Wisconsin Electric
POWER COMPANY

231 W Michigan, PO Box 2046, Milwaukee, WI 5320

(414) 221 2345

VPNPD-92-297
NRC-92-101

CERTIFIED MAIL

September 3, 1992

Director, Office of Enforcement
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, D.C. 20555

Attention: Document Control Desk

Gentlemen:

DOCKETS 50-266 AND 50-301
REPLY TO NOTICE OF VIOLATION
INSPECTION REPORTS 50-266/92014; 50-301/92014
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In a letter from Mr. A. Bert Davis dated August 4, 1992, the Nuclear Regulatory Commission forwarded to Wisconsin Electric Power Company, licensee for the Point Beach Nuclear Plant, a Notice of Violation and Proposed Imposition of Civil Penalty (Notice). The Notice describes violations identified during a special safety inspection conducted at Point Beach Nuclear Plant from May 27 to June 14, 1992.

We have reviewed this Notice and, pursuant to the provisions of 10 CFR 2.201, have prepared a written statement of explanation concerning each of the identified violations. The written statement is included as an attachment to this letter. We have also enclosed a check payable to the Treasurer of the United States in the amount of $50,000 for payment of the civil penalty imposed by the Notice.

The violations identified in the Notice pertain to steam generator crevice cleaning activities. In one case, the activities resulted in an excessive cooldown of the reactor coolant system and, in another case, resulted in the required decay heat removal system being removed from operation. The violations which were assessed a civil penalty relate to crevice cleaning of the PBNP Unit 1 steam generators and associated excessive cooldown of the reactor coolant system which occurred on May 27, 1992. As a result of this event, an internal investigation team was chartered to evaluate the causes of the event and to recommend corrective actions. We discussed the

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results of this incident investigation and the facts and circumstances associated with the proposed violations at an Enforcement Conference conducted by Mr. A. Bert Davis and other members of the NRC Region III staff on July 7, 1992.

The violations, taken collectively, represent a deficiency in the control of infrequently occurring control room activities. Our corrective actions taken or proposed respond to the specific factors associated with the violations and also address enhancements in planning and controlling of special infrequently performed tests or evolutions.

PBNP Procedure 3.4.19, "Infrequently Performed Tests or Evolutions/Special Test Procedures," provides the criteria and requirements for performing such evolutions. This procedure requires the validation of the evolution, use of the plant simulator as a training tool as appropriate, pre-evolution briefing, assessment of need for additional personnel, and the assignment of appropriate management oversight. We are in the process of revising this procedure to ensure effective and consistent application.

Along with this procedure revision, higher standards for the conduct of infrequently performed tests will be applied and enforced. Associated with these standards will be the requirement for applicable managers and supervisors to perform effective oversight and direction of control room operations during special or infrequently performed evolutions.

We believe that the attached reply is responsive to the concerns and fulfills the requirements identified in your August 4, 1992, letter. If you have any questions or require additional information regarding this response, please contact us.

Sincerely,

Bob Link

Bob Link
Vice President
Nuclear Power

Enclosures (Check 953787)

Copies to NRC Regional Administrator, Region III
NRC Resident Inspector

Subscribed and sworn to before me
this 3rd day of Sept., 1992.

Gloria J. Monsoor

Notary Public, State of Wisconsin
My Commission expires June 2, 199-

MFB/bjo

REPLY TO NOTICE OF VIOLATION AND
PROPOSED IMPOSITION OF CIVIL PENALTY

Wisconsin Electric Power Company
Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301

During a special inspection conducted at the Point Beach Nuclear Plant from May 27 to June 14, 1992, three violations of NRC requirements were identified. Two of the violations were classified in the aggregate as a Severity Level III problem and resulted in a civil penalty. The Notice of Violation and Proposed Imposition of Civil Penalty (Notice) transmitted to Wisconsin Electric on August 4, 1992, provided details regarding the three violations and the imposed civil penalty in the amount of $50,000. We agree that the events and circumstances described in the Notice are accurately characterized. We also agree that factors involving escalation or mitigation of the associated civil penalties have been correctly applied to these violations.

In accordance with the instructions provided in the Notice, our reply for each alleged violation includes (A) admission or denial of the alleged violation, (B) the reason for the violation if admitted, and if denied, the reasons why, (C) the corrective actions that have been taken, (D) the corrective actions that will be taken to avoid further violations, and (E) the date when full compliance will be achieved.

I. <u>VIOLATIONS ASSESSED A CIVIL PENALTY</u>

A. "Technical Specification 15.3.1.B.1 requires, in part, that during heatup, cooldown, criticality, and in-service leak and hydrostatic testing, the temperature of the reactor coolant system shall be limited to a maximum cooldown rate of 100°F in any one hour."

"Contrary to the above, on May 27, 1992, the maximum cooldown of 100°F in any one hour was exceeded during cooldown. Specifically, the reactor coolant system cooldown was approximately 141°F in one hour while Refueling Procedure RP-6B, 'Steam Generator Crevice Cleaning,' was performed."

B. "10 CFR Part 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by documented instructions or procedures of a type appropriate to the circumstances."

"Contrary to the above, as of May 27, 1992, Refueling Procedure RP-6B, 'Steam Generator Crevice Cleaning,' Revision 0, dated April 27, 1992, a procedure governing activities that affect quality, was not a type appropriate to the circumstances. Specifically, Refueling Procedure RP-6B, 'Steam Generator Crevice Cleaning,' did not prescribe adequate instructions to prevent the violation of maximum heatup and cooldown rates delineated in Technical Specification 15.3.1.B.1."

RESPONSE TO VIOLATIONS ASSESSED A CIVIL PENALTY

A. ADMISSION OR DENIAL OF ALLEGED VIOLATIONS

We acknowledge that these violations assessed a civil penalty are accurately characterized and agree that the described activities associated with the event constitute a violation of NRC requirements, as defined in Point Beach Nuclear Plant Technical Specifications and 10 CFR Part 50, Appendix B.

B. REASONS FOR THE VIOLATIONS ASSESSED A CIVIL PENALTY

A number of factors contributed to the May 27, 1992, excessive cooldown event. To identify and evaluate root causes, and to formulate corrective actions, an internal investigation team was chartered. Our review of the event and the findings of the incident investigation team identified the causes and contributing factors associated with these violations to be attributable to four major factors. Factors contributing to the event include (1) operator performance, (2) procedural inadequacies, (3) inadequate control room indications, and (4) training. The specific details relating to each of these factors are summarized in the following paragraphs.

1. Operator Performance

The operators performing the crevice cleaning evolution failed to adequately monitor reactor coolant system (RCS) temperature changes and control the cooldown rate to within the required PBNP Technical Specification requirements. A contributing factor to this failure was inadequate communication between the control operator performing the crevice cleaning operation and the Duty Shift Superintendent. This inadequate communication led the control operator to believe that he had permission to exceed the PBNP administrative cooldown rate limit of 50°F/hour. Exceeding the 50°F/hour administrative limit allows cooldown rates to approach the technical limits with less margin available for unanticipated cooling mechanisms.

An additional contributing factor affecting the performance of control room personnel was concurrent work activities which may have diluted the control operator's and the Duty Shift Superintendent's ability to provide adequate attention to the cooldown evolution in progress. These concurrent activities included (1) the control operator establishing an auxiliary charging path to facilitate testing of a charging line check valve, (2) personnel entering and exiting Unit 1 containment with containment integrity establisned, and (3) the Duty Shift Superintendent removing a Unit 2 radiation monitor from service and entering a limiting condition for operation on Unit 2 for the radiation monitor out of service.

2. Procedural Inadequacies

In accordance with procedure RP-6B, "Steam Generator Crevice Cleaning, Revision 0," the initial RCS temperature for the cleaning cycles was to be established within the range of 290°F - 300°F. Prior to the Fall 1991 Unit 2 refueling outage, the initial RCS temperature had been increased from an initial temperature of 250°F used in previously completed hot steam generator cleaning cycles. This higher initial RCS temperature increases the potential for a significant cooldown rate during the cleaning cycle. Several methods of heat removal from the RCS were in progress during the crevice cleaning cycles. The heat removal methods included boiling in the steam generators due to fully open atmospheric steam dumps, filling of the steam generators with auxiliary feedwater to maintain water level in the generators, residual heat removal (RHR) system leak-by through the RHR heat exchanger flow control valves, and use of the auxiliary charging system. Cooldown from the use of the auxiliary charging system differs from that of normal charging because the regenerative heat exchanger is not in the auxiliary charg g flow path to the RCS, as is the configuration for the normal charging flow path, and therefore, preheating of the auxiliary charging water does not occur. Procedure RP-6B did not adequately address the simultaneous use of these systems during the crevice cleaning evolution to provide the operator with sufficient guidance to establish adequate control of the cooldown.

Procedure RP-6B did not specify which temperature parameters were to be utilized by the control operator to monitor the cooldown or the proper method to evaluate the temperature values to monitor the cooldown rate. If appropriate guidance had been procedurally specified, such as manual plotting, computer trending, or strip chart recorder trending of specific RCS temperature data, the operators could have been alerted to the excessive cooldown rate in progress.

The PBNP Technical Specification cooldown rate limit of 100°F/hour was not specified prior to any of the cooldown steps in procedure RP-6B. Providing this limit in appropriate caution statements could have alerted the operators that the potential for a significant cooldown rate existed.

Procedure RP-6B also did not specify the PBNP administrative cooldown rate limit of 50°F/hour prior to all cooldown steps. Specifically, the step which requires opening of the atmospheric steam dumps for the cleaning cycle did not refer to this administrative limit. Where the limit was stated in the procedure, it was included as an informational "NOTE" rather than a more appropriate "CAUTION" statement.

Procedure PBNP 3.4.19, "Infrequently Performed Tests or Evolutions/Special Test Procedures," requires that, for these types of tests or evolutions, the procedure is validated prior to use, a pre-evolution briefing is performed, an assessment to determine if additional personnel are required to perform the evolution is completed, and independent line management oversight of the procedure is conducted. All the requirements contained in PBNP 3.4.19 were not applied to procedure RP-6B prior to this evolution.

An appropriate review of RP-6B was not accomplished during the revision process. Procedure RP-6B is a multi-disciplinary procedure involving both operations and chemistry technical information. As a multi-disciplinary procedure, an adequate technical review requires both groups to be involved. The revision of the procedure utilized during this evolution was written by a qualified individual from the Operations Group, and was formally reviewed for technical content only by a qualified individual from the Chemistry Group.

Difficulties controlling the cooldown rate during crevice cleaning had been experienced during the Unit 2 refueling outage in November 1991. At that time, Wisconsin Michigan Test Procedure (WMTP) 11.19 was used. During the cleaning evolutions, operators controlled the cooldown rate by a number of different methods. The methods used were throttling component coolant water flow to the RHR heat exchangers, throttling or shutting the RHR heat exchanger manual isolation valves, and shutting off the RHR pumps. After the November 1991 Unit 2 refueling outage, the Operations Group rewrote WMTP 11.19 into the Refueling Procedure (RP) format, specifically RP-6B. In addition to writing the procedure in the new format, a two-hour hot soak of the steam generators and notes preceding heatup and cooldown steps to remind the operators of the PBNP administrative limit on heatup and cooldown rates were added. While some of the problems identified during the use of WMTP 11.19 were addressed in RP-6B, the new procedure was not sufficiently enhanced to provide specific instructions regarding appropriate methods to control the cooldown rate. The operating experience gained during the Unit 2 crevice cleaning evolution was not effectively factored into procedure RP-6B.

Procedure RP-6B did not include a precaution regarding the initiation of additional systems which may cause an increase in the cooldown rate while the crevice cleaning cycles are in progress. For example, auxiliary charging was lined up during the crevice cleaning evolution. This established an additional cooldown mechanism as the regenerative heat exchanger is not in the auxiliary charging flow path to the RCS, as is the configuration for the normal charging flow path, and therefore, preheating of the auxiliary charging water does not occur.

3. <u>Control Room Indications</u>

The plant process computer system (PPCS) indication of the cooldown rate can be misleading because if the difference between any of the three calculated rates is greater than 10%, the displayed value is based on the highest cooldown rate calculated from the twenty-minute, five-minute, or five-second rate. For small step changes in the RCS temperature, high cooldown rates can be displayed, providing the operator with a value which is not representative of the actual cooldown rate.

Currently, there are no cooldown rate alarms installed in the PPCS software. If alarms had been available, the operator may have recognized the excessive cooldown rate and terminated the cooldown process.

4. Training

Training of the operating crews on Procedure RP-6B was not conducted prior to its use. The importance of providing training on this procedure was not recognized. In addition to the specific training on the procedure, knowledge and understanding of the thermodynamics of the RCS in the mode of operation established for the crevice cleaning evolution may not have been thoroughly understood.

C CORRECTIVE ACTIONS THAT HAVE BEEN TAKEN

In response to these violations, a number of corrective actions have been completed. A number of corrective actions were taken immediately following the identification of the excessive cooldown. Also, additional actions have been completed to date in response to the recommendations of our incident investigation team.

1. Immediate Corrective Actions

a) The steam generator crevice cleaning activities were suspended upon identification of the occurrence of the excessive cooldown. No additional crevice cleaning evolutions were performed for the remainder of the Unit 1 outage in order to ensure that adequate review of the causes and contributing factors could be performed and that appropriate corrective actions could be implemented.

b) We committed to not pressurizing Unit 1 above the Low Temperature Overpressure Protection limit until the engineering analysis of the structural integrity of the reactor vessel beltline region was completed. This action was taken to ensure no additional temperature or pressure transients would occur until the structural integrity of the reactor vessel beltline region was ensured.

c) We performed a review of the Fall 1991 PBNP Unit 2 refueling outage data to determine if any excessive temperature transients occurred during crevice cleaning evolutions on Unit 2. We concluded that no such temperature transient occurred during the Unit 2 outage.

d) An internal incident investigation team was formed to further evaluate the event. The team's charter was to review the event, identify the causes and contributing factors, and identify additional corrective actions as appropriate.

e) The control board operator was removed from main control board watchstanding duties until an evaluation of his abilities to perform licensed operator duties could be completed with satisfactory results. The control board operator successfully completed the evaluation and testing on June 20, 1992, and was allowed to resume watchstanding duties.

f) Disciplinary action was taken against the operators directly responsible for the cooldown event. This action was taken to reinforce management standards and expectations on watchstanding duties and principles.

g) The plant manager issued a memorandum on May 28, 1992, to all licensed operators to further emphasize the responsibility of licensed operators to be fully accountable for keeping several fundamental reactor safety concepts in mind while performing daily activities. The fundamentals stressed in the memorandum were to maintain reactivity control, core cooling, containment integrity, and reactor vessel integrity.

2. Additional Corrective Actions Taken to Date

a) An assessment of the need to continue performing steam generator crevice cleaning evolutions in the future has been completed. Cost, risk, and benefit realized were evaluated. This assessment was completed and reviewed by the PBNP Manager's Supervisory Staff on August 18, 1992. The Manager's Supervisory Staff agreed to continue to perform a maximum of three crevice cleaning cycles for the Unit 2 steam generators during the Fall 1992 outage. The staff also decided that further review of the need to continue the crevice cleaning evolutions on the Unit 1 steam generators was necessary. The Unit 1 review will be completed and the decision whether or not to continue the evolution will be completed by April 1, 1993, prior to the Spring 1993 Unit 1 refueling outage.

b) Procedure RP-6B, "Steam Generator Crevice Cleaning," was revised and issued as Revision 1 on August 13, 1992. Based on evaluations completed by individuals from the Chemistry and Operations Groups, this revision includes the following enhancements:

- The temperature range at which the cleaning evolution is initiated has been reduced to a range of 240°F - 250°F from the range of 290°F - 300°F specified in RP-6B, Revision 0. This revised initial temperature will reduce the potential for exceeding the technical specification cooldown rate. The final temperature at the completion of the cleaning cycle is specified in the procedure as 170°F - 180°F for a total maximum cooldown of 80°F.

- The proper methods for cooldown are now specified in the procedure. RP-6B, Revision 1, provides guidance to the operators that PBNP Technical Specifications require one method of decay heat removal to be in operation when the RCS temperature is between 140°F and 350°F. Temperature control during the 30 minute boiling period is to be maintained by throttling the atmospheric steam dumps if the cooldown rate will exceed 50°F/hour. Following the boiling period, the procedure requires the RCS temperature and steam generator levels to be stabilized before securing one reactor coolant pump. The procedure requires the cooldown of the RCS to 170°F - 180°F by maintaining one reactor coolant pump and one RHR pump in operation and controlling the temperature by adjusting RHR flow using the flow control valves.

- The problems associated with the undesirable cooling from RHR system flow control valve leak-through have been addressed in Procedure RP-6B in order to minimize the risk of another cooldown event. In order to minimize the effect of the RHR system flow control valve leak-through, one of the initial conditions required by Procedure RP-6B is that the component cooling water (CCW) heat exchanger outlet temperature be increased to and stabilized at approximately 95°F. The CCW system provides the heat sink for the RHR system and the elevated temperature provides for a more controlled cooling capacity due to the RHR system flow control valve leak-through.

- The proper temperature parameters to be monitored during the crevice cleaning evolution and the proper monitoring method have been specified in Procedure RP-6B. The procedure now requires tracking and manual plotting for each cooldown and heatup by the use of the "B" loop cold leg RTD (T451C) and the incore thermocouple average (TC_{ave}). The tracking and plotting instructions are contained in Operating Instruction (OI)-105, "RCS Heatup/Cooldown Plotting." OI-105 provides detailed directions for the set-up of manual plotting and requires temperature data to be obtained and plotted every 10 minutes as necessary dependent on the rate of the temperature change. Cooldown or heatup rates are also required to be calculated and recorded every 15 minutes.

- Procedure RP-6B now provides the operator with caution statements containing both the Technical Specification and PBNP administrative limits for heatup and cooldown rates. These caution statements appear prior to each heatup and cooldown step. The procedure has also been updated to include clarification of the requirements for actions to be taken if the administrative rate limit has been exceeded. If the administrative rate limit of 50°F/hour is exceeded, the operator is required to immediately stop the heatup or cooldown, stabilize the RCS temperature, and contact the assigned test coordinator for determination of reportability.

- PBNP 3.4.19, "Infrequently Performed Tests or Evolutions/Special Test Procedures," requirements will be applied to Procedure RP-6B. This requirement will be used to ensure that following significant revisions, the procedure will be validated by use of the simulator prior to performing the evolution. In addition, this requirement will ensure that a pre-evolution briefing is conducted, a second licensed operator is assigned to monitor the heatups and cooldowns, and that an independent line management senior reactor operator is assigned by the plant manager as the test coordinator. The test coordinator has the responsibility for oversight of the performance of the crevice cleaning evolution. The requirement to apply

PBNP 3.4.19 has been added as a prerequisite in the initial conditions of RP-6B.

- This revision of procedure RP-6B has received a technical review by qualified individuals from the Operations Group and by qualified individuals from the Chemistry Group.

c) Technical Specification 15.3.1.A.3, "Components Required for Redundant Decay Heat Removal Capability," has been evaluated for possible revision. The evaluation addressed the current requirement for RHR system flow at all times when the RCS is greater than 140°F, or when the RCS is less than 350°F with no reactor coolant pumps operating. Based on our review, we have concluded that a Technical Specification revision is warranted. This revision will specify RHR system operational requirements and clarify the "test" exception and associated basis. This Technical Specification revision will be submitted no later than February 28, 1993, as part of our ongoing Technical Specification Upgrade Project.

d) An engineering analysis of the temperature transient was initiated on the afternoon of May 27, 1992. A linear elastic fracture mechanics analysis was performed using the Babcock & Wilcox computer program PCRIT. The engineering evaluation was based on ASME Section XI, Appendix E, Article E-1300. The engineering analysis was performed using incore thermocouple temperature readings because these readings are believed to be most representative of the temperature transient experienced by the beltline region of the reactor vessel. The temperature transient as measured by the incore thermocouples was approximately 141°F during the one hour transient period.

The results of this engineering analysis concluded that the Appendix E acceptance criterion was met throughout the event. The engineering analysis resulted in a minimum ratio of fracture toughness to total calculated stress intensity factors of 1.27. Since this minimum ratio remained above the 1.0 ratio acceptance criterion, the analysis demonstrated that the structural integrity of the reactor vessel beltline region is assured and that acceptable margins of safety will be maintained during subsequent operations.

On July 3, 1992, a follow-up evaluation of the temperature data for the cooldown event was completed. Based on this evaluation, a decision was made to conduct an additional engineering analysis. The "B" cold leg wide range temperature monitor recorded a temperature transient of approximately 167°F during the one hour transient period. The "B" cold leg wide range temperature monitor is located adjacent to the RHR inlet and the auxiliary charging inlet nozzles. Because of the location of the cold leg wide range monitor and associated environment at this location due to the RCS configuration for steam generator crevice cleaning, these temperature readings are not believed to represent the transient experienced by the reactor vessel. The incore thermocouple readings are assumed to provide the most accurate representation of the temperature transient experienced by the reactor vessel.

Although the "B" cold leg monitor readings were not believed to best represent the reactor vessel transient, a decision was made to perform an engineering analysis using this temperature data to ensure that engineering analyses were bounded by the most conservative temperature transient data. A second engineering analysis was performed using the "B" cold leg wide range temperature [illegible]a. This analysis resulted in a minimum ratio of fracture toughness to total calculated stress intensity factor of 1.21. This result remained above the Appendix E acceptance criterion.

As a result of these analyses, it is concluded that the structural integrity of the vessel is assured and that acceptable margins of safety will be maintained during subsequent operation.

D. <u>CORRECTIVE ACTIONS THAT WILL BE TAKEN</u>

The following corrective actions are planned to be taken.

a) As previously stated, a review of the need to continue the crevice cleaning evolutions on Unit 1 steam generators will be completed by April 1, 1993, prior to the Spring Unit 1 refueling outage.

b) Our Plant Process Computer System (PPCS) will be enhanced to include heatup and cooldown rate alarms and improve the heatup and cooldown rate calculations to eliminate the potential confusion caused by the high instantaneous indications. These enhancements to the PPCS will be completed by December 31, 1992.

c) Training will be conducted on Procedure RP-6B, "Steam Generator Crevice Cleaning." This training will consist of both classroom and simulator training. Classroom and simulator training will also be conducted on the thermodynamics of the RCS when in the configuration required by Procedure RP-6B. Training will include the proper instrumentation and method to monitor heatup and cooldown evolutions. All of the above training in support of steam generator crevice cleaning and heatup and cooldown evolution monitoring will be completed by September 25, 1992.

We do not anticipate the use of related procedure RP-6A, "Steam Generator Crevice Flushing (Vacuum Mode)," during the Fall 1992 Unit 2 refueling outage. However, if it becomes necessary to utilize this procedure during subsequent refueling outages, classroom and applicable simulator training will be conducted prior to use. Procedure RP-6A will be revised by October 31, 1992, to require the subject training as a prerequisite condition prior to use of the procedure.

d) As previously stated, PBNP Technical Specification 15.3.1.A.3.a.(3) will be revised to specify RHR system operational requirements and clarify the "test" exception and associated basis. This Technical Specification change request will be submitted no later than February 28, 1993.

e) The requirements for PBNP 3.4.19, "Infrequently Performed Tests or Evolutions/Special Test Procedures," will be clarified to ensure that it is consistently applied to new or revised procedures. This clarification will be completed by September 25, 1992. In addition, existing procedures will be reviewed to determine if PBNP 3.4.19 should be exercised on the procedure prior to use. The review of the existing procedures will be completed by December 31, 1992.

Requirements contained in PBNP 3.4.19 will be strengthened to require an assessment of the need to perform training prior to each use of the procedure. PBNP 3.4.19 will also be strengthened to require evaluation of the use of the simulator to conduct before-use training when applicable. These enhancements will be added to the procedure by September 25, 1992.

f) Clarification of the requirements to perform a technical review of multi-disciplinary procedures in accordance with PBNP 2.1.1, "Classification, Review, and Approval of Procedures," will be completed by September 25, 1992. The clarification will be included to ensure that appropriate technical reviews are performed and that the individuals performing the review are qualified in each affected area.

E. DATE OF FULL COMPLIANCE

All corrective actions, identified above, which we believe will result in full compliance with NRC requirements have been completed or will be completed by April 1, 1993.

II. VIOLATION NOT ASSESSED A CIVIL PENALTY

"Technical Specification 15.3.1.A.3 requires, in part, that at least one decay heat removal method (one of the reactor coolant loops or residual heat removal loops) shall be in operation when the reactor coolant temperature is less than 350°F and greater than 140°F with one or more fuel assemblies in the core, except when required to be secured for testing."

"Contrary to the above, on November 10, 1991, all residual heat removal and reactor coolant loops were secured with one or more fuel assemblies in the core and reactor coolant temperature was greater that 140°F and less than 350°F, and the loops were not required to be secured for testing."

RESPONSE TO VIOLATION NOT ASSESSED A CIVIL PENALTY

A. ADMISSION OR DENIAL OF THE ALLEGED VIOLATION

We acknowledge that this violation is accurately characterized and agree that the described activities associated with the event constitute a violation of NRC requirements, as defined in Point Beach Nuclear Plant Technical Specifications.

B. REASONS FOR THE VIOLATION

During the PBNP Unit 2 Fall 1991 refueling outage, steam generator crevice cleaning was being performed using Wisconsin Michigan Test Procedure (WMTP) 11.19, Revision 7, "Steam Generator Crevice Cleaning." WMTP 11.19 directed the operator to adjust residual heat removal (RHR) cooling to try to maintain RCS temperature, as measured at the RHR heat exchanger outlet, within the desired range. However, the procedure gave no further guidance on cooldown control methods or requirements for maintaining the decay heat removal system in operation. As a result of the procedural inadequacies, three out of four operating crews accomplished this task utilizing different techniques. One of the crews secured both reactor coolant pumps, as required by the procedure, and secured both RHR pumps to control the cooldown rate. This operating configuration was in violation with PBNP Technical Specifications.

The securing of both reactor coolant pumps and both RHR pumps was evaluated by the operating crew and determined to be acceptable based on their interpretation of PBNP Technical Specifications. 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 in use for the crevice cleaning evolution was titled, "Wisconsin Michigan Test Procedure," the operators interpreted this procedure as applicable to the phrase "...except when required for testing," and believed that securing the operating RHR pumps was acceptable due to being in a test situation. Due to the Technical Specification allowance to temporarily secure both RHR pumps, the cooling action of the crevice cleaning, and the ability to quickly restart RHR flow by use of a control switch, the operators concluded that shutting off both RHR pumps was an allowed and prudent action. The basis for Technical Specification 15.3.1.A.3 contained no guidance on appropriate interpretation of the decay heat removal operational requirements.

Major factors contributing to this event included: 1) inadequate guidance in WMTP 11.19 regarding cooldown control methods, and 2) an interpretable technical specification and lack of guidance in the basis.

C. CORRECTIVE ACTIONS THAT HAVE BEEN TAKEN

a) Between the Fall 1991 Unit 2 refueling outage and the Spring Unit 1 refueling outage, WMTP 11.19, Revision 7, was converted to Refueling Procedure (RP) 6B, "Steam Generator Crevice Cleaning." Procedure RP-6B provided improvements to direct operators to minimize reactor coolant cooldown by operating one RHR pump in accordance with Technical Specification 15.3.1.A.3.a.(3) and by bypassing the RHR heat exchangers.

b) Technical Specification 15.3.1.A.3, "Components Required for Redundant Decay Heat Removal Capability," has been evaluated for possible revision. The evaluation addressed the current requirement for RHR system flow at all times when the RCS is greater than 140°F, or when the RCS is less than 350°F with no reactor coolant pumps operating. Based on our review, we have concluded that a Technical Specification revision is warranted. This revision will specify RHR system operational requirements and clarify the "test" exception and associated basis. This Technical Specification revision will be submitted no later than February 28, 1993, as part of our ongoing Technical Specification Upgrade Project.

D. CORRECTIVE ACTIONS THAT WILL BE TAKEN

This violation not assessed a civil penalty as described in the Notice, along with the May 27, 1992, event and associated violations assessed a civil penalty represents our need to enhance our pre-planning and oversight of steam generator crevice cleaning activities and, in general, other infrequently performed plant evolutions. We believe the corrective actions identified in the "Response to Violations Assessed a Civil Penalty" section of this reply are also applicable to this violation. Completed corrective actions and planned corrective actions identified for the violations assessed a civil penalty will also provide appropriate corrective measures for this violation.

E. DATE OF FULL COMPLIANCE

All identified corrective actions, which we believe will result in full compliance with NRC requirements, have been completed or will be completed by April 1, 1993. We expect the identified corrective actions will place us in full compliance with NRC requirements and prevent recurrence of similar violations.