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VPNPD-94-121
NRC-94-083

November 3, 1994

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

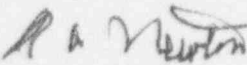
Gentlemen:

DOCKETS 50-266 AND 50-301
LICENSEE EVENT REPORT 94-011-00
REDUNDANT DECAY HEAT REMOVAL REQUIREMENTS
NOT MET DURING REFUELING SHUTDOWN
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Enclosed is Licensee Event Report 94-011-00 for Point Beach Nuclear Plant, Units 1 and 2. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(i), "Any operation or condition prohibited by the plant's Technical Specifications." The report describes a condition where redundant decay heat removal requirements were not met when the reactor was in the refueling shutdown condition.

If you require additional information, please contact us.

Sincerely,


for Bob Link

Bob Link
Vice President
Nuclear Power

KVA/jg

Enclosure

cc: NRC Regional Administrator
NRC Resident Inspector

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PDR ADCK 05000266
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JEH

LICENSEE EVENT REPORT (LER)

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

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1) Point Beach Nuclear Plant, Units 1 and 2	DOCKET NUMBER (2) 05000266	PAGE (3) 1 OF 5
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TITLE (4) Redundant Decay Heat Removal Requirements Not Met During Refueling Shutdown
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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
10	04	94	94	-- 011 --	00	11	03	94	PBNP UNIT 2	05000301
									FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9) N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)										
POWER LEVEL (10) 0	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
	20.405(a)(1)(iii)				X 50.73(a)(2)(i)				50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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 Kenneth V. Arneson, Senior Engineer - Licensing	TELEPHONE NUMBER (Include Area Code) (414) 221-3362

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)										
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPDs		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPDs

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 October 4, 1994, we discovered that on April 9, 1994, during the Point Beach Nuclear Plant Unit 1, annual maintenance and refueling outage, both residual heat removal pumps were secured when reactor coolant system temperature was <140°F with fuel in the core, a condition prohibited by Point Beach Technical Specifications. The cause of this event is inadequate communication to the operators concerning the interpretation of Technical Specification 15.3.1.A.3.b. Inadequacies in the temporary procedure change process are contributing factors. Both residual heat removal pumps were secured for less than two minutes and residual heat removal suction and discharge temperatures remained steady at approximately 74°F and 54°F, respectively. The health and safety of plant personnel and the public were not impacted by this event.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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Point Beach Nuclear Plant, Units 1 and 2	05000266	YEAR 94	SEQUENTIAL NUMBER -- 011 --	REVISION NUMBER 00	2 OF 5

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

Event Description:

On October 4, 1994, we discovered that on April 9, 1994, during the Point Beach Nuclear Plant (PBNP) Unit 1 annual maintenance and refueling outage, both residual heat removal (RHR) pumps were secured when reactor coolant system (RCS) temperature was <140°F with fuel in the core, a condition prohibited by PBNP Technical Specifications (TS). Technical Specifications Section 15.3.1, "Reactor Coolant System," Specification A.3.b(1) states that both RHR loops must be operable when RCS temperature is less than 140°F. Specification A.3.b(4) allows one RHR loop to be temporarily out of service to meet surveillance requirements.

On October 1, 1994, PBNP Unit 2 was in the refueling shutdown condition for the annual maintenance and refueling shutdown with RCS temperature <140°F. After Inservice Test Procedure IT-755, "Flow Test of Low Head Safety Injection Check Valves (Refueling), Unit 2," was initiated, the control operator recognized that completing the test as written would secure both RHR pumps. The test was suspended until a temporary change was made to ensure that one RHR pump was kept in operation. Condition Report 94-424 was written to document the event and initiate corrective action.

During the process of making the temporary procedure change, the procedure history file for IT-755 was reviewed. A temporary change to IT-755 dated October 1, 1993, was discovered. This temporary change was made to ensure at least one RHR pump would be kept in operation while the test was performed during the Unit 2 maintenance and refueling outage in the Fall of 1993. The temporary change form was marked to indicate that a permanent procedure change to this effect should be made. However, the permanent procedure change was not made.

As part of the corrective action for CR 94-424, the procedure change history file for the corresponding Unit 1 procedure, IT-750, "Flow Test of Low Head Safety Injection Check Valves (Refueling), Unit 1," was reviewed. It was discovered that on April 9, 1994, during the Unit 1 annual maintenance and refueling outage, IT-750 was performed as written. Thus, both RHR pumps were secured when RCS temperature was <140°F with fuel in the core, a condition prohibited by TS 15.3.1.A.3.b.

Additionally, a temporary change to IT-750 dated April 13, 1993, was discovered. This temporary change was made to ensure at least one RHR pump would be kept in operation while the test was performed during the Unit 1 maintenance and refueling outage in the Spring of 1993. The

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temporary change form was marked to indicate that a permanent procedure change to this effect should be made. However, the permanent procedure change was not made.

Component and System Description:

The Residual Heat Removal (RHR) System is a dual purpose system. During normal operation, it functions as a Low Head Safety Injection System. During normal cooldown of the RCS, it functions as a residual heat removal system to remove decay heat from the core and to reduce RCS temperature during the cooldown. When the reactor is shut down, the RHR system continues to remove decay heat.

The Unit 1 and Unit 2 RHR systems are identical. Each system contains two shell and U-tube heat exchangers and two centrifugal pumps. Reactor coolant flows through the U-tubes and is cooled by component cooling water (CCW) circulating through the shell.

The purpose of Procedures IT-750 and IT-755 is to perform a full stroke open test of the RHR pump refueling water storage tank (RWST) suction and RHR core deluge check valves as required by the ASME Section XI Code. Following the steps as written in IT-750/755 results in momentarily securing both RHR pumps.

Cause:

Technical Specification 15.3.1.A.3.b does not explicitly state that at least one RHR pump must be in operation when RCS temperature is less than 140°F with fuel in the reactor vessel. It requires that both RHR loops be operable in this condition and allows one loop to be out of service for surveillance testing. It also states that if no RHR loop is in operation, all operations causing an increase in decay heat load or a reduction in RCS boron concentration shall be suspended and corrective actions to return a decay heat removal method to operation shall be initiated immediately.

Prior to mid-1992, the interpretation of this specification was such that securing both RHR pumps during a surveillance test was not considered a violation of Technical Specifications. In the Spring of 1992, it was identified that redundant decay heat removal requirements were not met during the Fall 1991 maintenance and refueling outage for Unit 2. As part of the corrective actions for this event, the Manager's Supervisory Staff reevaluated this interpretation and subsequently required one RHR

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loop to be in operation at all times. This interpretation was not adequately implemented in operating procedures nor communicated to all operations crews.

The operators performing Procedure IT-750 on April 9, 1994, were not aware of this interpretation of TS 15.3.1.A.3.b and thus did not consider performing the test as written a violation of redundant decay heat removal requirements.

A contributing factor is that, due to administrative oversight, permanent procedure changes to IT-750 and IT-755 to ensure that one RHR pump remains in operation throughout the test were not made. The temporary procedure change forms for these procedures from the previous outages indicated that a permanent change was required. Had these permanent procedure changes been made, the event would not have occurred.

Corrective Action:

A root cause evaluation to determine the root cause and appropriate corrective actions for this event was completed on October 31, 1994.

Permanent changes will be made to IT-750 and IT-755 to ensure that one RHR pump is kept running throughout the test. The permanent changes will be made by December 16, 1994.

The interpretation of TS 15.3.1.A.3.b will be communicated to all operations crews by November 11, 1994.

We are considering a Technical Specification Change Request to clarify the redundant decay heat removal requirements when RCS temperature is less than 140°F.

Reportability:

This licensee event report is being submitted in accordance with the requirements of 10 CFR 50.73(a)(2)(i), "Any operation or condition prohibited by the plant's Technical Specifications."

Safety Assessment:

When the reactor is shutdown, the RHR system functions to remove decay heat from the core. There is no automatic safety function associated

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with the RHR system when the reactor is in the refueling shutdown condition. The reactor vessel head was removed and the refueling cavity was filled to the rod unlatching height. No fuel movement or reduction in boron concentration occurred during the test. The short time both RHR pumps were secured is not considered sufficient to allow stratification of boron in the reactor coolant. Throughout the performance of the test procedure, the operator had administrative control over the RHR pumps. There is no change in the valve line-up or heat exchanger configuration for the RHR system during this test. Thus, in the unlikely event of an accident requiring one or both RHR loops, the pumps could have been started immediately to perform their heat removal function.

Plant computer data for the time this test was conducted on April 9, 1994, was reviewed. Both trains of RHR were secured for less than two minutes and RHR suction and discharge temperatures remained steady at approximately 74°F and 54°F respectively. The health and safety of plant personnel and the public were not impacted by this event.

Similar Occurrences:

A review of licensee event reports was performed to identify other reports that describe violations of redundant decay heat removal requirements. The following licensee event reports were identified:

301/91-007-00 "Violation of Decay Heat Removal Requirements During Steam Generator Crevice Cleaning"