NRC FORM 366 (7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01	Licensee Code (2)	WIPBH2
01	License Number (3)	00-00000-00
01	License Type (4)	41111
01	CAT (5)	
01 (CON'T)	Report Source (6)	L
01	Docket Number (7)	05000301
01	Event Date (8)	040983
01	Report Date (9)	071583

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 On 03/25/83 Unit 2 was shut down for its annual refueling. The annual
03 Type "B" & "C" leakage tests were performed during the outage and on
04 04/09/83 the total as-found leakage exceeded the Tech. Spec. limit of
05 0.6 La because of high leakage through one specific valve. The "A" RCP
06 CCW supply line check had leakage greater than that allowed by TS
07 15.4.4.II.B & III.B. This event is reportable IAW TS 15.6.9.2.A.3 and
08 is similar to LER's 82-006/01T and 82-020/01T.

	System Code (11)	Cause Code (12)	Cause Subcode (13)	Component Code (14)	Comp. Subcode (15)	Valve Subcode (16)
09	SD	E	B	VALVEX	C	D

LER/RO Report Number (17)	Event Year		Sequential Report No.		Occurrence Code	Report Type		Revision No.
	83	–	004	/	01	T	–	0

Action Taken (18)	Future Action (19)	Effect on Plant (20)	Shutdown Method (21)	Hours (22)	Attachment Submitted (23)	NPRD-4 Form Sub. (24)	Prime Comp. Supplier (25)	Component Manufacturer (26)
B	X	Z	Z	0000	Y	Y	N	V085

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 Valve 755A, a 4" 150 lb. carbon steel Velan check, appeared to stick
11 open during its Type "C" test. The leakage through the valve could
12 not be quantified. Followup valve maintenance did not identify the
13 cause of the failure. After maintenance, the valve leakage was 188 sccm
14 and the total "B" & "C" leakage was 14,594 sccm.

	Facility Status (28)	% Power (29)	Other Status (30)	Method of Discovery (31)	Discovery Description (32)
15	H	000	N/A	B	Surveillance testing

	Activity Released (33)	Content of Release (34)	Amount of Activity (35)	Location of Release (36)
16	Z	Z	N/A	N/A

PERSONNEL EXPOSURES

	Number (37)	Type (38)	Description (39)
17	000	Z	N/A

PERSONNEL INJURIES

	Number (40)	Description (41)
18	000	N/A

LOSS OF OR DAMAGE TO FACILITY (43)

Line	Type (42)	Description
19	Z	N/A

PUBLICITY

Line	Issued (44)	Description (45)
20	N	N/A

8307220210 830715
PDR ADOCK 05000301
S PDR

NRC USE ONLY

NAME OF PREPARER C. W. Fay PHONE: 414/277-2811

GPO 917-926

ATTACHMENT TO LICENSEE EVENT REPORT NO. 83-004/01T-0

Wisconsin Electric Power Company
Point Beach Nuclear Plant, Unit 2
Docket No. 50-301

While performing refueling leakage tests of containment isolation valves on April 9, 1983, the "A" reactor coolant pump component cooling water supply containment isolation valve (755A) was found to have leakage such that the limit in Technical Specification 15.4.4.III.B may have been exceeded. Testing personnel action during the performance of the Type "C" test prevented meaningful quantification of the as-found leakage. The unit was in a refueling shutdown at the time of the test.

During the initial phase of the Type "C" test, pressurization of the test volume to the required test pressure could not be achieved. An indicated test volume pressure of 67 psia was obtained, however, this pressure is not a true indication of the actual test volume pressure due to test line losses associated with fluid flow. The required test pressure is 60 psig (75 psia).

Upon failing to obtain test pressure, it was noted that the subject valve was leaking as evidenced by the flow of air through a test connection used to provide a leakage flow path. At this time, prior to obtaining a leakage rate reading, the testing personnel tapped on the subject valve in an attempt to seat it. The attempt was successful, as no signs of air flow through the leakage path was then evident.

Following this action, pressurization of the test volume to full test pressure still could not be achieved. Further investigation revealed that another valve (761A), used to isolate the test chamber, was open. Valve 761A should have been closed, as it allowed leakage to the atmosphere through another portion of the system.

Upon closing valve 761A, valve 755A was tested and found to have insignificant leakage (3 sccm).

Since the initial leakage rate was not quantified prior to valve "manipulation", we are assuming the leakage past valve 755A, in itself, would violate the combined Type "B" and "C" limits set forth in the Technical Specifications.

The subject valve is a four-inch, 150 psig, carbon steel, swing check valve manufactured by the Velan Corporation. This check valve is located inside containment in an incoming line with a remotely operated valve located in series outside of containment available for additional isolation via operator action.

The valve was disassembled and inspected to determine the reason for its failure to close. No problems were identified during the maintenance. The valve was cleaned, reassembled, and retested. The retest of the valve resulted in a leakage rate of 188 sccm.

The total Type "B" and "C" leakage testing program for the Unit 2 refueling 9 was completed on June 29, 1983. The total as-found leakage, excluding the leakage through valve 755A, would have been 92,752 sccm or 40.1% of allowable. The total leakage after maintenance of 755A, and several other valves with greater than desirable leakage, was 14,594 sccm or 6.3% of allowable.

This event is reportable in accordance with Technical Specification 15.6.9.2.A.3, "Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment." The Resident Inspector has been notified of this event.

Since high leakage through this valve has been the subject of previous Licensee Event Reports, a review of the valve's leakage history and an evaluation of additional corrective action will be performed.

Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

July 15, 1983

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 NO. 50-301
LICENSEE EVENT REPORT NO. 83-004/01T-0
POINT BEACH NUCLEAR PLANT, UNIT 2

Enclosed is Licensee Event Report No. 83-004/01T-0 (a 14-day follow-up report) with an attachment which provides a description of an event reportable in accordance with Technical Specification 15.6.9.2.A.3, "Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment." The initiating event for this report occurred on April 9, 1983, however, submittal of this report was delayed until all Type "B" and "C" testing data were available. The Type "B" and "C" testing was completed on June 29, 1983.

Very truly yours,

CW Fay

Vice President-Nuclear Power

C. W. Fay

Enclosure

Copy to NRC Resident Inspector

JUL 18 1983

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