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	0 5 0 0 0 3 0 1	1 OF 0 5

TITLE (4): Containment Isolation Valve Leakage in Excess of Tech Spec 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 4	8 6	8 6	0 0 5	0 1	0 2	0 9	8 7	None	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9):
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 0

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	R 3 4 0	Yes					

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 leak rate allowed by Technical Specifications 15.4.4.II.B and III.B. Because of past leakage problems with this valve, it was removed and replaced with a new valve prior to unit startup.

On October 8, 1986, during a type "C" leak rate test of the excess letdown heat exchanger supply check valve (767), leakage also exceeded the Technical Specification limits cited above. This valve was disassembled and the disk O-ring and spring were replaced. Post-repair leakage was measured at 2 sccm.

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EVENT DESCRIPTION

On September 27, 1986, Unit 2 was shut down for its annual refueling. 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 (CCW) supply containment isolation valve (755A) and the excess letdown heat exchanger supply check valve (767) was found on October 3, 1986, and October 8, 1986, respectively. The leak rate limit of 0.6 La is required by Technical Specifications 15.4.4.II.B and III.B.

Valve 755A was tested according to procedure; however, the required test pressure of 65 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 sccm existed. Note that this leak rate is not the actual leak rate which would exist if the test volume could have been pressurized to 65 psig. Therefore, in view of the test findings, it is assumed that the leak rate of 755A by itself violates the total leakage limits set forth in the Technical Specifications.

755A is a four-inch, 150-pound, carbon steel, check valve model 73908 manufactured by Velan Corporation. This 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 operated remotely from the control room. Operator action would be required to shut this valve in the event it is needed to establish containment isolation. This valve (754A) was subjected to a type "C" test with satisfactory results. When valve 755A was last tested after its repair prior to the startup of Unit 2 in the fall of 1985, it had a leak rate of 1680 sccm. (See LER 85-002-01). As noted in the supplemental LER submitted May 19, 1986, it was anticipated that the 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.

An inspection of the old 755A check valve was performed after its removal. The disc would hang up slightly when it was manually supported and slowly lowered toward the closed position. It would hit the top of the seat first, and friction from this point of contact would hold the disc slightly away from the bottom portion of the seat. It is possible that after the system was secured as flow was coasting down, the closing motion of the disc was gentle enough that the disc would hang up in a manner similar to that manually simulated.

Another possible circumstance that could have caused the disc to hang up occurred when the system water was drained through the valve in preparation for the leak test. To simulate this, the valve and a short section of upstream and downstream piping was filled with water and then drained. Minimal movement of the disc occurred during this test and a very small amount of hangup appeared to occur. Because of the differences of the actual installed configuration and the simulated situation, it is possible that the valve response was different in the simulation than in the actual piping. In the simulation, the valve bonnet was removed and the piping system was open. During the draining of the valve for the refueling rest, which is when it failed, water slugs may have traveled through the valve when air was introduced in preparation for the leak test. An attempt was made to simulate this condition by intermittently pouring water into the upstream line as the line was draining. The disc hangup during the former simulation was much greater than the latter.

The reason for the valve disc hangup appears to be the result of too much clearance in the disc-to-hanger arm joint. This excessive clearance was due to wear. A complete disassembly of the valve revealed no other visible problems.

Two possible mechanisms which could have caused the valve hangup were investigated. Either gentle closure or the fluid dynamics of the test environment could have been the reason. In either case, we believe that during a design basis accident, the valve would have seated and limited penetration leakage.

Valve 767 was also tested and the required test pressure of 65 psig could not be achieved in the test volume. An indicated leak rate of greater than 200,000 sccm existed. However, the test volume could only be pressurized to 23 psig. The actual leakage through the valve would have been greater if the test volume could have been pressurized to 65 psig.

Valve 767 is a two-inch, 600- pound, carbon steel Rockwell Edward lift check valve, Figure 838YJ. The check valve is in the supply line to the excess letdown heat exchanger and is located inside of containment. A manual valve in the line outside of containment provides additional isolation capability, although the manual valve does not have to be leak tested in accordance with Appendix "J."

The previous leak rate test on 767 done in 1985 had an as-found leakage of 73 sccm. No repairs were done. The total Unit 2 as-left type "B" and "C" leakage in 1985 was 14,840 sccm, or 6.4% of allowable.

REPORTABILITY

This report is filed pursuant to 10 CFR 50.73(a)(2)(i), "Any operation or condition prohibited by the plant's Technical Specification."

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 of 755A, no actual backflow condition was created through the check valve as would be expected under hypothetical accident conditions. Therefore, there was 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 found during the test. Second, the CCW system is a closed system outside containment with a design pressure greater than containment accident pressure. Third, the existence of remote operating valve 754A, which passed its most recent leak rate test, allows the operator to isolate this CCW line should it become necessary to establish containment integrity.

As in the case of 755A, check valve 767 would probably limit the leak rate to a much less significant level than indicated by the test. In addition, the CCW system is a closed system outside containment with a design pressure greater than that expected during an accident inside containment. As discussed above, a manual isolation valve exists just outside containment which could be shut to provide 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 was acceptable. Leak testing of this valve prior to 1983 revealed no problems. See LERs 83-004/T-01, 84-008-00, 85-002-00, and 85-002-01.

Valve 767 has had a good history of successful type "C" tests in the past.

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

CORRECTIVE ACTIONS

755A has been replaced with a new Velan swing check valve with a slightly different disc/seat orientation. This seat is angled approximately 15 degrees from the vertical to allow gravity to assist the disc to seat properly in the valve.

Valve 767 has been repaired by disassembly and replacement of a disc O-ring and spring. The valve was returned to service with an as-left leak rate of 2 sccm.

ENERGY INDUSTRY IDENTIFICATION

The Energy Industry Identification for each of these valve is as follows:

	755A	767
System	BD	BD
Component	ISV	ISV

Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201 (414) 221-2345

VPNPD-87-062
NRC-87-016

February 10, 1987

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

Gentlemen:

DOCKET 50-301
LICENSEE EVENT REPORT 86-005-01
CONTAINMENT ISOLATION VALVE LEAK RATE
IN EXCESS OF TECHNICAL SPECIFICATIONS
POINT BEACH NUCLEAR PLANT, UNIT 2

Enclosed is Supplemental Licensee Event Report 86-005-01 for Point Beach Nuclear Plant, Unit 2. This report details the failure of two containment isolation valves to pass their Type "B" leak rate tests and the findings of our further investigation of the valve condition.

LER-86-005-01 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 Regional Administrator, Region III

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