Wisconsin Electric
POWER COMPANY

Point Beach Nuclear Plant
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(920) 755-2321

NPL 99-0707

December 2, 1999

10 CFR 50.73

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

Ladies/Gentlemen:

DOCKET NO 50-266
LICENSEE EVENT REPORT 1999-012-00
THROUGH WALL DEFECT ON STEAM
GENERATOR DRAIN VALVE WELD
POINT BEACH NUCLEAR PLANT, UNIT 1

Enclosed is Licensee Event Report 266/1999-012-00 for the Point Beach Nuclear Plant (PBNP), Unit 1. This report is provided in accordance with 10 CFR 50.73(a)(2)(ii)(B) as "any event or condition that resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded." This report describes the discovery of a through wall weld defect on the channel head drain isolation valve for the Unit 1 "A" Steam Generator. The weld has been replaced.

New commitments within this report are identified under the heading of "Corrective Actions" by italics.

Please contact us if you require additional information.

Sincerely,

A. J. Cayia

A. J. Cayia
Manager,
Regulatory Services & Licensing

CWK/tat

IE22

Enclosure

cc: NRC Resident Inspector
NRC Regional Administrator
NRC Project Manager
PSCW
INPO Support Services

NRC FORM 366 (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT

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 1	05000266	1 of 3

TITLE (4)
Through Wall Defect On Steam Generator Drain Valve Weld

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 NAME	DOCKET NUMBER 05000
11	04	1999	1999	- 012 -	00	12	02	1999	FACILITY NAME	DOCKET NUMBER 05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

OPERATING MODE (9)	N		20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	000		20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Charles Wm. Krause, Senior Regulatory Compliance Engineer	(920) 755-6809

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	AB	ISV-Weld	W120	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 15 single-spaced typewritten lines) (16)

This report discusses the discovery of an approximately 1/64 inch through-wall defect or flaw on the upstream weld for valve 1RC-526A, the isolation valve for the Unit 1 "A" steam generator channel head drain. This indication was discovered while conducting an informational liquid dye penetrant examination of that weld due to the visual identification of boric acid crystals on the weld. The unit was shutdown at the time of this discovery. A four hour non-emergency event notification was made to the NRC at 0307 CST on November 4, 1999, in accordance with 10 CFR 50.72(b)(2)(i) for an event discovered while the plant was shutdown that, had it been discovered while the reactor was operating, would have resulted in the principal safety barriers being seriously degraded. The weld has been replaced.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Event Description:

While conducting a liquid dye penetrant (LP) examination of the upstream weld on valve 1RC-526A, Unit 1 "A" Steam Generator Channel Head Drain Isolation Valve, licensee's NDE inspectors discovered an approximately 1/64 inch long crack indication in the weld metal (CR 99-2754). 1RC-526A is a 3/8 inch globe valve on the pipe which is used to drain the steam generator primary side channel head. Subsequent examination of this indication revealed that the crack was through-wall and was confined to the weld metal material. No flaws were noted in the valve or the base metal of the drain line piping. This indication was in a location that is non-isolable from the reactor coolant system.

At the time of discovery, Point Beach Nuclear Plant (PBNP) Unit 1 was in the sixteenth day of a refueling shutdown. A four hour non-emergency event notification was made to the NRC at 0307 CST on November 4, 1999, in accordance with 10 CFR 50.72(b)(2)(i) for an event discovered while the plant was shutdown that, had it been discovered while the reactor was operating, would have resulted in the principal safety barriers being seriously degraded. This report is provided in accordance with 10 CFR 50.73(a)(2)(ii) and the guidance in NUREG 1022, which identifies defects which cannot be found acceptable under ASME Section XI as reportable.

Upon a review of the maintenance history for the channel head drain valves, the system engineer noted that the same weld on the channel head drain line for the Unit 1 "B" steam generator was repaired in July 1990 (LER 266/90-008-00). At that time a NDE of the 1RC-526A valve weld was conducted with negative results. The PBNP Unit 2 steam generators do not have these valves or channel head drain piping.

An examination of recent Unit 1 operating records revealed that on October 8, 1999, an increase of unidentified RCS leakage had been observed (CR 99-2374). At that time total Unit 1 RCS leakage was measured as 0.2 gpm with known leakage of approximately 0.1 gpm. Steps were taken to identify the source, or sources, of the unidentified leakage, including containment entry and system walkdowns. These efforts were not conclusive in identification of any new leakage sources. However, given the presence of a minor buildup of boric acid crystals in the area of the valve and the through-wall nature of this weld defect, it is possible that this defect was the source of the unidentified RCS leakage. In that case, PBNP Unit 1 operated for some indeterminate time with a non-isolable RCS leak.

Cause:

The cause of this weld defect has not been identified. The defect is assumed to have originated with the original weld installation of this valve. The fabrication of this drain piping and isolation valve was performed during the Unit 1 steam generator lower assembly replacement in 1983-1984. The new Unit 1 steam generators were placed in service in April 1984. In 1990, at the time of the defect in the 1RC-526B valve weld, vibration testing at normal plant operating conditions revealed no significant abnormal vibration which could have caused that defect.

Corrective Actions:

The defective weld has been ground out and the valve re-welded to the drain line. A post installation NDE inspection was successfully completed

Pressure testing of the new weld will be accomplished in conjunction with the return to service from the current refueling outage.

The pipe stub and what remains of the weld will be sent out for analysis to determine the probable cause of the defect.

Safety Assessment:

Section 14.3.1 of the PBNP FSAR discusses the maximum break size for which the normal makeup system can maintain the pressurizer level. That section states that the makeup flow rate from two charging pumps is typically adequate to maintain pressurizer level long enough for the operators to respond and perform an orderly shutdown of the plant with a break through a 3/8 inch diameter hole. Thus even if the weld had completely failed, adequate control of the reactor coolant system inventory could have been maintained with the charging pumps. The actual observed unidentified leakage from the Unit 1 RCS was approximately 0.1 gpm. Therefore we have concluded that the health and safety of the public and the plant personnel was not affected by this event.

System and Component Identifiers:

The Energy Industry Identification System component function identifier for each component/system referred to in this report are as follows:

Component/System	Identifier
Reactor Coolant System	AB
Valve, Isolation	ISV
Steam Generator	SG

Similar Occurrences:

A review of past LERs identified the following event involving a similar weld defect on the Unit 1 "B" steam generator channel head drain:

LER NUMBER	Title
266/90-008-00	Reactor Coolant System Leakage