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Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

NRC 2003-0080

September 4, 2003

10 CFR 50.73

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

POINT BEACH NUCLEAR PLANT, UNIT 2
DOCKET 50-301
LICENSE NO. DPR-27
LER 301/2003-005-00; UNIT 2- MANUAL REACTOR TRIP AND
MANUAL SAFETY INJECTION DUE TO PRESSURIZER LOW LEVEL

Enclosed is Licensee Event Report 301/2003-005-00 for the Point Beach Nuclear Plant, Unit 2. This LER discusses the initiation of a manual safety injection signal, a manual reactor trip and a manual containment isolation signal as a result of a plant cooling transient during a reactor startup. These conditions were determined to be reportable under 10 CFR 50.73(a)(2)(iv) as; " Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section".

Corrective actions, completed and proposed, have been identified in the enclosed report. There are no new commitments in this report.

Please contact us if you have any questions or require additional information concerning this report.

A. J. Cayla
Site Vice President

CWK/kmd

Enclosure

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

IE22

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATOI COMMISSI(	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information 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 Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB

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TITLE (4)
MANUAL REACTOR TRIP AND SAFETY INJECTION DUE TO PRESSURIZER LOW LEVEL

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 05000
07	11	2003	2003 -	005 -	00	09	04	2003	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	3
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)	X	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)		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)
Charles Wm. Krause, Senior Regulatory Compliance	(920) 755-6809

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 (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16) On July 11, 2003, Point Beach Nuclear Plant (PBNP) Unit 2 was in Mode 3 at normal operating temperature and pressure in preparation for critical approach. Upon closure of the Reactor Trip Breakers in preparation for the critical approach, the main feed water regulating valves (MFRV) opened in response to steam generator levels slightly below programmed level. The (MFRV) controllers had been left in automatic mode following a reactor trip on July 10, 2003. The increased feedwater flow to the steam generators caused a cooldown of the RCS, which resulted in a decrease in Pressurizer level. With Pressurizer level not within 10% of programmed level, Operators initiated a manual reactor trip, a manual safety injection, and a manual containment isolation per AOP 1A. There was no actual safety injection flow into the RCS and the charging pumps made up for the RCS shrinkage due to the cooldown. Steam Generator levels remained within their normal operating range. The plant was maintained in Mode 3 while an initial investigation of this event was conducted. A four hour event notification (EN# 39993) was made to the NRC at 2357 pursuant to 10 CFR 50.72(b)(2)(iv)(A).

An investigation concluded that inadequate procedure guidance was the primary reason the controllers for the MFRVs were inappropriately left in automatic. Human performance and configuration control deficiencies were contributing factors. To prevent recurrence, procedure changes were made to ensure the MFRV controllers are in "manual" with valves "closed", prior to closing the Reactor Trip Breakers. All systems and equipment functioned as designed during this event. Accordingly, the impact on the health and safety of the public and the plant staff as a result of this event is negligible. Unit 2 was returned to power operation on July 13, 2003.

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

Event Description:

On July 11, 2003, at approximately 2006 (All referenced times are Central Daylight Time), Point Beach Nuclear Plant (PBNP) Unit 2 commenced a reactor startup following a previous reactor trip, which occurred on July 10, 2003. Details of that event are provided in a separate report, LER 301/2003-004-00. At the time the unit was Mode 3 "Hot Standby", with Reactor Coolant System (RCS)[1] at normal operating temperature and pressure in preparation for a critical approach. At about 2008 the Reactor Trip Breakers[2] were closed in accordance with the startup procedure OP-1B, "Reactor Startup". After the breakers closed, the Main Feedwater Regulating Valves (MFRVs)[3] immediately opened in response to the controllers[4] being in "automatic" and a demand to "open" based on steam generator[5] level, which was slightly below the programmed level. Opening the MFRVs caused the addition of relatively cool feedwater to the steam generators causing a level increase. The RCS temperature responded by a reduction of about 20 to 25 °F loop temperature and a Tave reduction of about 23°F. At approximately 2009, Operators received a Pressurizer[6] Level Setpoint Deviation Alarm[7] and entered the alarm response procedure (AOP-1A) based on decreasing pressurizer level. Due to pressurizer level not within 10% of program level, Operators initiated a manual reactor trip, a manual safety injection, a manual containment isolation and entered the Emergency Operating Procedure EOP-0, "Reactor Trip or Safety Injection." Operators placed both controllers for the MFRVs in "manual" and closed the valves. Steam Generator level was about 80% and decreasing at the time.

At 2022, Operators exited EOP-0 and transitioned to EOP-1.1, "SI Termination". At 2024, Operators reset the Unit 2 Safety Injection and Containment Isolation and shortly thereafter secured both Safety Injection Pumps[8] and both Residual Heat Removal (RHR) Pumps[9] in accordance with EOP-1.1. At 2106, Operators exited EOP-1.1 "SI Termination".

At 2044, the NRC Resident Inspector was notified of the event and a four-hour event notification was made to the NRC Operations Center via the Emergency Notification System at 2257. At 2319, a shutdown margin calculation was performed and verified satisfactorily that the unit was stable in Mode 3. This LER is provided in accordance with 10 CFR 50.73(a)(2)(iv) System Actuation; " Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section".

Following completion of the initial review of the event and completion of interim corrective actions (as described below), Unit 2 restart was authorized. On July 12, reactor startup commenced at 2100 and the reactor was critical at 2306. On July 13 at 0349, Unit 2 entered Mode 1 and on July 16, 2003 at 0445, full power operation was achieved.

[1] System Identifier: AB
[2] System Identifier: JC Component Identifier: 52
[3] System Identifier: SJ Component Identifier: FCV
[4] System Identifier: SJ Component Identifier: FIC
[5] System Identifier: SJ Component Identifier: HX
[6] System Identifier: AB Component Identifier: PZR
[7] System Identifier: AB Component Identifier: LA
[8] System Identifier: BQ Component Identifier: P
[9] System Identifier: BP Component Identifier: P

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

Event Analysis:

A root cause evaluation (RCE 207) team was appointed to determine the root cause or causes of the event and to identify the contributing factors. The analysis was focused on the areas of equipment performance, procedure adequacy and Human Performance. Engineering personnel performed an analysis of the equipment response during this event. The evaluation examined the logics related to the MFRVs to determine whether the equipment response was appropriate for the conditions experienced. From a detailed review of the logic diagrams a signal path was identified that satisfied the scenario observed and required the reactor trip breakers to close to initiate the event. A latch in path which required a previous low Tave and a reactor trip to inhibit the feedwater control signal from positioning the control valve was also necessary. These conditions were satisfied from the reactor trip on July 10, 2003, (See LER 301/2003-004-00). With the control signal inhibited, and the SG level below the program setpoint, the controller will integrate driving the valve open signal high. Once the reactor trip breakers were closed, the high control signal was no longer inhibited and the MFRVs were driven wide open.

A fault analysis was also performed to determine if a single control system fault or malfunction could have caused both MFRVs to open. This analysis concluded that there are no single control failures that could have caused the equipment to respond in the manner observed. There was no apparent problem with the equipment or the equipment response.

Cause:

The event could have been avoided had plant procedures directed the Operators to place feedwater control in "manual" with the valves "closed" prior to closing the reactor trip breakers, thus the proximate cause of the event was concluded to be a procedure inadequacy. Human performance and configuration control deficiencies were contributing factors to this event and are being investigated as part of the continuing root cause evaluation.

During the recovery phase of the previous MFP failure and Reactor Trip event which occurred on July 10th, Operators chose to leave the Feedwater Control System in "automatic" to maintain Steam Generator level. Procedure OP-3A "Power Operation to Hot Standby" permits operator preference as whether to control feedwater in automatic or manual at this time and a review of the procedures confirmed that levels were being controlled in "automatic". The Control Room Post Trip Checklist performed following the July 10th trip, also confirmed that feedwater control was in "automatic".

On July 11th, Operators commenced Unit 2 startup using procedure OP-1B, "Reactor Startup". A review of this procedure confirmed that it does not address the MFRVs or their controllers, thus their previous position of "automatic" was not altered prior to or during the plant startup. When the Reactor Trip Breakers were closed in preparation for critical approach, the MFRVs opened in response to the controllers being in automatic and steam generator levels slightly below programmed level (see Event Analysis). Opening of the MFRVs resulted in the addition of relatively cold feedwater to the steam generators, causing steam generator level to increase. The primary system temperature responded by a reduction in average temperature of about 23 degrees F. This cooldown of the RCS caused Pressurizer pressure to decrease from about 2230 to 1990 psig, and Pressurizer level to decrease off scale low. As directed by plant procedures, the operating crew initiated a manual reactor trip, a manual safety injection signal, and a manual containment isolation signal

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Corrective Action:

Immediate Corrective Action

The licensed control room operator placed the controllers for both MFRVs in manual/shut. Appropriate plant procedures were entered as discussed in the event description and the plant was stabilized in Mode 3.

Interim Review/Actions

The root cause team assembled after the event was tasked with performing an initial review to determine if there were any impediments to restart due to potential causes of this event. This initial review concluded that inadequate procedure guidance was the reason the MFRV controllers were in "automatic" when the reactor trip breakers were closed. This review did not identify any procedure use errors. To prevent recurrence during the subsequent startup, a temporary change to the reactor startup procedure OP-1B "Reactor Startup" was completed on July 12, to ensure the MFRV controllers are in "manual" with the valves closed prior to closing the reactor trip breakers.

Corrective Actions to Prevent Recurrence

A new procedure is planned to specifically address the transition from the emergency operating procedures to the operating procedures after a reactor trip. This procedure will be used to ensure that equipment is in the correct alignment for the hot shutdown condition. Additional corrective actions may be identified following completion of the RCE human performance evaluation. Any additional corrective measures will be tracked to closure under the PBNP corrective action program.

Safety Significance:

The plant response during and following this cooling transient and manual SI and manual reactor trip was as expected. Systems and equipment necessary to mitigate the consequences of this transient performed as designed and the plant remained in Mode 3 in a stable hot shutdown condition. There was no actual SI flow into the RCS. The charging pumps were able to maintain reactor coolant inventory during this transient. Although this event was an actuation of the reactor protection system, plant equipment necessary to maintain the plant in a stable configuration functioned as required. Therefore, the safety significance of this event was negligible and the safety and welfare of the public and the plant staff was not impacted by this event.

During this event and the subsequent recovery actions there was at no time a loss of a system, structure, or component related safety functions; therefore, this event did not involve a Safety System Functional Failure.

Previous Similar Events:

A review of LERs submitted in the past three years identified no other events which involved a reactor trip due to procedural inadequacies.