January 16, 2006

NRC 2005-0016
10 CFR 50.73

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

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

Licensee Event Report 266/301/2005-007-00
Control Rod Movement with Refueling Cavity Water Level Below TS 3.9.6 Limit

Enclosed is Licensee Event Report (LER) 266/301/2005-007-00 for the Point Beach Nuclear Plant Units 1 and 2. LER 266/301/2005-007-00 describes the discovery that control rods had been moved without refueling cavity water level being at the minimum value specified in Technical Specification (TS) LCO 3.9.6, "Refueling Cavity Water Level". This condition is reportable in accordance with 10 CFR 50.73(a)(2)(i)(B).

Summary of Commitments

NMC will revise Procedure RP-4A to clarify the water level requirements for checking control rods for freedom of travel and smooth operation by May 15, 2006.

for

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

Enclosure

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

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NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0066), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4)

CONTROL ROD MOVEMENT WITH REFUELING CAVITY WATER LEVEL BELOW TS 3.9.6 LIMIT

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: PT BEACH UNIT 2	DOCKET NUMBER: 05000301
11	15	2005	2005	-- 007 --	00	1	16	2006	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	5
POWER LEVEL (10)	0

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)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)	X	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)
T. Jessessky	920-755-6002

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

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

ABSTRACT

This report describes the November 15, 2005, discovery that Unit 1 control rods had been moved without refueling cavity water level being at the minimum value specified in Technical Specification (TS) LCO 3.9.6, “Refueling Cavity Water Level”. During performance of rod latching and unlatching in accordance with procedure RP-4A, “Full-Length Control Rod Drive Shaft Unlatching and Latching”, operators had also performed control rod drag testing, which involves grasping a coupled rod with the rod latch tool and raising the control rod assembly approximately 10 feet while monitoring a load cell for weight changes. This procedure was performed with the cavity water level at rod latch height, which is less than the required 23 feet of water above the fuel assemblies per TS LCO 3.9.6. This activity was performed on October 21, 2005 and during previous refueling outages.

Subsequent to this discovery, on November 16, 2005, operators identified that Unit 1 control rod F08 remained latched after completing of the procedure for unlatching. Difficulties had been experienced on the previous shift in moving the rod shaft with a special tool shortly after completing the initial unlatching procedure. Based on these difficulties, it appears that the rod may have relatched when the tool was set down on it. Although the condition could not be verified, the potential existed that the coupled rod may have been moved with the rod shaft, prior to discovery that it remained latched, while the cavity water level was less than the required 23 feet of water.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Event Description:

On November 15, 2005, Point Beach Nuclear Plant (PBNP) operators identified that Unit 1 control rods [AA] had been moved without refueling cavity [DF] water level being at the minimum value specified in Technical Specification (TS) LCO 3.9.6, "Refueling Cavity Water Level". During performance of control rod latching and unlatching in accordance with procedure RP-4A, "Full-Length Control Rod Drive Shaft Unlatching and Latching", operators had also performed control rod drag testing, which involves grasping a coupled rod with the rod latch tool and raising the control rod assembly approximately 10 feet while monitoring a load cell [LDC] for weight changes. This procedure was performed with the cavity water level at rod latch height, which is less than the required 23 feet of water above the fuel assemblies per TS LCO 3.9.6. This activity was performed on October 21, 2005 and during previous refueling outages.

Subsequent to this discovery, on November 16, 2005, operators identified that Unit 1 control rod F08 remained latched after completing of the procedure for unlatching. Difficulties had been experienced on the previous shift in moving the rod shaft with the normal latching tool. Shortly after completing the initial unlatching procedure, it appears that the rod may have relatched when the latching tool was set down on it to replace it with a special tool. Although the condition could not be verified, the potential existed that the coupled rod may have been moved with the rod shaft, prior to discovery that it remained latched, while the cavity water level was less than the required 23 feet of water.

LCO 3.9.6 is applicable during CORE ALTERATIONS, which is defined as the movement of any fuel, source, or reactivity control component within the reactor vessel with the reactor head removed. LCO 3.9.6 has one exception, for rod latching and unlatching, for which this water level does not apply. This exception allows for latching and unlatching control rods at the lower water level referred to as latch height.

The TS exception for rod latching and unlatching generally applies to the physical rod latching and unlatching activity as well as any activities associated with and required to perform this activity (e.g., physical rod movement required to latch/unlatch rods and associated rod movement necessary to verify latching/unlatching). Control rod drag testing is not required for latching/unlatching of control rods. Therefore, this exception does not appear to apply to the drag testing that follows the latching evolution. As such, the minimum water level for this activity is 23 feet above the top of the reactor vessel flange.

Component and system Description:

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, within containment requires a minimum water level of 23 feet above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident. Sufficient iodine activity would be retained to limit offsite doses from the accident to 10 CFR 50.67 limits.

LCO 3.9.6 is applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Event Analysis and Safety Significance:

The safety significance of this condition is minimal. During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment. With a minimum water level of 23 feet and a minimum decay time of 65 hours prior to fuel handling without the containment penetration requirements of LCO 3.9.3, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits.

However, the dropping of a control rod within its guide tube is actually a designed reactor protection activity; it is not a fuel handling accident initiator. Based, in part, on the insignificant safety consequences, a recent change to NUREG-1431, Standard Technical Specifications (STS 3.9.7), removed the applicability of the subject TS during CORE ALTERATIONS. The revised applicability is limited only to movement of irradiated fuel assemblies within containment. The identified condition would not have been a violation of the revised TS.

Cause:

The cause of control rod testing being performed with refueling cavity water level below the required height was due to inclusion of this test in the procedure for control rod latching and unlatching. No precautions regarding required water level existed prior to the step that directed testing. The procedure made the control rod test evolution appear to be a part of the rod latching process.

The most likely cause of control rod F08 being latched after completing the procedure for unlatching is that the rod relatched when the normal latching tool and a special lifting tool were set down on it.

Corrective Action:

No control rod activity was in progress when this condition was discovered. The immediate action taken to correct this condition was to ensure that future control rod testing was performed with cavity water level at the required height. Subsequent control rod testing was performed in accordance with the requirements of TS 3.9.6.

The longer term corrective action is to revise Procedure RP-4A to clarify the water level requirements for checking control rods for freedom of travel and smooth operation (drag testing) by May 15, 2006.

PBNP staff also intends to submit a license amendment request to the NRC to remove the CORE ALTERATIONS applicability from TS 3.9.6, in accordance with approved Technical Specification Task Force TSTF-51.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Previous Similar Events:

A review of recent LERs (past three years) identified two events that involved operation which was prohibited by the plant's Technical Specifications:

LER Number	Title
301/2003-003-00	Failure to Place Instrument Channel in Trip as Specified by LCO 3.3.1 Required Action D.1
301/2004-001-00	Safety Injection Accumulator Operated with Fluid Level out of Specification High