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NPL 97-0593

10 CFR 2.201

September 29, 1997

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

Ladies/Gentlemen:

DOCKETS 50-266 AND 50-301
REPLY TO A NOTICE OF VIOLATION
ENFORCEMENT ACTION 97-075
INSPECTION REPORTS 50-266(301)/96018(DRS) AND 50-266(301)/97005(DRP)
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In a letter from Mr. A. Bill Beach dated August 8, 1997, the Nuclear Regulatory Commission forwarded Enforcement Action EA 97-075. The enforcement action forwarded a Notice of Violation documenting violations of NRC requirements. The circumstances surrounding these violations are documented in NRC Inspection Reports 50-266(301)/96018 and 50-266(301)/97005. These issues had also been discussed at a predecisional enforcement conference in the Region III office on April 9, 1997.

On August 22, 1997, in a discussion with Mr. J. A. Grobe, NRC Region III, we discussed our desire for a 30-day extension to the required response due to our focus on returning Point Beach Nuclear Plant Unit 2 to full power operation and recovering from an extended refueling outage. Mr. Grobe indicated that such an extension would be acceptable. We formally requested the 30-day extension by letter dated August 25, 1997. The requested extension was confirmed by the NRC in a letter dated September 4, 1997, from Mr. M. Satorius, Deputy Director, Office of Enforcement.

We have reviewed the Notice of Violation and, pursuant to the provisions of 10 CFR 2.201, have prepared a written response of explanation concerning the identified violations of NRC requirements. Our written response is included as an attachment to this letter.

New commitments that have not been previously docketed are identified by italics.

If you have any questions or require additional information regarding this response, please contact us.

Sincerely,

Richard R. Grigg
President and Chief Nuclear Officer

Attachment

cc: NRC Regional Administrator, Region III
NRC Resident Inspector

DOCKETS 50-266 AND 50-301
REPLY TO A NOTICE OF VIOLATION
ENFORCEMENT ACTION 97-075
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

NRC letter EA 97-075, "Exercise of Enforcement Discretion", dated August 8, 1997, transmitted a Notice of Violation. Item A of the Notice contained fifteen violations involving our failure to promptly identify and correct conditions adverse to quality. Item B of the Notice contains two violations involving failure to perform adequate safety reviews in accordance with 10 CFR 50.59, "Changes, Tests and Experiments," such that unreviewed safety questions were created when our staff operated the Residual Heat Removal (RHR) and the Auxiliary Feedwater (AFW) systems in a manner that was not described in the Final Safety Analyses Report (FSAR). Item C of the Notice contains four violations involving failure to properly implement plant Technical Specifications (TS) requirements by not correcting inappropriate TS interpretations, failing to perform several tests required by the TS requirements for portions of the emergency power supply system, or perform the tests at the required frequency.

In accordance with the instructions provided in the Notice, our reply to the alleged violations includes: (1) the reason for the violation, or if contested, the basis for disputing the violation; (2) corrective action taken; (3) corrective action to be taken to avoid further violations; and (4) the date when full compliance will be achieved. In accordance with the instructions provided in the Notice, this reply makes reference to previously docketed correspondence as appropriate.

A. **Violations Associated with Breakdown of the Corrective Actions Program:**

10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the conditions is determined and corrective actions are taken to preclude repetition.

1. **Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding the number of transmission lines required during power operation. Specifically, on October 15, 1996, the licensee identified that Technical Specification Interpretation (TSI) 3.1.20 concerning the number of 345-kilovolt transmission lines required during power operation conflicted with Technical Specifications 15.3.7.A.1 and 15.3.7.B.1. The licensee concluded that this TSI should be removed from the Duty and Call Superintendent (DCS) Handbook. However, it had not been removed as of December 12, 1996.**

2. **Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding operation of a pressurizer power operated relief valve (PORV). Specifically, on October 15, 1996, the licensee identified that TSI 3.1.27 incorrectly stated that a PORV remained operable when the control switch was placed to close. The licensee concluded that this TSI should be removed from the DCS Handbook. However, it had not been removed as of December 12, 1996.**

3. **Contrary to the above, the licensee did not identify and promptly correct a condition adverse to quality regarding operation of a safety injection pump. Specifically, in April 1993, the licensee's test results indicated that the 1P-15B safety injection pump, powered from a lightly loaded emergency diesel generator with speed droop set, would run at higher frequency and current, potentially tripping on over current. As of February 1997, this condition had not been corrected.**

4. **Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding reactor trip circuit separation requirements. Specifically, on December 22, 1994, the licensee identified (open item design basis document (DBD) 27-001) that backup reactor**

trip circuits did not meet the safety-related train separation requirements of IEEE-279, "Nuclear Power Plant Protection Systems," as specified in section 7.2, "Protective Systems - Protective Systems Redundancy and Independence," of the Final Safety Analysis Report (FSAR). The licensee's assessment of the impact on system operability was not performed until December 16, 1996.

5. Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding circuit fault propagation. Specifically, on December 22, 1994, the licensee identified (open item DBD 27-002) that a single fault in the nonsafety-related backup reactor trip circuit could propagate into both reactor protection system (RPS) trains and disable the safety-related primary trip function. The licensee's assessment of the impact on system operability was not performed until December 16, 1996.

6. Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding reactor trip setpoints. Specifically, on December 22, 1994, the licensee identified (open item DBD 27-003) that installed instruments of lesser accuracy than accounted for in design calculations could result in nonconservative setpoints for five TS-required RPS trip functions. The licensee's assessment of the impact on system operability was not performed until December 19, 1996.

7. Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding accuracy of the containment condensate measuring system. Specifically, on January 3, 1996, the licensee identified (open item DBD 30-002) that the containment condensate measuring system was less sensitive than the 0.05 gpm value given in section 6.5 of the FSAR. The system may not have the capability to detect a 1 gpm RCS leak within four hours as described in the licensee response to GL 84-04, "SE of Westinghouse Topical Reports Dealing with the Elimination of Postulated Pipe breaks in PWR Primary Main Loops." The licensee's assessment of the impact of the identified insensitivity on system operability was not performed until December 16, 1996.

8. Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding analysis of containment back draft dampers. Specifically, on January, 3, 1996, the licensee identified (open item DBD 30-003) that the original containment back draft dampers had been analyzed to show that the dampers could withstand the dynamic forces following a loss-of-coolant accident (LOCA). However, replacement dampers that were installed during a previous refueling outage were not explicitly analyzed for their capability to withstand the post LOCA dynamic loads. The licensee's assessment of the impact on system operability was not performed until December 16, 1996.

9. Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding containment shield wall seismic analysis. Specifically, on January 6, 1995, the licensee identified (open item DBD 33-002) that previous calculations lacked evidence that a seismic analysis was considered in the original plant design for containment shield walls, intermediate concrete slabs and support steel. The licensee assessment of the impact on system operability was not performed until December 11, 1996.

10. Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding accident analysis. Specifically, on May 15, 1995, the licensee identified (open item DBD 35-002) that main feedwater flow would be lost immediately during a small break LOCA instead of the two seconds assumed in a licensing basis accident analysis. The licensee's assessment of the impact on system operability was not performed until December 13, 1996.

11. Contrary to the above, the licensee had identified but did not promptly correct a condition adverse

to quality regarding switchgear fault currents. On March 30, 1993, the licensee identified that fault currents for twenty-eight 4160-volt and 480-volt switchgear, including safety-related switchgear, could be larger than the demonstrated capability of the equipment. The licensee assessment of the impact on system operability was performed on April 2, 1993; however, as of December 12, 1996, the licensee had not implemented corrective action.

12. **Contrary to the above, the licensee did not promptly correct a condition adverse to quality regarding an operability assessment. Specifically, on December 19, 1996, as part of corrective actions for an NRC-identified error in a previous calculation, the licensee completed a prompt operability assessment for the loss-of-voltage relays associated with the reactor coolant pump under voltage trips using an incorrect trip breaker trip time. The 0.084-second trip time utilized for the assessment was not in accordance with procedure nor demonstrated to be statistically valid.**

13. **Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding evaluation of electrical fault propagation. Specifically, on June 9, 1993, the licensee identified that current limiting devices on safety-related inverters may not prevent a fault in one circuit from affecting other circuits. The licensee initiated an evaluation of the need for cable rerouting or the installation of current limiting fuses; however, completion of the evaluation was not prompt in that it was extended several times and was scheduled to be completed by April 15, 1997.**

14. **Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding an operability determination. Specifically, on June 23, 1994, the licensee documented in Justification for Continued Operation (JCO) 94-03, that some Unit 2 nonsafety-related cables of redundant trains were routed in the same raceways, possibly creating a common mode failure. It was concluded that the probability of such a fault was unlikely and the breakers would isolate the fault. However, the JCO did not examine the effect of losing DC buses. On January 13, 1997, during JCO review, the licensee identified that a fault associated with redundant, nonseparated cables for the Unit 2 rod drive motor generator could create a fault current greater than the thermal overload interrupts capability of the associated breakers. This could ultimately lead to the loss of the automatic closure of the Unit 2 main steam isolation valves and the automatic initiation of an engineered safety features actuation signal.**

15. **Contrary to the above, the licensee had identified but did not promptly correct a condition adverse to quality regarding containment penetration leak testing. Specifically, on October 14, 1996, the licensee identified that four spare containment penetrations (two for each unit) had not been leak tested (since 1985) in accordance with Appendix J of 10 CFR 50 and TS 15.4.4.I. However, corrective actions were not implemented promptly in that the Unit 1 penetrations were not tested until January 10, 1997.**

This is a Severity Level III problem (Supplement I)

Response

These examples, in the aggregate, as violations of 10 CFR 50, Appendix B, Criterion XVI, represent shortcomings in the corrective action process as implemented at the Point Beach Nuclear Plant. The examples document multiple instances where the corrective action process failed to identify, evaluate and correct, in a timely manner, potentially degraded or non-conforming conditions at PBNP. Our response to each specific example is provided below. Following our responses to the specific examples, under "Generic Considerations," is an assessment and discussion of initiatives underway to address these concerns as they relate to the corrective action process as a whole. These actions include a continued emphasis on conformance to NRC regulations and conditions of the PBNP license, a low threshold for identification of potential concerns, a redesign of our corrective action process organization, as well as training to improve our evaluations and determination of root cause.

Reply to Violation A Example 1:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

PBNP Technical Specification 15.3.7 serves two purposes. The first is to ensure two sources of power to an operating unit thus meeting 10 CFR 50, Appendix A, GDC 17, as evaluated for PBNP in NRC Safety Evaluation dated August 29, 1983. The second is to maintain continuity of unit operation if, under abnormal circumstances, the unit is connected to the offsite grid via one offsite line. In this case, the operating unit is limited to 50% power to provide reasonable assurance that the unit will remain critical and capable of self supporting operation if the remaining line is lost.

The non-conservative interpretation that is the subject of this violation, would have allowed the affected unit to remain at full power when connected to the offsite grid via one offsite power line. This configuration could occur during certain specific switchyard configurations that may be used to facilitate maintenance on the offsite power lines or the switchyard.

Wisconsin Electric personnel identified this non-conservative Technical Specification Interpretation as a result of reviews committed to during our September 12, 1996, pre-decisional enforcement conference related to enforcement action EA 96-273. The results of this review and its recommendations were not acted on in a timely manner, resulting in the interpretation remaining active, despite the identified non-conservatism. Adequate controls were not in place to ensure non-conservative Technical Specification Interpretations were corrected in a timely manner.

Corrective Action Taken:

Revision 3 to DCS 3.1.20, "Offsite Power Operability," was issued on June 27, 1997. This revision removed the non-conservatism contained in the interpretation.

Corrective Action To Prevent Recurrence:

Management continues to stress verbatim compliance with the Technical Specifications.

Administrative procedure NP 5.1.4, "Duty and Call Superintendent Handbook," has been revised to clarify the standards for Technical Specification Interpretations. The guidance specifically prohibits interpretations which contradict or change the wording, meaning or intent of any requirement. If a Technical Specification Interpretation is determined to be necessary, the interpretation will be temporary only. Interpretations will be canceled when conditions warrant, or until the appropriate Specification and/or bases is changed or clarified via the mechanisms provided by 10 CFR 50.90 and 10 CFR 50.59 as appropriate.

Date of Full Compliance:

We are presently in compliance for this example.

Reply to Violation A Example 2:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

Technical Specification 15.3.1.A.5 was implemented as a result of amendments requested in response to Generic Letter 90-06, "Power-Operated Relief Valve and Block Valve Reliability, and Additional Low-Temperature Overpressure Protection for Light-Water Reactors Pursuant to 10 CFR 50.54(f)." In the course of implementation, questions arose as to the requirements for operability of the power-operated relief valves (PORVs), and therefore, the appropriate application of the new Specifications.

The PORVs, as originally designed, installed and analyzed for PBNP were nonsafety-related. The PORVs were not considered necessary for accident mitigation. The safety-related function of the PORVs was to remain closed, thereby constituting part of the reactor coolant system pressure boundary during operation. Protection of the reactor coolant pressure boundary from overpressure is a function of the pressurizer safety valves when RCS temperature is greater than the Low Temperature Overpressure Protection (LTOP) enable temperature. Present safety analyses as documented in the FSAR maintain these assumptions.

The PORVs are credited for automatically relieving system pressure during operation in the LTOP mode of operation. This is considered a safety-related function of the PORVs.

During certain testing and surveillances, it is necessary to place the PORV control switch to close, thus defeating the automatic operation of the valves. Since, during operation at reactor coolant temperatures above the LTOP enable setpoint, the safety analyses in the FSAR do not credit the PORVs for automatic pressure relief, the PORVs can be considered operable under these conditions. However, during the LTOP mode of operation neither the Final Safety Analysis Report, nor the Technical Specification Bases allow substitution of operator action for this automatic function. Therefore, with the control switch in close during LTOP operation the PORVs are inoperable. DCS 3.1.27 incorrectly concluded that the PORVs remained operable for LTOP under this condition.

Our interpretation of Technical Specification 15.3.1.A.5 incorrectly reached this conclusion due to an inadequate questioning attitude resulting in a non-literal interpretation of the Technical Specification. The interpretation was not revised in a timely manner due to inadequate follow through on the recommendations of the evaluation.

Corrective Action Taken:

Technical Specification Interpretation DCS 3.1.27, Revision 1, was issued on June 27, 1997. This revision explicitly refers to the requirement of Technical Specification 15.3.15 as governing PORV requirements during LTOP operations.

Corrective Action To Prevent Recurrence:

Management continues to stress the importance of a questioning attitude and literal compliance with the Technical Specifications and other regulatory requirements. As discussed in our response to Example 1 above, the procedure controlling the Technical Specification interpretation process has been clarified to explicitly prohibit an interpretation that would change or contradict the meaning, intent or wording of any Technical Specification. Continued emphasis by management on conservative decision making will provide reasonable assurance that corrective actions followup occurs in timely manner.

Date of Full Compliance:

We are presently in compliance for this example.

Reply to Violation A Example 3:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

Under original PBNP design, emergency diesel generators (EDGs) G-01 and G-02 were equipped with mechanical governors which regulated the speed of the diesel engine and, hence, the output frequency of the attached generator, under varying load conditions. Each governor was set with a "speed droop" characteristic, which resulted in an engine speed/generator frequency which decreased with increasing EDG load. The purpose of the speed droop characteristic was to prevent EDG overload while the generator was operated in parallel with the electrical grid during monthly surveillance testing. However, the presence of speed droop could also result in EDG output frequencies significantly above the nominal value of 60 Hertz (Hz) when the generator was operating but not tied to the electrical grid. Operation of certain motor-driven loads, including pumps and fans, at frequencies above their nominal ratings can result in increased motor current draw. This is due to the fact that, at elevated frequencies, the pump or fan rotates faster, resulting in increased flow and an increased power demand on the prime mover (i.e. the motor). Increased motor power output corresponds to increased input current. Motor operation at increased current levels can result in long-term motor degradation due to excessive heating and can also result in undesired actuation (tripping) of motor overcurrent protective devices such as relays or circuit breakers. Inadvertent motor overcurrent device tripping is a particular concern for safety-significant loads.

At PBNP, at least two instances of unexpected overcurrent device actuation have been attributed to motor operation at elevated EDG frequencies. One instance involved a trip of a motor-driven auxiliary feedwater pump in 1996 (EA 97-075, Violation B, Example 2); the other involved a trip of a high-head safety injection (SI) pump in 1997.

Corrective Action Taken:

As described above, the problem of EDG overfrequency operation due to the presence of governor speed droop originally applied to the G-01 and G-02 EDGs. New EDGs G-03 and G-04, which were installed in the mid-1990's, were provided with electronic load-sharing governors which ensure generator operation at the nominal frequency of 60 Hz under all operating conditions (both islanded and paralleled to the electrical grid).

In 1993, a test was performed which demonstrated the ability of the high-head safety injection pumps to operate without tripping under worst-case flow and overfrequency conditions. In 1996, in response to the auxiliary feedwater pump trip described above, an analysis was completed to demonstrate that inadvertent overcurrent device actuation would not occur for any other safety-related loads, even under worst-case EDG overfrequency conditions. In 1997, the actuating setpoints for the overload alarm relays on the high-head safety injection pumps were raised to prevent unnecessary alarm actuation and distraction to the operators, and potential pump tripping under overfrequency conditions.

Corrective Action To Prevent Recurrence:

An electronic speed governor similar to those installed on EDGs G-03 and G-04 was installed on G-01, eliminating the potential for elevated frequency operation of the EDG. A modification is currently in progress to perform a similar installation for the remaining EDG (G-02).

Date Of Full Compliance:

We are presently in compliance for this example with EDG G01 aligned to supply A train emergency power.

Reply to Violation A Example 4:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

This issue was identified during the identification and consolidation of PBNP design basis information for the reactor protection system. This Design Basis Document was approved in late 1994. These documents undergo a rigorous review and approval process which includes review by appropriate disciplines and validation of the informational content. During this review, this issue was scrutinized and determined not to be an immediate concern. The issue was documented with the Design Basis Document and tracked by the Design Basis group with the intent of resolving the issue prior to the next scheduled update of the associated Design Basis Document. Because these reviews did not identify this condition as an immediate concern and the issue was being tracked, Wisconsin Electric personnel did not recognize that the issue should be handled within the formal corrective action process.

Corrective Action Taken:

Condition Report 96-1784 was generated on this condition, and an operability determination was completed on December 16, 1996. This operability determination showed that no failure of a backup trip circuit could disable the reactor trip function for primary trip parameters. The technical justification for this is described in DBD-27, the Reactor Protection Design Basis Document.

Corrective Action To Prevent Recurrence:

An FSAR change was made in June, 1997, which provides the technical justification for the IEEE-279 exceptions.

Date of Full Compliance:

We are presently in compliance for this example.

Reply to Violation A Example 5:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

This issue was identified during the identification and consolidation of PBNP design basis information in an approved Design Basis Document. These documents undergo a rigorous review and approval process which includes review by appropriate disciplines and validation of the informational content. During this review, this issue was scrutinized and determined not to be an immediate concern. The issue was documented with the Design Basis Document and tracked by the Design Basis group with the intent of resolving the issue prior to the next scheduled update of the associated Design Basis Document. Because of these reviews and the issue being tracked, Wisconsin Electric personnel did not recognize that the issue should be handled within the formal corrective action process.

Corrective Action Taken:

Condition Report 96-1783 was generated on this condition, and an operability determination was completed on December 16, 1996. The evaluation performed as part of this operability determination concluded that no failure mechanism existed that could disable both RPS trains due to a common mode failure in the backup trip circuitry, regardless of how the circuitry is separated in the field. Therefore, no adverse condition exists. The Condition Report and DBD open item have been closed.

Corrective Action To Prevent Recurrence:

There are no additional actions required for this example.

Date of Full Compliance:

We are presently in compliance for this example.

Reply to Violation A Example 6:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

This issue was identified during the identification and consolidation of PBNP design basis information in an approved Design Basis Document. These documents undergo a rigorous review and approval process which includes review by appropriate disciplines and validation of the informational content. During this review, this issue was scrutinized and determined not to be an immediate concern. The issue was documented with the Design Basis Document and tracked by the Design Basis group with the intent of resolving the issue prior to the next scheduled update of the associated Design Basis Document. Because of these reviews and the issue being tracked, Wisconsin Electric personnel did not recognize that the issue should be handled within the formal corrective action process.

Corrective Action Taken:

Condition Report 96-1775 was generated on this condition, and an operability determination was completed on December 19, 1996. This operability determination concluded that the affected protective functions would be accomplished in accordance with the assumptions in the safety analyses.

The Setpoint Verification Program is recalculating setpoints for each primary reactor trip setpoint and will determine the required instrument loop accuracy. Therefore, it will generically address the concern raised in this Condition Report. The Condition Report action will be closed when the Setpoint Verification Program is complete.

Corrective Action To Prevent Recurrence:

No additional action specific to this example is planned.

Date of Full Compliance:

We are presently in compliance for this example.

Additional setpoint verification efforts discussed above will ensure any similar issues are promptly identified, evaluated and corrected as appropriate.

Reply to Violation A Example 7:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

This issue was identified during the identification and consolidation of PBNP design basis information in an approved Design Basis Document. These documents undergo a rigorous review and approval process which includes review by appropriate disciplines and validation of the informational content. During this review, this issue was scrutinized and determined not to be an immediate concern. The issue was documented with the Design

Basis Document and tracked by the Design Basis group with the intent of resolving the issue prior to the next scheduled update of the associated Design Basis Document. Because of these reviews and the issue being tracked, Wisconsin Electric personnel did not recognize that the issue should be handled within formal corrective action process.

Corrective Action Taken:

Condition Report 96-1694 was generated on this condition, and an operability determination was completed on December 16, 1996. An evaluation of the capability of the condensate measuring system was performed by Wisconsin Electric personnel in response to this Condition Report. The conclusion of this evaluation was that the condensate measuring system performance capability has been determined to be within the limits of the leak-before-break criterion as documented for PBNP in an NRC Safety Evaluation dated June 1, 1984. The system is considered a viable leak detection method to fulfill the requirement of TS 15.3.1.D.7.

Corrective Action To Prevent Recurrence:

The FSAR was revised in June 1997, to clarify the requirements for the condensate measuring system.

Date of Full Compliance:

We are presently in compliance for this example.

Reply to Violation A Example 8:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

This issue was identified during the identification and consolidation of PBNP design basis information in an approved Design Basis Document. These documents undergo a rigorous review and approval process which includes review by appropriate disciplines and validation of the informational content. During this review, this issue was scrutinized and determined not to be an immediate concern. The issue was documented with the Design Basis Document and tracked by the Design Basis group with the intent of resolving the issue prior to the next scheduled update of the associated Design Basis Document. Because of these reviews and the issue being tracked, Wisconsin Electric personnel did not recognize that the issue should be handled within the formal corrective action process.

Corrective Action Taken:

Condition Report 96-1781 was generated on this condition, and an operability determination was completed on December 16, 1996. The operability determination concluded that the dampers remained operable. Work is currently being performed to address this condition, and is being tracked as an action item in the corrective action program. Sargent & Lundy has prepared a draft detailed evaluation showing the acceptability of the backdraft damper capability. That evaluation is currently in final review and comment by Wisconsin Electric personnel.

Corrective Action To Prevent Recurrence:

Actions taken for this example provide reasonable assurance this will not recur.

Date of Full Compliance:

We will be in compliance for this example upon approval of the Sargent & Lundy evaluation and any additional

corrective action that may result. Any additional action will be accomplished via the corrective action program commensurate with its importance to safety.

Reply to Violation A Example 9:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

This issue was identified during the identification and consolidation of PBNP design basis information in an approved Design Basis Document. These documents undergo a rigorous review and approval process which includes review by appropriate disciplines and validation of the informational content. During this review, this issue was scrutinized and determined not to be an immediate concern. The issue was documented with the Design Basis Document and tracked by the Design Basis group with the intent of resolving the issue prior to the next scheduled update of the associated Design Basis Document. Because of these reviews and the issue being tracked, WE personnel did not recognize that the issue should be handled within the formal corrective action process.

Corrective Action Taken:

Condition Report 96-1686 was generated on this condition, and an operability determination was completed on December 11, 1996. Bechtel calculation 10447-9611-001 was performed which concluded that seismic loads do not control the design of containment floor slabs and steel or primary and secondary shield walls. Therefore, these structures are adequate to perform their design function during or after design basis or maximum hypothetical seismic event. The Condition Report and DBD open item have been closed.

Corrective Action To Prevent Recurrence:

No further action is necessary specific to this example.

Date of Full Compliance:

We are presently in full compliance for this example.

Reply to Violation A Example 10:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

This issue was identified during the identification and consolidation of PBNP design basis information in an approved Design Basis Document. These documents undergo a rigorous review and approval process which includes review by appropriate disciplines and validation of the informational content. During this review, this issue was scrutinized and determined not to be an immediate concern. The issue was documented with the Design Basis Document and tracked by the Design Basis group with the intent of resolving the issue prior to the next scheduled update of the associated Design Basis Document. Because of these reviews and the issue being tracked, Wisconsin Electric personnel did not recognize that the issue should be handled within the formal corrective action process.

Corrective Action Taken:

Condition Report 96-1753 was generated on this condition, and an operability determination was completed on December 13, 1996. This operability determination included information from Westinghouse that the Small Break

LOCA analysis is insensitive to this assumption. A letter from Westinghouse has been received that formally documents this. Therefore, no adverse condition exists. The Condition Report and DBD open item have been closed.

Corrective Action To Prevent Recurrence:

No further action is required for this example.

Date of Full Compliance:

We are presently in compliance for this example.

Reply to Violation A Example 11:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

A calculation was completed in 1993 by contractor personnel which concluded that under certain conditions, the interrupting capability of certain 4160V and 480V breakers was insufficient to interrupt the worst case three phase "bolted" fault. This potential condition could affect both safety and nonsafety-related switchgear. A Condition Report was initiated (CR 93-137) to further evaluate this concern and take appropriate corrective action.

The conditions required were considered to be an extremely low probability occurrence and sensitive to the input assumptions and given a relatively low priority for evaluation. However, where the calculated overloads were determined to be the most significant, breaker replacements were made for Unit 2 in the Fall of 1993 and for Unit 1 during the Spring of 1994. In March of 1994, as a result of further review, a recommendation was made to evaluate each potentially affected breaker and switchgear. This additional action was given a relatively low priority and was scheduled for completion by June 30, 1996.

During 1993 and 1994, Wisconsin Electric completed significant modifications to the electrical system at PBNP. These modifications added two additional emergency diesel generators and reconfigured the emergency AC electrical distribution system. As a result of these modifications, Wisconsin Electric personnel recognized that the previous calculations would require revision and new calculations performed.

Corrective Action Taken:

An operability evaluation has been completed in accordance with the guidance in Generic Letter 91-18. This evaluation concluded that the affected systems and components remain operable under the identified potentially degraded conditions.

An evaluation of this condition and the effects on the safe shutdown capability of PBNP in accordance 10 CFR 50, Appendix R requirements has been completed. This evaluation and corrective actions necessary are documented in Licensee Event Report (LER) 97-032-00, dated July 30, 1997.

Corrective Action To Prevent Recurrence:

The additional calculations, evaluation and corrective action will be controlled through our corrective action process consistent with importance to safety.

Date of Full Compliance:

We will be in full compliance for this example following the completion of any additional actions identified by the ongoing reviews and evaluations.

Reply to Violation A Example 12:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

The complete Loss of Flow safety analysis documented in the PBNP FSAR, assumes that rod drop will commence within 1.5 seconds following a loss of voltage on non-safeguards buses A01 and A02. This is a primary trip variable. Calculation N95-0095, Revision 0, was completed to analytically verify that rod drop would occur within the 1.5 second assumption.

On December 18, 1996, the NRC OSTI team identified that a non-conservative value was used for the Reactor Trip Breaker cycle time within the calculation. Calculation N95-0095, Revision 0, used a value of 60 msec for the cycle time. A review of historical data from reactor trip breaker testing determined that this value did not bound all actual cycle times.

Based on these concerns, a prompt operability determination was completed demonstrating that the trip time was met with an assumed breaker cycle time of 84 msec. Further review by the NRC team determined that the this cycle time was also non-conservative because this value also did not bound all actual cycle times for this parameter.

These errors resulted from an incomplete review of existing information.

Corrective Action Taken:

On January 15, 1997, calculation N95-005, Revision 1, was approved based on a reactor trip breaker cycle time of 90 msec. This calculation demonstrated that the accident analysis assumptions were met, verifying the conclusions of the earlier operability determination. The results of this calculation were provided to the NRC via letter (NPL 97-0131). Additional reviews subsequently determined that this value, while bounding the majority of the test data, did not bound all existing information. A new calculation, performed by an outside contractor, was approved on June 30, 1997, superseding calculation N95-0095. This calculation assumes a cycle time of 100 msec and verified operability based on meeting the accident analyses assumptions.

Corrective Action To Prevent Recurrence:

Management continues to stress conservative decision-making and the need for thorough review of all relevant documentation with all personnel.

Date of Full Compliance:

We are presently in compliance for this example.

Reply to Violation A Example 13:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

A non-conformance report (NCR N-91-072) was initiated to document the fact that the instrument bus inverters are current limiting devices and may not be able to provide high enough fault currents to clear a fault quickly

enough to prevent a fault in one circuit from affecting other circuits. The non-conformance report was converted in June of 1993, to Condition Report CR 91-072A. Pursuant to this CR, Action Item 3 was initiated to evaluate whether or not adequate separation and isolation existed for all non-safety related loads supplied by the instrument buses.

The evaluation concluded that potential concerns existed and recommended, in 1993, that modifications be evaluated and performed to provide acceptable isolation and separation. This evaluation also recommended that fuses be used for this purpose.

A decision on installing fuses was delayed while waiting for the completion of the 120 VAC coordination study which was being performed by Sargent & Lundy. This study was initially scheduled to be completed in December 1995. However, when a completed coordination study was not received by May of 1996, the coordination study was brought in-house to be completed.

Calculations were completed to determine the acceptability of the intended fuses to provide for adequate coordination. The results of the calculations did not provide the expected results. Due to the limited resources within the evaluations group, a determination was again made in November 1996, to contract out for a 120 VAC coordination study.

Corrective Action Taken:

In December 1996, the OSTI review brought up the lack of separation on the 120 VAC Vital Instrument panels which was documented in Condition Report (CR) 96-1699. Electrical separation was provided for the Unit 2 vital instrument panels by modifications completed in April of 1997. *Separation will be provided for Unit 1 during the upcoming refueling outage. These modifications resolve the initial request of CR 91-072A, which was to provide isolation at the vital instrument panels.*

Corrective Action To Prevent Recurrence:

Controls within the modification process provide reasonable assurance that conditions similar to this will not recur.

Date of Full Compliance:

We are presently in compliance for this condition on PBNP Unit 2. We will be in compliance for this example following completion of modifications during the next Unit 1 refueling outage.

Reply to Violation A Example 14:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI. This occurrence, cause and corrective action are detailed in Licensee Event Report 50-266/97-004-00, dated February 12, 1997.

Reason for Violation:

On January 13, 1997, with Unit 1 operating at 90% power and Unit 2 in a refueling shutdown condition, licensee engineers were reviewing a Justification for Continued Operation (JCO) to support the restart of Unit 2. This JCO had justified plant operation with unreliable molded-case circuit breakers (MCCBs) in the VDC electrical distribution system, based on the belief that there were no credible single failures that could result in simultaneous faults on nonsafety-related circuits supplied from redundant DC trains. Further review of these circuits led to the discovery of a particular fault location that could result in coincidental failures of opposite-train safety equipment. Calculations showed that the magnitude of fault currents at this location would exceed the capability of the thermal elements of the associated MCCBs. Given the unreliability of the magnetic trip element to interrupt such fault current, it was determined that the associated breakers would not perform their required

safety function.

Engineers discovered that nonsafety-related cables downstream of 125 VDC breakers D-22-06 and D-19-09 are routed through several common raceways, including tray CB01. The potential therefore exists for a single initiating event to create simultaneous short-circuit faults on both cables. The maximum fault currents possible at these locations would exceed the maximum operating limits of the thermal trip elements in breakers D-22-06 and D-19-09. Failure of the thermal element along with the documented unreliability of the magnetic trip elements in these breakers could prevent the breakers from clearing their downstream faults and result in the loss of the VDC panels D-19 and D-22 when the upstream supply fuses to those panels open. The deenergization of D-19 and D-22 would result in the simultaneous loss of certain safeguards equipment of opposite trains.

Corrective Action Taken:

Immediately following the identification of the postulated fault in Unit 2, the breaker that feeds the A-train control rod drive motor-generator control circuit (D-22-06) was opened and danger-tagged to eliminate the potential for a common fault to cause failures in both safety-related trains of safeguards equipment.

With respect to the Unit 2 postulated fault between panels D-19 and D-22, the subject circuit breakers (D-22-06 and D-19-09) were replaced with Westinghouse EHD 2020 model breakers. The replacement provides assurance that the magnetic trip elements in both breakers will reliably function in the event of a fault downstream of either breaker.

An engineering review of similar circuit conditions in the VDC System was conducted and resulted in the discovery of only one other potential common mode failure in Unit 1 circuits. The breaker associated with this circuit was opened eliminating the immediate concern. The breaker was subsequently replaced with a breaker providing proper coordination and circuit protection.

Corrective Action to Prevent Recurrence:

No additional action specific to this example is required.

Date of Full Compliance:

We are presently in compliance for this example.

<u>Reply to Violation A Example 15</u>:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." The circumstances surrounding this occurrence, cause and corrective action specific to this occurrence are discussed in Licensee Event Report 50-266/97-003-00, dated February 6, 1997.

Reason For Violation:

The issue of failure to perform testing of spare penetrations was identified during a Wisconsin Electric internal audit and reported by QCR 96-066, "Flanges and Valves on Spare Containment Penetrations May Require Appendix J Testing." The audit identified ten penetrations that had potential for not being local leak rate tested as required by Appendix J. This was identified as a potential non-compliance with 10 CFR 50, Appendix J because the actual configuration of the penetrations was not verified by a field walkdown during the audit.

Review of the installed configuration of PBNP spare mechanical penetrations identified that two penetrations per unit were not being local leak rate tested as required. The remainder of the spare penetrations identified during the audit were found to be welded inside containment and therefore were not required to be local leak rate tested.

The spare penetrations were being tested by containment integrated leak rat ting. The penetrations requiring local leak rate testing have a bolted flange with a Flexitalic gasket inside ment. Since the penetrations would not have been disassembled between integrated leak rate tests, it determined that there as sufficient basis for concluding that leakage through these penetrations remained wit allowable limits and they were, therefore, operable.

Corrective Action Taken:

The Condition Reporting process has been revised so that new Condition Reports are reviewed at a plant morning meeting following review by an active SRO. This level of screening ensures that conditions that have potential to impact operability are identified, prioritized and corrected in a timely manner.

Additionally, PBNP now uses procedure NP 5.3.7 for performing and documenting operability determinations. Per this procedure, one of the types of conditions that should receive a written operability evaluation is a condition that is or may be outside the design basis description in the Technical Specifications, Final Safety Analysis Report, Design Basis Document, or design/purchase specifications. Had this process been in place and used during the resolution of QCR 96-066, the Technical Specification non-compliance and impact on operability would have been identified and promptly corrected upon completion of the evaluation. This would have resulted in the Unit 1 spare penetrations 12b and 30a being tested within 24 hours of completion of the evaluation of QCR 96-066 in October 1996.

Corrective Action To Prevent Recurrence:

As discussed above, steps have been taken to provide reasonable assurance that operability and reportability of potentially non-conforming or degraded conditions are promptly addressed for conditions reported via the PBNP Condition Reporting system.

Date of Full Compliance:

All affected penetrations requiring a local leak rate test have been tested and determined to be operable. We are presently in compliance for this example.

Generic Considerations:

These violations, in the aggregate, represent breakdowns in our corrective action program to identify, evaluate and ensure timely resolution of potentially degraded or non-conforming conditions. Wisconsin Electric has recognized the need and has undertaken initiatives to improve performance in this area.

Violation examples A.1 and A.2 concern non-conservative interpretations of Technical Specifications that were identified during Wisconsin Electric reviews conducted as a result of the previous escalated enforcement action EA 96-273. Wisconsin Electric recognizes the importance of literal compliance with the Technical Specifications, and has clearly communicated management expectations for literal compliance to all personnel. This new compliance philosophy minimizes the need to interpret Technical Specifications such that formal documented interpretations are minimized. In addition, these standards are expected to ensure any non-conservative or non-compliant determinations are promptly corrected.

Seven of the violation examples, A.4 through A.10, associated with breakdown of the Corrective Actions Program pertain to untimely operability determinations for Design Basis Document (DBD) open items. The Design Basis Documents are the result of an extensive voluntary ongoing initiative on the part of Wisconsin Electric personnel to collect the engineering design basis information for essential structures, systems, components and analyses at Point Beach into concise documents for use by personnel involved in operations, maintenance and engineering activities. This initiative was developed to closely follow the guidance in NUMARC 90-12, "Design Basis

Program Guidelines." During the generation of these documents, issues may be identified due to lack of, or incomplete, information. These items are tracked to ensure final resolution and closeout.

At the time of the NRC Operational Safety Team Inspection (OSTI), DBD open items (DBDOIs) for issued DBDs were tracked in the PBNP Nuclear Tracking System (NUTRK) under a DBDOI number if they were not determined to be Condition Reports. The review process discussed in the responses to the specific examples would have been expected to identify those items warranting Condition Reports. There were 94 DBDOIs at the time of this inspection. Full operability / reportability screenings were not given to these DBDOIs since they were not made Condition Reports.

In response to NRC inspectors' concerns, all 94 DBDOIs and 14 draft DBDOIs were reviewed by the DBD group, an active SRO, and a System Engineer in December, 1996, to determine if any operability or reportability concerns existed. 38 Condition Reports were generated from this review and 25 prompt operability determinations were completed. No operability issues were identified. However, one item prompted a 4-hour report to the NRC. This review revealed that: (a) a higher threshold than appropriate had been applied when the DBD group had previously reviewed DBDOIs for Condition Report applicability; and (b) the perspective of an SRO is valuable when reviewing DBDOIs for Condition Report applicability.

Several changes to the DBD open item management process have been put in place since December 1996, to address these concerns with the untimely assessment of impact on system operability for DBDOIs. In addition, the threshold applied by the DBD group when reviewing DBDOIs for Condition Report applicability has been lowered. The process changes include:

- All non-editorial DBDOIs shall receive review by an SRO and System Engineer with the DBD Engineer prior to DBD issuance. This change provides a more comprehensive review to provide assurance that Condition Reports are initiated as appropriate so that an operability or reportability concern is not overlooked. It also ensures that this operability / reportability review will not languish or be overlooked following DBD issuance.
- All DBDOIs that do not become Condition Reports are prioritized for resolution utilizing criteria based on safety and risk significance.
- All DBDOIs that do not become Condition Reports are reviewed every six months to verify appropriate work priority, status, and corrective action.

DBD Program Manual revisions, incorporating these process changes, were completed in April, 1997, and have been implemented. These program changes have resulted in the following:

- Since April, 1997, five new, non-editorial DBD open items have been created and reviewed with an SRO and System Engineer for operability / reportability concerns. This review will continue for all future non-editorial DBDOIs.
- The prioritization of DBDOIs was completed in March, 1997, and resolution of DBDOIs is now being performed based on the assigned priority. This prioritization will continue for future DBDOIs.
- The semi-annual review of DBDOIs was completed on June 19, 1997. This semi-annual review will continue in the future.
- The DBD group has issued over 20 additional Condition Reports since January 1, 1997, with at least two of these conditions resulting in reports to the NRC. This is evidence of a lower threshold being applied by the DBD group for Condition Report identification.

These process changes and their implementation have been discussed with the NRC Senior Resident and at an NRC / Wisconsin Electric management meeting in April, 1997. These changes were also reviewed by NRC staff during an inspection performed to verify the readiness of Wisconsin Electric to restart PBNP Unit 2.

Wisconsin Electric believes that these process changes, coupled with an improved awareness of the appropriate Condition Report threshold, will be effective in preventing recurrence of the problems reflected in these seven (A.4 through A.10) violation examples.

As discussed at the April 9, 1997, enforcement conference, initiatives in the areas of identification, assessment, and correction of degraded and non-conforming conditions have been undertaken as well as steps to ensure effective self assessment of our performance in the corrective action area. The desired outcome of these initiatives is the development and nurturing of a self-assessment culture which identifies, prioritizes, determines root causes, and corrects issues in a timely fashion. Successful implementation of these initiatives will reasonably ensure that conditions as documented in the cited violations are promptly detected, corrective action taken consistent with their importance to safety and are prevented from recurring.

To assess and determine the initiatives to be undertaken, a common cause evaluation was completed with the assistance of outside contractors experienced in the corrective action processes. The common cause evaluation was performed to ensure performance enhancement initiatives were successful in correcting program deficiencies; to identify organizational, programmatic and management issues that were root causes for deficiencies in our corrective action program; and to ensure appropriate, sustainable improvements are implemented. This evaluation was completed on April 28, 1997.

The common cause evaluation determined that the root cause of the programmatic breakdown was inadequate line ownership in the development and implementation of the corrective action program. This had resulted in the line organization not taking sufficient initiative to find and correct existing deficiencies. The common cause evaluation resulted in recommendations for improving our program. These recommendations were in the areas of program/administrative controls, organization, and skills and knowledge.

In the area of program/administrative controls, the prioritization process has been modified to more effectively identify and classify those conditions requiring more immediate and direct attention, including root cause evaluations. The prioritization system uses four basic categories (A, B, C, D) vice the numerical system previously in use which prioritized items on a scale of one to 99. Category A Condition Reports represent the most significant issues. All Condition Reports prioritized A or B, at a minimum, require a root cause evaluation.

The internal organization with responsibility for the day to day operation and maintenance of our corrective action program has been expanded and matrixed within the various functional areas of the Nuclear Power Business Unit. By matrixing these individuals into the functional areas, increased line ownership for the corrective action program is expected.

Human Error and Root Cause training has been conducted for this organization to ensure that the processes are understood by both a vertical and horizontal cross-section of the staff. This increases the effectiveness of the organization as a whole in ensuring the thoroughness and accuracy of the evaluations and resultant corrective actions. In addition, the individuals filling specific positions with the matrix corrective action process organization have received in-depth training in root/common cause analysis techniques. With this approach, expertise developed in the area of root cause analyses can be shared throughout the organization thereby increasing the overall effectiveness of corrective actions taken and the ability to prevent recurrence of identified issues. Additional in-depth training for other members of the Nuclear Power Business Unit is also planned to further broaden the knowledge base of our workforce in root cause analyses.

The threshold for the identification and reporting of issues has been lowered as a result of issues identified during the previous escalated enforcement action and initiatives identified in our Plant For Achievement Of Operational Excellence. This has resulted in a sustained, approximately four-fold increase in the number of Condition Reports (the vehicle by which issues and conditions are identified). This increase in Condition Reports is evidence of a broadened participation of staff in the Condition Reporting process and an improvement in the questioning attitude of the staff when potentially discrepant or non-conforming conditions are identified.

Initiatives have been undertaken to improve the assessment and evaluation of identified conditions. A daily meeting has been implemented at which management representatives review Condition Reports initiated since the previous meeting. At this meeting, management reviews and assigns priority if necessary and ensures ownership of the identified issue is assumed by the appropriate functional areas to ensure evaluation and resolution commensurate with the items' importance to safety.

New procedures have been developed to ensure prompt and thorough operability evaluations for degraded and non-conforming conditions. This guidance closely follows the guidance of Generic Letter 91-18, and implements standards for the timeliness of the evaluations.

To improve the effectiveness of the self-assessment process, thus ensuring the improvements in the process are sustained and evolve as necessary in the future, a new group within the organization has been formed with responsibility for this activity. This group, the Continuous Safety and Performance Assessment Group, is chartered to improved performance through self assessment of programs, processes and methods. The group will emphasize and support self-assessments by individual work groups and will complement Quality Assurance assessment activities.

B. <u>Violations Associated with Inadequate 10 CFR 50.59 Reviews</u>:

10 CFR 50.59(a)(l), "Changes, Tests and Experiments," states, in part, that the holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the Technical Specifications incorporated in the license or an unreviewed safety question.

10 CFR 50.59(a)(2)(i) defines, in part, that a proposed change shall be deemed to involve an unreviewed safety question if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased.

1. **Technical Specification (TS) 15.3.1.A.3.b(1), "Reactor Coolant System -Reactor Coolant Less Than 140F," states in part, with the reactor coolant temperature less than 140F, both residual heat removal (RHR) loops shall be operable except one RHR loop may be out-of-service when the reactor vessel head is removed and refueling cavity flooded, or one of the two RHR loops may be temporarily out-of-service to meet surveillance requirements. Section 9.3.2, "System Design and Operation - Residual Heat Removal," of the final safety analysis report (FSAR) stated that the inlet line of the RHR loops starts at the hot leg of one reactor coolant loop and the return line connects to the cold leg of the other loop.**

   Contrary to the above, during refueling outages between September 1987 and December 12, 1996, the licensee did not comply with TS 15.3.1.A.3.b(1) when they returned RHR flow to the reactor through the core deluge lines instead of the cold leg during reactor cavity flooding with the reactor coolant temperature less than 140F. This rendered both RHR loops inoperable. This created an unreviewed safety question that required prior Commission approval in that the licensee changed the RHR system configuration described in FSAR Section 9.3.2 and the licensee safety analysis concluded that this configuration may increase the probability of a dilution accident.

2. **TS 15.3.4.A, "Steam and Power Conversion System," requires, in part, that when the reactor coolant is heated above 350F the reactor shall not be taken critical unless 1) for Two Unit Operation - All four auxiliary feedwater pumps together with their associated flow paths and essential instrumentation shall be operable and 2) for One Unit Operation - Both motor driven auxiliary**

feedwater (MDAFW) pumps and the turbine driven auxiliary feedwater pump associated with that Unit together with their associated flow paths and essential instrumentation shall be operable.

FSAR Section 10.2, "System Design and Operation - Auxiliary Feedwater System" stated, in part, that after automatic start of the MDAFW pumps, automatic delivery of auxiliary feedwater flow to an affected Unit's steam generators occurs without operator action.

Contrary to the above, as of April 18, 1996, with Unit 1 or Unit 2 critical, the licensee created an unreviewed safety question when they changed the automatic operation of the train A motor-driven auxiliary feedwater system as described in FSAR Section 10.2 to manual operator action without prior Commission approval. The change required operator adjustment of the discharge pressure valve, AF-4012, to prevent flow from exceeding 200 gallons per minute to ensure the MDAFW pump motor would not trip on over current. This rendered the train A MDAFW pumps inoperable and may have increased the consequences of an accident described in the FSAR.

This is a Severity Level III problem (Supplement I)

Response

These examples, in the aggregate, are indicative of problems with the implementation of the requirements of 10 CFR 50.59 in the processes at PBNP. In response to EA 96-273, significant process improvement efforts were undertaken to improve the implementation of the requirements of 10 CFR 50.59 at PBNP. This process improvement effort was not yet complete at the time of identification of the above violations. Information on this effort, and actions taken are addressed in our April 25, 1997, supplemental response to EA 96-273. Process improvements as discussed in our April 25, 1997 supplemental response could reasonably have been expected to preclude this violation. Our response to each specific example is provided below. Additional assessment and action to address concerns represented by these violations in the aggregate, is provided under "Generic Considerations," following our responses to the specific examples.

Response to Violation B Example 1:

We agree that this is an example of a violation of 10 CFR 50.59.

Reason for the Violation:

The circumstances surrounding this violation are discussed in Licensee Event Report 50-266/97-019-00, dated May 2, 1997. This occurrence was attributed to insufficient conservative decision making. As a consequence, it was not recognized that the Technical Specification requirements, specifically the definition of operability as it relates to the RHR system operability, were not being met when operating the RHR system in this configuration. In addition, it has been Wisconsin Electric practice to maintain the PBNP Technical Specifications and Bases content and detail consistent with the original Specifications issued for PBNP. The PBNP Specifications provide less detail than the industry and NRC Standard Technical Specifications. The level of detail in the Specifications and Bases contributed to the need for interpretation and discouraged the submittal of necessary license amendments.

Corrective Actions Taken:

Procedures which allowed operation of the RHR system in this configuration were canceled. Operation of the RHR system in this configuration has been discontinued.

The evaluation that concluded the operation of the RHR system in this configuration was not an unreviewed safety question was canceled by the Manager's Supervisory Staff (onsite safety review committee) on July 1, 1997.

Corrective Actions To Prevent Recurrence:

Management has placed increased emphasis and established clear expectations of verbatim compliance with the Technical Specifications. This will ensure that Technical Specification interpretations are minimized. Existing or new interpretations will be appropriately conservative.

Technical Specification interpretations that have been determined to be non-conservative have been canceled or revised to ensure verbatim compliance with the Specification.

Wisconsin Electric is committed to upgrading the PBNP Technical Specifications by converting to the industry standards. By converting the Technical Specifications, detail in the Specifications and Bases will be developed based on the Standards and the PBNP design and licensing basis that will provide for more complete and succinct controls and limits on PBNP Operation. Work has been initiated on the Technical Specifications conversion project. Development of a formal program plan and schedule to ensure the efficient development and timely submittal of the required amendment requests is in progress. *After the program plan and schedule are finalized we will meet with NRC staff to discuss our plans.* In the interim, the focus on verbatim compliance will ensure requirements are appropriately met.

Date Of Full Compliance:

We are presently in compliance for this occurrence.

Response to Violation B Example 2:

We agree that this is an example of a violation of 10 CFR 50.59.

Reason for Violation:

This violation occurred due to insufficient conservative decision making which did not adequately consider the affects of operator action upon the operation of the Auxiliary Feedwater (AFW) System.

During performance of Operations Refueling Test (ORT) - 3A, Emergency Diesel Generator G02 was supplying power to 480 V Safeguards bus 1B03 via 4160 V safeguards bus 1A05. Auxiliary Feedwater Pump P38A was running supplied by 1B03. P38A ran for approximately six minutes at 280 gpm prior to its supply breaker tripping. The cause of the breaker trip was determined to be running P38A at full flow on a lightly loaded diesel generator. Under light load, the EDG governor controlled frequency at greater than 60 Hz. The increased frequency resulted in increased flow and pressure supplied by the pump. The increased current draw under these conditions subsequently resulted in the breaker trip.

The governors on the A train emergency diesel generators, G01 and G02, were set up to operate with a 4% speed droop. With the governors operating in this mode, the EDGs were set up to supply full load, 2850 kW at 60 Hz. Subsequently, under lightly loaded conditions, these EDGs supplied power at greater than 60Hz.

The motor-driven AFW pumps are started on low-low water level in any steam generator; trip or shutdown of both main feedwater pumps or closure of both feedwater regulating valves in one unit; or a safeguards actuation signal. As described in FSAR Section 10.2, the AFW system motor-driven pumps and discharge valves are configured to automatically deliver flow to the affected unit's steam generators without operator action. However, steam generator level is not controlled automatically when using the AFW system to supply steam generators. Operator action is ultimately required to control flow to prevent overfeeding the steam generators and potentially overcooling the reactor coolant system.

Due to a different governor design, the B train emergency diesel generators operate in the isochronous mode at 60

Hz and therefore, do not experience the same condition.

A dedicated operator was assigned in accordance with approved procedures to control the discharge flow from P38A to 200 gpm. This was intended as an interim measure until permanent corrective action could be taken. The dedicated operator was determined to be acceptable based on the design of the AFW system which requires operator action to control AFW flow following AFW initiation.

On April 18, 1996, Wisconsin Electric personnel completed an evaluation that concluded the use of the dedicated operator did not introduce an Unreviewed Safety Question as defined by 10 CFR 50.59. This evaluation did not appropriately apply and address human factor considerations associated with the use of manual action. Contributing to this violation was a failure to fully understand the distinction between nuclear safety considerations and the regulatory questions posed under 10 CFR 50.59.

Corrective Actions Taken:

Modifications to the EDG governor system to reduce the speed droop characteristics to reduce the potential for a pump trip were performed.

Corrective Actions To Prevent Recurrence:

The governor on train A EDG G01 has been replaced with a new electronic governor that operates in the isochronous mode, thus ensuring power is supplied at a nominal 60 Hz regardless of EDG loading. This eliminates this potential failure mechanism for the P-38A when power is supplied to the train A safeguards buses by this EDG. G01 is presently aligned to supply the A train safeguards buses in both Units.

The same modifications will be performed on train A EDG G02.

Date Of Full Compliance:

We are presently in full compliance for this occurrence based on the elimination of the potential failure mode and restoration of the AFW system to operation as described in the FSAR.

Generic Considerations:

Significant efforts have been undertaken to improve performance in the areas of 10 CFR 50.59 conformance, operability determinations and compliance with PBNP Technical Specifications. Process improvement efforts resulted in a major revision to our 10 CFR 50.59 process and enhancements in procedural guidance contained in NP 10.3.1, "Authorization of Changes, Tests, and Experiments (10 CFR 50.59 and 10 CFR 72.48) reviews. Significant training efforts were undertaken to communicate the revised standards to preparers and reviewers, including the Manager's Supervisory Staff. This enhanced guidance will support higher quality, appropriately detailed, and more consistent evaluations focused on the design and licensing bases of PBNP.

In addition, management expectations on the use of Technical Specification interpretations has been explicitly added to NP 5.1.4, "Duty and Call Superintendent Handbook." This guidance prohibits interpretation that changes the meaning, intent or wording of any Technical Specification.

C. Violations Associated with Inadequate Implementation of Technical Specifications:

1. **10 CFR 50, Appendix B, Crite[illegible] [illegible] I, "Corrective Actions," requires, in part, that measures be establ[illegible]shed to assure that cond[illegible] [illegible]dverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material [illegible]d equipment, and nonconformances are promptly identified and**

corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the conditions is determined and corrective actions are taken to preclude repetition.

a. Contrary to the above, the licensee did not promptly correct a condition adverse to quality regarding an analysis of values in their Technical Specifications. Specifically, around April 1995, the licensee concluded in an analysis that the 480 MWe (gross) value in Technical Specification (TS) 15.3.4.E, below which reactor power must be reduced for an inoperable crossover steam dump system, was not conservative and should be 450 MWe. As a result, TS 15.3.4.E did not accurately specify the lowest function capability or performance level of the crossover steam dump system required for safe operation of the facility. As of December 12, 1996, the licensee did not request an amendment to assure that the TS accurately reflected the minimum power level necessary for safe operation of the facility with an inoperable crossover steam dump system.

b. Contrary to the above, the licensee did not promptly correct a condition adverse to quality regarding Technical Specification relay setpoints. Specifically, on June 14, 1995, the licensee concluded in an analysis that the existing and proposed setpoints for the loss-of-voltage relays in Table 15.3.5-1 of Technical Specification 15.3.5.A did not electrically coordinate when the safety buses were heavily loaded. Consequently, the 480v undervoltage relays may not operate before the 4160 loss-of-power relays. Without load shedding the 480v loads, the potential existed to overload their associated emergency diesel generator during load sequencing. As of December 12, 1996, this condition had not been corrected.

2. Technical Specification (TS) 15.4.6.A.2, "Emergency Power System Periodic Tests - Diesel Generators," requires a test, during reactor shutdown for major fuel reloading of each reactor (annually), to assure that the diesel generator will start and assume required load in accordance with the timing sequence listed in FSAR Section 8.2, "Electrical System", after the initial starting signal.

   Contrary to the above, on the dates listed below for the specified diesel generators, the licensee did not verify that during refueling frequency testing, a safety injection pump and two containment fan cooler motors were properly shed from the buses and restored to operation upon automatic start of the diesel generators.

   a. From 1992 to 1994 and in 1996 for diesel generator G-01
   b. From 1991 to 1994 for diesel generator G-02
   c. In [illegible] for diesel generator G-03

3. Technical Specification (TS) 15.4.6.A.5. requires a monthly test to verify the operability of the emergency diesel generator fuel oil system.

   Contrary to the above, on the dates listed below for the specified diesel generators, the licensee did not verify the operability of the automatic start function of the diesel fuel oil system during monthly testing.

   a. Monthly from January to November 1996 for diesel generator G-01
   b. Monthly from March to November 1996 for diesel generator G-02
   c. Monthly from the Spring of 1995 to November 1996 for diesel generator G-03
   d. Monthly from the Fall of 1994 to November 1996 for diesel generator G-04

This is a Severity Level III problem (Supplement I)

Response

These examples of violations of 10 CFR 50, Appendix B, Criterion XVI, are indicative of lack of sensitivity to the implementation of Technical Specification requirements. Wisconsin Electric began addressing this issue in response to EA 96-273 with reviews to ensure the Technical Specification requirements are appropriately linked to PBNP procedures. A followon review is underway to assess and ensure the adequacy of the implementing procedures. The former, was ongoing at the time these examples were identified. Following our response to the specific examples below, under "Generic Considerations," is additional discussion of actions being taken to address in the aggregate, the concerns represented by these examples.

Response to Violation C Example 1.a:

We agree that this is an example of a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

Upon discovery that the Technical Specifications limits for operability of the cross-over steam dump system were non-conservative, administrative controls were established through a Technical Specification interpretation. This administrative control established appropriate power level reductions such that margins of safety consistent with those established by the Technical Specifications were maintained. We recognized that a change to the Technical Specifications was required. Since administrative controls were instituted which ensured system operability, a request for changing the Technical Specifications was considered a low priority.

Corrective Actions Taken:

Technical Specifications changes were proposed in our Technical Specifications Change Request 196, dated February 12, 1997, as supplemented March 11, 1997. The requested changes were approved and issued as amendments 176 and 180 to Operating Licenses DPR-24 for Unit 1 and DPR-27 for Unit 2, respectively, on August 6, 1997. These amendments authorize removal of the Technical Specification requirements to the FSAR and control under the requirements of 10 CFR 50.59. Implementation is required by June 1998.

Corrective Actions To Prevent Recurrence:

Management has established expectations that the Technical Specifications remain the controlling document for ensuring critical functions and parameters are maintained consistent with the safety analysis. When it is determined that the Technical Specifications are no longer controlling, expectations have been communicated to the plant staff that changes will be requested in a timely fashion.

Date Of Full Compliance:

We will be in full compliance for this occurrence following implementation of the authorized amendments.

Response to Violation C Example 1.b:

We agree that this example is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

In response to a internal QA audit conducted in early 1994, calculations which defined the basis for and the acceptability of the settings for the degraded grid voltage relays installed on the safety-related 4160 volt buses were revised. In completing this action Wisconsin Electric personnel realized that similar calculations did not exist for the loss of voltage relays installed on the same buses. These relays sense the loss of voltage on their respective

The completion of the modifications will restore both units to compliance for this example. The Technical Specification limits remain controlling following the modifications. No amendments to the Technical Specifications will be required.

Continued emphasis by management on compliance with the Technical Specifications will provide reasonable assurance that consideration of the Specifications is appropriately integrated into planning and prioritization of activities at PBNP.

Date Of Full Compliance:

We will be in full compliance for this occurrence by the completion of the Unit 2 1998 refueling outage.

Response to Violation C Example 2:

We agree that this occurrence is a violation of 10 CFR 50, Appendix B, Criterion XVI.

Reason For Violation:

Technical Specification 15.4.6.A.2 requires that during shutdown for major fuel reloading that each EDG be tested under actual interruption of AC power to the engineered safety system buses together with a safety injection signal. This test was conducted to assure that the diesel generator will start and assume required load in accordance with the timing sequence listed in FSAR Section 8.2. The test as performed at PBNP, did not require the loading of the equipment listed in Table 8.2, to the extent practical, on the EDG being tested. The test however, verified the timing sequence presented in the FSAR.

Point Beach Nuclear Plant was originally designed and constructed with two emergency diesel generators shared between the two units. In addition, there are a number of systems, such as Service Water, which are shared between the units. Since one unit is normally operating at power during refueling of the other unit, assumption of all the loads listed in Table 8.2 of the FSAR is not practical in that it may render redundant equipment necessary for the operating unit inoperable.

In addition, the Safety Injection and Residual Heat Removal (low-head safety injection) pumps were originally designed with a minimum recirculation line. Operation of the pumps during this required test on the original minimum recirculation could have resulted in pump damage. Therefore, it was not considered practical to start and run these pumps on the EDG during this test.

Full flow test lines were installed for the Safety Injection and Residual Heat Removal pumps in response to NRC Bulletin 88-04. Subsequent to these modifications, it became practical to load these pumps on the diesel generators. However, since it continued to be impractical to load all the FSAR Table 8.2 equipment on the EDGs during this testing due to shared system/operating unit concerns, Wisconsin Electric personnel did not recognize that testing under more realistic conditions was appropriate. This is attributed to an inadequate questioning attitude in the implementation of the Technical Specification requirements.

Corrective Actions Taken:

Testing has been performed on the EDGs as necessary to fulfill the Technical Specification requirements. Testing was completed with acceptable results.

Corrective Actions To Prevent Recurrence:

Management continues to emphasize the importance of a questioning attitude and verbatim compliance with the

4160 volt safety-related bus, open the normal supply breaker to the bus, start the associated emergency diesel generator, and allow closure of the diesel generator output breaker when the diesel comes up to speed and voltage. This evolution normally takes up to 10 seconds due to the time required for the diesel to start and accelerate. It is possible, however, for this transfer of the diesel to occur much more rapidly if the diesel generator is already up and running. In this scenario, the time between loss of the normal source to the bus and the reenergization from the diesel is limited only by the time delay associated with the operation of the 4160 volt loss of voltage relays and relay and breaker operating times. These relays are set to provide a time delay of 0.8 seconds. Technical Specification limits are 0.7 to 1.0 seconds.

In defining the acceptance criteria for the time delay associated with the 4160 volt loss of voltage relays, during the process of creating the calculation mentioned above, Wisconsin Electric personnel realized that one of the functions is to properly coordinate with the loss of voltage relays installed on the 480 volt safety-related buses supplied from the 4160 volt busses. Given a loss of the normal offsite supply, the 480 volt loss of voltage relays must act to strip loads from the 480 volt bus prior to it being reenergized from the 4160 volt bus and the associated diesel generator. Failure to strip such loads could result in diesel overload. Initially this was not thought to be a concern since the time delay settings for these relays are 0.4 seconds (Technical Specification limit is </= 0.5 seconds). Therefore, these relays would operate before the 4160 volt loss of voltage relays and thus before the closing of the diesel output breaker. It was also realized that the voltage on the 4160 volt bus and therefore, the associated 480 volt bus did not decay to zero instantaneously after a loss of supply.

The actual decay time for one of the buses was measured on April 7, 1995, and found to be significantly slower than previously thought. Since the 4160 volt loss of voltage relays are set at a higher voltage than the 480 volt loss of voltage relays on a per unit basis they will start timing out before the 480 volt relays start timing out. It was determined that given a slow enough voltage decay and the scenario that the diesel was already up and running it was theoretically possible for the 4160 volt relays to time out before the 480 volt relays. This could result in the diesel generator output breaker closing and reenergizing the 480 volt bus before load stripping had occurred. On approximately April 13, 1995, Wisconsin Electric personnel decided to add the determination of appropriate settings for the 480 volt loss of voltage relays to the calculation for the 4160 volt loss of voltage relays already being prepared.

Calculation N-94-130 was completed and approved on June 14, 1995. This calculation concluded that given operation of the 4160 volt safeguards relays at the extreme high end of their voltage operating range and at the extreme low end of their time delay operating range, the 4160 V relays could operate before the 480 V loss of voltage relays given they operated at the extreme low end of their voltage operating range and at the extreme high end of their time delay operating range. Given two of the three of each set of these relays would have to operate at these extremes for this scenario to occur and the low probability that the associated diesel would already be running the probability of this scenario occurring is very low.

As a result of the conclusions of calculation N-94-130, modification requests 95-048 and 95-049 were initiated to resolve this low probability potential problem. The modifications will ensure the existing Technical Specification limits remain controlling and no coordination problem will exist. Prioritization and scheduling of these modifications did not adequately account for the need to maintain the integrity of the Technical Specification limits.

Corrective Actions Taken:

Modification Requests 95-048 and 95-049 were scheduled to be completed during the 1997 refueling outages for each unit. Due to revisions in the Unit operating cycles as a result of the extended Unit 2 refueling and steam generator replacement outages, these modifications will be completed during the next outages after the date of this letter.

Corrective Actions To Prevent Recurrence:

Technical Specifications.

Date Of Full Compliance:

We are presently in compliance for this occurrence.

Response to Violation C Example 3:

We agree this is a violation of Technical Specification requirements. The circumstances surrounding this occurrence, cause and corrective actions taken are documented in Licensee Event Report 50-266/96-012-00, dated January 3, 1997.

Reason for Violation:

On December 5, 1996, while comparing emergency diesel generator (EDG) operational readiness test procedures to the PBNP Technical Specifications (TS), and after discussions with NRC inspectors, Wisconsin Electric personnel determined that the existing monthly tests of the EDGs did not adequately test the automatic features of the EDG fuel oil system. This was contrary to Technical Specification 15.4.6.A.5 which requires the EDG fuel oil system to be tested for operability on a monthly basis. A Condition Report was initiated to document this condition.

Corrective Action Taken:

The EDG fuel oil systems for EDG G-02 (Train A) and G-03 (Train B) were successfully tested within the 24-hour time period allowed by Technical Specification 15.4.0.3 to verify operability. Testing was subsequently successfully performed on the fuel oil systems for G01 and G04.

A review of Inservice Test Procedure IT-14, "Quarterly Inservice Test of Fuel Oil Transfer System Pumps and Valves," was also performed. Procedure IT-14 performs quarterly functional tests of the fuel oil transfer pumps (including pump flowrate determination), stroke tests of transfer pump discharge check valves, stroke tests of EDG day tank inlet motor-operated valves, and biennial valve seat leakage tests. This review determined that other required testing to ensure operability was performed.

Corrective Action to Prevent Recurrence;

Technical Specifications Tests TS-81, TS-82, TS-83, and TS-84 have been revised to include the testing of EDG fuel oil system automatic features on a monthly basis, including the fuel oil sump tank pumps for EDGs G-01 and G-02.

Generic Considerations:

Wisconsin Electric recognizes that verbatim compliance with the Technical Specifications is necessary and that rigorous application and conservative interpretation of the surveillance requirements is appropriate to ensure levels of safety are maintained. As a result reviews of the Technical Specifications have been, or will be performed to ensure complete implementation of the Technical Specification requirements.

In response to Enforcement Action EA 96-273, Wisconsin Electric undertook a review of administrative controls implementing or referencing the Technical Specifications to ensure the Technical Specification requirements are appropriately reflected in the administrative controls. This review encompassed approximately 700 plant procedures and concluded that, in general, all Technical Specification requirements were implemented by approved procedures. Potential discrepancies were documented in Condition Reports and are being evaluated and dispositioned within our Corrective Action Process.

This review also determined that the assessment should continue to review the technical adequacy of these procedures in implementing the Technical Specification requirements. This review will cover those surveillances and requirements that go beyond the reviews of instrumentation and logic testing being performed in response to Generic Letter 96-01. This review has commenced and is proceeding on a schedule based on a probabilistic ranking of safety significance. Discrepancies identified during this assessment will be documented and dispositioned via Condition Reports in accordance with approved procedures. These items will be evaluated for operability and reportability and action taken as appropriate.

Root Cause Evaluation 97-07 was conducted to determine the cause and recommend corrective action for failure to perform testing in accordance with the Technical Specification requirements. While the evaluation specifically considered examples C.2 and C.3, it also addressed the generic implications of not performing testing in accordance with the Technical Specification requirements. The evaluation was completed by a team of Wisconsin Electric personnel with review by an outside consultant. This evaluation determined that the root causes of these events were:

- Management philosophy of maintaining vague Technical Specifications in order to facilitate interpretation to support flexibility in addressing specific plant situations led to misunderstanding of the intent of some Technical Specifications.
- A history of discounting industry standards and performing evolutions based upon available time and resources led to a culture where we set our own standards with full belief that "we are doing the right thing."

Contributing factors related to issues of appropriate implementation of the Technical Specification requirements included:

- Management philosophy of performing the minimum testing required.
- Lack of a questioning attitude in the implementation of the requirements.
- Lack of appreciation for literal compliance with the Specifications
- Less than adequate performance of standards.

Corrective actions discussed in relation to Violation 3 as well as Violations 1 and 2 address these root and contributing causes. Wisconsin Electric is committed to converting the PBNP custom Technical Specifications to the Improved Standard Technical Specifications. This will eliminate much of the "vagueness" of the existing specifications as well as providing a more clear and complete basis for the Specifications. This will aid in the literal compliance to the Specifications as well ensuring the appropriate conservatisms are applied in implementation.

The reviews discussed above of the Technical Specifications and their implementation will ensure that the existing requirements are completely implemented and any discrepancies are detected, evaluated and corrected in a timely manner. This review will essentially rebaseline our compliance providing a strong base for continued, critical self-assessments and continued compliance with the requirements.