Wisconsin Electric
POWER COMPANY

Point Beach Nuclear Plant
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NPL 97-0131

10 CFR 2.201

April 2, 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
NRC INSPECTION REPORT NOS. 50-266/96018 AND 50-301/96018
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In a letter from Mr. John A. Grobe dated March 3, 1997, the Nuclear Regulatory Commission forwarded the results of an Operational Safety Team Inspection (OSTI) at our Point Beach Nuclear Plant which was concluded on February 7, 1997. The inspection report included a Notice of Violation which identified one apparent violation of Technical Specification requirements, an apparent violation of 10 CFR 50 Appendix B, Criterion V, an apparent violation of 10 CFR 50 Appendix B, Criterion III, and an apparent violation of 10 CFR 50.73.

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.

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

Sincerely,

for

Scott A. Patulski
Site Vice President

Attachment

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cc: NRC Regional Administrator, Region III
NRC Resident Inspector

DOCKETS 50-266 AND 50-301
REPLY TO A NOTICE OF VIOLATION
NRC INSPECTION REPORT NOS. 50-266/96018 AND 50-301/96018
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

During an Operational Safety Team Inspection (OSTI) commenced on December 2, 1996, and completed on February 7, 1997, four violations of NRC requirements were identified. Each of the violations were classified as Severity Level IV. Inspection Report 50-266/96018; 50-301/96018 and the Notice of Violation (Notice) transmitted to Wisconsin Electric on March 3, 1997, provide details regarding each violation.

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.

Violation 1:

Technical Specification 15.6.8.1 requires that the plant be operated and maintained in accordance with approved procedures.

a. Operations Manual (OM) 3.1, "Main Control Room Environment, Conduct and Access," Revision 5, states that "watchstanders are expected to monitor instrumentation, including computer screens, at frequent intervals consistent with plant conditions and evolutions in progress."

   Contrary to the above, on December 4, 1996, the inspectors observed that the Unit 1 reactor operator failed to monitor instrument 1FI-477, "Steam Generator Feed Water Flow Indication (Yellow)" during a period of about two hours after completion of reactor protection analog testing.

b. Test procedure TS-82, "Emergency Diesel Generator G-02 Monthly Technical Specification Surveillance Test," Revision 47, directed operators to watch for fluid discharge from diesel generator test ports during jacking of the engine flywheel one revolution.

   Contrary to the above, on December 5, 1996, the inspectors observed that operators did not watch for fluid discharge from diesel generator test ports during jacking of the G-02 engine flywheel one revolution.

Response to Violation 1 Example a:

Reason For Violation:

We agree that this example is a violation of Technical Specifications. The event occurred as described in the Notice of Violation. The reason for the incident was a particular lack of attentiveness by the operator and a lack of appropriate standards of conduct. Existing plant procedures and training have not sufficiently emphasized the need for observable behaviors as evidence of proper control room conduct.

Corrective Action Taken:

The individual involved in the incident was interviewed and acknowledged his poor attentiveness. He was counseled with respect to attentiveness. No further counseling with the individual was decided to be necessary.

Plant management has recognized the need for improvement in plant operations, and improved standards have been under development. On March 25, 1997, the Plant Manager issued a "Plan to Improve Conduct of Operations." This plan describes specific action in four critical areas to improve the performance of the Operations Group. The four focus areas include: resources, communications, shift turnover, and standards. Examples of specific actions include: the establishment of protocol for control room entry, improvement of control room environment during turnover, and development of an Operations Group staffing plan.

To better emphasize the need for attentiveness and formal control room decorum, plant management has directed the Operations Group to develop observable standards for control room conduct. With the input of industry consultants, a benchmarking trip to another nuclear station and examples of standards from another plant, the Operations Group has drafted a new procedure called OM 1.1, "Conduct of Plant Operations." This procedure was developed from industry best-practices and INPO Good Practices. Training on this procedure commenced on March 27, 1997.

Corrective Action To Prevent Recurrence:

Expectations and standards are being formally established in OM 1.1. The objective of these standards is to improve watchstanding and to ensure that plant status is appropriately monitored.

OM 1.1, "Conduct of Plant Operations," Revision 0, will detail the methods and practices to be used to monitor and control the plant to ensure safe, reliable operation of the nuclear units. This procedure will also detail the minimum standards that must be satisfied when performing activities directly related to watchstanding and operation of plant equipment and systems. Specifically, this procedure requires the designation of one licensed operator (Control Operator) to be the "Operator at the Controls," with specific, observable duties that include:

(a) Conducting at least three (3) walkdowns of the full control board during the shift.
(b) Conducting a control board walkdown of affected instruments after the completion of testing activities that might affect control board indicators.
(c) Scanning control boards, alarm panels and computer screens at least every 30 minutes. Scanning shall be conducted while standing.

Date of Full Compliance:

OM 1.1 will be approved by April 16, 1997, and fully implemented no later than May 1, 1997. Operations management will continue reinforcement of these expectations to ensure that performance continues to improve and that behaviors meet the expectations.

Response to Violation 1 Example b:

Reason For Violation:

We agree that the procedure was not followed as written. Personnel conducting the test have not historically complied verbatim with the procedure. Rather, they have used their judgment to conduct the test-port inspections after the jacking operation; knowing that it would be impossible for the assigned personnel to observe all 20 ports simultaneously during the jacking operation. They also believed that the intent of the test procedure would be satisfied if the inspection was conducted after the jacking operation. None of the test personnel proposed a change to the procedure to ensure that the procedure conformed with actual practice.

We have reviewed the basis for the test-port inspections in the procedure and have found no reason to require that the ports be inspected during the diesel jacking operation. Any fluid that is discharged during the jacking is evident one step later in the procedure when the port covers are replaced. Inspecting the ports during the operation would require many observers, which was not the intent of the procedure. In fact, the test-port inspections for emergency diesel generators G-03 and G-04 are required by procedure to be conducted after the step which jacks the engine.

Corrective Action Taken:

For the particular procedure step cited in the Notice, temporary changes were made to procedures TS-81 and TS-82, "Emergency Diesel Generator G-01/G-02 Monthly Tests," to allow the test-port inspection requirement to occur after the jacking operation, rather than during the jacking operation. The revised procedure is consistent with the existing inspection method used for the G-03 and G-04 emergency diesel generators, and allows for verbatim compliance. These temporary changes were made on March 7, 1997.

Corrective Action To Prevent Recurrence:

As discussed previously, OM 1.1 will be approved by April 16, 1997, and will be implemented by May 1, 1997. At that time, we will have further assurance that operators have sufficient standards with respect to verbatim compliance with procedures. This standard requires that, if a procedure cannot be followed as written, the activity will be stopped, the equipment will be placed in a safe condition, and the Duty Operating Supervisor (DOS) will be notified for further direction. The standard requires that incorrect or inadequate procedures are corrected prior to the continuation of work.

Date Of Full Compliance:

Appropriate standards will have been established when OM 1.1 is approved and fully implemented no later than May 1, 1997. Operations management will continue to monitor performance and reinforce these standards to ensure we remain in compliance.

Violation 2:

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states that "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

a. Contrary to the above, the inspectors identified that from July 24 to December 16, 1996, Operating Instruction OI-100, "Adjusting SI Accumulators Level and Pressure," did not provide appropriate instructions in that cross-connected accumulators were not considered inoperable.

b. Technical Specification 15.4.2.B. requires, in part, that inservice inspection of American Society of Mechanical Engineers (ASME) Code Class 1, Class 2, and Class 3 components be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda.

   ASME Code, 1986 Edition, Section XI, Table IWF-2500-1, required a VT-3 visual inspection on component supports that includes the support up to the building structure.

   ASME Code, 1986 Edition, Section XI, paragraph IWA-2213(a) stated that "The VT-3 visual examination shall be conducted to determine the general mechanical and structural condition of components and their supports, such as the verification of clearances, settings, physical displacements ..."

   Contrary to the above, the inspectors determined that as of December 13, 1996, procedure NDE 754, "Visual Examination (VT-3) of Nuclear Power Plant Components," Revision 3, lacked acceptance criteria to verify the acceptable clearances between the building structure and the pipe support baseplate.

c. Contrary to the above, the inspectors determined that around October 1996, emergency operating procedures were revised to include inappropriate instructions on adjusting auxiliary feedwater (AFW) flow to 200 gallons per minute in less than 250 seconds during an accident to prevent a trip of motor-driven AFW pump motor breaker. Specifically, operator actions were incorporated into caution statements and not made specific steps in the procedures.

Response to Violation 2 Example a:

Reason for the Violation:

We agree that the Technical Specifications and Operating Instruction OI-100, "Adjusting SI Accumulators Level and Pressure" could be more conservative in prescribing the compensatory action to take when safety injection (SI) accumulators are cross-connected. Although there was a documented basis for the existing compensatory actions (as discussed below), we agree that this basis may no longer be appropriate. The reasons that the Technical Specifications and OI-100 allowed cross-connected SI accumulators for a one-hour period are described below.

Years before the issuance of NRC Information Notice IN 96-31, "Cross-Tied Safety Injection Accumulators," Wisconsin Electric self-identified the potential safety significance of cross-connected safety injection (SI) accumulators. In 1989, a condition report (NCR 89-210) was initiated to determine if adequate SI requirements and core cooling requirements could be achieved with cross-connected SI accumulators. At that time, SI accumulators were routinely cross-connected when equalizing the water levels and the nitrogen cover gas pressures between accumulators. Because there was no existing Technical Specification Limiting Condition for Operation (LCO) for the cross-connected condition, the condition report was initiated to ensure that design basis accident-mitigation functions could be achieved in this condition.

The condition report concluded that cross-connected SI accumulators would not provide the degree of core cooling required in the accident analysis. However, the evaluation did conclude that cross-connected SI accumulators would provide better core cooling response than would a single isolated SI accumulator under the limiting conditions (i.e., a reactor coolant rupture occurring in the loop with the operable accumulator would result in no accumulator injection from the accumulators because the contents of the operable accumulator would be discharged out the break and the other accumulator would be isolated). At that time, TS LCO 15.3.3.A.2.a allowed one accumulator to be isolated for a one-hour period. On that basis, the condition report recommended a TS amendment to extend the existing LCO to include allowance for cross-connecting SI accumulators.

In response to the condition report recommendation, Wisconsin Electric submitted a license amendment request in 1990. These specifications, including the bases, were evaluated by NRC Safety Evaluation Report (SER) dated May 20, 1993, and approved via License Amendments 139/143. These documents indicate that the process of cross-connecting SI accumulators was explicitly reviewed, submitted, evaluated and approved. The restriction for cross-connected accumulators was determined to be acceptable because it was as restrictive as the "General Considerations" LCO in TS 15.3.0, and the 1-hour period is consistent with Standardized Technical Specifications.

As described in the Inspection Report, Operating Instruction OI-100, "Adjusting SI Accumulators Level and Pressure", Revision 5 provided guidance for cross-connecting SI accumulators and referred to the one-hour LCO. In that respect, OI-100 appropriately implemented the approved Technical Specifications. Based on the preceding rationale, it was our position that the TS provision for cross-connected accumulators was appropriate and procedure OI-100 was appropriate in implementing that provision, when necessary.

The NRC issued IN 96-31 on May 22, 1996 to identify the potentially unanalyzed condition that may be created when SI accumulators are cross-connected, citing another plant's conclusion that its plant would not be

protected if accumulators are cross-tied for some loss-of-coolant-accidents. The cited plant had procedures which allowed cross-connection and had Technical Specifications that require the condition. In evaluating this IN and the cited violation, we have determined that the industry has established a prudent precedent in prohibiting the practice of cross-connecting SI accumulators.

Corrective Actions Taken:

To ensure that prompt corrective action was taken, Temporary Information Record Sheet 96-138 was issued on December 16, 1996, to place tags on the control board to prohibit the practice of cross-connecting SI accumulators. In addition, Operating Instruction OI-100 has been revised (Revision 6 dated December 27, 1996) to prohibit cross-connecting both SI accumulators through the nitrogen inlet valves and/or the normal fill valves whenever the accumulators are required to be operable. In addition, operators were trained on this prohibition during Training Cycle 97-1, which concluded on March 3, 1997.

Corrective Actions To Prevent Recurrence:

Applicable sections of Technical Specification 15.3.3 will be reviewed to assure that Technical Specification provisions clearly prohibit cross-connecting SI accumulators. If necessary, a license amendment will be submitted to the Commission to incorporate changes to the Technical Specifications.

Date Of Full Compliance:

Based on the OI-100 revision to prohibit cross-connecting SI accumulators, we are presently in full compliance with 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings."

Response to Violation 2 Example b:

Reason for Violation:

We agree that procedure NDE-754, "Visual Examination (VT-3) of Nuclear Power Plant Components," Revision 3, lacks acceptance criteria to quantitatively verify clearances between the building structure and pipe support baseplates. Although WE believes that procedure NDE-754 has some qualitative acceptance criteria for gaps below baseplates, we acknowledge that the existing guidance is not explicitly identified and is therefore not readily apparent to the procedure users. The development of specific criteria or additional qualitative guidance would be an improvement to our inservice inspection (ISI) program.

Detailed acceptance criteria for clearances between the building structure and the pipe support baseplate have not been previously included in NDE-754 because the baseplate is not within the boundary of the inservice inspection. ASME Section XI, Figure IWF-1300-1 illustrates that the extent of the examination boundary includes the connection (e.g. weld) to the building structure. Figure NF-1132-1 in ASME Section III, Division I - Subsection NF, "Component Supports," illustrates typical examples of jurisdictional boundaries between pipe supports and the building structure. The figure explicitly states that the baseplate, bolts (i.e. expansion anchors), and nuts shall be considered building structure.

In order to put the existing NDE-754 acceptance criteria for plate bearing surfaces in the right perspective, classically designed baseplates need to be differentiated from baseplates commonly used to support various nuclear power plant components. Classically designed baseplates are associated with structural columns. Their purpose is to distribute large structural loads (i.e., tens of thousands of pounds) through bearing to an underlying foundation. The governing load in classically designed baseplates is compression. In contrast, component baseplate assemblies are used to support secondary (i.e., non-structural) loads. Examples of these include pipe supports, conduit supports, and cable tray supports. The typical load for these assemblies is a

fraction of the load seen by column baseplates, and the governing load case is tension. The main concern in component baseplates is catastrophic failure of the expansion anchors

Baseplates supporting secondary loads are typically analyzed by assuming the plate to be a rigid beam spanning between the expansion anchors. Reactions, shears and moments are calculated using simple statics. Concrete compression under the plate is rarely considered in the design of the baseplate because, in order to account for the support attributed to concrete, a finite element analysis would have to be done; modeling the concrete as a one-way spring. The bearing surface between the concrete and the baseplate is not an important design parameter for component baseplate assemblies. Thus, any loss of bearing due to a gap is of no concern because support from the underlying concrete was not part of the original design.

10 CFR 50, Appendix B, Criterion V requires that instructions, procedures, or drawings include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. ASME Section XI rules require a mandatory program of examinations, testing, and inspection to evidence adequate safety. Section XI covers individual components and complete power plants that have met all of the requirements of the Construction Code. The purpose and emphasis of Section XI is to inspect components for visual indications of degradation caused by operating (i.e. inservice) loads. The rules associated with Section XI are not intended to be used to reconcile the as-designed component with the as-built component.

From an inservice inspection perspective, what is important are those baseplate attributes that indicate baseplate distress (i.e. yielding) or degradation of the baseplate attachment to the supporting structural member (e.g. welds or expansion anchors). In accordance with 10 CFR 50, Appendix B, Criterion V, WE procedure NDE-754 has provided some qualitative acceptance criteria for the clearance between the baseplate and the concrete support surface. Recordable discrepant conditions include:

- Section 9.1.1.1 - Loose parts (bolting, pins, tie wires, etc.), empty holes, or movement that might indicate loose attachments. Where accessible bolting shall be hand checked for tightness.
- Section 9.1.1.8 - Assure that building structure attachments do not show severely chipped concrete or grouting, loose nuts, or broken welds.

Based on the limited and subjective criteria presented above, Wisconsin Electric believes that procedure NDE 754 can be substantially improved by establishing additional acceptance criteria. We acknowledge that there is a general lack of disseminated design information on baseplate gaps. As such, a condition report has been initiated to evaluate acceptance criteria for support baseplate bearing surfaces.

Corrective Actions Taken:

We have initiated Condition Report QCR 97-0004 to evaluate acceptance criteria for support baseplate bearing surfaces. Resolution of the QCR will formally document design information as it relates to gaps. Resolution of the QCR also requires that we formally update all applicable procedures where appropriate.

Corrective Actions To Prevent Recurrence:

The establishment of additional guidance for gaps behind baseplates in NDE-754 will prevent recurrence of the cited condition.

Date Of Full Compliance:

We will be in full compliance when *baseplate acceptance criteria are established by procedure or guideline no later than June 30, 1997.*

Response to Violation 2 Example c:

Reason For Violation:

On April 25, 1996, safety evaluation report SER 96-027 was approved. This SER approved a change to the existing caution statements in the Emergency Operating Procedures (EOPs). The existing EOP caution statement warned that the motor-driven auxiliary feedwater (AFW) pump breaker may trip at an AFW flow rate greater than 320 gallons per minute (gpm). The revision changed the flow rate value in the caution statement from 320 gpm to 200 gpm. This change was based on the discovery that a potentially higher electrical bus frequency on A-Train could cause the A-Train motor-driven AFW pump (P-38A) to produce greater flow rates, which if not limited by operator action, could lead to a pump-motor trip. The present caution statement reads as follows:

Caution: MOTOR-DRIVEN AFW PUMP DISCHARGE FLOW IN EXCESS OF 200 GPM MAY RESULT IN MOTOR BREAKER TRIP DUE TO OVERCURRENT.

The supporting evaluation (SER 96-027) explicitly declared that this caution statement did not direct any operator action. In essence, the evaluation was considering that the operator was already aware that operator action from the control room is necessary following actuation, and this caution statement was a reminder. As described in the evaluation, "auxiliary feedwater is a system which requires "skill of the craft" control for a variety of postulated events to limit the consequences." On that basis, the evaluation considered that the caution statement specifically complied with NUREG-1358, Supplement 1, in that it did not dictate operator action.

Although the basis for the original evaluation appeared appropriate at the time, we agree that this type of caution statement is not an appropriate method for ensuring that required operator actions are taken.

Corrective Actions Taken:

In 1996, we completed an EOP Upgrade Project to ensure compliance with NUREG-1358 guidelines. This NUREG included a guideline to prohibit the inclusion of action steps in caution steps. During the upgrade, several caution statements were removed or revised on this basis.

The inspection report described our original intent to remedy the cited condition by removing the action statements from caution statements and creating distinct steps in the appropriate procedures. Since that time, we have evaluated the motor-driven auxiliary feedwater (MDAFW) pump condition and have established reasonable alternatives to creating distinct action steps to accommodate it. Therefore, we will remove the caution statements from the appropriate procedures and implement other compensatory measures for the condition.

Corrective Actions To Prevent Recurrence:

We will remove the cited caution statements from the Unit 1 and Unit 2 EOPs prior to Unit 2 startup from Refueling 22 (U2R22). These EOPs include:

EOP-0	*"Reactor Trip or Safety Injection"*
EOP-0.1	*"Reactor Trip Response"*
ECA-0.0	*"Loss of All AC Power"*
CSP-S.1	*"Response to Nuclear Power Generation / ATWS"*
SEP-3.0	*"Loss of All AC Power to a Shutdown Unit"*

Date Of Full Compliance:

We will be in full compliance when the appropriate procedure revisions are issued prior to startup from U2R22.

Violation 3:

10 CFR 50, Appendix B, Criterion III requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, the inspectors identified that as of December 18, 1996, design basis information for the reactor trip breaker trip time was not correctly translated into a calculation for the reactor coolant pump undervoltage trip setpoint.

The basis is further described as follows, in the Inspection Report:

"Calculation No. N-95-0095 was performed by the licensee to determine the response time associated with a reactor trip caused by a loss of AC voltage to the 4160-V busses, which was an initiating event assumed in the complete loss of flow accident analysis. This calculation was used to demonstrate the adequacy of the setpoint used for the reactor coolant pump undervoltage (RCP UV) trip..."

On December 18, the inspectors questioned the validity of the input value of 0.06 seconds for the reactor trip breaker trip time used in the calculation. The engineering staff had selected this time based on the longest time of 0.058 second recorded during U1R22 (Unit 1 Refueling 22) reactor trip breaker testing and had recorded this value as conservative. However, the inspectors identified that a value of 0.15 second had been assumed for this parameter in the Accident Analysis Basis Document DBD-T-35, "Loss of Forced Reactor Coolant Flow," Revision 0, for the complete loss of flow accident. The inspectors reviewed additional data for Unit 1 and Unit 2 reactor trip breaker trip times recorded during the 1995 and 1996 outages, and identified a breaker with a 0.0733 second trip time, which confirmed that the assumed value of 0.06 second was inappropriate and nonconservative.

On December 19, the licensee completed a prompt operability determination for the loss-of-voltage relays associated with the RCP UV trip, and concluded that the relays were operable. This determination was based on an assumed value of 0.084 second for reactor trip breaker trip time, which yielded a 1.474 seconds total delay time for the RCP UV trip, which was less than the 1.5 seconds assumed in the accident analysis (FSAR Table 14.1.8-1). The operability evaluation stated that the 0.084-second trip time was the maximum allowed by procedure RMP 26, "Reactor Trip and Bypass Breaker Maintenance," Revision 14. However, the inspectors identified that the maximum time allowed by the procedure was 0.167 second. This disparity prompted the engineering staff to commit to change the trip time in RMP 26 to 0.084 second. . . . The 0.084-second trip time

was not bound by procedure nor demonstrated to be a statistically bound value, and thus the inspectors concluded that the use of this number to demonstrate operability was inappropriate and inadequate."

Response to Violation 3:

Reason For Violation:

We agree that this violation occurred as described in the Notice. The cited calculation used an engineering input that was not bounded by all historical data. This was caused by an incomplete review of the operating history of the reactor trip breakers and a failure to evaluate and document the statistical basis for the input. Subsequent review of reactor trip breaker operating history confirmed that 0.090 second (90 milliseconds) is an appropriate calculation input from a statistical viewpoint.

This violation was caused by failure to adequately implement the established PBNP calculation program, and is not indicative of a program failure. In assessing the conclusions of our previous audits, assessments, and inspections of the PBNP calculation program, we have not identified a significant trend in the use of non-conservative or statistically inappropriate calculation inputs. Therefore, we have determined that the use of a non-conservative input in this instance was the result of poor judgment in implementing the existing program.

Corrective Actions Taken:

On March 25, 1997, the Senior Project Engineer - Electrical conducted a presentation to the Nuclear Engineering Section in Milwaukee. This presentation focused on the appropriate use and documentation of calculation inputs. This presentation was also given to the site engineering group on April 2, 1997.

A thorough review of historical testing data was conducted and demonstrated that 90 milliseconds is a statistically bounded value for PBNP reactor trip breaker trip time. Calculation N-95-095 was revised to include a maximum reactor trip breaker opening time of 90 milliseconds rather than the previous value of 60 milliseconds. The 90 millisecond bounding value in the calculation does not bound the maximum limit established in current version of RMP-26 (10 cycles or 167 milliseconds), but it does bound all known historical test data. Following the revision of RMP 26 to provide a 5-cycle acceptance criterion (84 milliseconds), the calculation assumption will bound the test acceptance criterion as well. The revised calculation also includes new discussion of the conservatism in the methods used to determine bus voltage decay time.

Corrective Actions To Prevent Recurrence:

During the inspection, we committed to revise Routine Maintenance Procedure RMP 26 to require a more restrictive reactor trip breaker opening time of 5 cycles (84 milliseconds). We will complete this procedure change prior to the startup of Unit 2 or Unit 1 from their present outages.

To ensure that the basis for the RMP 26 acceptance criterion is most appropriate, *we have initiated an evaluation of alternative calculation methodologies, including use of a less restrictive reactor coolant pump decay emf methodology. Use of an alternative methodology may require a change to the design basis as described in the Design Basis Document (DBD-T-35).*

Date Of Full Compliance:

We will be in full compliance for this occurrence when RMP 26 is revised to provide a 5-cycle acceptance criterion. That revision, in combination with the revision made to calculation N-95-095, and the presentations made to the Nuclear Engineering personnel, will fully address the cited violation..

Violation 4

10 CFR 50.73(a)(2)(i)(B) requires the licensee to submit a Licensee Event Report (LER) within 30 days after discovery of the event for any operation or condition prohibited by the plant's Technical Specifications.

Contrary to the above, the inspectors identified that as of December 20, 1996, an LER had not been submitted for a condition prohibited by the plant's Technical Specifications identified by the licensee on October 14, 1996. Specifically, the licensee identified that two spare containment penetrations (P-12b and P-30a) on each unit had not been "Type B" tested in accordance with Technical Specification 15.4.4 since 1984.

Response to Violation 4:

Reason For Violation:

Potentially reportable conditions at PBNP are first identified in condition reports (CRs). Within 24 hours of receipt, the condition is screened by an SRO-licensed individual to ensure that operability and reportability are evaluated. Within the three following working days, the Regulatory Services group performs a more detailed assessment of the condition and enters a description of the condition into a condition report database record. If a condition is decided to be reportable per 10 CFR 50.73, it is customary that Regulatory Services creates an action item to issue an LER within 30 days of the discovery date. We believe that the failure to implement this last step contributed to the untimely submission of an LER.

On September 12, 1996, condition report CR 96-795 was initiated to evaluate the reportability and operability of a particular condition in the component cooling water (CC) system related to its qualification as a "closed system outside containment." This issue was raised during Quality Assurance (QA) audit A-P-96-23. On September 14, it was determined that this particular issue was reportable via 10 CFR 50.73, but subsequent actions were not complete and definitive. The documented basis for the reportability determination stated that "this may be a violation of the design basis described in the FSAR, and may require a 30-day LER". An action item to issue an LER was not created until September 19, and the scope of the action item was poorly defined.

On September 16, 1996, the same QA audit identified nine spare containment penetrations with blank flanges in which the associated penetration diagrams did not clearly indicate if the flanges were welded or bolted. According to the requirements of 10 CFR 50, Appendix J, no local leak testing is required for spare containment penetrations with welded flanges. However, if bolted flanges or caps are used, then Appendix J Type B or C leak testing is required. At this time, a new condition report was initiated (QCR 96-066) to document the potential missed surveillances, and an action item was created to ascertain whether the penetrations were welded. This condition was determined to be reportable if any of the cited penetrations were not welded. Further evaluation of reportability was deferred to an action item being created under CR 96-795.

On September 19, an action item was created under CR 96-795 to "determine the reportability of any of the issues surrounding the deficiencies identified in audit A-P-96-23 concerning the lack of, or missed Appendix J leak testing." The action item did not specifically state that an LER was required, nor did it clearly identify the containment penetration issues. Therefore, not all of the required LERs were issued. On October 15, 1996, LER 266/96-009-00 was issued for the identified CC system condition being outside the design basis.

On October 14, 1996 (pursuant to QCR 96-066), a review of containment penetration drawings determined that Penetrations P-12b (both units) and P-30a (both units) made use of non-welded (bolted) flanges inside containment, but were not Type B leak tested in accordance with 10 CFR 50, Appendix J. Although corrective actions were recommended to leak test Penetrations P-12b and P-30a and revise the supporting documentation, no action item was created to issue an LER (because reportability had been deferred to the other condition report -- CR 96-795).

During the OSTI, on or about December 20, 1996, the NRC identified that an LER should have been submitted regarding this event. A subsequent discussion on the subject resulted in the detailed re-evaluation of all of the specific items associated with the audit report, and the conclusion was reached that this event was also reportable. On January 9, 1997, it was determined that the failure to Type B test the flanges on penetrations P-12b and P-30a was a violation of Technical Specification 15.4.4, "Containment Tests." Within 30 days of this determination, LER 266/97-003-00 was issued on February 7, 1997.

Corrective Actions Taken:

LER 266/97-003-00 was issued on February 7, 1997. Corrective actions related to the missed surveillances of the containment penetrations are described in that LER.

Corrective Actions To Prevent Recurrence:

Procedure NP 5.3.1, "Condition Reporting System" will be revised prior to June 1, 1997, to require that an action item be created to clearly document the need and scope of an LER when a reportability determination has concluded the condition is reportable.

Date Of Full Compliance:

Based on the issuance of LER 266/97-003-00, we were in full compliance with 10 CFR 50.73 on February 7, 1997.