April 28, 2000

Mr. M. Reddemann
Site Vice President
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH INSPECTION REPORT 50-266/2000003(DRP); 50-301/2000003(DRP)

Dear Mr. Reddemann:

This refers to the inspection conducted on February 29 through April 1, 2000, at the Point Beach Nuclear facility.

The two units operated normally and there were no safety significant performance issues identified during this inspection period. Operators continued to be attentive to plant status and their assigned duties. However, an example of maintenance staff not promptly entering a degraded material condition issue into the corrective action program was identified. Several other examples of the failure to document degraded or nonconforming equipment conditions had been identified during previous inspection periods. None of these equipment problems resulted in any in-service failures and the Maintenance Department was developing broad corrective actions at the conclusion of this inspection period.

Based on the results of this inspection, the NRC has determined that three violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations consistent with Section VII B.1.a of the Enforcement Policy. These Non-Cited Violations are described in the subject inspection report. If you contest the violations or severity level of these Non-Cited Violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region III, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you choose to provide one, will be placed in the NRC Public Electronic Reading Room (PERR) link at the NRC homepage, http://www.nrc.gov/NRC/ADAMS/index.html.

Sincerely,

/RA/

Roger Lanksbury, Chief
Reactor Projects Branch 5

Docket Nos. 50-266; 50-301
License Nos. DPR-24; DPR-27

Enclosure: Inspection Report 50-266/2000003(DRP);
50-301/2000003(DRP)

cc w/encl: R. Grigg, President and Chief
Operating Officer, WEPCo
M. Sellman, Senior Vice President,
Chief Nuclear Officer
R. Mende, Plant Manager
J. O'Neill, Jr., Shaw, Pittman,
Potts & Trowbridge
K. Duveneck, Town Chairman
Town of Two Creeks
B. Burks, P.E., Director
Bureau of Field Operations
J. Mettner, Chairman, Wisconsin
Public Service Commission
S. Jenkins, Electric Division
Wisconsin Public Service Commission
State Liaison Officer

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-266; 50-301
License Nos:	DPR-24; DPR-27
Report No:	50-266/2000003(DRP); 50-301/2000003(DRP)
Licensee:	Wisconsin Electric Power Company
Facility:	Point Beach Nuclear Plant, Units 1 & 2
Location:	6610 Nuclear Road Two Rivers, WI 54241
Dates:	February 29 through April 1, 2000
Inspectors:	F. Brown, Senior Resident Inspector R. Powell, Resident Inspector N. Shah, Reactor Inspector
Approved by:	R. Lanksbury, Chief Reactor Projects Branch 5 Division of Reactor Projects

EXECUTIVE SUMMARY

Point Beach Nuclear Plant, Units 1 & 2
NRC Inspection Report 50-266/2000003(DRP); 50-301/2000003(DRP)

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 5-week inspection period of resident inspection.

Operations

- Conduct of operations, including control room and auxiliary operator activities, was considered good. Overall, the plant operators were focused on operational safety parameters and plant status. (Section O1.1)

- The licensee identified that the Unit 1 containment personnel air lock was inoperable without applicable action statements being implemented within the times specified by the Technical Specifications. The licensee attributed this status control problem to human performance. One Non-Cited Violation was identified. (Section O8.1)

Maintenance

- Plant maintenance staff observed foreign material inside a safety-related motor-operated valve (MOV) limit switch housing. The foreign material was the result of an interference within the MOV and was observed prior to performance of maintenance activities. This as-found condition was not promptly documented or completely corrected, nor was an extent of condition evaluation for similar MOVs performed until the MOV failed a post-maintenance valve stroke test. Previous foreign material induced MOV failures and previous examples of the failure to enter degraded or nonconforming equipment into the corrective action program had been documented in NRC inspection reports in the preceding 6 months. The Maintenance Department was developing broad corrective actions at the conclusion of the inspection period. One Non-Cited violation was identified. (Section M1.1.)

Engineering

- The inspectors concluded that the licensee's emergency diesel generator surveillance tests satisfied the plant's current licensing basis. However, while evaluating the licensee's design controls for demonstrating emergency diesel generator and safety-related switchgear functionality under accident conditions, the inspectors asked questions that led the licensee to identify two examples of an inadequate design control. One Non-Cited Violation was identified. (Section E1.1)

Plant Support

- The radiological protection controls and work practices for the Unit 1 steam generator primary-side inspection were good. (Sections E 2.1 and R1.1)

- The inspectors observed poor radiological controls for a leaking primary sample system valve. Specifically, a catch containment had been removed prior to the valve leakage being stopped. (Section R1.2)

Report Details

Summary of Plant Status

Unit 1 entered the inspection period in a mid-cycle shutdown to investigate indications of a loose part in the "A" steam generator and to perform other corrective maintenance. The unit was restarted on March 4, 2000. Unit 2 remained at full power throughout the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 Control Room Activities and Auxiliary Operator Walkdowns

a Inspection Scope (Inspection Procedure (IP) 71707)

The inspectors reviewed routine control room conduct and auxiliary (station) operator activities. This inspection consisted of observations in the Units 1 and 2 control rooms and during plant walkdowns with operators. Specific procedures reviewed included Operations Manual 2.15, Revision 7, "Operations Organization and Responsibilities," and Operations Manual 1.1, Revision 5, "Conduct of Operations."

b. Observations and Findings

The conduct of operations in the control room was considered good. Station procedures and three-way communications were properly used to control major activities, including a Unit 1 restart. Reactor operators (ROs) were focused on plant parameters. Control room roles and responsibilities were clearly stated and shift supervision was observed performing periodic panel walkdowns. Shift turnover briefings clearly covered plant status. Planned or ongoing work, infrequently performed tasks and, as applicable, industry operating experience were stressed. For example, during the March 15, 2000, afternoon shift briefing, operations supervision reviewed Operations Manual 3.4, Revision 1, "Operations Self-Checking Expectations," with the operators.

The control room activity level increased during the inspection period due to the presence of a training crew. The normal operating crew was typically augmented by two RO trainees and two Senior Reactor Operator (SRO) trainees. The control exercised over both RO and SRO trainees was appropriate. With the exception of one instance, the Duty Operating Supervisors and Duty Shift Superintendents minimized operator distractions and kept the number of people in the control room to a minimum.

On several occasions, the inspectors questioned the ROs regarding changes in plant parameters. Each time, the operators correctly explained the reasons for the change and referenced the appropriate log entries.

The inspectors observed station operators performing walkdowns of the Units 1 and 2 turbine buildings, the primary auxiliary building, and the water treatment buildings. Overall, the walkdowns were well conducted and appropriate communications were maintained with the control room staff. For example, the primary auxiliary building auxiliary operator kept the control room aware of system status while performing Periodic

Check 26, “Recirculate the Boric Acid Storage Tanks,” Revision 6. The inspectors observed that the operators used the appropriate controlled checklists and that plant housekeeping (including radiological conditions) was not a significant impairment.

c. Conclusions

Conduct of operations, including control room and auxiliary operator activities, was considered good. Overall, the plant operators were focused on operational safety parameters and plant status.

O1.2 Quarterly Functional Test of P-32F Service Water (SW) Pump

a. Inspection Scope (IP 71707)

The inspectors observed the quarterly functional test of the P-32F SW pump conducted on March 16, 2000. Additionally, the inspectors reviewed procedure In-service Test (IT) 07F, “P-32F Service Water Pump (Quarterly),” Revision 4.

b. Observations and Findings

An SRO trainee conducted a pre-job briefing immediately prior to the test. The briefing was closely monitored by a licensed SRO. The briefing adequately covered the procedure and emphasized precautions/limitations and lessons learned from previous performance of activities listed in the procedure.

In general, the inspectors determined that the crew’s performance of activities listed in IT 07F was adequate. Two procedural problems, however, were noted. The operating crew correctly identified an error in Attachment E of IT 07F. Specifically, the test involved the north SW flow header, but the attachment referenced the south header. The error was determined to be an obvious editorial error, and was corrected in accordance with the requirements of Nuclear Power Business Unit Procedure (NP) 1.2.3, “Temporary Procedure Changes,” Revision 7. The operating crew did not, however, identify an error on Attachment A of IT 07F. Per the attachment, the SW header flow was to be recorded at Step 5.18 of the test procedure and used for comparison with flow acceptance criteria. Step 5.18 did not, however, require recording the flow in Attachment A nor was a requirement to record the data contained elsewhere in the procedure. The inspectors observed that the crew recorded the data at Step 5.30.1. When questioned by the inspectors, the Duty Operating Supervisor provided a reasonable explanation for the crew’s action, and acknowledged the procedural inadequacy. The inspectors noted that incorrect data could have been obtained at other steps in the procedure, so correcting Attachment A was of potential importance, although it was minor in this case.

On March 20, 2000, the inspectors requested an information copy of Form PBF-0026p, Procedure Feedback, from the licensee’s procedure feedback coordinator. Both NP 1.2.3 and NP 1.1.4, “Use and Adherence of Procedures and Work Plans,” Revision 5, stated that Form PBF-0026p should be initiated for the type of minor technical deficiencies identified above. The coordinator informed the inspectors that he did not have a Procedure Feedback Form associated with the March 16, 2000, performance of Procedure IT 07F. The coordinator informed the inspectors that he would pursue the issue. A Procedure Feedback Form, dated March 20, 2000, was subsequently provided

to the inspectors. The Procedure Feedback Form did not address the two technical deficiencies noted above. This observation was shared with plant management.

c. Conclusions

The licensee adequately performed activities specified in procedure IT 07F. The inspectors were concerned, however, that the licensee's administrative process for correcting minor procedural technical deficiencies was not followed.

O8 Miscellaneous Operations Issues

O8.1 (Closed) Licensee Event Report (LER) 50-266/1999-014-00: Containment Upper Hatch Outer Door Vent Valve Found Open. This LER described a licensee-identified failure to close the upper hatch outer door vent valve following a December 24, 1999, containment entry. An auxiliary operator observed the valve misposition on December 28, 1999. The operator then closed the vent valve and notified the control room. The outer hatch door was considered to be inoperable while the vent valve was open. Technical Specification (T/S) 15.3.4.A.1.d.(1) required that the licensee lock the operable door in the affected air lock within 24 hours, if one door in the containment air lock is inoperable. Although the inner door was closed and operable, the failure to lock the inner door within 24 hours was a condition prohibited by T/S. The failure to lock the inner door as required by T/S 15.3.4.A.1.d.(1) is being treated as a Non-Cited Violation (NCV) (NCV 50-266/00003-01(DRP)), consistent with Section VII.B.1.a of the NRC Enforcement Policy. The LER documented the licensee's corrective actions and was being tracked in the licensee's corrective action program.

O8.2 (Closed) LER 50-266/2000-001-01: Manual Reactor Trip Due to Decreasing Circulating Water Fore Bay Level. Licensee Event Report 50-266/2000-001-00 was discussed and closed in Inspection Report (IR) 50-266/2000001(DRP); 50-301/2000001(DRP). The LER revision corrected statements in the event narrative concerning reactor coolant system temperature and pressurizer level transients after the manual trip. The corrections were necessary because a second review of the post-trip data indicated a primary system cooldown in excess of the anticipated value. The inspectors concluded that the cooldown in excess of the anticipated value was on minimum safety significance. The licensee attributed the cooldown to problems with the condenser steam dump system. Problems with the condenser steam dump control system had previously been identified, and corrective modifications were planned.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Foreign Material in Safety-Related Valve Motor Operator

a. Inspection Scope (IP 71707 and IP 62707)

A safety-related motor-operated valve (MOV) failed a stroke test following modification work. The inspectors observed portions of the licensee's troubleshooting efforts, discussed the condition with involved plant staff, and reviewed the licensee's corrective action program record of the issue.

b. Observations and Findings

The licensee was installing a redundant safety-related SW system MOV on the supply line to a nonsafety-related load. The installation work plan for the modification required removal of the cover for the limit switch housing on the existing SW isolation MOV (valve SW-2816). When the cover was removed, the mechanics noted that control wires were frayed, and that particles of the frayed insulation were loose in the limit switch compartment. The mechanics cleaned the visible foreign material prior to reassembling the limit switch housing on March 16, 2000. The housing was then reassembled without taking action to repair the frayed wires, prevent further wire fraying, or evaluate long-term valve operability and the extent of the condition. Licensee Procedure NP 5.3.1, "Condition Reporting System," Revision 15, Attachment A, "Guidance for Initiation of Condition Reports (CRs)," stated that CRs should be initiated for the degradation or damage of plant equipment that is not the result of normal wear. A CR was not promptly generated for the frayed wire or the foreign material in the MOV limit switch. An independent industry evaluator questioned a maintenance department manager on the handling of the as-found condition. The manager directed the observer to a staff level maintenance quality verification specialist on March 17, but no immediate corrective actions were taken for the observer's concerns.

Subsequently, on March 20, valve SW-2816 failed to stroke closed (its safety-related function) during post-maintenance testing (PMT). While troubleshooting the stroke failure, the licensee found foreign material from the frayed wires in a contact for the MOV limit switch. The licensee attributed the valve stroke failure to this dirty contact. The inspectors observed portions of the trouble shooting performed after the March 20, stroke failure, and were informed of the dirty contact and frayed wire. The inspectors were not informed (at that time) that the frayed wire and foreign material had been identified during the initial MOV inspection on March 16. The licensee maintenance staff documented the dirty contact and the frayed wire in CR 00-0985. This CR identified that a vendor letter dated July 1989, allowed machining of the limit switch housing to prevent interference with the internal wiring. A component engineer stated to the inspectors that at least two other utilities had experienced similar fraying. The CR stated that the contact had been cleaned, the MOV housing machined, and the frayed wires repaired prior to returning valve SW-2816 to service.

On March 24, the maintenance department quality verification specialist (who had been contacted by the independent evaluator) initiated CR 00-1017 to document the as-found foreign material in valve SW-2816. This CR was reviewed by a licensed SRO on March 24, and by the plant management team on March 27. On March 29, the inspectors asked plant management whether an extent of condition review for potential operability concerns had been completed. The inspectors were told that the extent of condition review would be performed as part of the apparent cause review of CR 00-0985. The inspectors will review the licensee's apparent cause evaluation when it is completed.

Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B, requires that conditions adverse to quality be promptly identified and corrected. The as-found presence (prior to performance of maintenance work) of a frayed wire and foreign material within the limit switch housing of a safety-related valve was a condition adverse to quality. Attempting to return the valve to service without correcting the frayed wire and eliminating the source of foreign material was not an effective corrective action. The

failure to promptly document the foreign material and correct its source was a violation of Criterion XVI. This violation is being treated as a Non-Cited Violation (NCV 50-266/2000003-02(DRP); 50-301/2000003-02(DRP)) consistent with Section VII.B.1.a of the NRC Enforcement Policy. Because the SW-2816 valve failed its PMT and the condition adverse to quality was corrected prior to actually returning the valve to service, the inspectors evaluated whether this violation should be treated as a minor violation. Paragraph 3.5.c of the NRC Enforcement Policy states that a violation is not minor if there is a realistic potential for impact on safety. The inspectors concluded that there was a realistic potential for foreign material in an MOV limit switch housing to not cause a PMT failure, but to later cause an in-service failure following a seismic event or even normal thermal and vibrational cycling. Another instance in which internally generated foreign material affected the operation of a safety-related MOV was discussed in Section M1.3 of IR 50-266/99019(DRP); 50-301/99019(DRP). In that case, foreign material in a relay cabinet caused a PMT valve stroke failure, but the source of the foreign material was not identified and corrected until a subsequent in-service failure occurred. Paragraph 3.5.c, and supplemental guidance provided by the NRC Office of Enforcement, further state that violations are not minor if they are not isolated. Prior examples of the plant maintenance staff's failure to initiate corrective actions for conditions adverse to quality within the preceding 6 months were described as minor violations. Examples are found in Section M1.2 of IR 50-266/99019(DRP); 50-301/99019(DRP), and Section M1.2 of IR 50-266/99018(DRP); 50-301/99018(DRP). For the above reasons, this violation was characterized as being an NCV, not a minor violation. The potential scope of this issue was discussed with plant management, who acknowledged the observation. The Maintenance Department was developing broad corrective actions at the conclusion of the inspection period.

c. Conclusions

Plant maintenance staff observed foreign material inside a safety-related MOV limit switch housing. The foreign material was the result of an interference within the MOV, and was observed prior to performance of maintenance activities. This as-found condition was not promptly documented or corrected, nor was an extent of condition evaluation for similar MOVs performed until the MOV failed a post-maintenance valve stroke test. Previous foreign material induced MOV failures and previous examples of the failure to enter degraded or nonconforming equipment into the corrective action program had been documented in NRC inspection reports in the preceding 6 months. The Maintenance Department was developing broad corrective actions at the conclusion of the inspection period. One NCV was identified.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) LER 50-266/2000-002-02; 50-301/2000-002-02: Technical Specification Surveillance Requirement to Verify Emergency Core Cooling System Valve Position not Fully Implemented. This issue was dispositioned in Section M1.1 of IR 50-266/20001(DRP); 50-301/20001(DRP).

III. Engineering

E1 Conduct of Engineering

E1.1 Transient Load Analysis for Emergency Diesel Generators (EDGs)

a. Inspection Scope (IP 92903)

While observing a refueling outage frequency EDG surveillance test, the inspectors had been concerned that the conditions of the test were substantially less challenging than accident conditions would be. Inspection Follow-up Item (IFI) 50-266/99018-03(DRP); 50-301/99018-03(DRP) had been opened to evaluate the significance of this difference. During this period, the inspectors completed their review of this issue.

b. Observations and Findings

As described in Section E1.1 of IR 50-266/99018(DRP); 50-301/99018(DRP), the EDGs at Point Beach are each designed to carry the accident loads on one unit and the hot shutdown loads of the other unit. However, the refueling outage frequency surveillance test performed on the EDGs did not include the non-accident unit loads. This was because one unit was typically at full power when the refueling outage frequency surveillance test was performed on the other unit. The inspectors discussed the practical limitations of increasing the surveillance test loads with the licensee and with technical staff in the Office of Nuclear Reactor Regulation of the NRC. Based upon these conversations, the inspectors concluded that the surveillance tests as they have been performed satisfied the plant's current licensing basis.

Appendix B, Criterion III, "Design Control," of 10 CFR Part 50, required that design basis and regulatory requirements were correctly translated into specifications and instructions. This requirement applied to post-construction design changes such as the modification of EDG governors and the installation of new or different electrical loads or switchgear. The second part of the IFI was a review to determine how the licensee bridged the gap between practicable test conditions and functional performance requirements. The inspectors determined that the only practicable way to ensure that EDGs would successfully perform their required functions under accident conditions following design changes was to evaluate the effect of the change using computational models. These models could predict the maximum and minimum bus voltages, based upon EDG governor and engine response, and the resultant maximum current in-rush for safety-related switchgear and breakers. The inspectors requested a copy of the Point Beach computational analysis for EDG loading under accident conditions.

While evaluating the inspectors' concern, the licensee identified that the most current transient loading calculation for the "A" train EDGs had not been appropriately reviewed, accepted, or placed in the licensee's controlled calculation program. Specifically, Calculation E-09334-369-DG.1, Revision O, had been received by the licensee on November 19, 1998, but review and acceptance had not been completed as of January 20, 2000. The existing calculations of record had not been kept current with changes made to "A" train EDGs during the interim period. This condition was documented in CR 00-0210. While reviewing Calculation E-09344-369-DG.1, the inspectors observed that a sensitivity evaluation for changes to the EDG's electronic governors contained "acceptance" criteria for changes to the "gain" and "reset" values.

The “gain” and “reset” values were used in the calculation to model the response of the EDG to load changes. This response affected the predicted bus voltages, and thus the switchgear and breaker currents. The inspectors questioned the licensee as to whether the most recent changes to the G-01 EDG’s governor was within the “acceptance” criteria of the controlling calculation. The licensee determined that the as-left “gain” and “reset” settings had not previously been evaluated relative to the calculation, and that the as-left “gain” setting was outside the existing sensitivity studies acceptance criteria. This condition was documented in CR 00-1057. The licensee evaluated both of the above conditions and determined that the operability of safety-related equipment was not affected. The inspectors identified no obvious problems with the licensee’s evaluation. The failure to maintain current and controlled safety-related calculations for changes to the “A” train EDGs was a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Controls.” This violation is being treated as an NCV (NCV 50-266/2000003-03(DRP); 50-301/2000003-03(DRP)) consistent with Section VII.B.1.a of the NRC Enforcement Policy. The licensee informed the inspectors that the EDG transient loading calculations for both trains of safety-related emergency power and electrical loads were being revised. This revision was part of a broad electrical system upgrade project. The upgrade project would ensure that as-installed equipment was accurately reflected in a large number of inter-related calculations. The licensee’s priority for the revised calculations was reasonable.

c. Conclusions

The inspectors concluded that the licensee’s EDG surveillance tests satisfied the plant’s current licensing basis. However, while evaluating the licensee’s design controls for demonstrating EDG and safety-related switchgear functionality under accident conditions, the inspectors asked questions that led the licensee to identify two examples of inadequate design control. One NCV was identified.

E2 Engineering Support of Facilities and Equipment

E2.1 Visual Inspection of Steam Generator Hot Leg Channelhead (IP 37551 and IP 71750)

The inspectors observed the performance of activities specified in the engineering work plan for work order (WO) 9924390. This WO covered the visual inspection of the Unit 1 “A” steam generator (HX-001A) hot leg channelhead, and the retrieval of any observed loose parts. The activities specified in the WO work plan were performed by a contractor under the direction of a member of the licensee’s engineering staff.

The inspectors determined that the WO work plan was executed in a safe, timely manner. The inspectors noted that the radiological protection practices of all individuals involved were particularly good. After initial problems with the contractor’s camera equipment, a detailed inspection of the accessible reactor coolant system piping, the interior of the channelhead, the tubesheet surface, and the tube-to-tubesheet welds was conducted. The inspection, and subsequent review of video tapes taken during the inspection, determined that no damage was present. The inspection also did not identify any loose parts.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) IFI 50-266/99018-03(DRP); 50-301/99018-03(DRP): Design Controls Over Changes to the Emergency Diesel Generators and Electrical Distribution Systems. This item is discussed in Section E1.1.

E8.2 (Closed) IFI 50-266/99013-03(DRP); 50-301/99013-03(DRP): Licensee's Evaluation of SW System Silt on Auxiliary Feedwater Pump Operability. The inspectors verified that the licensee received vendor confirmation that the silt in the SW system would not affect the safety function of the auxiliary feedwater pumps. The inspectors concluded that the licensee was making reasonable progress toward completion of an updated SW system monitoring program.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Radiological Support of Activities (IP 71750)

The radiological protection support of the containment entries during the unit 1 forced outage was good. Direct radiological support of the steam generator HX-001A visual inspection was very good.

R1.2 Drip Catch Maintenance

a. Inspection Scope (IP 71750 and IP 71707)

The inspectors observed radiological conditions associated with primary plant systems in the radiologically controlled area.

b. Observations and Findings

During a routine tour on March 13, 2000, the inspectors observed that a drip catch installed under Valve 1SC-938 in the Unit 1 Primary Sample Room was full of liquid. The inspectors noted that the drip catch did not have a tag or identification number on it. Active catches were typically tagged and tracked. The material condition of the valve was observed to be poor, in that an excessive buildup of boric acid was present. The inspectors notified radiological protection personnel of the observed conditions. In response to the inspectors concerns, the drip catch was promptly drained by plant personnel.

On March 15, 2000, the inspectors observed that the drip catch had been removed from under Valve 1SC-938; however, the valve was leaking (dripping) onto the floor. The inspectors notified the licensee of the observed condition. The licensee later informed the inspectors that plant personnel had adjusted the valve threaded joint and believed the leakage had been stopped prior to removing the drip catch.

The inspectors had two concerns regarding the incident. First, the leaking valve had not had an open WO or CR associated with it. The presence of a catch containment and dried boric acid indicated that the leak was more than a minor threaded joint leak that

could be addressed by adjusting the valve "on the spot." This conclusion was supported by the unsuccessful attempt to adjust the valve threaded joint on March 13, 2000. Secondly, a potentially similar radiological control problem was observed in the Unit 2 Primary Sample Room. The inspectors observed that the material condition of several valves on the sample panel, most notably 2SC-989, 2SC-983, and 2SC-961B, was poor in that excess boric acid buildup was present. The inspectors noted that the tray under the sample panel had been removed as part of the drip catch reduction program. The inspectors observed absorbents on the floor under the sample panel, but no active leaks. This observation was also communicated to plant management.

c. Conclusions

The inspectors observed poor radiological controls for a leaking primary sample system valve. Specifically, a catch containment had been removed prior to the valve leakage being stopped.

R4 Staff Knowledge and Performance in Radiation Protection and Chemistry

R4.1 Primary Chemistry Sampling (IP 71750)

The inspectors observed chemistry personnel performing Unit 1 primary water sampling. Overall, the sampling was well conducted. The technicians used good health physics practices during the sampling to minimize contamination spread. Two technicians were involved, one performing the sampling and the other providing independent verification. Both were knowledgeable of the procedures and had prior sampling experience.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on March 31, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Wisconsin Electric Power Company

M. E. Reddemann, Site Vice President
R. G. Mende, Plant Manager
B. J. O'Grady, Operations Manager
V. M. Kaminskas, Maintenance Manager
R. P. Farrell, Radiation Protection Manager
A. J. Cayia, Regulatory Services and Licensing Manager
C. R. Peterson, Director of Engineering
D. D. Schoon, System Engineering Manager

NRC

B. A. Wetzel, Point Beach Project Manager, NRR

INSPECTION PROCEDURES USED

IP 37551	Onsite Engineering
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92903:	Followup - Engineering

ITEMS OPENED AND CLOSED

Opened

50-266/2000003-01(DRP)	NCV	Failure to lock the inner door as required by technical specification.
50-266/2000003-02(DRP) 50-301/2000003-02(DRP)	NCV	Failure to promptly document the presence of foreign material and correct a condition adverse to quality.
50-266/2000003-03(DRP) 50-301/2000003-03(DRP)	NCV	Failure to maintain current and controlled safety-related calculations for changes to the “A” train emergency diesel generators.

Closed

50-266/1999-014-00	LER	Containment upper hatch outer door vent valve found open.
50-266/2000003-01(DRP)	NCV	Failure to lock the inner door as required by technical specification.
50-266/2000-001-01	LER	Manual trip due to decreasing circulating water fore bay level.
50-266/2000003-02(DRP) 50-301/2000003-02(DRP)	NCV	Failure to promptly document the presence of foreign material and correct a condition adverse to quality.
50-266/2000-002 50-301/2000-002	LER	Technical specification surveillance requirement to verify emergency core cooling system valve position not fully implemented.
50-266/2000003-03(DRP) 50-301/2000003-03(DRP)	NCV	Failure to maintain current and controlled safety-related calculations for changes to the “A” train emergency diesel generators.
50-266/99018-03(DRP) 50-301/99018-03(DRP)	IFI	Design controls over changes to the emergency diesel generators and electrical distribution systems.
50-266/99013-03(DRP) 50-301/99013-03(DRP)	IFI	Licensee’s evaluation of service water system silt on auxiliary feedwater pump operability.

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
CR	Condition Report
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
IFI	Inspection Follow-up Item
IP	Inspection Procedure
IR	Inspection Report
IT	In-service Test
LER	Licensee Event Report
MOV	Motor-Operated Valve
NCV	Non-Cited Violation
NP	Nuclear Power Business Unit Procedure
NRC	Nuclear Regulatory Commission
PMT	Post-Maintenance Testing
RO	Reactor Operator
SRO	Senior Reactor Operator
SW	Service Water
T/S	Technical Specification
WO	Work Order