Wisconsin Electric
POWER COMPANY
231 W Michigan, P.O. Box 2046, Milwaukee, WI 53201 (414) 221-2345

VPNPD-92-249 10CFR50.73
NRC-92-077

July 14, 1992

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

Gentlemen:

DOCKET 50-301
LICENSEE EVENT REPORT 91-007-00
VIOLATION OF DECAY HEAT REMOVAL REQUIREMENTS
DURING STEAM GENERATOR CREVICE CLEANING
POINT BEACH NUCLEAR PLANT, UNIT 2

Enclosed is Licensee Event Report 92-002-00 for Point Beach Nuclear Plant, Unit 2. This report is provided in accordance with 10 CFR 50.73(a)(2)(i)(B), "The licensee shall report...any operation or condition prohibited by the plant's Technical Specifications."

This report describes the violation of Point Beach Technical Specification 15.3.1, "Reactor Coolant System," Specification A.3.a.(3), in which one required method of decay heat removal was not maintained in continuous operation during steam generator crevice cleaning.

Please contact us if any further information is required.

Sincerely,

Bob E. Link
Vice President
Nuclear Power

Enclosure

Copies to NRC Resident Inspector
NRC Regional Administrator

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Point Beach Nuclear Plant, Unit 2	0 5 0 0 0 3 0 1	1 OF 0 7

TITLE (4)
Violation of Decay Heat Removal Requirements During Steam Generator Crevice Cleaning

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
1 1	1 0	9 1	9 1	0 0 7	0 0	0 7	1 4	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

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

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 (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
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	TELEPHONE NUMBER AREA CODE	
Norm Hoefert, Manager - Operations	4 1 4	7 5 5 - 2 3 2 1

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

ABSTRACT

On November 10, 1991, Unit 2 was in its seventeenth annual refueling outage. While performing Wisconsin Michigan Test Procedure (WMTP) 11.19, Revision 7, "Steam Generator Crevice Cleaning," RHR pump 2P-10A was secured, leaving the unit without at least one method of decay heat removal in operation as required by Point Beach Nuclear Plant Technical Specification 15.3.1, "Reactor Coolant System," Specification A.3.a.(3), which states, "At least one of the above decay heat removal methods shall be in operation except when required to be secured for testing."

This event was identified as a potential Technical Specification violation during a special safety inspection conducted by the NRC Resident Inspectors on May 27 through June 14, 1992. Wisconsin Electric management personnel discussed this issue with the NRC Resident Inspectors on June 15, 1992, during the NRC exit meeting.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Point Beach Nuclear Plant, Unit 2	0 5 0 0 0 3 0 1	9 1	— 0 0 7	— 0 0	0 2 OF 0 7

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

EVENT DESCRIPTION

During an NRC special safety inspection (NRC Inspection Reports 50-266/92014 and 50-301/92014) of a Unit 1 RCS excessive cooldown rate Technical Specification violation (LER 266/92-005-00), it was discovered that on November 10, 1991, while performing steam generator crevice cleaning following the fall 1991 Unit 2 refueling outage, an operating residual heat removal (RHR) pump was secured without an additional method of decay heat removal being in operation. RHR pump 2P-10A was secured to limit the RCS cooldown rate during the third steam generator crevice cleaning cycle, leaving the unit without at least one of the required methods of decay heat removal in operation as required by Point Beach Nuclear Plant Technical Specification 15.3.1, "Reactor Coolant System," Specification A.3.a.(3), which states, "At least one of the above decay heat removal methods shall be in operation except when required to be secured for testing." The decay heat removal methods referenced in specification A.3.a.(3) are contained in specification A.3.a.(1). These methods are:

1. Reactor Coolant Loop A, its associated steam generator and either reactor coolant pump.

2. Reactor Coolant Loop B, its associated steam generator and either reactor coolant pump.

3. Residual Heat Removal Loop (A).

4. Residual Heat Removal Loop (B).

Although the pumps associated with the above decay heat removal loops had been secured, the pumps and their associated loops remained operable.

Because the procedure being performed was titled, "Wisconsin Michigan Test Procedure," plant operators interpreted this procedure as applicable to the Technical Specification 15.3.1.A.3.a.(3) phrase "...except when required to be secured for testing." Because the operators were performing a test procedure, they concluded that the test exception could be exercised to allow securing the RHR pumps. Due to the Technical Specification allowance to temporarily secure both RHR pumps, the cooling action of the crevice cleaning, and the ability to quickly restore RHR flow by turning a control switch, the operators concluded that shutting off both RHR pumps was an allowed and prudent action.

This event was identified as a potential Technical Specification violation during the special safety inspection conducted by the NRC Resident Inspectors on May 27 through June 14, 1992. Wisconsin Electric management personnel discussed this issue with the NRC Resident Inspectors on June 15, 1992, during the NRC exit meeting. It was also discussed at the July 7, 1992, Enforcement Conference at Region III.

EQUIPMENT AND EVOLUTION 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 WMTP 11.19 is to remove soluble materials from the steam generator tubesheet crevice areas. Preparation for crevice cleaning begins by filling the steam generators to a level of approximately 30 inches with the atmospheric steam dumps shut. The RCS is heated to 290-300°F and maintained at that temperature for 30 minutes to dissolve contaminants in the steam generators. Both atmospheric steam dumps are then fully opened, and the steam generators are allowed to boil for 60 minutes. Both atmospheric steam dumps are then shut, and the RHR system is used to perform a controlled cooldown to 175-190°F in accordance with required cooldown limits. After obtaining chemistry samples, the steam generators are completely drained to remove the water containing the dissolved contaminants. The steam generators are then refilled. The necessity to perform additional crevice flush cycles is determined by chemistry sample results.

The RCS cooldown rate can be controlled using the following techniques:

1. Varying the flow of CCW to the RHR heat exchangers.

2. Manually adjusting the position of the RHR heat exchanger outlet valves.

3. Cycling an RHR pump (however, this is not allowed by PBNP Technical Specifications, except for testing).

4. Throttling the supply of service water to the CCW heat exchangers to vary the CCW temperature.

When the desired final temperature is reached, it is maintained by reducing RHR flow through the heat exchangers and, if necessary, by reducing the number of pumps and heat exchangers in service. Flow

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Point Beach Nuclear Plant, Unit 2	0 5 0 0 0 3 0 1	9 1	— 0 0 7	— 0 0	0 4 OF 0 7

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

through the heat exchangers is then controlled to compensate for the decreasing heat input from decay heat following plant shutdown.

CAUSE

The causes and contributing factors to this event include:

1. WMTP 11.19, Revision 7, directs the operator to adjust RHR cooling to try to maintain RCS temperature (measured at the RHR heat exchanger outlet) within a desired range. However, the procedure gives no guidance on cooldown control methods or methods required for maintaining decay heat removal. As a result, three out of four operating crews accomplished this task differently, with one crew concurrently securing both reactor coolant pumps (as specifically required by the procedure) and both RHR pumps.

2. To control the RCS cooldown rate during the third cycle of WMTP 11.19, one operating crew secured the remaining RHR pump. Technical Specification 15.3.1.A.3.a.(3) states, "At least one of the above decay heat removal methods shall be in operation except when required to be secured for testing." Because the procedure being performed was titled, "Wisconsin Michigan Test Procedure," the operators interpreted this procedure as applicable to the phrase "...except when required to be secured for testing." Hence, this was interpreted as justification for securing the remaining RHR pump. Also, the basis to Technical Specification 15.3.1 contained no information to support an interpretation of this requirement.

CORRECTIVE ACTIONS

The following corrective actions have been completed or are planned to be taken:

1. The need to continue performing the steam generator crevice cleaning procedure will be assessed. This assessment will be performed by Chemistry personnel by July 31, 1992.

2. Between the fall 1991 Unit 2 refueling outage and the spring 1992 Unit 1 refueling outage, WMTP 11.19, Revision 7, was converted to Refueling Procedure (RP) 6B, "Steam Generator Crevice Cleaning." This procedure was improved, as it now directs operators to minimize RCS cooldown by operating one RHR pump (in accordance with Technical Specification 15.3.1.A.3.a.[3]) and bypassing the RHR heat exchanger. However, this procedure will be further revised to (1) specify the cooldown control method and temperature monitoring

location, (2) include decay heat removal Technical Specification requirements, and (3) require implementation of Procedure PBNP 3.4.19, "Infrequently Performed Tests or Evolutions/Special Test Procedures." Procedure RP-6B will be revised by Operations personnel and issued by August 13, 1992 (prior to the fall 1992 Unit 2 refueling outage).

3. An evaluation to assess the need to revise Technical Specification 15.3.1.A.3.a.(3) and to clarify its basis will be conducted. This evaluation will be conducted by Operations and Regulatory Services personnel by August 28, 1992.

4. Classroom and simulator training will be conducted on Procedure RP-6B or related Procedure RP-6A, "Steam Generator Crevice Flush (Vacuum Mode)," depending on the crevice cleaning method planned for the fall 1992 Unit 2 refueling outage and will be completed by September 25, 1992.

5. We are currently committed to a PBNP Technical Specification upgrade project which will revise a select group of Technical Specifications to include Limiting Conditions for Operation (LCOs) and Surveillances. Although the major emphasis of the project is to add LCO and surveillance requirements, those Specifications which are recognized as requiring clarification or interpretation will be revised as determined appropriate. We have previously committed to submit Technical Specification Change Request packages required to support this upgrade project by February 28, 1993.

REPORTABILITY

This event is being reported under the requirements of 10 CFR 50.73(a)(2)(i)(B), "The licensee shall report...any operation or condition prohibited by the plant's Technical Specifications."

SAFETY ASSESSMENT

Circulation through the reactor core is desirable during steam generator crevice cleaning for the following reasons:

1. Mixing boron in the RCS to prevent stratification in the event of RCS boration or dilution. Because the basis for PBNP Technical Specification 15.3.1 contains no information to support an interpretation of Specification A.3.a.(3), the basis for Westinghouse Standard Technical Specification 3.4.1, "Reactor Coolant Loops and Coolant Circulation," is referenced as guidance. The basis for Westinghouse Standard Technical Specification 3.4.1 states:

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Point Beach Nuclear Plant, Unit 2	0 5 0 0 0 3 0 1	9 1	— 0 0 7	— 0 0	0 6 OF 0 7

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

"The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control."

During this event, boration or dilution was not occurring during the time the RHR pumps were secured. Therefore, the intent of Westinghouse Standard Technical Specification 3.4.1 and, we believe, PBNP Technical Specification 15.3.1 was satisfied, and reactivity changes associated with boron concentration fluctuations were not a major safety concern.

2. Obtaining accurate core temperatures. With both reactor coolant pumps secured and the RHR system in operation, the most representative indicator of core temperature is RHR heat exchanger inlet temperature. However, the RHR pump had been secured, reducing flow through the RHR heat exchanger. Thus, indicated and actual core temperature may not have corresponded. However, natural circulation was occurring through the RCS loops and incore thermocouples were available for accurate temperature indication.

3. Decay heat removal. At the time of this event, Unit 2 was shut down and had just been refueled, generating minimal decay heat. The plant was in a controlled evolution with four decay heat removal loops operable. Therefore, decay heat removal was not a primary safety concern at that time.

GENERIC IMPLICATIONS

A thorough understanding of the Technical Specification bases must be maintained. Not only are some systems required to operate to perform their primary functions, they may also serve collateral functions which are inherent in the bases of other Technical Specification requirements.

SIMILAR OCCURRENCES

Although there are no past occurrences documented of both RHR pumps being secured during steam generator crevice cleaning, there have been two recent events related to RCS temperature control during this evolution.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Point Beach Nuclear Plant, Unit 2	0 5 0 0 0 3 0 1	9 1	— 0 0 7	— 0 0	0 7 OF 0 7

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

On May 10, 1991, while performing steam generator crevice cleaning on Unit 1, the primary coolant temperature was not maintained in accordance with Technical Specification requirements. During this evolution, primary system temperatures exceeded 200°F for approximately 17 minutes without containment integrity being set, contrary to Technical Specification 15.3.6.A.a. This event was reported in LER 266/91-004-00 dated September 16, 1991.

On May 27, 1992, during the performance of Refueling Procedure (RP) 6B, "Steam Generator Crevice Cleaning," the temperature cooldown rate limit of 100°F per hour, as specified in PBNP Technical Specification 15.3.1.B.1.b, was exceeded. This event was reported in LER 266/92-005-00 dated June 26, 1992.