NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

January 14, 2005

NRC 2005-0010
10 CFR 50.73

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
License Nos. DPR-24 and DPR-27

Licensee Event Report 301/2004-003-00
Reactor Shutdown Required by Containment Penetration TSAC 3.6.3.C

Enclosed is Licensee Event Report (LER) 301/2004-003-00 for the Point Beach Nuclear Plant Unit 2. This LER describes the discovery of a pinhole steam leak in the body of a steam flow transmitter root valve which required the unit to be shutdown and cooled down for repairs. This condition is reportable in accordance with 10 CFR 50.73(a)(2)(i)(A).

This letter contains no new commitments and no revisions to existing commitments.

Dennis L. Koehl
Site Vice-President, Point Beach Nuclear Plant
Nuclear Management Company, LLC

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC

IE22

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NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

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LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
POINT BEACH NUCLEAR PLANT UNIT 2	05000301	1 of 5

TITLE (4)

REACTOR SHUTDOWN REQUIRED BY CONTAINMENT PENETRATION TSAC 3.6.3.C

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Not Applicable	
11	19	2004	2004	-- 003 --	00	01	14	2005	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR .: (Check all that apply) (11)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)	X	50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
C. W. Krause, Senior Regulatory Compliance Engineer	920-755-7656

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	SB	V	V135	Y					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT

On November 19, 2004, a pinhole steam leak was discovered inside containment on the body of valve 2MS-465D, the first-off low side root isolation valve for the Point Beach Nuclear Plant Unit 2 "A" steam generator steam line flow transmitter. The leak location could not be isolated from the main steam line. The affected portion of the main steam line is credited as a closed system boundary for containment penetration P-1, the "A" main steam line penetration. Accordingly, Technical Specification (TS) Condition 3.6.3.C, which is applicable to penetration flow paths with only one containment isolation valve and a closed system, was entered and Required Action C.1 was initiated. The required action for this condition is to isolate the affected penetration flow path with a completion time of 72 hours. The repair of the leak would require a shutdown and cooldown of Unit 2 and could not be accomplished within that 72 hour time frame; therefore, a reactor shutdown and cooldown was promptly initiated. This was considered to be a TS required shutdown and was reported in accordance with 10 CFR 50.72.(b)(2)(i). Unit 2 entered Mode 5 on November 21, 2004, and the valve body leak was repaired. Unit 2 returned to full power operation on November 25, 2004. The cause of the leak was determined to be either a material defect in the valve body or an improper welding process, which provided a stress riser that failed due to cyclic loading. An assessment of the safety significance of this event concluded that the impact of this event on the health and safety of the public was not significant.

Event Description: On November 19, 2004, the Point Beach Nuclear Plant (PBNP), Unit 2, was operating at full power. A Unit 2 containment [EIIS System Code: NH] entry was conducted on that date by Nuclear Management Company, LLC (NMC) personnel. The purpose of that entry was to inspect the unit in an attempt to identify the source of a suspected secondary system steam leak inside the containment. The possibility of a fluid leak in containment had been initially detected earlier in October 2004, based on having to pump the Unit 2 containment "A" sump [EIIS Component Code: DRN] (volume of approximately 44 gallons) as often as once per day (CAP 60247). A sample of the sump on October 27, 2004, indicated that the leakage did not appear to be primary coolant, service water, or component cooling water, but was consistent with condensation. An earlier containment entry on November 8, 2004, was not successful in identifying the source of the leakage.

During the containment inspection on November 19, 2004, operations personnel located a source of sump "A" leakage when a pinhole leak was found on the body of valve 2MS-465D [EIIS Component Code: RTV]. This valve is the first-off low side root isolation valve for the PBNP Unit 2 "A" steam generator [EIIS Component Code: SG] steam line flow transmitter [EIIS Component Code: FT], 2FT-465. The inspection also revealed a packing leak on valve 2MS-465B, which is the second-off high side root isolation of 2FT-465.

Due to the leak location on the valve body of 2MS-465D, NMC determined that the leakage could not be isolated from the "A" steam generator main steam [EIIS System Code: SB] line. For containment isolation considerations, the main steam system is considered to be a closed system inside containment. Since the leak could not be isolated from the containment penetration [EIIS Component Code: PEN] for the "A" main steam line, the main steam system inside containment was no longer considered to be a closed system and was declared out of service at 15:44 (all times are CST) on November 19, 2004.

Technical Specification LCO 3.6.3 states that "Each containment isolation valve shall be operable." The modes of applicability for this LCO are 1 through 4. Technical Specification Action Condition (TSAC) 3.6.3.C is specifically applicable to penetration flow paths having only one containment isolation valves and a closed system. TSAC 3.6.3.C was entered for "One or more penetration flow paths with one containment isolation valve inoperable". Since the Unit 2 "A" main steam line inside containment was declared out of service and was no longer considered to be a closed system, Required Action C.1 was applicable. This action directs that the affected penetration flow path be isolated with a completion time of 72 hours. Since the repair or replacement of the 2MS-465D valve would require a shutdown and cooldown of PBNP Unit 2, NMC concluded that the 72 hours completion time for required action 3.6.3.C.1 could not be met. TSAC 3.6.3.D would then be applicable and directs that the unit be shut down to Mode 3 within 6 hours and Mode 5 within 36 hours of the 72-hour completion time of required action 3.6.3.C.1.

At 16:22, a Unit 2 load reduction was initiated in accordance with plant procedures from power operations to hot standby. This normal reactor shutdown was considered to be a Technical Specification required shutdown. At 18:01 a four-hour Emergency Notification System report was made to the NRC in accordance with 10 CFR 50.72(b)(2)(i) (EN# 41212). The NRC senior resident inspector had been notified at 15:54 on November 19, 2004, of the planned shutdown.

The shutdown was completed with no significant problems noted. Unit 2 entered Mode 3 at 21:10 on November 19, 2004, and Mode 5 at 09:22 on November 21, 2004.

Following the repair of valve 2MS-465D (See discussion under Corrective Action) Unit 2 was returned to service. Unit 2 entered Mode 1 at 09:07 on November 24, 2004, and full power was achieved for Unit 2 at 07:00 on November 25, 2004.

Safety Significance:

An assessment of the safety significance for the containment pressure boundary leak was performed shortly after identification of the steam leak on 2MS-465D. Prior to discovery of the steam leak, the containment sump "A", which has a volume of about 40 gallons, was being pumped less than once per day. Assuming that the condensation in the sump was coming from the steam leak alone, the leakage from the valve body hole was assumed to be bounded by approximately 40 gallons of water per day. Using a factor of 1,000 to account for the steam volume being added to containment (1,000:1 expansion ratio for water boiling at atmospheric pressure), this equates to approximately.40,000 gallons per day of steam leakage or about 5,350 cubic feet per day. This leakage was occurring with a differential pressure of approximately 800 psid.

Under accident conditions, the design pressure of the containment building (60 psid) is not exceeded in a loss of coolant accident (LOCA), and therefore presents an upper limit on the potential leakage through the closed system containment boundary into the main steam system. To adjust the flow rate for comparing this lower differential pressure to the higher differential pressure of the observed leakage, the square-root relationship was used to give an adjustment factor of 0.274. Multiplying the leakage by this factor gives an estimated potential steam leakage through the closed system inside containment boundary following a design basis accident of 1,460 cubic feet per day.

Using the containment net free volume 970,000 cubic feet, the estimated leakage would equate to 0.15%/day containment leakage. This is less than the FSAR containment design limit of 0.4%/day and our currently committed administrative limit of 0.2%/day. Based on this assessment, the leakage through the valve wall was bounded by the existing accident analyses.

Under a LOCA scenario, the steam generators remain at a much higher pressure than the containment pressure until well into the event. Therefore, leakage from containment out into the main steam (MS) system would be significantly delayed and not occur at the onset of the accident. Additionally, the MS system is not directly vented to the environment at that point in the accident scenario. Any steam releases at that time would be via steam dumps or safety valves. Steam dumps and safety valves are not effective or used with steam pressures below 60 psig, and residual heat removal [EIIS System Code: BP] is used to remove decay heat at these low pressures (even in the event of a LOCA). In aggregate, the increase in risk of a radiological release due to the leak was deemed minimal. Accordingly, the potential impact of this event on the health and safety of the public and the plant staff was considered to be insignificant.

NMC further concludes that the through-wall steam leak, which prompted this Technical Specification required shutdown, did not constitute a loss of any system, structure or component safety function; therefore, the degradation of this containment closed system boundary did not constitute a safety system functional failure.

Cause:

A review of the component history of the valve revealed that the instrumentation line off the main steam header including the two isolation valves and condensing pot had previously been identified as a location of vibration (CR 97-2440). This main steam flow instrument line was not supported as were all other similar main steam flow indication lines. An evaluation of the vibration had been performed and the fatigue stress determined to be acceptable. Although fatigue failure was initially suspected, the defect on 2MS-465D valve was in the valve body in the heat affected zone of the weld. A dye penetrant test (PT) revealed a horse-shoe shaped indication in the valve body at the socket weld location suggesting that this failure was the result of a manufacturing defect in the valve body and not a fatigue failure. Excavation of the indication during repair resulted in removal of the weld and valve body in the socket location down to the pipe material. After the horse-shoe shaped indication was removed, another PT exam did not reveal any additional circumferential cracking, which would be expected if fatigue was the mode of failure. The horse-shoe shape of the PT examinations indicates the existence of a possible stress riser due to a material defect either from the valve body or the welding process used during original installation. The manufacturing defect was likely opened by the stress of repeated transients. Had the defect not been present, a failure would not be expected. Because of the repair technique used, destructive testing and material examination for further analysis of the failure was not possible.

Corrective Action:

Repairs were made to the affected area by excavation and welding the affected area. PT examinations of the surface were conducted during the grinding to ensure that the flaw had been completely removed. The affected area was then repaired by laying in new weld material. A PT of the final weld was performed and documented as satisfactory. A final successful leak check of the weld was completed with the unit at normal operating temperature and pressure. The packing leak on valve 2MS-465B was also corrected.

The other accessible welds on the main steam header instrumentation lines for Unit 2 were PT examined. No other weld defects were noted. In addition, all the welds on the 2FT-465 instrumentation line were examined with no additional indication identified. NMC is also evaluating a modification to provide a support for the existing condensing pot similar to the supports provided on the other main steam flow instrument lines.

Previous Similar Events:

A review of recent LERs (past three years) identified two events which involved a reactor shutdown required by the Technical Specifications and one event involving a defect in a pressure boundary:

LER NUMBER	Title
301/2002-001-00	Completion of Unit 2 Shutdown Required by TS LCO 3.5.2 Action B.1
301/2003-002-00	Reactor Shutdown Required Due To Technical Specification TSAC 3.1.6.B.2 Not Met
266/2004-001-00	Reactor Vessel Upper Head CRDM Penetration 26 Flaw Indications