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, Unit 2 | 0 5 0 0 0 3 0 1 | 1 OF 0 3 |

TITLE (4)
Containment Isolation Valve Leakage in Excess of Technical Specifications

| 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) |
| | | | | | | | | | None | 0 5 0 0 0 |
| 1 0 | 0 5 | 8 4 | 8 4 | 0 0 8 | 0 0 | 1 2 | 1 1 | 8 4 | None | 0 5 0 0 0 |

| OPERATING MODE (9) | N | THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11) | | | |
|---|---|---|---|---|---|
| POWER LEVEL (10) | 0 0 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 |
|---|---|
| C. W. Fay, Vice President - Nuclear Power | AREA CODE 4 1 4 2 7 7 - 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) | | 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 09/28/84, Unit 2 was removed from service for its tenth refueling. Type "B" and "C" local leakage tests were performed during the outage and on 10/05/84. the total as-found leakage exceeded the Technical Specification limit of 0.6 La because of high leakage through one specific valve. The "A" reactor coolant pump component cooling water supply line check valve (755A) had leakage greater than that allowed by Technical Specifications 15.4.4.II.B & III.B.

Valve 755A, a 4-inch, 150 lb, carbon steel, Velan, swing check, appeared to stick open during the initial Type "C" test. The required test pressure could not be achieved, thus, the leakage could not be quantified.

Followup maintenance revealed that one of the disc's hinge pin bushings had migrated toward the disc's swing arm resulting in the valve disc intermittently sticking. The subject valve was repaired and satisfactorily retested.




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

While performing refueling leakage tests of containment isolation valves on October 5, 1984, 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 was exceeded. The high leakage through the subject valve was solely responsible for violating the Technical Specification limit. Test equipment limitations 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 38 psia was obtained. The associated indicated flow rate was 136 slpm. 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, the testing personnel tapped on the subject valve in an attempt to seat it. The attempt was successful as pressurization of the test volume to 80.7 psia was achieved. The associated leakage flow was insignificant (5 sccm).

It should be noted that the "premanipulation" indicated pressure and flow are not a true representation of the test volume parameters due to significant test line losses associated with high flow rates. Thus, in view of the "premanipulation" test findings, and that accurate quantification of the leakage flow is not possible, we are assuming that the leakage past the subject valve, in itself, would violate the combined Type "B" and "C" limits set forth in the Technical Specifications.

The subject valve is a 4-inch, 150 psig, carbon steel, swing check valve manufactured by the Velan Corporation. This check valve is located inside containment in an incoming component cooling water line. Additional isolation capability was available, via operator action, with a remotely operated valve located in series to 755A, but outside of containment. This secondary valve was also subjected to a Type "C" test, with satisfactory results. In addition, a third barrier to containment leakage through this penetration was available. This barrier is provided by the component cooling water system itself, which is a closed system outside of containment.

Followup maintenance on valve 755A revealed that one of the disc's hinge pin bushings had migrated toward the disc's swing arm resulting in the valve disc intermittently sticking. The cause of the bushing migration is not known. The subject valve was cleaned, lapped, repaired, reassembled and retested. The retest of the valve resulted in a leakage rate of 449 sccm.

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

The total Type "B" and "C" leakage testing program for the Unit 2 Refueling 10 was completed on 11/14/84. The total as-found leakage, excluding the leakage through valve 755A, was 18,424 sccm or 8.0 percent of allowable. The total as-left leakage, following the Unit 2 Refueling 10 outage, was 15,188 sccm or 6.6 percent of allowable.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(i)(B), "Any operation or condition prohibited by the plant's Technical Specifications." The Resident Inspector has been notified of this event.

This particular valve has been the subject of previous Licensee Event Reports associated with similar leakage failures. During the previous events, we were not able to identify a cause for the failures. In Licensee Event Report 83-004/01T-0, WE acknowledged that the valve's performance history was not satisfactory and identified that additional evaluations and corrective actions would be performed. In view that detailed maintenance inspection/investigation revealed and corrected a definite mechanical problem, no further corrective action is planned.

DMB

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

December 11, 1984

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. 84-008-00
CONTAINMENT ISOLATION VALVE LEAKAGE IN EXCESS
OF TECHNICAL SPECIFICATIONS
POINT BEACH NUCLEAR PLANT, UNIT 2

Enclosed is Licensee Event Report No. 84-008-00 which provides a description of an event reportable in accordance with 10 CFR 50.73(a)(2)(i)(B), "Any operation or condition prohibited by the plant's Technical Specifications." The initiating event for this report occurred on October 5, 1984, 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 November 14, 1984.

Very truly yours,

C. W. Fay

Vice President-Nuclear Power

C. W. Fay

Enclosure

Copy to NRC Resident Inspector

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