DCP

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NRC-91- 048

10 CFR 50.73

May 21, 1991

U. S. NUCLEAR REGULATORY COMMISSION
Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Gentlemen:

DOCKETS 50-266
LICENSEE EVENT REPORT 91-003-00
CONTAINMENT ISOLATION VALVE LEAKAGE IN EXCESS
OF TECHNICAL SPECIFICATION LIMITS
POINT BEACH NUCLEAR PLANT, UNIT 1

Enclosed is Licensee Event Report 91-003-00 for Point Beach Nuclear Plant, Unit 1. This report is provided in accordance with 10 CFR 50.73 (a)(2)(i), "Any operation or condition prohibited by the plant's Technical Specifications."

This report details the failure of a containment isolation valve to pass local leak rate tests at a level required by Technical Specifications.

If further information is required, please contact us.

Very truly yours,

C. W. Fay
Vice President
Nuclear Power

Enclosure

Copies to NRC Regional Administrator, Region III
NRC Resid~nt Inspector

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MAY 23 1991

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): POINT BEACH NUCLEAR PLANT

DOCKET NUMBER (2): 0 5 0 0 0 2 6 6

PAGE (3): 1 OF 0 4

TITLE (4): CONTAINMENT ISOLATION VALVE LEAKAGE IN EXCESS OF TECH SPEC LIMITS

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 4	2 9	9 1	9 1	0 0 3	0 0	0 5	2 1	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

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: C. W. Fay, Vice President, Nuclear Power

TELEPHONE NUMBER: AREA CODE 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)

ABSTRACT:

On April 29, 1991, during Appendix J, Type C containment leak rate testing, charging system check valve 1-370 was discovered with leakage in excess of limits cited in Technical Specifications 15.4.4.II.B and 15.4.4.III.B. In this case, the required test pressure could not be achieved; therefore, a leak for design basis conditions could not be quantified.

The valve was repaired, tested and restored to operation. This report is filed pursuant to 10 CFR 50.73(a)(2)(i), "Any operation or condition prohibited by the plant's Technical Specifications."

EVENT DESCRIPTION:

On April 6, 1991, Unit 1 was shut down for annual refueling and maintenance outage No. 18. On April 29, 1991, excessive leakage was discovered through the charging system regenerative heat exchanger supply check valve (1-370). The leakage past valve 1-370 appeared to exceed the leak rate limit of 0.6 La, the maximum leakage allowed by Technical Specification 15.4.4.II.B and 15.4.4.III.B.

Valve 1-370 was tested according to Operations Refueling Test 34. As the piping under test reached 20 psig, the leakage was quantified at 227,000 sccm. It is conservatively assumed that the leakage would have been greater had the required test pressure of 65 psig been achieved.

Leakage from all remaining valves tested within Technical Specification limits. Excluding valve 1-370, the total "as-found" leakage for Type "B" and "C" local leak rate testing amounted to 6924 sccm or 3% of the Technical Specification allowance.

COMPONENT AND SYSTEM DESCRIPTIONS:

Valve 1-370 is a 3-inch, 1500 pound class, Gr 316 stainless steel, swing check valve. It is a Model B10-4114C-11NB, Drawing 78409 and was manufactured by Velan Engineering Companies. It is located inside of containment in the charging supply line to the regenerative heat exchanger. It has been in service since commercial operation began in December 1970. Piping on the supply and discharge sides of the check valve is stainless steel and has a design rating of 2500 psig at 130°F. Additional isolation capability for this piping is provided by either remotely operated valves 1-1298 and 1-296 inside of containment, or manually operated valves 1-384B and 1-323B outside of containment. Operator action would be required to shut each of these valves if they were needed to establish containment isolation.

CAUSES AND CORRECTIVE ACTIONS:

Engineering and Maintenance personnel conducted a detailed root cause analysis. The results of the analysis indicated the disc arm bushing steps had worn during the course of normal operation. The wear allowed binding at the hinge connection which prevented disc arm movement and caused the disc to remain open. There was no notable wear on the hanger bracket bushing steps.

Corrective action consisted of cladding the worn disc arm bushing steps with E308 weld material and hand filing the steps to their design contour, a repair method that was reviewed and approved by the valve manufacturer's engineering department. The weld integrity was verified by liquid penetrant examination. The valve was reassembled and tested. Post maintenance leakage was found to be 32 standard cubic centimeters per minute. The check valve was restored to operation May 3, 1991.

In addition, preventive maintenance inspections for this valve and the corresponding valve in Unit 2 are planned at 5 year intervals to detect and correct any future degradation.

GENERIC IMPLICATIONS:

There have been no generic implications identified at this time. Point Beach Unit 2 has an identical component and application. Operations testing of this component will be conducted during the refueling outage scheduled to commence September 27, 1991.

REPORTABILITY:

The licensee event 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 and system names of each valve and system referred to in this LER are:

Valve No.: 1-370
System: BD
Component: ISV

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
POINT BEACH NUCLEAR PLANT	0 5 0 0 0 2 6 6	9 1	— 0 0 3 —	0 0	0 4 OF 0 4

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 significant safety hazard to the general public or to the employees of the Point Beach Nuclear Plant. Alternate means of remotely isolating the piping system inside containment was available to operators. The capability of local, manual isolation outside containment also exists.

SIMILAR OCCURRENCES:

Investigations of plant maintenance, the Nuclear Plant Reliability Database (NPRDS) and past LERs have revealed that valve 1-370 had three previous failures. The most recent occurred in April 1989 and contributed to the total as-found leakage cited in LER 266-89-002. One other involving leakage past valve 1-370 is described in LER 266-83-009. The remaining Unit 1 failure involved a violation of plant leakage administrative limits and is the subject of an NPRDS report filed in September 1978.

Similar occurrences in Unit 2 included a single routine corrective maintenance action in response to leakage in excess of plant administrative limits. This event is also the subject of an NPRDS report filed in November 1990.