April 3, 2002

EA-02-031

Mr. M. Warner
Site Vice President
Kewaunee and Point Beach Nuclear Plants
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH SPECIAL INSPECTION - NRC INSPECTION REPORT 50-266/01-17(DRS); 50-301/01-17(DRS), PRELIMINARY RED FINDING

Dear Mr. Warner:

Your staff notified the NRC of a potential common mode failure, discovered by the Nuclear Management Company, of auxiliary feedwater pumps at the Point Beach Nuclear Plant. In response to the notification, the NRC conducted a Special Inspection at the facility. The reported potential common mode failure met the NRC Management Directive 8.3, "NRC Incident Investigation Program," threshold for a Special Inspection in that the potential common mode failure could have led to a loss of safety function. The Special Inspection was conducted December 3, 2001, through February 28, 2002, in accordance with Inspection Procedure 93812, "Special Inspection." On February 28, 2002, the NRC discussed with you and members of your staff, by telephone, the results of the Special Inspection. The enclosed report presents the results of that inspection.

This report discusses an issue that appears to have high safety significance. As described in Section 4OA3.1 of this report, your staff identified a potential common mode failure of the auxiliary feedwater pumps due to inadequate operator actions in response to a loss of instrument air. Although your staff identified this issue in November 2001, the inspection identified that inadequate procedure guidance had existed for many years and that there were seven prior opportunities to identify the issue. The failures to provide adequate procedural guidance and to take appropriate corrective actions were both apparent violations of 10 CFR Part 50, Appendix B, Criteria V and XVI. This issue was assessed using the applicable Significance Determination Process and was preliminarily determined to be Red, an issue with high safety significance that may result in additional NRC inspection. This issue is of high safety significance because a common mode failure of auxiliary feedwater pumps would result in substantially reduced mitigation capability for safely shutting down the plant in response to certain transients. Your staff took prompt corrective actions to revise procedures and train operators to address the immediate safety concerns associated with the issue. Additionally, you recently installed backup pneumatic supplies for the recirculation valves to improve the safety of the auxiliary feedwater system design.

Two apparent violations of NRC requirements were identified during the inspection and are being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current Enforcement Policy is included on the NRC's website at www.nrc.gov.

Before the NRC makes a final decision on these matters, we are providing you an opportunity to request a Regulatory Conference where you would be able to provide your perspectives on the significance of the findings, the bases for your position, and whether you agree with the apparent violations. If you choose to request a conference, we encourage you to submit your evaluation and any differences with the NRC evaluations at least one week prior to the conference in an effort to make the conference more efficient and effective. If a conference is held, it will be open for public observation. The NRC will also issue a press release to announce the conference.

Please contact Mr. John M. Jacobson at (630) 829-9736 within seven days of the date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision and you will be advised by separate correspondence of the results of our deliberations on these matters.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for these inspection findings at this time. In addition, please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your responses will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

J. E. Dyer
Regional Administrator

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

Enclosure: Special Inspection Report 50-266/01-17(DRS); 50-301/01-17 (DRS)

See Attached Distribution

<u>Distribution</u>

cc w/encl:

R. Grigg, President and Chief
Operating Officer, WEPCo
R. Anderson, Executive Vice President
and Chief Nuclear Officer
T. Webb, Licensing Manager
D. Weaver, Nuclear Asset Manager
T. Taylor, Plant Manager
A. Cayia, Site Director
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A. Bie, Chairperson, Wisconsin
Public Service Commission
S. Jenkins, Electric Division
Wisconsin Public Service Commission
State Liaison Officer

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Sincerely,
/RA/
J. E. Dyer
Regional Administrator

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

Enclosure: Special Inspection Report 50-266/01-17(DRS);
50-301/01-17 (DRS)

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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:	50-266/01-17(DRS); 50-301/01-17(DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Point Beach Nuclear Plant, Units 1 & 2
Location:	6610 Nuclear Road Two Rivers, WI 54241
Dates:	December 3, 2001, through February 28, 2002
Lead Inspector:	R. Langstaff, Senior Reactor Inspector Mechanical Engineering Branch
Inspectors:	S. Burgess, Senior Reactor Analyst Division of Reactor Safety A. Dunlop, Senior Reactor Inspector Mechanical Engineering Branch G. O'Dwyer, Reactor Inspector Mechanical Engineering Branch R. Powell, Resident Inspector Point Beach Nuclear Plant
Approved By:	J. Jacobson, Chief Mechanical Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000266-01-17(DRS), 05000301-01-17(DRS), on 12/03/2001-02/28/2002, Nuclear Management Company, LLC, Point Beach Nuclear Plant. Special Inspection.

This Special Inspection was conducted by a team of three Region III inspectors, a Region III senior reactor analyst, and a resident inspector. The inspection identified one finding preliminarily of high safety significance (Red) with two associated apparent violations. The significance of this finding is indicated by the color Red using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Findings

Cornerstone: Mitigating Systems

- TBD. Units 1 and 2. The licensee identified a potential common mode failure of the auxiliary feedwater pumps due to operator actions specified in plant procedures. The team identified that procedural guidance provided to operators was inadequate to prevent such a common mode failure. In addition, the team identified that the licensee had seven opportunities, from 1981 through 1997, to identify the problem and take appropriate corrective actions. The failures to provide adequate procedural guidance and to take appropriate corrective actions are both apparent violations of 10 CFR Part 50, Appendix B, Criteria V and XVI.

  This issue has been preliminarily determined to have high safety significance (Red). A common mode failure of the auxiliary feedwater pumps would result in substantially reduced mitigation capability for safely shutting down the plant in response to certain transients. The significance was determined to be high largely due to the relatively high initiating event frequencies associated with the involved transients and the high likelihood of improper operator actions due to the procedural inadequacies. (Section 4OA3.1)

Report Details

Summary of Plant Status:

At the beginning of the inspection period, Unit 1 was being operated at approximately 98 percent power for work associated with the plant process computer system (PPCS). Unit 1 continued to be operated at 98 percent power until December 18, when power was reduced to 30 percent to reduce the potential dose to workers for a containment entry to isolate a small leak on the sensing line for 1PT-420, reactor coolant system (RCS) wide range pressure detector. Unit 1 was returned to 98 percent power on December 19 and to 100 percent power on December 24 after the PPCS modification was accepted for Rated Thermal Power calculation purposes. Unit 1 continued to be operated at or near full power throughout the remainder of inspection period.

At the beginning of the inspection period, Unit 2 was being operated at approximately 98 percent power for work associated with the PPCS. Unit 2 continued to be operated at 98 percent power until December 7, when power was reduced to 92 percent for condenser steam dump testing. Unit 2 was returned to 98 percent power on December 19 and to 100 percent power on December 24 after the PPCS modification was accepted for Rated Thermal Power calculation purposes. Unit 2 was shutdown on February 22, 2002, to meet a Technical Specification action statement regarding a safety injection pump. A rotating assembly for a safety injection pump was replaced and Unit 2 was returned to criticality on February 25, 2002. Unit 2 continued to be operated at or near full power throughout the remainder of inspection period.

4. OTHER ACTIVITIES (OA)

4OA3 Event Follow-Up (93812)

.1 Potential Common Mode Failure of Auxiliary Feedwater Pumps Due To Operator Actions

.a Inspection Scope

The potential common mode failure of auxiliary feedwater pumps, reported by the licensee on November 29, 2001, met the NRC Management Directive 8.3, “NRC Incident Investigation Program,” threshold for a Special Inspection in that the potential common mode failure could have led to a loss of safety function. The team performed inspection activities as specified by the charter for the Special Inspection. The charter was outlined in NRC memorandum from John M. Jacobson to Ronald A. Langstaff, dated November 30, 2001. The charter directed review of the following areas:

- Timeline development relating to contributors and discovery of the potential common mode failure of the auxiliary feedwater (AFW) system due to the loss of instrument air.

- Adequacy of licensee's operability evaluation and immediate corrective actions for addressing impact of the loss of instrument air on AFW.

- Preliminary determination of risk significance.

- Apparent cause of condition resulting in potential loss of AFW upon loss of instrument air.

- Evaluation of pressurizer power operated relief valve (PORV) modifications impact on operational capability in response to loss of feedwater.

- Extent of condition of the adequacy of engineering review of instrument air system, other air operated valves, and failure modes.

- Failure of the original individual plant examination (IPE) to consider AFW recirculation valve function.

.b Findings

One finding involving two apparent violations was identified regarding the potential common mode failure of the AFW pumps due to operator actions. An apparent violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for failure to have adequate guidance in emergency operating procedures to prevent damage to AFW pumps. The second apparent violation was of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," and was identified for failure to promptly identify and correct the significant condition adverse to quality relating to potential common mode failure of AFW pumps. The finding associated with the violations was preliminarily determined to be of high safety significance (Red).

(1) Event Description

The licensee probabilistic risk analysis (PRA) staff identified a vulnerability associated with AFW recirculation valves. The recirculation valves were air operated valves which failed closed upon a loss of instrument air. Consequently, in certain transients, such as a loss of instrument air, a loss of off-site power, a loss of service water, or a seismic event, the flow path via the recirculation lines would be lost due to the recirculation valves failing closed upon a loss of instrument air. Closure of the recirculation valves could result in pump failure under low flow conditions such as when AFW flow was throttled back to control steam generator level or mitigate RCS overcooling.

The PRA staff identified the vulnerability while updating the Point Beach PRA model for internal events. The PRA staff originally considered the vulnerability to be a procedural weakness associated with abnormal operating procedure (AOP) 5B, "Loss of Instrument Air." The original concern was that the steps to restore AFW pump recirculation flow did not occur sufficiently early in the procedure. Condition Report (CR) 01-2278 was initiated on July 6, 2001, to document the concern. The PRA staff continued discussions with operations personnel over the next several months with regards to the vulnerability. In November 2001, the PRA staff completed their internal events modeling and determined that the vulnerability resulted in a substantial increase in risk. On

November 28, 2001, the PRA staff, engineering personnel, and operations personnel met to discuss the significance of the vulnerability and potential courses of action. On November 29, 2001, operations personnel concluded that temporary information tags and operator briefings were necessary to address the vulnerability. CR 01-3595 was initiated to document the increased risk and to address the vulnerability. The NRC was also formally notified (Event Notification 38525) on November 29, 2001. The issue was subsequently reported by Licensee Event Report (LER) 266/2001-005-00, submitted on January 28, 2002.

(2) System Description

Point Beach Nuclear Plant is a two unit site. Each unit has a turbine driven AFW pump (pumps 1P29 and 2P29) which can supply water to both steam generators. Additionally, the plant has two motor driven AFW pumps (pumps P39A and P39B) each of which can be aligned to a steam generator in each unit. The recirculation valves for both the turbine driven and motor driven pumps would open for the initial 45 seconds after pump start and would open on low flow conditions. However, the recirculation valves were air operated valves which failed closed upon a loss of instrument air. The control room had valve position indication for the recirculation valves, flow indication to individual steam generators, and flow indication to the steam generators from each pump. However, the flow element for providing flow indication for each pump was downstream of where the recirculation line branched off from the discharge line. Consequently, the flow indication for each pump would not indicate recirculation line flow.

The AFW recirculation lines were installed, as part of original construction, to ensure the pump would have a flow path to prevent dead-heading the pump, which would damage the pump. Discussions with licensee engineering staff indicated that a pump could be damaged within minutes under insufficient flow conditions due to lack of cooling. The initial lines installed included an orifice that allowed a 30 gallons per minute (gpm) flow rate. This flow rate was determined by the pump vendor, Byron Jackson, to be sufficient to prevent pump damage based on pump heat-up when on recirculation flow. The recirculation lines were subsequently modified in 1988, in response to Bulletin 88-04, “Potential Safety-Related Pump Loss,” to accommodate a greater recirculation flow rate and protect the pump from low flow instabilities.

(3) Procedural Guidance

Emergency Operating Procedure (EOP)-0.1, “Reactor Trip Response,” directed operators to control feedwater flow early in the procedure. Procedure EOP-0.1 was the procedure which operators would use for most transients. Response not obtained (RNO) column step 1.c of the procedure directed operators to reduce feed flow if reactor coolant system (RCS) temperatures were less than 547 degrees (°) Fahrenheit (F) and trending lower. Step 4.b directed operators to control feed flow to maintain steam generator levels between 29 percent and 69 percent. RNO step 4.b directed operators to stop feed flow to intact steam generators if level continued to rise. If instrument air had been lost, damage would occur to the AFW pumps by these operator actions to control feedflow due to the low flow conditions created. The team noted that procedure OM 4.3.1, “AOP and EOP Writers’ Guide,” step 5.4.2 stated, “A caution is used to present information regarding potential hazards to personnel or equipment associated with the subsequent step(s)." The emergency operating procedures steps did not provide any such cautions prior to November 30, 2001.

Portions of the EOP-0.1 steps one and four are illustrated below:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Verify RCS Temperature Control:	Perform the following:
	a. Check RCS wide range cold leg temperatures: • LESS THAN OR EQUAL TO 547° F AND • STABLE	1. IF RCS cold leg temperature less than 547° F AND RCS temperatures are trending lower, THEN stabilize RCS temperature as follows: a) Stop dumping steam. b) Ensure S/G blowdown isolations - SHUT c) IF cooldown continues, THEN control feed flow: 1) Reduce total feed flow. 2) Maintain total feed flow greater than or equal to 200 gpm until level greater than 29 percent in at least one S/G.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	Stabilize S/G Levels:	
	a. Check S/G/ levels - GREATER THAN 29 percent	a. Maintain total feedflow greater than 200 gpm until level greater than 29 percent in at least one S/G
	b. Control feed flow to maintain S/G levels between 29 percent and 65 percent	b. IF level in intact S/G continues to rise, THEN stop feed flow to that S/G.

Based on discussions with licensee engineering staff, the team determined that the time that the AFW recirculation valves would fail closed due to loss of instrument air could vary. The engineering staff had determined that the recirculation valves would begin to drift shut when instrument air header pressure was reduced to 40 pounds per square inch gauge (psig) and would be fully closed at 25 psig. The instrument air header pressure was nominally maintained at 100 psig with some variation due to cycling of air compressors. Based on observations of instrument air header pressure drop between cycling of air compressors, the engineering staff determined that the instrument air head pressure would drop approximately 13.5 pounds per square inch in one minute under normal loads. The engineering staff estimated that the AFW recirculation valves would begin to drift shut approximately six to eight minutes after loss of all air compressors with complete valve closure one to two minutes thereafter. A loss of instrument air due to a leak in an airline versus a loss of air compressors would result in different bleed down rates, depending on the size of the break. Additionally, the instrument air bleed down rate could be faster due to greater demands on the instrument air system in response to the transient.

Based on discussions with operating licensee personnel, the preferred method for controlling AFW flow was by throttling or closing the AFW flow control valves (for the motor driven AFW pumps) or discharge valves (for the turbine driven AFW pumps) rather than securing the pumps. The team noted that Section 14.1.12, "Loss of All AC Power to the Station Auxiliaries," of the original Final Facility Description and Safety Analysis Report (FFDSAR)," stated, "The reactor operator in the control room can monitor the steam generator water level and control the feedwater flow with remote operated AFW control valves." The FFDSAR did not discuss securing AFW pumps as a means to control steam generator levels. Additionally, the team noted that in some loss of instrument air scenarios (e.g., those involving RCS overcooling), the recirculation valves could remain open at the time that operators throttle or close flow control and discharge valves due to remaining air header pressure. However, the recirculation valves would subsequently close due to decreasing air pressure. Consequently, the valves could reposition at a time when an operator's attention would not be directly focused on the AFW pumps.

Operating experience demonstrated that operators would drastically reduce AFW flow within several minutes of pump start due to RCS overcooling under some transient conditions. For example, on June 27, 2001, the Unit 2 reactor was manually tripped due to low and decreasing water level in the Unit 2 circulating water pump bay (reported in LER 05000301/2001-002-00). Due to subsequent low steam generator water levels, the Unit 2 turbine driven AFW pump and both motor driven AFW pumps initiated and began feeding the Unit 2 steam generators. One steam generator in a unit nominally requires 200 gpm feedwater flow for decay heat removal. However, with three AFW pumps running, approximately 800 gpm of feedwater flow - approximately four times the required flow, was provided to the Unit 2 steam generators. Consequently, the reactor coolant system was cooled down at an excessive rate. Approximately three minutes after the reactor was tripped, operators closed either the flow control valves or the discharge valves to stop flow from the motor driven AFW pumps. Approximately four minutes after the reactor was tripped, operators closed the discharge valves from the Unit 2 turbine driven AFW pump stopping all AFW flow to the steam generators. The AFW pumps were not secured until approximately eight minutes after the reactor was tripped when feed flow using main feedwater was partially restored. In this particular event, the AFW recirculation valves were functional because instrument air had not been lost. However, had instrument air not been available, as would happen in transients such as loss of instrument air, loss of off-site power, and loss of service water events, all AFW pumps could have been damaged.

Procedure AOP-5B, "Loss of Instrument Air," provided operators guidance for loss of instrument air. However, the team noted that, during these transients, operators would typically be using emergency operating procedures, such as EOP-0.1, in their initial response to a transient. After plant conditions stabilized, abnormal operating procedures, such as AOP-5B, would be used to restore equipment. The team reviewed procedure AOP-5B and determined that procedural steps were provided to secure open the AFW pump recirculation valves. However, guidance to secure open the valves did not appear until step 1 of Attachment R, "Auxiliary Feed," located on page 36 of the procedure. Operators were directed to Attachment R by step 26 (located on page 14) of the procedure. Step 26 simply directed operators to check plant systems status per attachments A through Z. Consequently, although procedure AOP-5B had steps which addressed the recirculation valves, the team determined that operators would likely damage all AFW pumps by following the emergency operating procedures given the transient timelines described above.

(4) Regulatory Issue Associated With Procedure Guidance

10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. As of November 29, 2001, procedures EOP-0.1 Unit 1, "Reactor Trip Response," Revision 24, and EOP-0.1 Unit 2, "Reactor Trip Response," Revision 23, addressing activities affecting quality, were not of a type appropriate to the circumstances. Specifically, the procedures did not provide adequate guidance to operators regarding the potential to damage AFW pumps while controlling AFW flow upon low instrument air header pressure, which would cause the recirculation valves to fail closed. Because the procedures did not include instructions to ensure the recirculation valves were open, the AFW pumps could be damaged under low flow conditions such as when the flow is throttled back to control steam generator level or to mitigate RCS overcooling. This issue is considered an apparent violation (AV 50-266/01-17-01; 50-301/01-17-01).

(5) Operator Training

The team reviewed licensed operator training lesson plans and simulator scenarios for training conducted prior to November 29, 2001, and interviewed licensed reactor operators and senior reactor operators to evaluate the emphasis placed on the effect of loss of instrument air on AFW operability prior to November 29, 2001. The team determined:

- Operators were trained on and knowledgeable of the "fail-safe" position of air operated valves including the AFW recirculation valves. No emphasis, however, was placed on the consequence of the fail-closed AFW recirculation valves.

- Lesson Plan 2672, "Instrument Air and Service Water Review," outlined training on the loss of instrument air PRA initiating event. The outline addressed the loss of instrument air effect on secondary cooling. The lesson plan stated that the turbine-driven AFW pump would be available for feeding the steam generator - "loss of instrument air had no effect" and that the pressure control valves for the motor-driven AFW pumps fail open upon a loss of pneumatic supply providing a flow path from the motor-driven AFW pumps to the steam generators. The team noted that the training only addressed the forward flow aspect of AFW to feed the steam generators. The training did not address the consequences of the "fail-closed" recirculation valves causing pump damage.

- No simulator training scenario, including loss of offsite power and loss of instrument air, had included the failure of an AFW pump due to loss of recirculation flow. The licensee's training staff informed the team that the simulator, as modeled, would not fail an AFW pump due to low flow conditions as would likely occur in the plant.

(6) Operability Evaluation

The team reviewed the licensee's initial operability determination screen completed by a senior reactor operator on November 29, 2001. The documented basis for system operability was satisfactory completion of required surveillance testing. The team noted that the operability determination screen did not address the potential simultaneous failure of all AFW pumps due to loss of instrument air and procedurally directed operator actions (the specific issue identified by the CR). The site resident inspectors engaged licensee management (duty shift supervisor and operations manager) on the adequacy of the operability determination screen. Licensee management assured the resident inspectors that extensive discussions of system operability were conducted involving both operations and engineering and the operability determination was adequate. After questions by the resident inspectors, the licensee initiated a formal engineering operability determination on November 30, 2001. The team reviewed Revision 1 of the formal engineering operability determination. The operability determination concluded that the AFW system was operable but nonconforming and specified necessary procedural revisions.

(7) Licensee Corrective Actions

The licensee revised procedures EOP 0, “Reactor Trip or Safety Injection,” and EOP 0.1, “Reactor Trip Response,” on November 30, 2001, to provide additional guidance to operators. The foldout pages for both procedures were revised to state:

> IF any AFW pump mini-recirc valve fails shut, THEN maintain minimum flow or stop the affected AFW pump as necessary to control S/G levels.
> - P-38A minimum flow - GREATER THAN 50 GPM
> - P-38B minimum flow - GREATER THAN 50 GPM
> - P-29 minimum flow - GREATER THAN 75 GPM

The above guidance addressed overfilling of steam generators which would, generally, take longer than 10 minutes after the transient initiated. Consequently, under such circumstances, had instrument air failed, it would have likely bled down to the point of failing the recirculation valves shut before operators would have taken actions to drastically control AFW flow. As such, the operators would have had the opportunity, when controlling AFW flow, to observe that the recirculation valves had failed shut.

The licensee PRA staff subsequently identified that operator action to control AFW flow could be required much earlier in a transient due to RCS overcooling before the recirculation valves would shut due to a loss of instrument air. In response to this issue, the licensee revised the foldout page for procedures ECA-0.0, "Loss of all AC Power," EOP 0, and EOP 0.1, on December 20, 2001, to state:

> IF any AFW pump mini-recirc valve fails shut OR annunciator C01 A 1-9, INSTRUMENT AIR HEADER PRESSURE LOW in alarm, THEN monitor and maintain minimum AFW flow or stop the affected AFW pump as necessary to control S/G levels.
>
> - P-38A minimum flow - GREATER THAN 50 GPM
> - P-38B minimum flow - GREATER THAN 50 GPM
> - P-29 minimum flow - GREATER THAN 75 GPM

The team reviewed an operations notebook entry, dated December 27, 2001, and determined that operations staff had also changed the annunciator for low instrument air header pressure to a green color. The change was made to make the annunciator tile stand out if a large number of alarms are received at one time. The majority of annunciator tiles were the color white.

(8) Failures to Identify Significant Condition Adverse to Quality

The team identified a number of opportunities which the licensee had prior to 2001 to identify that the failure mode of the AFW recirculation valves conflicted with operating practice. The specific instances were as follows:

1981 In Generic Letter (GL) 81-14, "Seismic Qualifications for Auxiliary Feedwater Systems," the NRC requested that the licensee perform a walk-down of the non-seismically qualified portions of their AFW systems to identify apparent and practically correctable deficiencies that may exist. The GL specifically identified instrument air for AFW control valves as a potential issue. In Attachment 1, Section IV, of their response, dated May 4, 1982, the licensee documented that "the AFW recirculation valves are now normally open and fail close." The licensee did not address the impact that the valves failing closed could have on the system.

1988 GL 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment," requested that licensees perform a design verification of the entire instrument air system including an analysis of current air operated component failure positions to verify that they were correct for assuring required safety functions. The licensee's response, dated February 20, 1989, stated under action item 2 that Abnormal Operating Procedure AOP-5B, "Loss of Instrument Air," provided operators with a listing of component failure positions due to loss of instrument air and the actions that might be necessary for various systems and/or components. The licensee failed to recognize that the emergency operating procedures did not include appropriate guidance.

1989 In their April 17, 1989, submittal to the NRC in response to 10 CFR 50.63, "Loss of All Alternating Current," (i.e., the station blackout rule), the licensee stated that no air-operated valves are required to operate to cope with a station blackout for one hour.

1991 The original PRA performed in response to GL 88-20, “Individual Plant Examination for Severe Accident Vulnerabilities,” did not model the recirculation valves failing closed upon loss of instrument air. Consequently, the interaction between the instrument air system and the AFW system was not fully evaluated.

1994 The design basis document (DBD) for the AFW system, DBD-01 dated April 1994, stated that recirculation valves had a safety function to open and remain open. However, the identified safety function for the valves to open was not reconciled with the valves failure mode to fail closed upon a loss of instrument air.

1997 In March 1997, the licensee identified an AFW system failure mode due to instrument air (reported by LER 97-014-00). Specifically, the flow control valves for the motor driven AFW pumps were air operated valves which failed open. In certain scenarios, such as a main steam line break coincident with a loss of instrument air, the motor driven AFW pumps could be in a run-out condition and trip the circuit breakers for the pumps. As a result of identifying this vulnerability, the licensee installed nitrogen back-up for the motor driven pump flow control valves. However, the licensee did not adequately review the function of other air operated valves in the AFW system such as the recirculation valves.

1997 In October 1997, a contractor working on the revision of the licensee’s inservice testing (IST) program identified the discrepancy between the IST background document and the AFW system DBD for the safety function of the valves in the recirculation line. The IST background document stated the check valves did not have a safety function to open since there was always adequate flow to the steam generator such that the recirculation flow path was not needed to protect the pump. The AFW system DBD stated that the recirculation valves, and, hence, the recirculation lines, did have a safety function to open to protect the pumps. The issue was documented on CR 97-3363 and investigated. In their investigation, the licensee focused on ensuring that the AFW system would provide adequate flow to the steam generators. For the turbine-driven pumps, the valve lineup was such that there was normally a flow path to the generator. The only power-operated valves in the line were motor-operated valves (MOVs) to each steam generator that were normally in the throttled position. For the motor-driven pumps, although there were normally closed valves (one control valve and an MOV to each steam generator) in the discharge path, these valves received an open signal on pump start to provide an adequate flow path. The dead-heading of the motor-driven pump could occur if the control valve or MOV failed to open. Based on single failure criteria, this type of failure would only affect one of the two motor-driven pumps. Based on this evaluation, the licensee deleted the open safety-function of the recirculation valves from DBD-1. However, the licensee failed to address operator actions which could be taken to control AFW flow to prevent overcooling of the RCS or overfilling the steam generators. As such, the licensee failed to identify that multiple AFW pumps (both turbine driven and motor driven) could be damaged by the failure mode of the AFW recirculation valves.

(9) Regulatory Issue Associated With Failures to Identify Significant Condition Adverse to Quality

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition. As of November 2001, the licensee failed to identify that the AFW system was not capable of performing its safety function under certain conditions. Specifically, all AFW pumps would be subject to a common mode failure involving dead-heading of AFW pumps following a loss of instrument air, loss of offsite power, loss of service water, or a seismic event due to the closure of the recirculation valves upon loss of the non-safety grade, non-seismically qualified instrument air system and prescribed operation actions to control feedwater flow in response to transient conditions. On seven occasions between 1981 and 1997, the licensee was made aware of the susceptibility of the AFW system to this type of vulnerability, but the licensee failed to identify this significant condition adverse to quality. This issue is considered an apparent violation (AV 50-266/01-17-02; 50-301/01-17-02).

(10) Pressurizer PORV Impact on Operational Capability

The pressurizer PORVs were air operated valves which were provided with a backup nitrogen supply. However, since 1979, the back-up nitrogen supply has been isolated, by procedure, during power operation. A containment entry was required to restore the back-up nitrogen supply. Consequently, upon a loss of instrument air, the PORVs would not be available. The safety injection pumps do not provide sufficient discharge pressure to lift the reactor coolant system safety relief valves. Although the positive displacement charging pumps provide sufficient discharge pressure to lift and pass coolant through the code safety relief valves, the charging pumps do not provide sufficient flow for adequate decay heat removal. Consequently, a loss of instrument air would result in the loss of effective feed and bleed capability. A loss of auxiliary feedwater combined with a loss of instrument air, which would also involve a loss of main feedwater, would result in a loss of decay heat removal capability.

(11) Extent of Condition

The team reviewed the configuration of other significant systems, such as safety injection, to verify that the recirculation lines did not have air-operated valves which failed closed upon of loss of instrument air. The team did not identify any other systems in which a similar vulnerability existed.

(12) Safety Significance

The team evaluated the finding using the Phase 2 process described in Inspection Manual Chapter 609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The site specific worksheets for the Point Beach Nuclear Plant were used. These site specific worksheets had been benchmarked against the licensee's current PRA model for the plant. Based on this review, the team determined that the most limiting scenario was loss of instrument air. The following assumptions were made:

- The exposure time was greater than 30 days.

- The initiating event frequency was 1×10^{-3} for loss of instrument air.

- No credit for any AFW was applied. The emergency operating procedures used by operators did not provide adequate guidance to address recirculation valve closure. Additionally, operators were not trained to specifically recognize the potential for AFW recirculation valve closure and the consequences. The licensee's PRA staff performed informal calculations which showed overall human error probabilities in the range of 0.3 to 0.5 (depending on the calculational method used) for operator actions in response to steam generator overfill. For operator response to RCS overcooling, the licensee's PRA staff assumed that operator actions would result in failure of the AFW pumps.

- For loss of instrument air, feed and bleed capability using the pressurizer PORVs was not credited because of the reliance upon instrument air.

Based on use of the Point Beach Nuclear Plant site specific worksheets for loss of instrument air, the finding was preliminarily determined to be of high safety significance (Red). The dominate sequences involved the loss of instrument air and the loss of feedwater.

The team also evaluated the finding using the loss of off-site power site specific worksheets. The assumptions used were similar to those above with the following exceptions:

- The initiating event frequency was 1×10^{-2} for loss of off-site power.

- Instrument air would be initially lost upon loss of off-site power because the air compressors would be automatically stripped from the safeguards power buses. The initial loss of instrument air would result in damage to the AFW pumps due to operator actions.

- Credit for feed and bleed capability was applied because instrument air could be restored by manual operators actions. The necessary operator actions were proceduralized.

- Credit for high pressure recirculation was applied because neither safety injection nor residual heat removal was affected by the finding.

Based on use of the Point Beach Nuclear Plant site specific worksheets for loss of off-site power, the finding was preliminarily determined to be of high safety significance (Red). The dominate sequences involve the loss of off-site power and AFW with either feed and bleed capability or high pressure recirculation being available.

In addition, the finding was evaluated using the loss of service water worksheets. However, the significance due to loss of service water was not as great as the loss of instrument air and loss of off-site power transients as described above.

.2 Licensee Event Reports

LER 50-266/2001-05; 50-301/2001-05 (Open): PRA assessment of AFW system reveals procedural vulnerability related to loss of instrument air. The subject of this LER is discussed in Section 4OA3.1 and two apparent violations were identified. This LER will remain open pending future inspection review.

4OA6 Meeting(s)

Exit Meeting

On December 13, 2001, at the conclusion of the on-site inspection activities, the lead inspector presented the initial findings to Mr. Reddemann and other members of licensee management at Point Beach Nuclear Plant. On February 28, 2002, the team presented the findings to Mr. Warner and other members of licensee management. The licensee representatives acknowledged the findings presented. The team identified the proprietary information reviewed during the inspection and noted that the information would be handled accordingly. The licensee did not identify any other material reviewed during the inspection as being proprietary.

KEY POINTS OF CONTACT

Licensee

J. Anderson, Manager, Production Planning
F. Cayia, Director, Kewaunee - Point Beach Site
R. Mende, Director, Kewaunee - Point Beach Engineering
M. Reddemann, Vice President - Engineering, Nuclear Management Company
J. Strharsky, Assistant Manager, Operations
M. Warner, Vice President, Kewaunee - Point Beach Site
T. Webb, Manager, Kewaunee - Point Beach Regulatory Affairs

NRC

J. Grobe, Director, Division of Reactor Safety, Region III
J. Jacobson, Chief, Mechanical Engineering Branch, Region III

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-266/01-05 50-302/01-05	LER	PRA Assessment of Auxiliary Feedwater System Reveals Procedural Vulnerability Related to Loss of Instrument Air
50-266/01-17-01 50-301/01-17-01	AV	Potential Common Mode Failure of Auxiliary Feedwater Pumps Due to Inadequate Procedural Guidance
50-266/01-17-02 50-301/01-17-02	AV	Failure to Identify and Correct Problem Associated With Potential Common Mode Failure of Auxiliary Feedwater Pumps

LIST OF ACRONYMS USED

°	Degrees
AFW	Auxiliary Feedwater
AOP	Abnormal Operating Procedure
AV	Apparent Violation
CFR	Code of Federal Regulations
CR	Condition Report
DBD	Design Basis Document
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EA	Enforcement Action
EOP	Emergency Operating Procedure
F	Fahrenheit
FFDSAR	Final Facility Description and Safety Analysis Report
GL	Generic Letter
GPM	Gallons Per Minute
IPE	Individual Plant Examination
IR	Inspection Report
IST	In Service Testing
LER	Licensee Event Report
LLC	Limited Liability Company
MOV	Motor Operated Valve
NMC	Nuclear Management Company, LLC
NRC	U.S. Nuclear Regulatory Commission
PORV	Power Operated Relief Valve
PPCS	Plant Process Computer System
PRA	Probabilistic Risk Analysis
PSIG	Pounds per Square Inch Gauge
RCS	Reactor Coolant System
RNO	Response Not Obtained
SDP	Significance Determination Process
S/G	Steam Generator

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC team reviewed the documents in their entirety, but, rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort.

Number	Title	Revision/Date
Procedures		
AOP-5B	Loss of Instrument Air	Revision 18
AOP-10A Unit 1	Safe Shutdown - Local Control	Revision 32
CSP-H.1 Unit 1 Red	Response to Loss of Secondary Heat Sink	Revision 21
ECA-0.0 Unit 1	Loss of All AC Power	Revision 29
ECA-0.0 Unit 2	Loss of All AC Power	Revision 30
EOP-0 Unit 1	Reactor Trip or Safety Injection	Revision 35
EOP-0 Unit 2	Reactor Trip or Safety Injection	Revision 36
EOP-0.1 Unit 1	Reactor Trip Response	Revision 24
EOP-0.1 Unit 2	Reactor Trip Response	Revision 23
IT 10	Test of Electrically-Driven Auxiliary Feed Pumps and Valves (Quarterly)	July 5, 2001
OM 4.3.1	AOP and EOP Writers' Guide	Revision 3
Temporary Changes		
2001-0871	EOP-0 Unit 1, Reactor Trip or Safety Injection	November 30, 2001
2001-0872	EOP-0 Unit 2, Reactor Trip or Safety Injection	November 30, 2001
2001-0873	EOP-0.1 Unit 1, Reactor Trip Response	November 30, 2001
2001-0874	EOP-0.1 Unit 2, Reactor Trip Response	November 30, 2001
2001-0911	ARP C01 A 1-9, Instrument Air Header Pressure Low	December 20, 2001

2001-0912	ECA-0.0 Unit 1, Loss of All AC Power	December 20, 2001
2001-0913	EOP-0.1 Unit 2, Reactor Trip Response	December 20, 2001
2001-0914	EOP-0 Unit 2, Reactor Trip or Safety Injection	December 20, 2001
2001-0915	EOP-0 Unit 1, Reactor Trip or Safety Injection	December 20, 2001
2001-0916	EOP-0.1 Unit 1, Reactor Trip Response	December 20, 2001
2001-0917	ECA-0.0 Unit 1, Loss of All AC Power	December 20, 2001

<u>Design Basis Documents</u>

DBD-01	Auxiliary Feedwater System	Revision 0
DBD-01	Auxiliary Feedwater System	Revision 1
DBD-06	Instrument & Service Air	Revision 2
DBD-T-46	Station Blackout	Revision 0

<u>Updated Final Safety Analysis Report Sections</u>

4.2	RCS System Design and Operation	June 2000
8.8	Diesel Generator (DG) System	June 2000
9.7	Instrument Air (IA) / Service Air (SA)	June 2000
10.2	Auxiliary Feedwater System (AF)	June 2000

<u>Calculations</u>

N-91-007	Steam Generator Inventories 5 Minutes After an Earthquake	November 7, 1991
N-91-031	1 & 2 P29 Mini-Recirc Line System Characteristics	March 19, 1991
N-91-032	Comparison of Nominal Flow Rates from 2P-29 to 2HX-1A and 2HX-1B with the Recirc Line Open	March 19, 1991

Correspondence

	NRC Generic Letter No. 81-14, Point Beach Nuclear Plant, Units 1 and 2	July 16, 1981
	Additional Response to NRC Generic Letter 81-14, Point Beach Nuclear Plant, Units 1 and 2	May 4, 1982
	Seismic Qualification of the Auxiliary Feedwater System, Point Beach Nuclear Plant, Units 1 and 2	December 15, 1982
	Final Resolution of Generic Letter 81-14, Seismic Qualification of Auxiliary Feedwater System, Point Beach Nuclear Plant, Units 1 and 2	April 26, 1985
VPNPD-88-335	Response to NRC Bulletin 88-04	June 28, 1988
VPNPD-88-090 NRC-89-021	Response to Generic Letter No. 88-14, Instrument Air System Problems Affecting Safety-Related Equipment, Point Beach Nuclear Plant, Units 1 and 2	February 20, 1989
VPNPD-89-216 NRC-89-043	Response to 10 CFR 50.63, Tac. Nos. 68586 and 68587, Loss of All Alternating Current Power, Point Beach Nuclear Plant, Units 1 and 2	April 17, 1989
	Response to NRC Bulletin 88-04	May 26, 1989
	Minimum Flow Analysis	August 7, 1989
VPNPD-95-056	Generic Letter 88-20, Supplement 4, Summary Report on Individual Plant Examination of External Events for Severe Accident Vulnerabilities	June 30, 1995
NPL 97-0186	Licensee Event Report 97-014-00, Auxiliary Feedwater System Inoperability Due to Loss of Instrument Air	April 18, 1997
NRC 2001-057	Licensee Event Report 301/2001-002-00, Manual Reactor Trip Due to Decreasing Water Level in Circulating Water System	August 17, 2001
	NRC Memorandum From John M. Jacobson to Ronald A. Langstaff, Special Inspection Charter for Point Beach Potential Common Mode Failure of Auxiliary Feedwater	November 30, 2001

NRC 2002-0012	Licensee Event Report 266/2001-005-00, PRA Assessment of Auxiliary Feedwater System Reveals Procedural Vulnerability Related to Loss of Instrument Air	January 28, 2002

Condition Reports

97-0930	Questions and Concerns About the use of Operator Action to Control AFW Flow	March 20, 1997
97-3363	IST Program Design Basis for AFW Minimum Flow Recirculation Valves	October 15, 1997
98-2575	P-38A AFW Pump Recirc Valve Found Failed Shut	June 29, 1998
QCR 99-0115	Code Testing Conflict with the Aux Feedwater Mini-Flow Recirc Check Valves	May 24, 1999
99-3091	Aux Pump Recirc Line Leakage Acceptance Criteria Questioned	December 3, 1999
01-2278	Auxiliary Feedwater Probabilistic Risk Assessment (PRA) Model for Loss of Instrument Air	July 6, 2001
01-3595	Potential common mode failure for all auxiliary feed pumps under certain initiating events.	November 29, 2001
01-3641	Modifications that had potential to identify concern	December 4, 2001
01-3654	The development and revision of DBD-01 appears to have been a missed opportunity to identify a design weakness in the AFW system.	December 6, 2001

Evaluations

RCE 98-148	P-38A AFW Pump Recirc Valve Found Failed Shut	January 29, 1999

Drawings

M-201, sheet 1	Main & Reheat Steam System	January 20, 2001
M-209, sheet 1	Service Air	May 12, 2001
M-209, sheet 2	Service Air	November 18, 2000
M-209, sheet 3	Instrument Air	November 18, 2000

M-209, sheet 4	Instrument Air	October 25, 2001
M-209, sheet 11	Instrument Air	January 19, 1998
M-217, sheet 1	Auxiliary Feedwater System	September 29, 2001
M-217, sheet 2	Auxiliary Feedwater System	February 3, 2001
M-2201, sheet 1	Main & Reheat Steam System	January 20, 2001
<u>Modifications</u>		
97-038*A	AFW Motor Driven Pump Discharge Control Valve Modification	March 24, 1998
97-038*B	AFW Discharge Valve AF-04012 & AF-04019 Modification	June 26, 1998
88-099, Common	AFW Recirc Line Modification	March 27, 1991
88-099*A, Common	AFW Recirc Line Modification	February 14, 1992
88-099*C, Common	AFW Recirc Line Modification	February 14, 1992
88-099*D, Common	AFW Recirc Line Modification	July 1, 1992
<u>Miscellaneous Documents</u>		
LP3178	Auxiliary Feedwater System	June 15, 2001
Mod Request IC-274	Making AFW Recirc Valves Normally Open	
Procedure Change OP-1A,Major	Cold Shutdown to Low Power Operation	December 26, 1978
Procedure Change OP-1A,Major	Cold Shutdown to Low Power Operation	July 26, 1979
IST Background Valve Data Sheet	Auxiliary Feedwater	May 17, 2000
97-201	Setpoint Change to the Auxiliary Feedwater By-pass Control Valves Time Delay Relay Setpoints (1/2-NC005, 62-P38A and 62-P38B)	December 4, 1997
	Action Items Associated with GL 88-14	September 17, 1991
	Final Facility Description and Safety Analysis Report	Original