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EXECUTIVE SUMMARY

Prairie Island Nuclear Generating Plant, Units 1 & 2
NRC Inspection Report 50-282/97008, 50-306/97008

This report includes the results of an announced System Operational Performance Inspection by regional inspectors and NRR of plant operations, maintenance, and engineering for the auxiliary feedwater (AFW) system and parts of the control room ventilation and safeguards chilled water systems.

Operations

- Operations' performance during an observed startup of Unit 1 was good (Section O1.1).

- The emergency operating, operating, and alarm response procedures provided acceptable instructions for operating the AFW system during all aspects of plant operation (Section O3.1). While overall, the checklists and drawings reviewed were acceptable, the inspectors identified that AFW pre-start checklists did not reflect the current plant configuration (Section O3.2).

- While the operators' performance of the AFW surveillance was considered good, the operating shift did not identify, prior to commencing the surveillance, that current plant conditions would have resulted in the inability to perform specific sections within the special procedure (Section O4.1).

- The inspectors concluded that the control room operators were very knowledgeable concerning the recent AFW system modifications (Section O4.2) and observed operations training concerning the recent AFW pump modifications was considered good (Section O5.1).

Maintenance

- With a few exceptions, maintenance was being performed according to approved procedures. Work packages were well planned and contained adequate instructions (Section M1.1).

- Overall, the observed material condition of the plant was good (Section M2).

- Maintenance procedures were technically adequate and sufficiently detailed to perform the required maintenance and inspection tasks and had the necessary provisions to identify and evaluate deficiencies. The procedures reviewed also satisfied or exceeded vendor recommendations (Section M3.1).

- Based on examination of available maintenance history, performance indicators, and trending data, plant components were being appropriately maintained to provide assurance of operating when called upon (Section M8.1).

Engineering

- The AFW pump surveillance test procedure acceptance criteria could have allowed the AFW pumps to degrade below design requirements. This was an apparent violation of test control requirements. The latest test results were close to the design requirement values (Section E1.1).

- The failure to accomplish corrective action from 1991 of reviewing safety related pump test acceptance criteria was an apparent violation of corrective action requirements (Section E1.1).

- The failure to correct the inaccurate 400 gpm AFW flow rate in the USAR, despite two opportunities to do so in December 1993 and 1995, was considered an apparent violation of Accuracy of Information requirements and also an apparent violation of Maintenance of Records requirements (Section E1.2).

- The failure to report that the plant was outside its design basis when it was determined that the main feedwater line rupture analysis used a 400 gpm AFW flowrate was considered an apparent violation of Reportability requirements. The failure to perform a safety evaluation for this defacto change to the facility as described in the USAR and to verify that no unreviewed safety question existed was considered an apparent violation of 10 CFR 50.59 requirements (Section E1.2).

- Design changes and modifications reviewed, including documentation revisions and post-modification testing, for the AFW system were acceptable (Section E1.3).

- The basis for the unfiltered inleakage rate assumption in the control room habitability dose analysis was considered weak because it had not been validated through testing of the control room envelope or testing of the isolation dampers (Section E1.4).

- While many of the calculations reviewed were considered acceptable, the inspectors noted weaknesses in the calculation verification program based upon the errors found in the mechanical calculations, some of which were introduced during the verification process. These errors were considered a violation of design control requirements (Section E3.1).

- Identification of discrepancies in system drawings indicated a weakness in the drawing control program to assure plant drawings accurately reflect plant status (Section E3.4).

- The Safety Audit Committee and Operations Committee meetings fulfilled their Technical Specification requirements and provided the necessary oversight function for which they were intended (Section E7.1).

- The licensee's corrective actions for cable trays not meeting separation criteria were inadequate in that it took over 4 years to determine reportability and additional cable trays were not identified until NRC inspectors noted them. This was considered a violation of corrective action requirements (Section E8.4).

Report Details

I. Operations

O1 Conduct of Operations

O1.1 Observation of Unit 1 Startup

a. Inspection Scope

On April 27, 1997, inspectors observed operator actions during the startup of Unit 1. The plant startup was conducted using procedure 1C1.2 "UNIT 1 STARTUP PROCEDURE," Revision 16.

b. Observations and Findings

While overall the operator actions observed by the inspectors during the startup were good, an issue with control of steam generator (SG) level was noted. Temporary Memo TMA-1997-0059 added Limitation 4.6 "Steam Generator Level" to the Unit 1 startup procedure, 1C1.2, which stated: "WHEN RCS temperature is greater than 350°F AND reactor power is less than 5%, THEN do NOT exceed 38% steam generator narrow range level." However, during the transition from auxiliary feedwater to main feedwater, steam generator water level exceeded the 38% narrow range level on the 11 SG for approximately four minutes. Operators responded appropriately to maintain steam generator level below 40%. In response to this issue, the licensee formed a multi-disciplined task force to review the restrictions and determine potential actions required or available to increase the limited margin.

c. Conclusions

Operations' performance during the observed startup of Unit 1 was good. However, the inspectors noted a weakness in operators not being able to maintain steam generator level below an administratively imposed limit.

O3 Operations Procedures and Documentation

O3.1 Review of Operating Procedures

a. Inspection Scope

The inspectors reviewed the adequacy of emergency operating procedures (EOPs), operating procedures (OPs), and alarm response procedures (ARPs) for the AFW system, as listed at the end of this report, for event sequences requiring AFW initiation.

b. Observations and Findings

The inspectors observed that recent AFW modifications were incorporated into the OPs and ARPs, both through the use of procedure changes or the facility's temporary memo process.

The inspectors reviewed the ARPs located in the simulator at the Prairie Island Nuclear Generating Plant (PINGP) Training Center, and noted that the ARPs did not reflect the current condition of the simulator. Specifically, ARP C47010-0205, "11 TD AFWP LO OR DISCH PRESS TRIP," Revision 30, indicated a setpoint of < 200 PSIG for initiating the "Discharge Pressure Low" annunciator and alarm. The inspectors determined through Simulator Change 97I-002, dated March 17, 1997, that the setpoint for the AFW low discharge trip had been changed to 800 PSIG prior to testing during the weeks of February 9 and 16, 1997. The simulator ARP was subsequently updated on April 29, 1997.

c. Conclusions

The inspectors concluded that while some delay occurred in updating ARPs in the simulator, the EOP, OP, and ARP procedures provided acceptable instructions for operating the AFW system during all aspects of plant operation.

O3.2 Review of AFW System Prestart Checklists

a. Inspection Scope

The inspectors reviewed previously completed checklists on both Unit 1 and Unit 2 auxiliary feedwater systems, and performed a walkdown with checklists and system flow drawings.

b. Observations and Findings

During a walkdown on the AFW system using the prestart checklist, C28-2 (Unit 1, Revision 34) and C28-7 (Unit 2, Revision 37), four valves were discovered in mid-position, that is, 45° open, contrary to the required "OPEN" position detailed on the checklists. In addition, operations personnel (including shift managers) indicated the valves had been in the "throttled" position since the modified piping system was installed in 1994.

The four valves in question, AF-39-1(3) and 2AF-39-1(3), are suction vent loop seal drain valves. The valves maintain a continuous flow of condensate water through the suction piping of the AFW pumps to flush possible cooling water leakage past the cooling water system suction supply motor-operated isolation valves. The four valves are throttled to limit the condensate inventory loss, but are also adjusted to maintain weekly sodium samples less than 1 part per billion (ppb).

Also, a review of previous checklists performed on both units indicated that the previous checklists either incorrectly documented the valves as OPEN and not THROTTLED or the checklists were crossed out and initialed to indicate "throttled."

While the safety consequences of the valves' position was negligible, the checklist did not reflect the current plant configuration, and operators had not identified this condition on a number of previous checklists. The inspectors considered it a weakness that plant procedure reviews and operator performance did not identify the need for a procedure deviation in excess of two years, the approximate time the piping had been installed in the system. In response, the licensee initiated a procedure submittal form to formally change the required "STATUS" position of the drain valves located on the checklists.

c. Conclusions

While overall, the checklists and drawings reviewed were acceptable, the inspectors identified that AFW pre-start checklists did not reflect the current plant configuration, and noted that operators had not identified this condition on a number of previous checklists. This was considered a weakness.

O4 Operations Staff Knowledge and Performance

O4.1 AFW Operability Surveillance Test

a. Inspection Scope

The inspectors witnessed the operating shift crew perform a post-modification operability test on the Unit 1 turbine-driven auxiliary feedwater pump (TDAFW) following a recent modification to the AFW system, and prior to the Unit 1 startup.

b. Observations and Findings

During performance of surveillance procedure (SP) 1102, "11 Turbine-Driven Auxiliary Feedwater Pump Test," Revision 58, the inspectors observed the operators stationed locally at the 11 TDAFW pump read through the procedure steps prior to the performance of each step required by the surveillance. The inspectors observed good communication between operators in the control room and locally in the AFW pump room. However, the inspectors observed a number of procedure errors and procedure steps not applicable for the plant condition identified by the operating crew while the test was being performed.

Specifically: (1) Step 7.2.3 was identified as a procedure error for referencing "steps 5.3.2.A and 5.3.2.B" of C28.1; the correct reference was 1C28.1, Section 5.6; (2) Step 7.2.5 was identified as a procedure error for referencing C28.1, which does not exist; (3) Step 7.32.2 was not performed because the test was normally performed at 100% power with the 12 motor-driven AFW pump (MDAFW) idle. Plant conditions at the time of the test had the 12 MDAFW pump running for control of steam generator water level, and the step could not be completed. In addition, the "CAUTION" statement immediately prior to Step 7.32.2 identified the 12 MDAFW pump as "IDLE" for the four steps within Section 7.32.2; (4) Steps 7.19 and 7.20 could not be completed due to the plant conditions present at the time of the test, namely, the other train of AFW was inservice and the steam generator blowdown would remain inservice throughout the performance of

SP 1102. The operations crew was able to address these discrepancies and successfully complete the test.

These procedure issues were considered a weakness as the operators should have identified, prior to commencing the surveillance, that current plant conditions would have resulted in the inability to perform specific sections within the special procedure. In response, the licensee stated that the procedure discrepancies were noted by the previous operating shift but the shift turnover was inadequate.

c. Conclusions

While the operators' performance of the AFW surveillance was considered good, the operating shift did not identify, prior to commencing the surveillance, that current plant conditions would have resulted in the inability to perform specific sections within the procedure.

O4.2 Review of Operations Staff Knowledge via Questioning of Operations Personnel Regarding The Auxiliary Feedwater System (AFW)

a. Inspection Scope

The inspectors randomly questioned on-shift personnel to determine their level of knowledge regarding the AFW system, including the recent AFW system modification, 96AF01, "AFW PUMP RUNOUT PROTECTION."

b. Observations and Findings

The inspectors questioned on-shift personnel from different operating crews, focusing on specific details of the modification relating to control room switch positions and the associated TDAFW pump trips. Each operator responded with answers consistent with the AFW modification.

In addition, various on-shift personnel were questioned on procedures developed to monitor the AFW pump discharge piping during each shift. The procedures were developed to assist in the detection of backleakage of steam generator water through system check valves, which could lead to steam binding of the AFW pumps. Each operator was knowledgeable of the steam binding issue and the requirement for AFW pump discharge piping monitoring during each shift.

c. Conclusions

Based on sample interviews, the inspectors concluded that the control room operators were very knowledgeable concerning the recent AFW system modifications.

O5 Operations Staff Training and Qualification

O5.1 Operator Training on the Auxiliary Feedwater System (AFW)

a. Inspection Scope

The inspectors observed on-shift training and licensed operator requalification training to determine the adequacy of training on the AFW system.

b. Observations and Findings

The inspectors observed on-shift training conducted in the main control room by the applicable shift managers regarding the recent AFW modification to protect against AFW pump runout. The training was administered to all crews over a two-week period, and detailed the major changes to the AFW pump operational logic. The observed training was considered good.

Additionally, the inspectors observed a licensed operator requalification training session. Included in the training was a discussion of the recent AFW pump runout protection modification and other AFW operational issues. The instructor detailed the major changes to the AFW pump operational logic incorporated by the modification. Good feedback was observed from the operators concerning recent changes to the unit startup operating procedure, C1.2, which limits steam generator water level during certain plant conditions.

c. Conclusions

The inspectors concluded that operations training concerning the recent AFW pump modifications was good. This conclusion was supported by the results of random questioning of control room operators detailed in Section 04.2.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Work Observed

a. Inspection Scope

The team observed maintenance and surveillance work activities involving selected plant equipment. Maintenance and surveillance activities observed and reviewed are listed at the conclusion to this report.

b. Observations and Findings

The observed instrumentation and controls (I&C) and electrical maintenance and surveillance work activities were adequately performed. The procedures contained necessary acceptance criteria. The surveillance results were acceptable. The measuring and test equipment used were noted to be in calibration. The I&C

technicians and the maintenance craft were experienced and knowledgeable in the areas observed.

Work order packages for electrical, instrumentation, and mechanical related work appeared to be well planned and included sufficient instructions to assure work was accomplished according to procedure. Tagging instructions were clearly noted in the work packages. In addition, quality verification hold points were identified. Post-maintenance testing requirements and responsibility for conducting the test were included in the procedure, when applicable.

Work Scheduling Weaknesses

During the performance of the diesel generator (D5) 18 month preventive maintenance activities, the team noted that the I&C, electrical, and mechanical test procedures were being performed simultaneously. With 3 procedures causing alarms in the D5 control room, there was confusion as to which procedure was causing the alarm. This was most evident while the I&C team and the electrical relay team were both causing numerous lockout relay actuation alarms that resulted in workers from each team unsure of which team had caused the alarm. The licensee recognized the potential for coordination errors and revised the testing.

c. Conclusions

The team concluded that, with a few exceptions, maintenance was being performed according to approved procedures and that work packages were well planned and contained adequate instructions.

M2 Material Condition of Plant

a. Inspection Scope

The team walked down selected areas of the plant to review the material condition.

b. Observations and Findings

The team walked down accessible areas of the AFW system, control room (CR) ventilation system, and the diesel generator rooms to review the material condition of the equipment. Equipment material condition, and housekeeping were good in almost all cases. Several minor discrepancies were brought to the licensee's attention and were corrected.

The inspectors noted during walkdowns that the licensee had installed yellow-colored plastic chains on the front of many of the plant's switchgear and motor control center cabinets as bump hazard warning barriers. These barriers served to remind breaker maintenance crews and other plant personnel that the electrical equipment was energized and that a bump to the cabinet could cause a device or relay to trip. The inspectors considered this to be a simple yet innovative design feature to enhance safety and prevent undesired breaker trips.

c. Conclusions

The team concluded that, overall, the material condition of the plant observed was good.

M3 Maintenance Procedures and Documentation

M3.1 Review of Maintenance Procedures

a. Inspection Scope

The team reviewed selected maintenance procedures for the systems selected for inspection. The reviews were to determine technical adequacy and that they satisfied vendor requirements and recommendations.

b. Observations and Findings

The licensee's maintenance procedures reviewed during this inspection appeared to be technically adequate to perform the specific maintenance task and provided for the identification and evaluation of equipment and work deficiencies. The inspectors' review of sample modifications to equipment or systems determined that the maintenance procedures had been revised to incorporate the modifications.

Maintenance procedure content was compared against manufacturer's maintenance and inspection recommendations for the auxiliary feed pumps, auxiliary feed pump turbines, MDAFW motors, circuit breakers, motor-operated valves, control room chillers and control room air handlers. The procedures appeared to satisfy, and in some cases exceed, the manufacturer's maintenance and inspection requirements. Vendor manuals appeared to be complete and up-to-date.

The team also reviewed the calibration records of several instruments on these systems and noted that the instrumentation was generally well maintained. With few exceptions, the reviewed measuring and test equipment used for surveillance tests were in calibration.

Discrepancy Report Not Completed for Out-of-Tolerance Data

The inspectors' review of surveillance procedure, SP-2224, dated March 1996, indicated that the control room recorders, 2TR-450 and 2TR-451 (wide range RCS temperatures), were out of tolerance yet a surveillance procedure discrepancy report (SPDR) had not been written. This was in conflict with work procedure, SWI-STE-10, "Evaluation of Out-of-Tolerance Calibration Data in I&C Procedures," which specified that a SPDR be completed when as-found data did not meet the specified tolerance of the acceptable value. The issue was of minimal safety consequence as the recorders were brought back into calibration (when initially identified) and were considered operable. In response, the licensee issued nonconformance reports (NCRs) Nos. 2010746 and 2010747 to address the issue. The licensee's failure to generate the SPDRs was considered a weakness.

c. Conclusions

The team concluded that, overall, the licensee's procedures were technically adequate and sufficient to perform the required maintenance and inspection tasks and had the necessary provisions to identify and evaluate deficiencies. The procedures also satisfied or exceeded vendor recommendations for maintenance and inspection of vendor supplied equipment.

M8 Miscellaneous Maintenance Issues

M8.1 Maintenance-Related Unavailability

a. Inspection Scope

The team reviewed maintenance history on selected components, performance indicators, and trending to determine whether equipment was being adequately maintained to assure its operability under all conditions.

b. Observation and Findings

Review of performance indicators from April 1996 through March 1997, provided the following information:

- Average monthly corrective action backlog: less than 50 work orders
- Licensee event reports directly attributed to maintenance during the past year: 1
- Reactor trips initiated by maintenance: none
- Repeat work requests generated: 16
- Power block Priority 1 average backlog: 4
- Overdue preventive maintenance January 1994 - February 1997: none

The data reviewed indicated that the maintenance and preventive maintenance programs appeared effective in assuring equipment operability. Based on examination of the available data as well as field walkdowns, the inspectors noted that plant components were adequately maintained such that equipment had a high degree of assurance of operating when called upon.

c. Conclusion

Based on examination of available maintenance history, performance indicators, and trending data, plant components were being appropriately maintained to provide assurance of operating when called upon.

III. Engineering

E1 Conduct of Engineering

E1.1 Inadequate AFW Pump Surveillance Testing Acceptance Criteria

a. Inspection Scope

The inspectors reviewed the Updated Safety Analysis Report (USAR), the Technical Specifications and Bases, and other licensing and design basis documents to identify and quantify the functions and performance requirements for the AFW system. The inspectors reviewed the completed procedures for the four previous performances of the refueling outage (RFO) functional tests for each of the four AFW pumps and the monthly AFW pump surveillance procedures. The inspectors also reviewed applicable engineering calculations.

b. Observations and Findings

The licensee had designated the minimum acceptance criteria for the AFW pump tests as 10% degradation from the reference pump curve which satisfied ASME code, Section XI. However, based on review of the design basis accident (DBA) requirements, the inspectors raised a concern that the licensee had not evaluated whether the pumps, at 10% degradation, would meet the DBA requirements. The licensee had not calculated the minimum pump performance requirements necessary for the pumps to meet minimum design requirements but instead based the test acceptance criteria only on Code requirements of allowing up to 10% degradation. From USAR Section 11.9, the AFW pumps' minimum DBA requirement was to provide a flowrate of at least 200 gpm to one steam generator (SG) at 1100 psig.

Of particular concern was the inspectors' observation that the 3% actual degradation of the most limiting AFW pump (21) appeared to be near the minimum design flow requirement. The licensee promptly documented in calculation ENG-ME-315 that assuming worst case conditions, worst case instrument inaccuracy combinations and other conservatisms even the most limiting AFW pump (21) would deliver at least 200.8 gpm to one SG at 1142.6 psig. The calculation used empirical test data and a computer model of the AFW system. Some parts of the model still needed to be validated and the licensee intended to accomplish that validation testing during the next refueling outages (RFOs) (October 1997 for Unit 1 and February 1998 for Unit 2). A preliminary team review found that the calculation provided reasonable assurance that the pumps would perform the AFW safety functions during any DBA. The licensee believed that improved test equipment and calculations would demonstrate that the pumps actually have more margin. Detailed NRC review of the calculation and verification of the model will be tracked as inspection followup item (IFI 50-282/306-97008-01(DRS)).

Further, the licensee promptly initiated non-conformance report NCR 2010728 which documented that the ASME acceptance criteria (10% from the reference curve) for all the AFW pump tests could have allowed the pumps to degrade below minimum design requirements. The team confirmed that the acceptance criteria were inadequate. 10 CFR Part 50, Appendix B, Criterion XI, Test Control, requires, in part, that testing shall be performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The failure of AFW test procedures to have incorporated the design requirements contained in applicable design documents is an apparent violation of 10 CFR 50, Appendix B, Criterion XI, Test Control (EEI 50-282/306-97008-02).

The non-conformance report also documented that all AFW pump test procedures would be corrected by May 31, 1997, or before the test was reperformed, whichever was sooner. While onsite, the team confirmed that the tests that were performed had corrected acceptance criteria.

The licensee informed the team that there was reasonable assurance that even the most limiting AFW pump (21) would not degrade below safety function capacity before the next RFO test because there were numerous conservatisms in calculation ENG-ME-315. A team review confirmed the existence of substantial conservatisms in the calculation. The team also reviewed the last four tests for each AFW pump and found that the degradation between tests was small enough to assure that the AFW pumps would not degrade below the safety function capacity.

The licensee assured the team that their preliminary review found that all safety related pumps were performing above minimum design requirements.

Failure to Complete Corrective Action on Similar Issue

In response to team questions on the acceptability of the acceptance criteria of other safety related pumps, the licensee stated that the cooling water pumps' performance was reviewed prior to an NRC service water operational performance inspection (SWOPI) performed in the early 1990s. The pumps' performance was found adequate and the lowest test acceptance criteria were also found to be adequate. The licensee also stated that the safety injection (SI) pumps were reviewed during a 1991 modification and found to be performing above design requirements but the acceptance criteria had to be corrected. The licensee stated in NCR 2010728 that the acceptance criteria for the remaining safety related pumps would be reviewed by July 1, 1997.

However, an operational experience assessment (OEA) action item was generated in 1991 to review the acceptance criteria of all of the ASME Section XI pumps other than the cooling water and safety injection pumps. This review was not given proper priority and was never accomplished. This review would likely have identified that the AFW and other pump tests had inadequate acceptance criteria. The failure to complete this corrective action was not identified until prompted by NRC questions. The licensee's corrective action process for industry operating experience issues was separate from the corrective action tracking process for other nonconformances and as a result did not have adequate controls to ensure proper action was taken on an item open for several years. In response, the licensee stated that all OEA open items, priorities, and schedules would be reviewed by June 30, 1997.

10 CFR 50, Appendix B, Criterion XVI, Corrective Action, required that "Measures shall be established to assure that conditions adverse to quality. . .are promptly identified and corrected." Contrary to this requirement, since the original identification in 1991 of the above described condition adverse to quality, the licensee did not promptly act to correct this condition. The failure to accomplish the review of other ASME Section XI pumps is an apparent violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action (EEI 50-282/306-97008-03).

c. Conclusion

The inspectors concluded that the AFW pumps' test procedure acceptance criteria did not include the design requirements from the USAR. The acceptance criteria could have allowed the AFW pumps to degrade below required design flows. This was an apparent violation of test control requirements.

The licensee failed to accomplish corrective action from 1991 of reviewing safety related pump test acceptance criteria and this was an apparent violation of corrective action requirements.

E1.2 Required AFW Flow Rates Following a Design Basis Accident

a. Scope

The inspectors reviewed the AFW design basis document (DBD), the follow-on items (FOI) resulting from validation of the DBDs, and the USAR to determine the most limiting required flow rates.

b. Observations and Findings

USAR Section 11.9.3 "Performance Analysis [Condensate, Feedwater, and Auxiliary Feedwater Systems]" specified that 400 gallons per minute (gpm) of AFW flow were available to the intact steam generator within 10 minutes of a main feedwater line rupture (MFLR). Based upon the nameplate rating of the AFW pumps, both AFW pumps would have to supply water to one steam generator to achieve this value. If there was a single failure of one AFW pump, then the required flow rate could not be achieved. In the DBD, the inspectors noted that the issue of the required flow rate following a MFLR had been designated a FOI.

The FOI had been issued in December 1992 to resolve a discrepancy between the USAR required value and the capability of a single pump. The FOI, 781, also stated that the MFLR was not discussed in the accident analysis section of the USAR, Section 14, although it was the accident which placed the most limiting conditions upon the AFW system.

The licensee's initial evaluation in early 1993 confirmed that the MFLR scenario was based upon a guillotine rupture of the feedwater piping after the AFW system joined the line. A simultaneous loss of offsite power would require AFW flow to mitigate the accident. The assumed single failure was the loss of the AFW pump to the unbroken loop. The remaining pump would feed the break until manually realigned. The operator was required to take action to realign the remaining AFW pump to the unbroken loop within 10 minutes. However, this evaluation confirmed that only one AFW pump would be available to provide AFW flow to the steam generator. Since each pump provides approximately 200 gpm, the 400 gpm flow rate listed in the USAR would not be achievable.

In July 1993, the licensee concluded that the nuclear analysis department (NAD) should confirm that the appropriate AFW flow rate (200 gpm) was used in the main feedwater line break analysis. If so, NAD was to take steps to appropriately revise the USAR. If not, NAD was to perform the necessary analysis to show that 200 gpm was acceptable. At the same time, the licensee performed an operability evaluation and concluded there was a reasonable basis for considering 200 gpm acceptable. This conclusion was based partially upon a 1969 letter from the nuclear steam supply vendor and relied upon a less conservative initiating reactor trip scenario than was stated in the USAR. Because the 400 gpm value was considered a "paperwork" issue, the licensee did not establish a high priority for confirming that 200 gpm was an acceptable value.

Although the licensee considered the issue to be one where the USAR was incorrect, the schedule for updating the USAR was not taken into account in setting a resolution date. The USAR was updated in late December 1993 and was supposed to reflect changes to the USAR as of six months previous (i.e., up through June 1993). Although the incorrect USAR value was identified in November 1992, and the operability analysis performed in June 1993 declared 200 gpm to be the correct number, the USAR was not changed in the 1993 update.

Two years later, in June 1995, the licensee questioned the status of the FOI and whether the USAR should be updated. At that time, NAD had determined that the main feedwater line break analysis did assume a 400 gpm AFW flow rate, but had not yet redone the analysis to confirm that 200 gpm would be sufficient. Therefore, the licensee decided to not update the USAR, because acceptability of a 200 gpm AFW flow rate to mitigate the MFLR was not proven. It appeared the licensee did not fully consider the dichotomy of this decision: if 200 gpm was not an acceptable number for the USAR, then the plant was no longer within its design basis and the operability evaluation should have been revisited to ensure that AFW was still capable of performing its safety related function following a MFLR. The licensee also did not recognize or report that the plant was in an unanalyzed condition, since the 400 gpm flow rate assumed by the MFLR analysis was not achievable by the pumps, and the available 200 gpm flow rate was not analyzed. Nor did the licensee perform a safety evaluation to justify a "de facto" modification to the facility as described in the USAR.

The inspectors questioned the licensee about the status of the FOI. A member of the licensing staff responded that the licensee intended to update the USAR during the December 1997 update, but acknowledged that, as of the time of the inspection, the information necessary to support the update was not available. The inspectors then discussed the issue with the responsible technical engineer. The inspectors were informed that NAD had performed the analysis and concluded that 200 gpm was acceptable. However, the calculation was still undergoing the review and approval process. The licensee engineer stated that a June 30, 1997, date had been established for NAD to complete the review and approval process. The inspectors questioned whether the engineer had considered the USAR update schedule in establishing this date; the licensee responded that they had not, but would ensure that it would be taken into account.

10 CFR 50.9(a), "Completeness and Accuracy of Information," requires, in part, that information provided to the NRC by a licensee or information required by regulation to be maintained by a licensee shall be complete and accurate in all material respects.

10 CFR 50.71(e), "Maintenance of Records, Making of Reports," requires, in part, that each licensee periodically update the final safety analysis report (FSAR) to assure that the information included in the FSAR contains the latest material developed. Subsection 4 requires, in part, that revisions be filed such that the intervals between successive updates to the FSAR do not exceed 24 months. It further states that the revisions must reflect all changes up to a maximum of 6 months prior to the date of filing.

10 CFR 50.73(2)(ii)(B) requires, in part, that the licensee report any event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant.

10 CFR 50.59, "Changes, Tests and Experiments," permits the licensee, in part, to make changes to the facility as described in the safety analysis report without prior Commission approval provided the change does not involve an unreviewed safety question. It requires, in part, that the licensee maintain records of changes in the facility and that these records include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question.

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

The failure to correct the inaccurate 400 gpm AFW flow rate in the USAR, despite two opportunities to do so in December 1993 and 1995, is considered an apparent violation of 10 CFR 50.9 and of 10 CFR 50.71(e) (EEI 50-282/306-97008-04a and -04b).

The failure to report that the plant was outside its design basis when it was determined that the MFLR analysis used a 400 gpm AFW flowrate was considered an apparent violation of 10 CFR 50.73 (EEI 50-282/306-97008-05a). The failure to perform a safety evaluation to make permanent this change to the facility as described in the USAR and to verify that no unreviewed safety question existed was considered an apparent violation of 10 CFR 50.59 (EEI 50-282/306-97008-05b).

The failure to take prompt corrective actions to resolve the above described significant condition adverse to quality is considered an apparent violation of

10 CFR, Part 50, Appendix B, Criterion XVI, "Corrective Action" (EEI 50-282/306-97008-06).

c. Conclusions

Based upon knowledge of the required AFW flows for similar nuclear power plants, the inspectors considered the preliminary (unverified) results of the licensee's MFLR analysis to provide reasonable assurance that the AFW pumps were operable and could handle the main feedwater line rupture accident. However, the licensee did not take prompt and appropriate actions to confirm that the 200 gpm flow rate was acceptable and to correct the USAR.

E1.3 Modifications and Design Changes

a. Inspection Scope

The team reviewed several mechanical, electrical, and instrumentation and control design changes. The inspectors reviewed the design changes for an adequate description of the design change, necessary interdepartmental reviews for technical adequacy, 50.59 evaluations, adequate supporting calculations, adequate implementation of the design change, quality control (QC) reviews, post-modification testing, adequate documentation, and training on the design change, as needed. Design Changes reviewed are listed in back of this report.

b. Observations and Findings

The inspectors reviewed a sample of modifications from 1982 through 1996 and observed that the modifications generally made only minor changes and did not affect the design basis. The inspectors reviewed the associated safety evaluations in accordance with 10 CFR 50.59. The licensee showed a definite improvement in the quality of safety evaluations over the years, with the latter evaluations being much more comprehensive and in-depth. Based on reviews of safety evaluations and screenings, the inspectors did not identify any examples where an unreviewed safety question existed, although Section E3.2 discusses a concern of failure to generate a safety evaluation. The inspectors concluded that the modifications, including documentation, revisions, and post-modification testing, on the AFW system were acceptable.

c. Conclusions

Design changes and modifications reviewed, including documentation revisions and post-modification testing, on the AFW system were acceptable.

E1.4 Lack of Validation of Control Room (CR) Habitability Analysis Assumptions

a. Inspection Scope

The inspectors reviewed the control room ventilation system including original and recent calculations related to control room (CR) habitability. The team also reviewed design and licensing basis documents related to the system, equipment testing procedures, and the CR ventilation's compliance with regulations.

b. Observations and Findings

Background

The calculated radiation exposure to the CR operators is dependent on several factors including the flow rate of unfiltered air inleakage to the CR envelope assumed in the safety analysis. These assumed values are based on system design and are typically fixed but bounding values in the safety analysis. However, industry experience, as documented in NUREG/CR-4960, "CR Habitability Survey of Licensed Commercial Nuclear Power Generating Stations," indicates that air inleakage rates are commonly found to be significantly greater than the assumed values. This may be due to wear on dampers and door seals and degradation of duct and penetration seals.

As discussed in NUREG-4960, in evaluating CR habitability for inleakage of potentially contaminated unfiltered air, attention should be focused on penetration of the CR envelope, (ducts, piping, cabling, and doors), particularly system dampers. Air inleakage at these locations can occur for all types of CR habitability system designs, including those such as Prairie Island's that do not rely on maintenance of positive pressure relative to adjacent areas. In systems where positive pressure is not maintained, penetrations of the CR envelope may be the source of significant inleakage and a periodic test would demonstrate that the radiological analysis has not been negated due to increased inleakage. This testing had not been done at Prairie Island Nuclear Generating Plant (PINGP).

Prior to the inspection, the NRC resident inspection staff had raised several questions related to inconsistencies in assumptions between different control room dose calculations. Partly as a result of these questions, the licensee generated nonconformance report (NCR) 2010713 to address the inconsistencies. Subsequently, the licensee revised the CR personnel post-LOCA dose analysis (GEN-PI-023, Addendum 1) in an attempt to bound the identified non-conservative inputs in the original calculation. The revised inputs included use of control room volume values that added the Safeguards Chilled Water Rooms and the Relay Room as part of the control room envelope. The CR volume in the analysis changed from approximately 44,000 ft^3 to a volume of 164,000 ft^3.

Assumption for CR Unfiltered Inleakage Rate not Validated

The revised calculation concluded that the thyroid, whole body, and beta skin doses to the control room operators continued to satisfy the General Design Criteria (GDC) 19 criteria, namely 5 rem whole body or equivalent. However, the inspectors noted that the analytically determined total thyroid dose of approximately 27.6 rem provided little margin to the GDC 19 limit of 30 rem. The inspectors were concerned that a pivotal assumption made in the revised calculation, the unfiltered control room inleakage, assumed to be 165 cfm, had not been verified or validated by testing. Higher inleakage values could readily place the plant outside of the regulatory limit.

Based on reviews of the documentation and interviews with the licensee, the inspectors considered that the licensee had relied on generic guidance, without fully demonstrating that the inleakage value was appropriate. While no regulatory requirement was identified requiring validation of the assumption, the weak technical basis was a concern.

In response to the inspectors' concerns, the licensee discussed their regulatory and technical basis for concluding that the use of 165 CFM inleakage valve was appropriate.

The regulatory basis relied on use of NRC Standard Review Plan 6.4, Section II.3.d.2. In order to obviate testing of the inleakage value, the licensee assumed a leakage value just above the value that the SRP would require validation via testing. Further, the licensee noted other license basis documents where the NRC had referenced the subject SRP section. It appeared that the licensee was using the SRP guidance in a "piecemeal" fashion. For example, contrary to the discussion in the SRP, the gross leakage (calculated or measured) was not based on test data. Also, discussion with NRR indicated that correlating the CR volume to unfiltered CR leakage as the licensee was doing, was used as a starting point assumption during the licensing process. The actual inleakage may differ significantly and continued use of the SRP values should have a technical basis.

The licensee's technical bases for the adequacy of the assumed unfiltered inleakage rate were also discussed. The licensee staff stated they had confidence in the conservativeness of the assumed inleakage value based on arguments such as

(1) physical CR location which has minimal unsealed openings,
(2) a relative negative pressure in the auxiliary building during a LOCA (from the auxiliary building special ventilation system),
(3) sealing quality design of the isolation dampers, and
(4) use of an additional inleakage value (unverified) for post accident CR egress and ingress.

The inspectors noted the technical arguments continued to rely on the assumption that all penetrations are adequately sealed, that the assumed inleakage is in fact bounding and that degradation over the years has been minimal. A periodic test, which would demonstrate that the radiological analysis has not been negated due to increased inleakage, was not required and had never been conducted.

Although the Prairie Island Nuclear Generating Plant CR isolation dampers are inspected annually, the inspection consists only of a visual examination of damper mating surfaces and visual checks of closure. There are no minimum leaktightness performance requirements. The licensee staff stated that the louver style dampers were designed for maximum leakage of approximately 15 cfm at 4-inch pressure differential, and per the vendor, would maintain outstanding sealing characteristics through a broad range of pressure differentials. However, as noted by NRC inspections documented in NUREG/CR-4960, of the various damper styles in use for isolation purposes, based on industry empirical testing, louver-style dampers appear to have the highest potential for significant leakage. Louver-style dampers were found to be very poor at maintaining air tightness especially when exposed to a differential pressure of several inches of water.

The licensee stated that it would be prudent to confirm the amount of isolation damper degradation that may have occurred since installation and would further evaluate the need to confirm this assumed value. However, the licensee did not give a time frame for this evaluation. The licensee also planned to re-perform the control room dose analysis using the Dose Conversion Factors (DCF) from International Commission on Radiological Protection (ICRP) 30 instead of from ICRP 2 which were used in the latest calculation. It was expected that the ICRP 30 values would increase the margin between the analytical values and the GDC 19 limits.

c. Conclusions

For the CR habitability dose analysis, the inspectors considered that the licensee had a weak basis for concluding that the unfiltered inleakage rate assumption was conservative. PINGP relied on industry guidance and non-validated technical arguments without demonstrating that the actual inleakage value had not changed or that the CR envelope had not degraded. While no regulation or license condition appeared to require testing of the CR envelope or of the CR isolation damper, the low margin to the GDC 19 thyroid dose limit and the effects of the unfiltered inleakage on the analytical doses were of concern.

E1.5 Safeguards Chilled Water Piping

a. Inspection Scope

The team reviewed the design of the control room chilled water system piping to ascertain whether the piping would perform its intended function under plant design basis conditions.

b. Observations and Findings

The safeguards chilled water system was originally design class III and during the original design a detailed seismic analysis was not performed on the piping system. The Prairie Island USAR did not classify the piping system as design class 1, which at Prairie Island required a seismic evaluation. However, the piping provides cooling to several safety-related rooms through unit coolers or air conditioners. These rooms are:

4kV Safeguards Switchgear rooms
480V Safeguards Switchgear rooms
Relay room
Control room
Event Monitoring Equipment rooms
Residual Heat Removal Pits

In May 1996, the licensee questioned the architect/engineer regarding the seismicity of the safeguards chilled water system. The architect/engineer was able to locate seismic documentation for system components but not for the piping.

In response to the team's concerns that the piping was not design class I, the licensee produced documentation describing the safeguards chilled water piping walkdown, calculation ENG-ME-309, "Seismic Adequacy Review of Safeguards Chilled Water Piping," Revision 0, March 4, 1997, and safety evaluation, SE No. 21, Revision 2, May 2, 1997. This documentation qualitatively demonstrated that the safeguards chilled water system piping should maintain the pressure boundary during a seismic event. Heat load analysis qualifying equipment in the above rooms had been generated. The Safe Shutdown Earthquake (SSE) at Prairie Island was relatively small.

Horizontal acceleration	SSE	0.12g
Vertical acceleration	SSE	0.08g

The team's review of licensing requirements and the USAR found no requirement for the safeguards chilled water piping to be design class I piping.

c. Conclusions

The safeguards chilled water system was not seismically designed; however, the team did not identify any requirement in the USAR or licensing documents that required the piping to be seismic design class I.

E3 Engineering Procedures and Documentation

E3.1 Review of Calculations

a. Inspection Scope

The inspectors reviewed calculations in electrical, instrumentation and mechanical disciplines (see list at end of inspection report) for technical adequacy, verification of assumptions and overall correctness of conclusions.

b. Findings and Observations

The calculations ranged from those performed during initial construction of the plant in the early 1970's to some as late as 1995. The inspectors had minimal comments with the electrical, instrumentation, HVAC and pipe stress analyses reviewed. These calculations were considered acceptable with respect to assumptions, methodology, and conclusions. However, the inspectors noted minor discrepancies in many of the pump and hydraulic related mechanical calculations reviewed.

For example, during initial construction, a calculation was performed to determine the AFW pump discharge pressure. The controlled copy of the calculation did not show the calculation as being independently verified, showed numbers crossed out with new numbers written in, contained mathematical errors, and did not reflect changes made to the plant during installation. Similarly, a calculation for determining the total dynamic head did not use conservative assumptions (in regard to water temperature) and was not independently verified. Additionally, the inspectors determined that the assumed friction head losses were less than half of those in the installed system; however, the calculation was not revisited when actual piping information became available. In both cases, although the numerical results were incorrect, the overall conclusions of the calculations were not affected.

In 1990, during the station blackout project modifications, the Unit 2 condensate storage tanks (CSTs) were moved further away from the plant. A calculation, M-376-CD-001, was performed to determine the effects of this move on the net positive suction head (NPSH) available for the AFW pumps. The independent reviewer identified some errors in the original calculation, and performed an alternate calculation to correct those errors. However, the alternate calculation by the independent reviewer actually introduced more significant errors. For example, the independent reviewer did not calculate the worst case NPSH (from the #22 CST to the #11 AFW pump); instead, the reviewer calculated the line losses from the #22 CST to the #12 pump (which removed approximately 30 feet of line losses from the calculation). Additionally, the independent reviewer ignored the head loss from the pipe nozzle and through contractions in the pipe diameter, left out approximately 16 feet of pipe between the CSTs and the header, and made incorrect assumptions about head losses through elbows. The inspectors performed an independent calculation and determined that the NPSH available was about 27 feet, well above the required NPSH of 13 feet. Therefore, the calculational errors did not affect the AFW pump operability. The licensee acknowledged the errors in the calculation and was considering a revision to the calculation.

In 1992, the licensee performed calculation SYS-AF-002 to determine how quickly condensate would build up in the steam supply line to the TDAFW pump. The purpose of the calculation was to determine if the TDAFW pump could be considered operable if the steam line drains were isolated. The inspectors noted that the calculation was performed in January 1992, but the calculation was not validated until December 1992. Additionally the inspectors noticed that both the preparer and the independent reviewer used an incorrect formula for calculating the Nusselt number for the horizontal runs, both overlooked 11 feet of piping, and, in correcting a pipe length error in the original calculation, the independent reviewer introduced a new error by performing the calculations on the wrong diameter piping. Finally, the independent reviewer's alternate calculation contained mathematical errors: in calculating the Raleigh number, the reviewer forgot to convert one of the terms from feet per second squared to feet per hour squared. This introduced a conversion error equal to 12,960,000 seconds squared per hours squared. These errors had no impact on the calculation's conclusions, since the licensee had determined that the TDAFW pump must be considered inoperable if the drains were closed. However, the licensee acknowledged that the calculation needed revising to correct the errors.

In October 1992, the licensee performed calculation ENG-ME-292 to determine if sufficient cooling water flow could be passed through a half-open gate valve to the AFW pumps. Similar to the other calculations, errors were discovered by the inspector, including an incorrect number of elbows in the pipe and a non-conservative cooling water header pressure. Additionally, during review of the isometric drawings while performing an NPSH calculation, the inspectors noted that the isometric showed the cooling water connection to the AFW pumps to be 1½-inches in diameter versus the 4-inches claimed in the calculation. The errors resulted in the numerical value being significantly decreased; however, it still appeared to be above the required flow rate. The licensee prepared a nonconformance report and planned to revise the calculation.

In 1995, the licensee revised calculation ENG-ME-148 which evaluated the effects of flooding in the AFW pump room. During review of ENG-ME-148, Revision 1, the inspectors noted that it claimed (on page 4) that "supporting calculations performed by NSP's Nuclear Analysis Department [Reference 7] show that this flow rate can be readily handled by the floor drains, trench, and the gap under the doors leading the AFW rooms with less than 3 inch rise in water level." However, when the inspectors reviewed "Reference 7," which was the corporate Nuclear Analysis Department calculation V.SMN.94-006, the following errors were discovered: First, the NAD calculation made no attempt to estimate flow through the drains. During an inspection during the first week onsite, the inspectors observed that several of the small floor drains were clogged with dust and debris. The inspectors asked if the drains received periodic cleaning. The licensee's response was "no;" however, the drains were clear by the last week of inspection. The inspectors also noted that there was one large rectangular grated sump which led to a drain which, due to the water flow observed, appeared to be clear.

Second, the NAD calculation assumed that the trench running through the room was uncovered and then calculated various percentages of blockage, down to 10 percent open, due to the cover normally over the trench. However, during the walkdown, the inspectors observed that the trench was completely covered, with only three small (less than 2-inches in diameter) openings - one on the Unit 1 side and two on the Unit 2 side. These openings provided an access to the trench of less than 1 percent; considerably less than assumed in the calculation. Finally, the calculation evaluated the flow of water under the door. However, a mathematical mistake was made in that the preparer calculated a 1.25-inch gap across the length of the door rather than the actual condition of a ½-inch gap for 2.3 feet and ¾-inch gap for the remaining 4.5 feet of the door length. Ignoring the majority of the drains, due to the chance of their being clogged, the inspectors independently calculated the flow into the su[illegible]d normally open drain, along with more realistic flows under the door and into the trench. The inspectors found that the water buildup in the room would probably not exceed 6-inches, which was the height of several electrical connections.

10 CFR Part 50, Appendix B, Criterion III "Design Control," requires, in part, that design control measures shall provide for verifying or checking the adequacy of the design, such as by performance of design reviews or by use of alternate or simplified calculations. In the above calculations, the design control measures failed to verify the adequacy of the design in that the above errors were not identified during the verification or new errors were introduced by the verification review. This is considered a violation of 10 CFR Part 50, Appendix B, Criterion III (VIO 50-282/306-97008-08(DRS)).

c. Conclusions

While many of the calculations reviewed were considered acceptable, the inspectors noted weaknesses in the calculation verification program based upon the errors found in the mechanical calculations; some of which were introduced during the verification process. These errors were considered violations of design control. However, the inspectors acknowledged that, if taken individually, the errors had only minor safety significance, due to the conservative actions taken based upon the calculations or the margin available.

E3.2 Effect of Loss of Instrument Air on the Chilled Water System

a. Inspection Scope

The inspectors reviewed the licensee's actions regarding installation of a nitrogen bottle and use of operator action on an air-operated valve in the cooling water return line from the chilled water system. These actions were necessary to compensate for the consequences following a loss of instrument air. The inspectors reviewed work order 9505565, licensee event report (LER) 95-013, JSAR Section 10.3.3, and the abnormal operating procedures for loss of instrument air, and earthquakes.

b. Observations and Findings

During a walkdown of the control room chilled water system, the inspectors noted that nitrogen bottles were installed in the chilled water system room and airline tubing was staged to an air-operated valve (AOV) on the cooling water return line from the chilled water condenser. The licensee explained that during the licensee-conducted service water system operational performance inspection in August 1995, engineers had discovered that the cooling water return line valve failed closed on loss of instrument air. This resulted in the environmental qualification of some control room equipment being exceeded.

At that time, the licensee installed the nitrogen bottle and changed procedures to require operator action to connect the nitrogen supply to the AOVs following loss of instrument air. Additionally, the licensee determined the issue was reportable, and issued LER 95013.

During review of this issue, the inspectors determined that the nitrogen bottle was added to the rooms under a work order, using a standard anchor bolt installation procedure. The licensee justified use of a work order rather than a design change, primarily based upon the fact that the nitrogen bottle was not actually connected to the cooling water system. After further questioning by the inspectors, licensee engineers stated that they did not believe a safety evaluation was performed for the change, but they believed that appropriate procedures were revised.

The inspectors acknowledged that the installation of the nitrogen bottle, in itself, did not modify the system configuration, but they were concerned that the use of operator action to hook up the nitrogen supply to the air-operated valve constituted a change in the way the system was designed to operate following an event.

The inspectors noted that the USAR Section 10.3.3.1 stated that the chilled water system was "designed to provide a reliable means of cooling and filtering air supplied to the Control and Relay Rooms under both normal and post-accident conditions." The inspectors ascertained that the USAR statement could not be met under "both normal and post-accident conditions" based upon the licensee's determination that operator action was necessary following a loss of instrument air. Since the function of the system, as described in the USAR, was changed, the inspectors considered that a safety evaluation should have been performed under 10 CFR 50.59.

During a subsequent walkdown, the inspectors noted that the pressure gauge for the nitrogen bottle was normally closed. The inspectors questioned whether there was any surveillance procedure to ensure that the nitrogen bottles were regularly verified to be pressurized. Although the licensee believed that the bottles were checked as part of routine operator duties, this was not confirmed by the end of the inspection.

The licensee had alternate plans to cool the control room (such as by propping open doors) following an earthquake, and would probably have sufficient time to take those actions before equipment environmental qualifications were exceeded. However, the inspectors were concerned about other scenarios that might result in a loss of instrument air. The licensee noted that there were three instrument air compressors, each of which was fed from a different emergency diesel generator, although the system was non-safety related. Therefore, it would be unlikely that loss of offsite power would cause a loss of instrument air.

In 1996, the Office of Nuclear Reactor Regulation (NRR) reviewed the acceptability of the licensee modifying the design basis to take credit for operator actions for an inadequate intake line issue. The NRR staff concluded that, for the particular case, an unreviewed safety question existed for two reasons: The change to the licensee's design basis of requiring operator actions: (1) might increase the probability of a malfunction of equipment important to safety previously evaluated in the USAR because operator intervention was now being relied upon for effective performance of systems important to safety and (2) might result in the possibility for creating an accident or malfunction of a different type than evaluated previously in the USAR because making the effective performance of systems important to safety reliant upon human intervention could potentially introduce unanalyzed failure modes caused by operator acts of omission or commission.

c. Conclusions

The inspectors determined that the nitrogen bottle installation and resultant dependence on operator action appeared to be a change to the system function as described in the USAR. The issue is considered an Unresolved Item (URI 50-282/306/97008-09) pending coordination with NRR to determine if this example of use of operator actions involves an unreviewed safety question.

E3.3 Instrumentation Setpoint Methodology Review

a. Inspection Scope

The inspectors reviewed design basis document follow on item FOI 0060, "Evaluate Basis for Precautions, Limitations and Setpoints (PL&S)," dated May 18, 1990, which was still open and required further licensee review. This follow on item questioned the lack of a clear basis for existing setpoints. Also reviewed were Technical Specification setpoint values and corresponding values used in plant procedures.

b. Observations and Findings

Follow-on Item 0060, "Evaluate Basis for PL&S," dated May 18, 1990, questioned the existing basis for various plant setpoints and stated that a project should be started to clearly establish the status of the PINGP setpoint methodology and handling of calculations and safety evaluations versus current regulatory expectations. A review of the existing plant correspondence and discussions between the inspectors and licensee indicated that the technical bases for some of the plant's limiting safety system settings and other safety-related setpoints may not exist or may be inadequate. The setpoints may be inadequate in that no margin to account for instrumentation uncertainties existed between some Technical Specification (TS) setpoints and corresponding values used in plant accident analyses.

In response to this concern, but subsequent to the inspectors leaving the site, the licensee stated that the basis for the plant's existing setpoints and limiting safety system settings was the plant-specific PL&S document developed by Westinghouse and backed up by channel uncertainty calculations also performed by Westinghouse.

The credibility of the Westinghouse PL&S-based setpoints was to be verified by the plant-specific setpoint calculations to indicate that a margin exists to assure that the plant's analytical limits and safety limits would not be exceeded during normal operation and design basis accidents. The results of this effort to date were provided to the inspectors in the form of a table comparing actual plant setpoints, TS setpoints, safety analysis setpoints, and instrument uncertainties assumed in the PL&S or design specifications. The inspectors noted that the table was not comprehensive because not all of the limiting safety system settings (LSSS) and limiting setpoints from the plant's TS were encompassed by the table. Further, for some of the setpoints listed in the table, including LSSS such as overtemperature delta T and overpower delta T, no margin existed between the setpoint values from the TS and the corresponding setpoints used in the safety analyses. However, the actual setpoints were consistently more conservative than the T.S. setpoints.

The inspectors were not able to determine the acceptability of the Prairie Island setpoint methodology process but did note that the licensee was working with other utilities and appeared to be following industry guidance such as ANSI/ISA-S67.04, "Setpoints for Nuclear-Related Instrumentation." The concern regarding lack of margin to account for instrumentation uncertainties between some TS setpoints and corresponding values used in plant accident analyses may be contrary to 10 CFR 50.36, "Technical Specifications." 10 CFR 50.36 states, in part, that LSSS must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. This issue of setpoint adequacy is considered an Unresolved Item pending further review by NRR and Region III (URI 50-282/306/97008-10(DRS)).

c. Conclusions

The technical bases for some of the plant's limiting safety system settings and other safety-related setpoints may not exist or may be inadequate. The inspectors were not able to determine the acceptability of the PI setpoint methodology process but did note that PINGP was working with other utilities and was following industry guidance. This issue remains unresolved pending further review by the NRR and Region III.

E3.4 Drawing Control

a. Inspection Scope

The team performed system walkdowns on the selected systems, reviewed the system configuration for consistency with design drawings, and assessed the material condition of the systems.

b. Observations and Findings

The team noted errors in the control room air flow diagram on drawing NF-39603-1, Revision AH. Damper NFD-23 was shown on the 3,000 CFM duct, but was installed in the 12,000 CFM duct. The drawing shows device TE 15781 on the discharge of the train A clean up filter fan; however, device TE 15781 was installed on the suction side of the fan. A damper on the discharge duct of the control room air handler in Unit 1, train A was not shown on the drawing.

On flow diagram NF-39603-3, Revision AE, on the chilled water system, temperature transmitter TT-17402 was shown on the cooling water line between manual valves CL-16-8 and CL-16-9. In the plant, the transmitter was between valve CL-16-9 and the flexible connection.

On condensate makeup piping isometric drawing X-HIAW-106-188, Revision B, butterfly valve C-41-2 was shown on the condensate line between auxiliary feedwater pumps 12 and 21. However, the internals had been removed from this valve. Incorrect drawing information on this valve impacted both the flow modeling and net positive suction head calculations.

In response to the inspectors' question, the licensee stated a walkdown of the system was planned for within two weeks of the team's exit date.

c. Conclusions

The team's identification of the above discrepancies in system drawings indicated a weakness in the drawing control program to assure plant drawings accurately reflect plant status.

E7 Quality Assurance in Engineering Activities

E7.1 Review of Safety Audit Committee Meeting Minutes and Operations Committee Meeting Minutes

a. Inspection Scope

The inspectors reviewed the safety audit committee (SAC) meeting minutes for June, September, and December 1996. The inspectors also reviewed the Operations Committee (OC) meeting minutes for October 1996 through April 1997, and witnessed portions of an OC meeting on April 18, 1997.

b. Observations and Findings

In general, based upon review of the meeting minutes, the SAC meetings appeared to have an appropriate focus and to accomplish the requirements of TS 6.2. The inspectors noted that the OC meeting minutes were extremely short, merely listing the items discussed during the meeting. The inspectors observed that it was difficult to determine from the meeting minutes what was accomplished during the OC meeting. During the OC meeting witnessed by the inspectors, the inspectors determined that the required OC members were present, that the members were prepared for the meeting, and that there was a good discussion of the issues presented to the OC members.

c. Conclusions

The inspectors concluded that the SAC and OC meetings fulfilled their TS requirements and provided the necessary oversight function for which they were intended.

E7.2 Quality Audits

a. Inspection Scope (40500)

The team reviewed licensee quality assurance audits and assessments and the licensee's corrective action relative to deficiencies identified during the audits.

b. Observations and Findings

The licensee's quality assurance program updated in 1996 included an audit plan or schedule based on the four SALP functional areas. The licensee audit teams were normally composed of quality personnel from both Prairie Island and Monticello plus specialists as needed.

The team reviewed four recent audits and numerous quality surveillances performed at Prairie Island. Findings were documented and presented to plant line management for initiation of appropriate corrective action. Correction of deficiencies identified by the findings appeared to be thorough and timely. Corrective actions were reviewed by Quality Assurance to assure all aspects of the finding were addressed and properly corrected.

c. Conclusions

Based on the sample examined, the team considered the licensee's quality verification program to be adequately designed and implemented. Corrective actions on recent QA findings were appropriate; however, corrective actions violations for older issues were identified in Sections E1.1, E1.2, and E8.4 of this report.

E8 Miscellaneous Engineering Issues

E8.1 Closed LER 282/306/96010: Auxiliary Feedwater Pumps Not Protected Against Runout for All Conditions. This event was previously discussed in Inspection Reports 50-282/306/96007 and 50-282/306/96010 and a non-cited violation was issued. During the SOPI, the inspectors witnessed portions of the licensee's setpoint modification for Unit 1, including the post-modification test. No problems were observed. As all corrective actions for this modification are now complete, this LER is closed.

E8.2 (Closed) LER 282/306/97003: Discovery That the Auxiliary Feedwater Pumps Will Trip on Low Steam Generator Pressure During a Complete Loss of Feedwater ATWS Event. During review of a safety evaluation being prepared to resolve the issue described in LER 96010, the licensee identified that the increased discharge pressure setpoints would result in an AFW pump trip during an anticipated transient without scram (ATWS). The licensee identified that an AFW pump trip was not considered during the generic ATWS analysis used by the plant. Following identification of the issue, the licensee obtained a plant-specific analysis assuming tripping of the AFW pumps. The inspectors discussed the results of the analysis with the licensee and reviewed the vendor information describing the assumptions and results of the analysis. The inspectors concluded that the licensee had appropriately resolved this issue. The inspectors concluded that the finding constituted a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Due to the licensee identifying the issue and promptly and adequately correcting it, the violation is being treated as a Non-Cited Violation (NCV 50-282/306/97008-11), consistent with Section VII.B.1 of the NRC Enforcement Policy. This LER is closed.

E8.3 (Closed) LER 282/306/97004: AMSAC Actuation Blocking Setpoint Inadvertently Set Non-Conservatively High. During a system review, a licensee engineer discovered that the AFW pump anticipatory start signal setpoint upon an ATWS did not agree with the USAR value. The licensee determined this was because a previous setpoint calculation assumed that first stage turbine impulse pressure was linear, when it was not. The licensee promptly determined the correct values and reset the setpoints. The inspectors reviewed the licensee's actions and determined that the corrective actions taken were acceptable. The inspectors concluded that the finding constituted a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Due to the licensee identifying the issue and promptly and adequately correcting it, the violation is being treated as a Non-Cited Violation (NCV 50-282/306/97008-12), consistent with Section VII.B.1 of the NRC Enforcement Policy. This LER is closed.

E8.4 (Open) LER 50-282/306/96-13; Unresolved Item (50-282/96008-09): Cable Trays Not Meeting Separation Criteria. On July 31, 1996, the licensee reported that several cases of cable trays did not meet the separation criteria in Section 8.7.2 the Updated Safety Analysis Report (USAR). This issue was previously discussed in Inspection Reports 50-282/306/96008 and 50-282/306/96014. The inspectors concluded that the licensee's evaluation of this issue was untimely and narrowly focused. It took over four years to complete the safety evaluation and to determine that the configurations were outside the plant's design basis and, therefore, reportable. After making the report, pursuant to 10 CFR 50.72, the licensee's investigation of the issue involved only those tray interactions listed in the original findings, until prompted by additional NRC findings, despite evidence in the original list that the interactions might not be limited to original findings. This is considered a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which requires, in part, that measures be established to assure that conditions adverse to quality, are promptly identified and corrected. (VIO 50-282/306/ 97008-13).

The inspectors also reviewed portions of the licensee's modifications and actions in response to this issue and interviewed licensee staff working on the issue's resolution. The final review of the operability evaluation and the final modifications will be coordinated with NRR to verify acceptability of use of recent IEEE guidance and use of a 1971 Pioneer technical document to justify cable separation distances greater than described in the USAR. The Unresolved Item will remain open.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented a summary of preliminary findings to members of Northern States Power management at the exit meeting on May 16, 1997. In addition, a telephone exit was conducted on June 13, 1997, to notify the licensee of additional examples of violations. 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

K. Albrecht, General Superintendent Engineering
T. Amundsen, General Superintendent Engineering
J. Curtis, Superintendent, Electrical Systems Engineering
J. Goldsmith, General Superintendent, Engineering
S. Heideman, Superintendent Mechanical Systems Engineering
J. Hill, Manager Quality Services
G. Lenertz, General Superintendent Plant Maintenance
J. Leveille, Licensing & Management Issues
C. Mundt, Superintendent, I&C Systems Engineering
R. Pearson, Superintendent, Mechanical Systems Engineering
R. Peterson, Design Standards, Principal Engineer
T. Silverberg, General Superintendent Plant Operations
J. Sorensen, Plant Manager
M. Wadley, Vice President, Nuclear Generation

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 90712:	In Office Review of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92903:	Followup - Engineering
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors
IP 93801:	Safety System Functional Inspection
TI 2515/118:	SW System Operational Performance Inspection

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

282/306/97008-01	IFI	Review of AFW Flow Model
282/306/97008-02	EEI	Apparent Viol. of Test Control involving AFW Acceptance Criteria
282/306/97008-03	EEI	Apparent Viol. of Corrective Action involving failure to review acceptance criteria of other ASME pumps
282/306/97008-04a	EEI	Apparent Viol. of 50.71(e) involving failure to update the USAR AFW accident flowrate

ITEMS OPENED, CLOSED, AND DISCUSSED (cont'd)

282/306/97008-04b	EEI	Apparent Viol. of 50.9 involving failure to provide accurate USAR AFW accident flowrate
282/306/97008-05a	EEI	Apparent Viol. of 50.73 involving failure to report the USAR MFLR AFW accident flowrate was outside DB
282/306/97008-05b	EEI	Apparent Viol. of 50.59 involving failure to perform SE to address change to the facility as described in the USAR resulting from incorrect AFW flow rate
282/306/97008-06	EEI	Apparent Viol. of Corrective Action involving failure to correct USAR AFW flowrate
282/306/97008-07	URI	Determination of seismicity requirements for safeguards chilled water piping
282/306/97008-08	VIO	Design Control violation involving inadequate calculation verification
282/306/97008-09	URI	Determination of acceptability of manual action installing N_2 bottle on Loss of IA for SCW system
282/306/97008-10	URI	Determination of acceptability of instrumentation setpoint uncertainties and of administrative control of setpoints
282/306/97008-11	NCV	Design control non-cited violation for AFW trip on Lo SG Press during Loss-of-FW-ATWS
282/306/97008-12	NCV	Design control non-cited violation for non-conservative setting of AMSAC Actuation Blocking Setpoint
282/306/97008-13	VIO	Design Control violation involving Untimely corrective action on cable tray separation issue

Closed

282/306/96-010	LER	Determination that the Auxiliary Feedwater Pumps are not Protected Against Runout for all Accident Conditions
282/306/97008-11	NCV	Design control non-cited violation for AFW trip on Lo SG Press during Loss-of-FW-ATWS
282/306/97003	LER	Discovery that AFW Pumps will trip on Low SG Pressure during a complete Loss-of-FW-ATWS Event
282/306/97008-12	NCV	Design control non-cited violation for non-conservative setting of AMSAC Actuation Blocking Setpoint
282/306/97004	LER	Non-conservative setting of AMSAC Actuation Blocking Setpoint

Discussed

EA 96-402	VIO	Failure to Identify an Unreviewed Safety Question Existed in a Safety Evaluation of the Emergency Cooling Water Intake Line
282/306/96013	LER	Cable Trays Not Meeting Separation Criteria
282/306/96008-09	URI	Cable Trays Not Meeting Separation Criteria

LIST OF ACRONYMS USED

AB	Auxiliary Building
AFW	Auxiliary Feedwater
AMSAC	ATWS Mitigating System Actuation Circuitry
ANSI	American National Standards Institute
AOV	Air-Operated Valve
ARP	Alarm Response Procedure
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
CFM	Cubic feet per minute
CFR	Code of Federal Regulations
CL	Cooling Water
CR	Control Room
CST	Condensate Storage Tank
DBA	Design Basis Accident
DBD	Design Basis Document
DCD	Dose Conversion Factor
DRS	Division of Reactor Safety
EA	Enforcement Action
EEI	Escalated Enforcement Issue
EOP	Emergency Operating Procedure
EQ	Environmentally Qualified
FOI	Follow-On Item
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GPM	Gallons Per Minute
HVAC	Heating, Ventilation and Air Conditioning
I&C	Instrumentation and Controls
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronic Engineering
IFI	Inspection Followup Item
IP	Inspection Procedure
ISI	Inservice Inspection
IST	Inservice Testing
ISTS	Improved Standardized Technical Specifications
LCO	Limiting Conditions for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LSSS	Limiting Safety System Settings
MDAFW	Motor Driven Auxiliary Feedwater Pump
MFLR	Main Feedwater Line Rupture
NAD	Nuclear Analysis Department
NCR	Nonconformance Report
NCV	Non-cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSP	Northern States Power Company
OC	Operations Committee

LIST OF ACRONYMS USED (cont'd)

OOT	Out-cf-Tolerance
OP	Operations Procedure
PINGP	Prairie Island Nuclear Generating Plant
PDR	Public Document Room
PL&S	Precautions, Limitations and Setpoints
PPB	Part Per Billion
QC	Quality Control
RCS	Reactor Coolant System
RFO	Refueling Outage
SAC	Safety Audit Committee
SALP	Systematic Assessment of Licensee Performance
SE	Safety Evaluation
SER	Safety Evaluation Report
SI	Safety Injection
SG	Steam Generator
SOPI	System Operational Performance Inspection
SP	Surveillance Procedure
SPDR	Surveillance Procedure Deviation Report
SSE	Safe Shutdown Earthquake
SWOPI	Service Water Operational Performance Inspection
TDAFW	Turbine Driven Auxiliary Feedwater
SRP	Safety Review Plan
TS	Technical Specifications
URI	Unresolved Item
USAR	Updated Safety Analysis Report
VIO	Violation
WC	Water Column
ZH	Safeguards Chilled Water System
ZN	Control room ventilation system

PARTIAL LIST OF PROCEDURES USED AND DOCUMENTS REVIEWED

Calculations

- Auxiliary Feedwater Pump Room Heatup Analysis, Tenera 194001-2.2-004 (NSP ENG-ME-021), Rev. 0, 11/22/91
- Calculation of Total Dynamic Head for Auxiliary Feedwater Pumps, Pioneer Services & Engineering Initial Plant Design, Rev. 0, 6/18/68
- Cooling Water Header Pipe Failure Causing Flooding in the Auxiliary Feedwater Pump/ Instrument Air Compressor Room, NSP ENG-ME-148, Rev. 0, 12/16/94 and Rev. 1, 8/8/95
- Condensate Storage Tank Piping Friction Loss NPSH, Fluor Daniel M-376-CD-001, Rev. 0, 10/5/90
- Control Room Loss of Ventilation, Tenera 192210-2.2.001, Rev. 0, 1/14/92
- Control Room Ventilation System Design, NSP ENG-ME-188, Rev. 0, 5/18/95
- Control Room Volume, NSP ENG-ME-314, Rev. 0, 4/16/97
- Detailed Analysis of Auxiliary Feedwater Pump Room Internal Flooding, NSP V.SMN.94.006, Rev. 0, 4/7/94
- Determination of Possible Flow Rate in Cooling Water (CL) to Auxiliary Feedwater Pump Piping with Gate Valve Half Open to Verify Design Flow Will Pass Thru Half Open Gate Valve, NSP ENG-ME-292, Rev. 0, 10/23/92
- Determine Auxiliary Feedwater Pump Discharge Piping Design Pressure, Pioneer Services & Engineering Initial Plant Design, Rev. 0, 6/25/70
- Maximum Out-of-Service Time for Steam Line Drains Upstream of the Auxiliary Feedwater Pump Steam Supply Control Valves CV-31998 & CV-31999, NSP SYS-AF-002, Rev. 0, 1/13/92
- Reload Safety Evaluation Methods Applicable to Prairie Island Units, NSP NSPNA-8102-A, Rev. 6, 8/95
- Replacement Valve Evaluation - Auxiliary Feedwater Pump Drive Turbine Steam Supply System, Fluor Power Services 217450-269, Rev. 0, 2/3/81
- Safeguards Chilled Water Evaluation, NSP ENG-ME-028, Rev. 1, 5/12/94
- Engineering calculation ENG-ME-315, Rev. 0
- 4160 Volt Safeguards Degraded Bus Voltage Setpoint, SPC-EA-006, Rev. 1.
- NSP Prairie Island Nuclear Generating Station, Setpoint Methodology, Revision 1
- Unit 1 4 KV Bus Minimum Voltage, ENG-EE-061, Rev. 0
- 480 Switchgear Branch Breaker Settings, E-385-EA-21, Rev. 2
- Degraded Voltage Relay Drop-out, E-415-EA-3, Rev. 1
- Cable Sizing Calculation for Mod #96EB01, ENG-EE-095, Rev. 0
- 480 VAC Supplemental Coordination Study, ENG-EE-014, Rev. 0
- Justification for Low Voltage Concerns (230 VAC), ENG-EE-052, Rev. 0
- Diesel Generator Steady State Loading for a LOOP Coincident with a SBO, ENG-EE-045, Rev. 2
- Safeguards Low Voltage Power Systems Ground Fault Current Calculation, ENG-EE-092, Rev. 0
- Cable Ampacity for Control & Power Cables for Mod #96EB01, ENG-EE-089, Rev. 0
- Medium Voltage Ground Fault Calculations, ENG-EE-093, Rev. 0
- PI Offsite and CR Habitability LOCA dose for Vantage Plus Fuel, Calculation M-834532
- Control Room Personnel Post-LOCA Dose, Calc. GEN PI-023, Addendum 1

PARTIAL LIST OF PROCEDURES USED AND DOCUMENTS REVIEWED (cont'd)

Design Basis Documents

DBD-SYS-28B, Rev. 1, "Auxiliary Feedwater System Design Basis Document,"
DBD-TOP-01, Rev. 1, "Accident Analysis Topical Design Basis Document," 12/5/95
DBD-STR-02, Rev. 1, "Auxiliary Building"

Drawings

"Auxiliary Feedwater System, Unit 1," Flow Diagram NF-39222, Rev. AW
"Auxiliary Feedwater System, Unit 2," Flow Diagram NF-39223, Rev. AU
"AFW Logic Diagrams" NF-40312 and NF-40767
"Cooling & Chilled Water Systems & Fire Protection for Vent Filters in Auxiliary & Containment Buildings," Flow Diagram NF-39603-4, Rev. T
"Lab & Service Area A/C & Chilled Water Safeguard System," Flow Diagram, NF-39603-3, Rev. AE
"12-inch Condensate Makeup AFW Pump Suction Piping," Isometric, NQ-118234, Rev A
"Condensate Makeup to AFW Unit 1," Isometric X-HIAW-1106-188, Rev. B
"Condensate Makeup to AFW Unit 2," Isometric X-HIAW-1106-261, Rev. D
"30-foot Diameter and 29-foot High Dome Roof Condensate Storage Tank," Isometric Detail X-HIAW-74-56, Rev. 1
"Condensate Storage Tank 12-inch Diameter Shell Nozzle (Butt Welded)," Isometric Detail X-HIAW-74-57, Rev. 1
"Main & Aux. Steam Flow Diagrams," NF-39218, NF-39219

Miscellaneous

Tank Book, pages for the Condensate Storage Tank, 7/1/93

Modifications

Auxiliary Feedwater Pump Flush Strainer, 89A0089, 11/23/94
Auxiliary Feedwater Pump Suction Cooling Water Vent Loop Seal, 92L369, 2/8/94
Chilled Water Heat Removal Hanger and Piping Modification, 82Y230, 1/6/82
Chlorine Monitor Removal, 89Y060, 4/14/93
Document the As-Found Condition of 2-AFWH-42, 89A0110, 4/27/89
Prevent Auxiliary Feedwater Pump's Shaft Driven Lube Oil Pump from Becoming Air-Bound, 90A193, 11/30/90
Relocate 11/22 Turbine Driven Auxiliary Feedwater Pump Steam Valves, 84L838, 1/18/88
Replacement of 122 Control Room Air Handler Cooling Coil, 88A0002, 2/8/88
Install Flow Meters for Chilled Water Pumps 121 and 122, 79L401
Alarm in the Control Room for TD Auxiliary Feedwater Pump Over Speed Trip, 79L564
Provide Lo-Lo Level Annunciators for 11 and 21 CST on AFW Panels, 79L566
AFWP Low Discharge Pressure and Low Suction Pressure Trip, 80L579
Add Phase to Phase PT's to Safeguard 4 KV Busses, 93L421, Rev. 0
480 V Common Loads, 96EB01, Rev. 0
Install Battery Disconnect Switches, 93L415, Rev. 0
Load Sequencer Source Breaker Interlock, 95L485, Rev. 0

PARTIAL LIST OF PROCEDURES USED AND DOCUMENTS REVIEWED (cont'd)

Removal of Automatic Start of AFW Pumps, 77L397
AFW Pump Runout Protection, 96AF01

Purchase Specifications

Auxiliary Feedwater Pumps, 10/1/70
Miscellaneous Reactor Plant Control Valve, 12/21/70
Miscellaneous Valves for Nuclear Service, 12/7/70

Technical Manuals

"Auxiliary Feedwater Pumps," X-HIAW-258-23
"Auxiliary Feedwater Pump Turbine," X-HIAW-258-24

QA - Committee Meeting Minutes

Safety Audit Committee Meeting Minutes, 6/7/96, 9/19/96, and 12/14/96
Operations Committee Meeting Minutes #2158 - 2237, 10/2/96 - 4/8/97

Surveillance Procedures Reviewed/Observed

SP 1100, 12 Motor-Driven Auxiliary Feedwater Pump Monthly Test,
SP 1101, 12 Motor-Driven Auxiliary Feedwater Pump Once Every RFO Test
SP 1102, 11 Turbine-Driven Auxiliary Feedwater Pump Monthly Test
SP 1103, 11 Turbine-Driven Auxiliary Feedwater Pump Once Every RFO Test
SP 2100, 21 Motor-Driven Auxiliary Feedwater Pump Monthly Test
SP 2101, 21 Motor-Driven Auxiliary Feedwater Pump Once Every RFO Test
SP 2102, 22 Turbine-Driven Auxiliary Feedwater Pump Monthly Test
SP 2103, 22 Turbine-Driven Auxiliary Feedwater Pump Once Every RFO Test

SP 2216, 4.16 KV Safeguards Bus 25 Undervoltage Relay Calibration
SP 2218, Monthly 4 KV Bus 25 Undervoltage Relay Test
SP 2150, D5 Diesel Generator Functional Test
SP1002A, Analog Protection System Calibration
SP1024, Reactor Water Storage Tank Level for Unit 2
SP1035A, Reactor Protection Logic Test at Power
SP2150-D5, Diesel Generator Functional Test

Emergency Procedures Reviewed

1FR-S.1, Response to Nuclear Power Generation/ATWS
2E-0, Reactor Trip or Safety Injection, and Basis

Operating Procedures Reviewed

C28-2, System Prestart Checklist, AFW System, Unit 1, dated 2/21/96
C28-2, System Prestart Checklist, AFW System, Unit 1, dated 3/1/96
C28-7, System Prestart Checklist, AFW System, Unit 2, dated 3/23/97

PARTIAL LIST OF PROCEDURES USED AND DOCUMENTS REVIEWED (cont'd)

1C28.1, AFW System Unit 1
2C28.1, AFW System Unit 2
C28.1 AOP1, Steam Binding Of An AFW Pump
5AWI 1.5.0, Procedure Control
5AWI 1.5.1, Procedure Deviation Process
5AWI 1.5.3, Periodic Procedure and Checklist Review
5AWI 1.5.4, Temporary Memos
5AWI 3.10.5, Plant Equipment Labeling
5AWI 4.4.0, Drawing Control
PINGP 196, Turbine Bldg Data - Unit 2
NSP Work Order 9702379, Pre-Op Test on 22 TD AFWP Low Pressure

Alarm Response Procedures Reviewed

ARP C47009
ARP C47010

Training Documents Reviewed

Job Performance Measures AF-1 through AF-5
Job Performance Measures AF-5F
Job Performance Measures AF-5F-1
Job Performance Measuras AF-6S
Job Performance Measures AF-7
AFW System Lesson Plan, P8180L-007, R4
AFW System Lesson Plan, P8440L-507, R3
Simulator Continuing Training Course Outline, P9160S
License Requalification Training Program Description, P9100
Simulator Change #97I-002
PINGP 1224, Crew Training on AFW System changes dated 4/15/97

Miscellaneous Licensee Documents Reviewed

Licensing Commitments N-964, N-965, and N-794
USAR Input Item 90-098
Safety Evaluation 470, AFW Pump Runout Protection
Safety Evaluation 472, AFWP Operability with Auxiliary LO Pump OOS
Temporary Memo TMA 1997-0022
Temporary Memo TMA 1997-0028
Temporary Memo TMA 1997-0035
Temporary Memo TMA 1997-0041
Temporary Memo TMA 1997-0042
Temporary Memo TMA 1997-0059
Temporary Memo TMA 1997-0065
H3.1, Outplant Equipment Labeling
PINGP Updated Safety Analysis Report, Various Section
PINGP Technical Specifications