaining in the line would cause the valve disc to seat resulting in a less significant leak rate than that found during the test. Second, the CCW system is a closed system outside containment with a normal operating pressure greater than containment accident pressure. Third, the existence of remote operating valve 754A, which passed its most recent leak test, allows the operator to isolate this CCW line should it become necessary to establish containment integrity.

SIMILAR OCCURRENCES

Valve 2-755A has failed its leak test several times in the past. Each time the valve was disassembled and repaired such that the after-repair leak rate of the valve was acceptable. Leak testing of this valve prior to 1983 revealed no problems. See Licensee Event Reports 83-004/T-01, 84-008-00, 85-002-00, and 85-002-01.

Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201 (414) 277-2345

VPNPD-86-447
NRC-86-105

October 31, 1986

Mr. J. G. Keppler, Regional Administrator
Office of Inspection and Enforcement
Region III
U. S. NUCLEAR REGULATORY COMMISSION
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

DOCKET 50-301
LICENSEE EVENT REPORT 86-005-00
CONTAINMENT ISOLATION VALVE LEAK RATE IN EXCESS OF TECHNICAL SPECIFICATION LIMITS
POINT BEACH NUCLEAR PLANT, UNIT 2

Enclosed is Licensee Event Report 86-005-00 for Point Beach Nuclear Plant, Unit 2. This report details a containment isolation valve which had a leak rate in excess of the Technical Specification limit of 0.6 La during its Type "C" test.

LER 86-005-00 is filed pursuant to 10 CFR 50.73(A)(2)(i), "Any operation or condition prohibited by the plant's Technical Specifications."

If any further information is required, please contact us.

Very truly yours,

CW Fay

C. W. Fay
Vice President
Nuclear Power

Enclosure

Copies to NRC Resident Inspector
NRC Document Control Desk,
Washington, DC (with original)

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