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VPNPD-91-394 10 CFR 50.73
NRC-91-133

November 15, 1991

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

Gentlemen:

DOCKET 50-301
LICENSEE EVENT REPORT 91-004-00
CONTAINMENT ISOLATION VALVE LEAKAGE
IN EXCESS OF TECHNICAL SPECIFICATION LIMITS
POINT BEACH NUCLEAR PLANT, UNIT 2

Enclosed is Licensee Event Report 91-004-00 for Point Beach Nuclear Plant, Unit 2. 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 containment isolation valve CC-767 to pass the seat leakage tests required by 10 CFR 50, Appendix J for containment isolation valves. Valves RC-595 and SC-966B are also discussed in this report as a result of leakage exceeding the administrative limit established at Point Beach Nuclear Plant. Total Type C leakage exceeded the Technical Specification limit of sixty percent of the maximum allowable leak rate (0.6La).

In order to allow a more complete description of the three separate events to be performed, the NRC Resident Inspector stated that the thirty-day time requirement for the submittal of this Licensee Event Report could be extended.

Please contact us if there are any questions.

Very truly yours,

James Zach

James J. Zach
Vice President
Nuclear Power

Enclosure

Copies to NRC Regional Administrator, Region III
NRC Resident Inspector

IE22
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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)	DOCKET NUMBER (2)	PAGE (3)
Point Beach Nuclear Plant, Unit 2	0 5 0 0 0 3 0 1	1 OF 0 6

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 8	9 1	9 1	0 0 4	0 0	1 1	1 5	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	TELEPHONE NUMBER (AREA CODE)
Norman L. Hoefert, Manager - Operations	4 1 4 7 5 5 - 2 3 2 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	C C	I S V	X 9 9 9	Y	X	B D	I S V	M 1 2 6	Y
X	B D	I S V	G 2 5 7	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 October 1, 1991, during the Unit 2 refueling shutdown, Operations Refueling Test (ORT) 67, "Component Cooling Water to and from the Excess Letdown Heat Exchanger, Unit 2," was in progress. This test, as well as the other tests discussed in this report, was being performed to meet the requirements of 10 CFR 50, Appendix J for Type C containment isolation valves. During the initial performance of this seat leakage test, check valve CC-767 failed to shut. Because the valve failed to shut, the required test pressure could not be achieved; therefore, the as-found leak rate could not be quantified. A second attempt was made to pressurize this line. This time the line pressurized to the required pressure. The leak rate measured during this second attempt was within the limits of Technical Specifications Section 15.4.4.III.B.

EVENT DESCRIPTION

On October 1, 1991, Operations Refueling Test (ORT) 67, "Component Cooling Water to and from the Excess Letdown Heat Exchanger, Unit 2," was being performed during the Unit 2 refueling shutdown. During the performance of the seat leakage test on CC-767, the required test pressure of 65 psig could not be achieved. The maximum pressure achieved was 28.9 psig. This indicated that the check valve did not shut as required.

Following the initial test failure, an operator disconnected the low volume test rig, hooked up a high volume test rig and attempted the test a second time. During this attempt, the line initially pressurized to only 28 psig but then subsequently increased to 65 psig. This indicated that the check valve had shut. The leak rate measured at this time was 4 standard cubic centimeters per minute (sccm). This is below the applicable Technical Specification and Point Beach Nuclear Plant administrative limits.

The information on the as-found leakage and corrective actions on the following two valves is not reportable but is being presented as information.

On October 8, 1991, OPT 31, "Nitrogen Supply to the Pressurizer Relief Tank, Unit 2," was being performed during the Unit 2 refueling shutdown. During performance of the seat leakage test on RC-595, an as-found leak rate of 9730 standard cubic centimeters per minute (sccm) was measured. This is in excess of the administrative limits established at Point Beach Nuclear Plant but less than the limits specified in Technical Specification Section 15.4.4.III.B. An inspection of the valve was performed, the problem was corrected, and the test was satisfactorily completed with no leakage detected.

On October 17, 1991, ORT 36, "Pressurizer Liquid Sample Line, Unit 2," was being performed. During the performance of the seat leakage test on SC-966B, an as-found leak rate of 2520 sccm was measured. This leak rate is in excess of the administrative limits but less than the limits specified in Technical Specification Section 15.4.4.III.B. An inspection of the valve was performed, the problem was corrected, and the post-maintenance testing was completed satisfactorily.

COMPONENT AND SYSTEM DESCRIPTION

Valve CC-767 is a two inch, carbon steel, lift check valve with a dual seat manufactured by the Rockwell-Edwards Corporation. This valve is rated for a design pressure of 600 psig. This check valve is located inside containment on the component cooling water supply line to the shell side of the excess letdown heat exchanger. Further isolation of this supply line can be provided by shutting manual valve CC-865A inside containment and by shutting manual valve CC-766 outside containment.

Valve RC-595 is a 3/4 inch, 304 stainless steel, diaphragm valve manufactured by Grinnell Industrial Piping Incorporated. The valve has a design pressure of 150 psig. RC-595 is located outside containment in the nitrogen supply line to the Pressurizer Relief Tank. Additional isolation capability for this supply line is provided by RC-441 (a remotely operated valve) and NG-1662 (a manual valve), both located outside containment; and by manual valve RC-532 which is located inside containment.

Valve SC-966B is a 3/8 inch, 316 stainless steel, pneumatically operated globe valve manufactured by Masoneilan International Inc. The valve has a design pressure of 1500 psig. SC-966B is located outside containment in the pressurizer liquid sample line. Additional isolation capability for this line is provided by SC-953, a remotely operated valve located inside containment, and by SC-956B, a manual valve located outside containment.

The Energy Industry Identification System component function identifier, system, and manufacturer for these valves are as follows:

Valve:	CC-767	RC-595	SC-966B
Component:	ISV	ISV	ISV
System:	CC	BD	BD
Manufacturer:	X999	G257	M126

CAUSE AND CORRECTIVE ACTION

The test of CC-767 initially failed because the check valve did not shut. The valve shut during the second attempt while using the high volume test rig. The leak rate measured during this second attempt was below both the administrative and Technical Specification limits.

A maintenance work request was initiated to inspect and clean the internals of CC-767. The inspection revealed some dirt on the valve seating surface. This dirt was removed and a new O-ring was installed in the valve. The post-maintenance testing was performed, but the leakage exceeded the administrative limit of 2000 sccm. The valve was inspected a second time, and the new O-ring was found to be torn. This tear is believed to be a result of a manufacturing defect. On October 25, 1991, another O-ring was installed, and the inspection was completed. The post-maintenance testing was completed satisfactorily on November 4, 1991.

Following the initial test failure of RC-595, a thorough inspection of the valve was performed. This inspection revealed that the stem travel jack nut on the valve was too tight. This prevented the valve from shutting fully, resulting in excessive seat leakage. The jack nut was loosened and the test was completed satisfactorily. The stem travel jack nut is installed on the valve to prevent the valve from being shut with an excessive amount of force. This ensures that the diaphragm inside the valve body does not get damaged. A procedure currently in place details the method for properly setting the height of the jack nut following valve maintenance or installation.

This procedure adequately ensures that the jack nut height is properly set. It is still possible for this nut to move and thus prevent the valve from functioning properly. This movement may be caused by excessive vibration or by accidental operator contact with the jack nut during valve manipulation. To prevent this from occurring in the future, a maintenance work request has been initiated which calls for the installation of a lock nut on the valve that will prevent the stem travel jack nut from moving once its position has been properly set.

A review of the maintenance history of valve SC-966B indicated that this valve has had a history of leakage problems. This leakage has been mainly caused by problems with the valve seat. Several attempts to repair the valve seat have been made since 1989. An inspection of the valve following the test failure revealed a crack in the valve body starting below the valve seat and extending to the inlet side of the valve.

In order to correct the immediate problem, a new valve body manufactured by Masoneilan International Inc. was installed, replacing the cracked body. This installation took place on November 8, 1991, and was immediately followed by the successful completion of the post-maintenance testing.

As a long-term corrective measure, a modification request has been initiated to replace the entire valve. We intend to use a Valtec, Mark II, 3/8 inch, 316 stainless steel, globe valve with a design pressure of 1500 psig. This valve is similar to valve SC-953, the pressurizer liquid sample isolation valve, and SC-955, the Reactor Coolant System hot leg sample isolation valve. These two valves were installed in 198_ and neither has exhibited any seat leakage problems. This modification request is scheduled for completion during the 1992 Unit 2 refueling outage.

REPORTABILITY

This Licensee Event Report is filed in accordance with 10 CFR 50.73(a)(2)(i), "Any operation or condition prohibited by the plant's Technical Specifications." Since the as-found condition of CC-767 precluded a quantification of leakage, total leakage measured by the Type C test would have exceeded sixty percent of the maximum allowable leak rate (0.6La).

Because this Licensee Event Report discusses three events (the first of which was determined to be reportable) that occurred over a span of three weeks, we asked the NRC Resident Inspector if we could submit the LER within thirty days of the last event rather than the reportable event date. This extension allowed a complete description of the actions taken for all three valves (both the reportable one and the two non-reportable) to be discussed in a comprehensive report.

SAFETY ASSESSMENT

There are no safety consequences from this event. The corrective actions for valves CC-767 and RC-595 have been completed. Although not strictly required for compliance with Technical Specifications, a modification request is in progress to replace SC-966B. In addition to the affected valves, several other means of containment isolation are available to isolate the component cooling water supply line to the excess letdown heat exchanger, the nitrogen supply line to the Pressurizer Relief Tank, and the pressurizer liquid space sample line. The health and safety of plant personnel and the public were not endangered.

SIMILAR OCCURRENCES

A review of Licensee Event Reports was performed. Several LERs discussing valve CC-767, or valves similar to CC-767, were identified. One LER concerning a valve similar to RC-595 was also found. No LERs were located that concerned valve SC-966B.

LER 86-005-01 for Unit 2 concerns the excessive leakage of valve CC-767. Following the unsuccessful completion of ORT 67, the valve was disassembled and repaired by replacing the disc O-ring, spring, and flex gasket. The test was then completed satisfactorily.

LER 87-003-00 for Unit 1 concerns excessive leakage of valve NG-1713, an inlet check valve on the nitrogen supply line to the vent header of the Reactor Coolant Drain Tank. This valve is a one inch, 600 psig, carbon steel, lift check valve manufactured by the Rockwell-Edwards Corporation. The Rockwell-Edwards Corporation also manufactured valve CC-767. The leakage was caused by foreign material found in the piping, preventing the valve from properly seating.

LER 88-004 for Unit 1 concerns the excessive leakage of RC-528, the nitrogen supply check valve to the Pressurizer Relief Tank. This is a two inch, 600 psig, carbon steel, lift check valve manufactured by the Rockwell-Edwards Corporation. Inspection of the valve internals following the test failure revealed rust particles on the valve seat that prevented the valve from properly seating.

LER 87-006 for Unit 2 concerns valve IA-3048, a Fisher Governor air-operated containment isolation valve located in the instrument air supply line. This LER was written because of the suspected lack of operability of this valve due to a partially engaged handwheel gag. Having the gag in this position would have prevented it from fully shutting.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Point Beach Nuclear Plant, Unit 2	0 5 0 0 0 3 0 1	9 1	— 0 0 4	— 0 0	0 6 OF 0 6

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

A search was also conducted of the Nuclear Plant Reliability Database System (NPRDS). No entries concerning the three valves were identified.