December 20, 2000

EA 00-282

Mr. J.Sorensen
Site General Manager
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT - NRC INSPECTION REPORT 50-282/00-13(DRS); 50-306/00-13(DRS)

Dear Mr. Sorensen:

On November 3, 2000, the NRC completed the first baseline safety system design and performance capability inspection at your Prairie Island Nuclear Generating Plant. On November 3, 2000, the results were discussed with Mr. Schuelke and other members of your staff. Additional information was provided to you and other members of your staff during a conference call on December 7, 2000. The enclosed report presents the results of the inspection.

The inspection was a detailed examination of design activities and records as they related to ensuring that the cooling water system was capable of performing required post-accident functions, and to verify compliance with the Commission's rules and regulations and the conditions of your license. Within these areas, the inspection consisted of observations of activities, discussions with cognizant personnel and a selective examination of procedures, design documents, and representative records.

This report discusses a design issue that appeared to have more than low safety significance. The issue involved an inadequacy in the original design of the three safety related deep draft cooling water (service water) pumps and an inappropriate design change to the pumps in 1977. The original design and installation failed to require safety related electrical power for the filter backwash system used for the pump drive shaft lubricating and cooling water. This could have resulted in the pumps becoming inoperable due to clogging of the system filters during a loss of offsite power. In addition, a design change in 1977 inappropriately reclassified the bearing lubricating water source from safety related to non-safety related. These issues resulted in the installation of non-safety related bearing lubrication water sources for the three safety related vertical cooling water pump drive shaft bearings. The problems were assessed, using the applicable Significance Determination Process, as a potentially safety significant finding that was preliminarily characterized by the significance determination process as having substantial safety significance (Yellow).

After review of this issue, licensee personnel declared all three safety related cooling water pumps inoperable on November 1, 2000. Both units were affected and steps were initiated to shut down both reactors. In order to avoid shutdown of both units, a request for enforcement discretion was requested and a 14 day Notice of Enforcement Discretion (NOED) was granted on November 1, 2000 (EA 00-282).

The cooling water pump issue appeared to be an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," (Enforcement Policy) NUREG-1600. The current Enforcement Policy is included on the NRC's website at www.nrc.gov/OE.

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

Please contact Ron Gardner at (630) 829-9751 within seven days of the date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision, and you will be advised by separate correspondence of the results of our deliberations on this matter.

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

In addition, the NRC inspectors identified two other issues, which were considered to be of very low safety significance. These issues are listed in the summary of findings and are discussed in the report details. These issues were entered into your corrective action program and are being treated as Non-Cited Violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy.

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

Sincerely,

/RA/

John A. Grobe, Director
Division of Reactor Safety

Docket Nos. 50-282, 50-306
License Nos. DPR-42, DPR-60

Enclosure: Inspection Report 50-282/00-13(DRS);
50-306/00-13(DRS)

cc w/encl: Plant Manager, Prairie Island
M. Wadley, Chief Nuclear Officer
G. Eckholt, Site Licensing Manager
S. Northard, Nuclear Asset Manager
J. Malcolm, Commissioner, Minnesota
Department of Health
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Dakota Community
J. Silberg, Esquire
Shawn, Pittman, Potts, and Trowbridge
A. Neblett, Assistant Attorney General
Office of the Attorney General
S. Bloom, Administrator
Goodhue County Courthouse
Commissioner, Minnesota Department
Of Commerce

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos: License Nos:	50-282; 50-306 DPR-42; DPR-60
Report No:	50-282/00-13(DRS) and 50-306/00-13(DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Prairie Island Nuclear Generating Plant
Location:	1717 Wakonade Drive East Welch, MN 55089
Inspection Dates:	October 16 - November 3, 2000
Inspectors:	H. Walker, Team Leader A. Dunlop, Mechanical Inspector J. Gavula, Mechanical Inspector G. Hausman, Electrical Inspector K. O'Brien, Mechanical Inspector D. Prevatte, Mechanical Inspector (contractor)
Approved by:	Ronald N. Gardner, Chief Electrical Engineering Branch Division of Reactor Safety

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas) reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

IR 05000282-00-13(DRS); IR 05000306-00-13(DRS), on 10/16 - 11/03/2000, Nuclear Management Corporation, LLC, Prairie Island Nuclear Generating Plant. Design activities and records related to the cooling water system and the ability of the system to perform it's design function.

The inspection was conducted by region based inspectors. One Yellow finding, one Green finding, and one No Color finding were identified during the inspection. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP did not apply are indicated by "No Color."

Inspector Identified Findings

Cornerstone: Mitigating Systems

- Yellow. The original design and installation of the three safety related deep draft cooling water (service water) pumps failed to require safety related electrical power for the filter backwash system for the water source for bearing lubrication and cooling of the pump drive shaft bearings. During a loss of offsite electrical power, this could have resulted in the clogging of the filters after a short time and, with the loss of shaft bearing lubrication water, inoperable cooling water pumps. In addition, a design change in 1977 inappropriately reclassified the safety related bearing lubricating water source for the pumps from safety related to non-safety related. This resulted in the installation of non-safety related bearing lubrication water sources for the safety related drive shaft bearings. Licensee personnel investigated the issue on November 1, 2000, and declared all three of the safety related cooling water pumps inoperable. Both units were affected and a Notice of Enforcement Discretion was granted on November 1, 2000, to allow both units to continue operations. This issue was identified as an apparent violation of Criterion III of 10 CFR Part 50, Appendix B (Section 1R21.2).

- Green. A Non-Cited Violation was identified regarding the potential failure of one of the air/vacuum valves associated with the cooling water pumps that would result in significant flooding in the area of the safety related cooling water pumps, and could result in the flooding and subsequent in-operability of all three cooling water pumps. After reviewing this issue, on October 26, 2000, licensee personnel declared all three cooling water pumps inoperable. An operator was stationed in the pump area within two hours to shutdown the problem pump should the valve failure occur. Because of this mitigating action the licensee declared the pump operable and both units of the plant continued to run. This issue was identified as a violation of Criterion III of 10 CFR Part 50, Appendix B (Section 1R21.3).

- No Color. A Non-Cited Violation was identified during the review of a 1995 modification, installed in the cooling water system supply for the Auxiliary Feedwater Pumps. The design change review process did not consider the increased failure rate of the AFW system due to the increased probability that the AFW pump would trip on low suction pressure with the modification installed. Criterion III of 10 CFR Part 50, Appendix B requires that design changes be subject to design control measures commensurate with those applied to the original design, including verifying or checking the adequacy of the design by the performance of design reviews, calculations, or testing. The failure to determine the effect of a design change on the AFW system performance was identified as a violation of Criterion III of 10 CFR Part 50, Appendix B (Section 1R21.5).

Report Details

Baseline Inspection Procedure: IP 711111.21, "Safety System Design and Performance Capability," dated April 3, 2000.

Summary of Plant Status: Both Units of the Prairie Island Nuclear Generating Plant operated at or near 100 percent power throughout the inspection period.

1. REACTOR SAFETY

Cornerstones: Mitigating Systems and Barrier Integrity

1R21 Safety System Design and Performance Capability (71111.21)

The cooling water (CL) system was selected for review during this safety system design and performance capability inspection at the Prairie Island Nuclear Generating Plant. The purpose of the inspection was to assess whether the design bases had been correctly implemented and to ensure that the system could be relied upon to meet functional requirements. The inspection was performed in accordance with the new Nuclear Regulatory Commission (NRC) regulatory oversight process, which uses a risk-informed approach for selecting the risk significant areas and attributes to be inspected.

.1 System Requirements

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report (FSAR), Technical Specifications (TS), and available design basis information to determine the performance requirements of the CL system. The reviewed attributes of the system included process medium (water, air, electrical signal, or the atmosphere being processed), energy sources (electrical and air), control systems, and equipment protection. The inspectors also evaluated operator actions by review of normal, abnormal, and emergency operating procedures and by verification that instrumentation and alarms were available to operators for making necessary decisions. The review included a consideration of requirements and commitments identified in the FSAR, TS, design basis documents, and plant design documents.

b. Findings

No findings of significance were identified.

.2 System Condition and Capability

a. Inspection Scope

The inspectors reviewed operation of the CL system by review of normal, abnormal, and emergency operating procedures, review of system records, and discussions with cognizant licensee personnel. Records for three system critical components were selected for in-depth inspection and records review. The components selected for specific review were the diesel driven CL pumps, the motor driven CL pump and intake hardware and structures. In addition, selected records of periodic testing and calibration procedures and results were reviewed to verify that the design requirements of calculations, drawings, and procedures were incorporated in the system and were demonstrated by test results. Test results were also reviewed to ensure automatic initiations occurred within required times and that testing performed to validate the procedures were consistent with design basis information.

b. Findings

Cooling Water Pump Drive Shaft Bearing Lubrication Water Supply

During the review of design documents and records for the safety related CL pumps, the inspectors noted that a January 3, 1977, safety evaluation changed the classification of the source of drive shaft bearing lubrication water supply from safety related to non-safety related. This water provided lubrication and cooling for the pump shaft bearings and seals of the three safety related deep draft CL pumps. The design change was approved based on the unsubstantiated belief that the bearing lubrication water was not required for pump operation.

The original bearing lubrication water supply was classified as safety related and consisted of filtered water supplied from the CL system pump discharge header. Although classified as safety related, the original design and installation failed to require safety related electrical power for the system used to backwash the shaft bearing lubricating and cooling water filter. Power for the backwash system would be lost if a loss of offsite power (LOOP) occurred, which would result in clogging of the filters, making the lubrication water supply unavailable. After a short period of time, this could make the safety related CL pumps inoperable. In addition, since the 1977 safety classification change, several changes had been made to the drive shaft bearing lubrication water supply including the substitution of the non-safety plant well water system as the preferred water source with the filtered water source, down-graded to non-safety, as a back-up. The inspectors noted that since 1977, licensee personnel had replaced portions of the plant well water system with non-seismic polyvinylchloride (PVC) piping.

In response to questions by the inspectors, licensee personnel determined that lubricating water for the pump drive shaft and impeller bearings was required for pump operation. Based on the fact that the non-safety related lubricating water systems might not be available after a seismic event, a LOOP and possibly other failures, licensee personnel declared all three safety-related pumps inoperable. Both units were affected. A formal telephone notification was made to the NRC on this problem and Licensee Event Report (LER) 1-00-04 was later submitted on this issue.

Compensatory measures were developed to ensure continued system availability of the lubricating water until a temporary modification could be developed and installed. On November 1, 2000, a Notice of Enforcement Discretion (NOED) was requested and granted to allow the units to continue to operate (EA 00-282). Licensee personnel initiated condition report (CR) 20004776 to put the lubricating issue for the CL pump bearings and seals in the corrective action program.

Criterion III, “Design Control,” of 10 CFR Part 50, Appendix B requires, in part, that measures be established to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Design changes shall be subject to design control measures commensurate with those applied to the original design, including verifying or checking the adequacy of the design by the performance of design reviews, calculations, or testing.

Inadequate design control measures for the CL system resulted in the installation of non-safety related shaft and bearing lubrication water sources for the safety-related vertical pump drive shaft bearings. The issue involved an inadequacy in the original design of the three safety related cooling water pumps and an inappropriate design change in 1977. The failure to assure that the original pump lubricating water filter backwash systems were provided with electrical power from a vital bus and the failure to insure that a 1977 change, lowering the quality standard for the safety-related CL vertical pump lubricating water system, was consistent with the original design is an apparent violation of Criterion III (EEI 50-282/00-13-01;50-306/00-13-01).

Analysis of Significance

The significance determination process (SDP) was used to evaluate the risk significance of the inadequacy of design and design changes associated with the safety-related CL vertical pump lubricating water supply. The NRC staff evaluated the risk significance of the inspection finding in terms of the contribution from both internal and external initiating events. Consistent with the guidance for the SDP in the Revised Oversight Process, the change in core damage frequency (CDF) was evaluated stemming from the identified plant design deficiency. External initiating events including earthquake, fire, and tornado/high wind were individually evaluated. A brief description of the SDP evaluation process follows:

<u>Phase 3 SDP Risk Evaluation</u>

A Phase 3 SDP analysis was performed using the NRC’s Standardized Plant Analysis Risk (SPAR) model, Revision 3i for Prairie Island Unit 1 and 2. The SPAR model permits a reasonable estimate of the significance of operational events/issues, including human and system interactions. Details of the analysis assumptions, methods, and results are provided below.

<u>Probabilistic Safety Assessment Modeling Approach</u>

The assessment of this condition involved considering the following aspects: (1) LOOP at Unit 1 which degrades into a station blackout, (2) a seismic event which affects the Filter Water (small diameter) PVC piping, and (3) eventual loss of component cooling water at Unit 2.

Based on the analysis results, a seismic event and the loss of component cooling water at Unit 2 event each had an average annual CDF increase of less than 1E-6 and were characterized as having very low safety significance (Green).

LOOP at Unit 1 which degrades into a station blackout:

Assumptions

1. Time periods exist in which a LOOP and river water conditions result in filter plugging in three hours. The three hour plugging factor was based on information provided to the analyst by the licensee based on quarterly water samples from the system. The time period accounted for worst-case flows in the system and used the worst-case sample results. The time period that such river water conditions have occurred is unknown, and was varied to assess its impact on the Unit 1 SPAR model internal events annual average CDF.

2. Causes of LOOP could be extreme severe weather or could result from plant-centered, or grid-centered events.

3. If filter #121 is plugged in three hours, operators have one hour to switch to filter #123 (the licensee can provide vendor analysis to confirm that the Cl pumps can run for one hour without bearing cooling water). Failure to switch to filter #123 and no offsite power recovery in four hours results in the loss of Unit 1 EDGs due to loss of cooling. (Recovery of offsite power allows recovery of automatic backwash.) If operators successfully switch to filter #123, it will also plug in three hours. Assuming that the CL pumps can operate for one hour after loss of bearing cooling, offsite power recovery in seven hours will prevent station blackout. There is a filter bypass capability using manual valve CL-19-23, but the alarm procedure does not direct the operator to bypass. This analysis assumes the operators will follow the alarm procedure and switch to filter #123; however, no credit was given for bypassing the filters. No manual backwash capability of the strainers exist to recover or to prevent plugging of the filters.

4. The SPAR human error worksheet was consulted for evaluating the probability that operators fail to switch from filter #121 to filter #123. The operator action nominal base case probability was increased using assumptions that the stress level would be high and that experience/training is low especially in light of the fact that the system's importance was downgraded to QA Type III because an erroneous analysis determined that bearing cooling was not needed for pump operability.

   Baseline HEP = 1E-3 x 2 (high stress) x 3 (training low/infrequent) = 6E-3

5. If Unit 1 EDGs are failed during a LOOP, the Unit 2 EDGs can be cross-tied.

6. If station blackout occurs at Unit 1 (i.e., Unit 1 and the Unit 2 cross-tie failed), there is no credit for recovering the Unit 1 EDGs independent of recovering offsite power since offsite power would need to be recovered to wash off the filters. Also, no credit was assigned to recovering the Unit 2 cross-tie. This is

considered a difficult task to perform in the field in the required time, and not much credit could be assigned to this potential recovery.

7. Station blackout parameter assumptions include that PI reactor coolant pump (RCP) seals are of the Westinghouse old O-ring design and battery lifetime is 2 hours. The two hour battery lifetime is consistent with the 1994 Individual Plant Examination report. The RCP seal design was assumed to still be the old O-ring design; however, it is believed that implementation of the new O-ring design would not significantly change the results of this analysis.

Method

A LOOP event tree specific to the observed conditions was incorporated into the Prairie Island SPAR Revision 3 model.

Results: The annual average internal CDF increase to Unit 1 from this potential station blackout scenario was calculated for various fractions of the year that filter plugging occurs in three hours. Based on the calculation results, if the river water conditions were unfavorable for approximately seven days, the risk was predicted to be 1E-6/yr. If such river water conditions existed for approximately 59 days out of a given year, the risk was predicted to be 1E-5/yr. Because a safety-related power supply had never been provided for the backwash function of the filtered water strainers since original plant operation, this analysis determined that the river water conditions were potentially unfavorable for greater than 59 days and; therefore, had an internal CDF increase of greater than 1E-5/yr. The loss of cooling water due to the loss of the filtered water supply to all three of the vertical cooling water pumps' line shaft bearings is characterized as having substantial safety significance (Yellow).

<u>Single Failure Vulnerability in Emergency Dump Flow Pathway</u>

Portions of the CL return pathways for both divisions were routed through non-seismically qualified piping in the non-seismically qualified turbine building. For a seismic event, the potential existed that these pathways could be blocked. An alternate safety-related return flowpath (emergency dump) was provided in the auxiliary building. This flowpath was normally isolated by motor-operated-valve MV32038 and operator action was required to open this valve when necessary. A single failure of this valve to open when required could leave all safety-related heat loads in both divisions with no CL flow. The flow through the CL pumps could be reduced to below minimum flow adequacy.

Although Operators might be dispatched to manually open the dump valve, the inspectors considered it unlikely that this condition could be diagnosed and the valve manually opened locally in time to prevent failures of the Unit 1 diesel generators. Additionally, if the failure were mechanical, manual operation could be precluded.

Licensee personnel maintained that considering single failure for "external events," such as a seismic event, was outside their licensing basis. No documentation was identified to support this position, and all USAR statements reviewed by the inspectors concerning single failure indicated that consideration of single failure was required for all events requiring plant safe shutdown, with no exceptions for "external events." It should also be noted that "external events," such as earthquake, tornado, or flood, could also cause a LOOP, which was discussed in USAR Section 14.4.11, "Loss of All AC Power to the Station Auxiliaries (LOOP)." This USAR section explicitly required single failure consideration for a LOOP.

For this particular example, the most vulnerable components cooled by the CL system were the Unit 1 diesel generators. Without CL, the diesels could fail in several minutes. Their loss would result in loss of the ability to remove decay heat from the core by all pathways except the turbine driven auxiliary feedwater pump. However, since the CL system was the only safety-related water source for the auxiliary feedwater system, if the CL pumps failed as a result of operating near shutoff head, this decay heat pathway could also be lost. Therefore, the risk of core damage appeared to be high.

During discussions of this issue with licensee personnel, the inspectors learned that the CL emergency dump valve was only one of many examples throughout the plant where the single failure criteria had not been considered for "external events." Based on statements by licensee personnel, this was a practice which had been accepted by the NRC for the Prairie Island plant. Since licensee personnel considered this issue to be an acceptable practice, no CR was written on this concern.

This issue was an example of a broader concern regarding application of single failure analysis techniques. Several other minor examples were also identified. The inspectors were unable to determine the validity of this approach to using the single failure criteria for problems when external type events occurred. Pending review and determination of the validity of this approach by NRC headquarters, this issue is considered an Unresolved Item (URI 50-282/00-13-02;50-306/00-13-02).

.3 System Walk-downs

a. Inspection Scope

The inspectors performed walk-downs of the CL system, portions of the CL support systems and components in systems cooled by the CL system. The walk-downs focused on the installation and configuration of piping, components, and instruments; the placement of protective barriers and systems; the susceptibility to flooding, fire, or other environmental concerns; physical separation; provisions for seismic concerns; accessibility for operator action; and the conformance of the currently installed configuration of the systems with the design and licensing bases.

b. Findings

Cooling Water System Pump Air/Vacuum Valve Failure

During the initial walk-down of the CL system, the inspectors noted that each of the three normally idle, deep draft safety-related pumps were equipped with float-actuated, air/vacuum valves to vent air from the pump columns on a pump start. The inspectors questioned the impact of a failure of one of the valves when the associated safety related pump started. No records of analyses or tests were found to exist to address this type failure.

Licensee personnel determined that if one of the valves failed to automatically close after venting, water would be discharged into the CL pump area and would result in significant flooding in the area of the safety related CL pumps. It was uncertain if any of the three safety related CL pumps would survive this failure and the operability of the pumps on October 26, 2000, was indeterminate. As a result, licensee personnel declared all three pumps inoperable and established a continuous watch in the pump area within two hours to immediately secure any of the pumps if this problem should occur. Because of this compensatory action the licensee declared the pumps operable and both units of the plant continued to run. A formal telephone notification was made to the NRC on this problem and LER 1-00-03 was later submitted on the issue.

The failure to verify the adequacy of design or to demonstrate the capability of the CL system to perform its safety functions with the failure of a CL pump air vent/water trap valve could credibly affect the function of the CL system and is considered an example of a violation of 10 CFR Part 50 , Appendix B, Criterion III, "Design Control." The inspectors evaluated the risk significance of the finding using the SDP. No failure of these valves had actually occurred and the air/vacuum valves had functioned successfully in over 700 starts of the cooling water pumps. The air/vacuum valves were capable of performing their intended function. The violation is associated with an inspection finding that is characterized by the SDP as having very low risk significance (Green) and will not be cited in accordance with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-282/00-013-03; 50-306/00-013-03).

.4 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed a sample of CL system problems, which had been previously identified and documented by licensee personnel, and had been placed in the corrective action program. The corrective action system used Prairie Island assessment process (CR) documents for problem identification and tracking. Where possible, the inspectors verified that CL problems were appropriately documented. The purpose of the review was to evaluate the adequacy and effectiveness of the identification and correction of CL system problems. Inspection Procedure 71152, “Identification and Resolution of Problems,” was used as guidance for inspection in this area.

b. Findings

No findings of significance were identified.

.5 Design Control

b. Inspection Scope

The inspectors reviewed selected areas of CL design to verify that the system and components would function as required under accident conditions. The review included a review of the design basis, design changes, design assumptions, calculations, boundary conditions, and models as well as a review of selected modification packages. Instrumentation was reviewed to verify appropriateness of applications and set-points based on the required equipment function. Additionally, the inspectors performed analyses in several areas to verify that design values were correct and appropriate. Documentation reviewed included drawings, procedures, calculations, plant modifications, and maintenance work orders, as well as the TS and the FSAR. The purpose of the reviews was to determine if the design bases of the systems were met by the installed and tested configurations.

c. Findings

CL System LOCA Analysis Incomplete

During normal operation, CL flow to the component cooling (CC) water heat exchangers was controlled by air-operated temperature control valves (TCVs) that would fail open on a loss of air. The instrument air supply was non-safety-related, and therefore, potentially unavailable for a design basis event. Although full open was the safe position for the valves with respect to CL flow to the CC heat exchangers, calculation ENG-ME-244, 11/17/95, "CC Heat Exchanger TCV Position Range," indicated that this reduced the necessary CL flow to other safety-related components to below design basis requirements during the injection phase of a loss of coolant accident (LOCA).

To address this concern, modification 95CL04 was installed to add adjustable mechanical stops to the valves to limit the flow of CL to the CC heat exchangers to the minimum required to meet the LOCA injection phase heat loads on the loss of instrument air. This would assure adequate flow to the other safety related components.

After the injection phase, accident response procedures required that operators remove the stops to allow the valves to go full open in order to handle the much higher CC heat loads associated with the recirculation phase of the LOCA. The inspectors questioned how other safety related heat loads would be affected and how adequate CL flow could be assured for these loads during the recirculation phase.

During discussions with licensee personnel on this issue, the inspectors were told that there was no formal written analysis to demonstrate that adequate CL flow would be provided to the other safety related heat loads. Licensee personnel further stated that, with the stops removed, the low header pressure would result in the isolation of the non-safety related turbine building heat loads and the flow gained from this isolation would more than offset the additional flow required by the CC heat exchangers. No evaluation or analysis was available to substantiate this.

One of the most vulnerable heat loads on the CL system was the Unit 1 emergency diesel generators (EDGs). For a LOCA with a LOOP, reduced CL flow to the Unit 1 EDGs could result from removal of the stops and both Unit 1 diesel generators could be lost. Loss of the Unit 1 EDGs would result in loss of the ability to remove decay heat from the Unit one reactor core by all pathways except the turbine driven auxiliary feedwater pump. Additionally, reduced heat removal by the FCUs in both units due to reduced CL flow and loss of the FCU fans in Unit 1 due to the loss of the EDGs could cause the containment design parameters to be exceeded. Finally, reduced cooling for the control room heat removal systems and the other safety-related equipment could threaten their operability.

Licensee personnel initiated CR 20004834 to enter this issue into the corrective action program. The safety significance of this unanalyzed condition could not be determined until the CL flow for the recirculation phase of a LOCA is analyzed to verify that recirculation phase heat loads could be removed. This matter is considered an unresolved item pending completion of this analysis (URI 50-282/00-13-04; 50-306/00-13-04).

Cooling Water Supply to the Auxiliary Feedwater Pumps

The inspectors reviewed modification 92L369 which was installed on the safety-related CL supply for the auxiliary feedwater (AFW) pumps. Based upon a review of the project description and safety evaluation for the change, the inspectors determined that the design was installed as a non-quality class addition to the Quality Class 1, seismically qualified CL piping. The modification also required that a normally closed vent valve on the safety related piping be left in the open position.

In the safety evaluation, the licensee engineering staff concluded that installation of the modification would not affect operation of the AFW pumps. In reaching this conclusion, licensee personnel relied upon the AFW pump low suction trip to protect the pump in the event of low suction pressure caused by a failure of the modification with the vent valve open. The safety evaluation did not consider the increased failure rate of the AFW system following a seismic event, due to the increased probability the AFW pump would trip on low suction pressure with the suction vent valve open to the atmosphere. As a result, the licensee engineering staff concluded that the modification could be maintained as a Quality Class 3, non-seismic piping run.

During discussions on this issue, licensee engineering staff concurred that the AFW low suction pressure trip only protected the AFW pump from damage following a seismic event and did not ensure that the AFW system would operate as designed. The licensee’s staff also determined that the piping modification should be controlled as Quality Class 1 and should be seismically mounted. The issue was entered into the corrective action program. The licensee’s engineering staff performed an initial seismic evaluation of the installed components and determined that the equipment would meet current seismic mounting requirements.

Criterion III, “Design Control,” of 10 CFR Part 50, Appendix B requires, in part, that measures be established to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Design changes shall be subject to design control measures commensurate with those applied to the original design, including verifying or checking the adequacy of the design by the performance of design reviews, calculations, or testing.

Inadequate design control measures for the CL and AFW systems resulted from the failure to maintain the quality class and associated seismic qualification of the pressure boundary for the CL supply to the AFW system and could have affected the operability and reliability of all trains of the AFW system during a design basis seismic event. Licensee personnel were able to demonstrate that the installed modification could meet current seismic mounting requirements and no other modifications had been made to the installed piping run. The violation is associated with an inspection finding that is more than minor in that the lack of analysis affected all trains of AFW and is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-282/00-013-05; 50-306/00-013-05).

Assumptions Regarding Failure or Leakage of Non-Seismic Piping

In the CL system flow model, licensee personnel assumed that non-seismically designed piping would not completely break, but instead would fail by means of a "through-wall leakage crack." This approach was based on methodology derived through the NRC's Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," for moderate energy systems.

The licensee's previous design approach had assumed that the system could tolerate a complete severance of a 3-inch pipe in the screenhouse or a 4-inch pipe in the turbine building. During a self assessment in the early 1990s, licensee personnel determined that the ability of the CL system to tolerate the above pipe failures was based on incorrect assumptions. As an alternate approach, licensee personnel concluded that a complete rupture of non-seismically designed piping at Prairie Island was not considered a credible event and that the assumption of a through wall leakage crack would be very conservative.

In reviewing this issue, the inspectors questioned the use of this design approach for non-seismically designed piping and were unable to determine the validity of the licensee personnel's assumption. Pending review and evaluation of this assumption by headquarters personnel, this issue is considered an unresolved item (URI 50-282/00-13-06; 50-306/00-13-06).

4. OTHER ACTIVITIES

4OA6 Management Meetings

Exit Meeting Summary

The inspection results were presented to members of licensee management at the conclusion of the inspection on November 3, 2000. During the inspection no documents or information were identified to the inspectors as proprietary. Licensee personnel acknowledged the results presented during the exit and agreed that no additional proprietary information was discussed or provided. Additional inspection information was provided to members of licensee management during a conference call on December 8, 2000.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Amundson, General Superintendent of Engineering
D. Anderson, Regulatory Issues Manager
T. Breene, Manager, Nuclear Performance Assessment
L. Gard, General Superintendent Plant Maintenance
J. Goldsmith, General Superintendent of Engineering, Nuclear Generation Services
S. Heidemann, Superintendent of System Mechanical Engineering
T. Lillehei, Electrical Design Engineering
R. Peterson, Design Standards Group Supervisor
D. Schuelke, Plant Manager
C. Seipp, Systems Engineer
T. Silverberg, General Superintendent Plant Operations
J. Sorenson, Site General Manager
G. Sundberg, Electrical Design Engineering
S. Thomas, Senior Engineer
M. Thompson, Electrical Design Engineering
R. Williston, Systems Engineer

NRC

S. Burgess, Senior Reactor Analyst
R. Caniano, Deputy Division Director, Division of Reactor Safety
R. Gardner, Chief, Electrical Engineering Branch, DRS
J. Jacobson, Chief, Mechanical Engineering Branch, DRS
S. Ray, Senior Resident Inspector
S. Thomas, Resident Inspector

ITEMS OPENED, CLOSED AND DISCUSSED

Opened

50-282-00-13-01 50-306-00-13-01	Apparent Violation	Inadequate design control Measures, which resulted in a potential failure of the cooling water pumps due to a lack of lubricating water for shaft bearings.
50-282-00-13-02 50-306-00-13-02	URI	Unable to determine the validity of the failure to use single failure criteria in circumstances caused by external events.
50-282-00-13-04 50-306-00-13-04	URI	Unable to determine the safety significance of the failure to analyze cooling water flow to assure adequate cooling during the re-circulation phase of a loss of coolant accident.
50-282-00-13-07 50-306-00-13-07	URI	Unable to determine the validity of the practice of, after a seismic event, using assumptions for through wall leakage of non-seismically designed piping rather than complete pipe severance.

Opened and Closed During This Inspection

50-282-00-13-03 50-306-00-13-03	NCV	The failure to verify the adequacy of design and determine that the failure of the cooling water pump air/vacuum control valve could result in the possible failure of the three safety related cooling water pumps due to flooding in the pump area.
50-282-00-13-05 50-306-00-13-05	NCV	The failure to determine during a design change of the cooling water supply to the auxiliary feed water system that the change could have affected the operability and reliability of the auxiliary feedwater system during a design basis seismic event.

50-282-00-13-06 50-306-00-13-06	NCV	A modification to supply control air to the cooling water backwash valves from a single source could result in the failure of both divisions of cooling water.

Previous Items Closed

No items from previous inspections were closed.

Previous Items Discussed

No items from previous inspections were discussed and not closed.

LIST OF DOCUMENTS REVIEWED

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

Licensing Documents

USAR Section 1.5, General Design Criteria
USAR Table 1.3-2 , Shared Components Required for Normal Pant Operation
USAR Section 2, Plant Site and Environs
USAR Section 6.3, Containment Air Cooling System
USAR Section 6.4, Containment Vessel Internal Spray System
USAR Section 8, Plant Electrical Systems
USAR Section 10.4, Plant Cooling System
USAR Table 10.4-1, Cooling water Requirements for Single Unit Operation in GPM
USAR Table 10.4-2, Component Cooling System Component Data
USAR Section 11.9, Condensate, Feedwater, and Auxiliary Feedwater Systems
USAR Table 11.1-1, Steam and Power Conversion System Component Design Parameters
Technical Specification 4.5, Engineered Safety Features
Technical Specification 3.3.D, Cooling Water System and Bases
Technical Specification 3.4, Steam and Power Conversion System and Bases
NSP letter to NRC dated 1/28/97, Re: Response to Generic Letter 96-06
NSP letter to NRC dated 1/14/90, Re: Response to Generic Letter 96-06
NSP letter to NRC dated 7/11/00, Re: Response to Generic Letter 96-06
NSP letter to NRC dated 2/20/89, Re: Response to Generic Letter 88-14
NRC letter to NSP dated 12/5/89, Re: Response to Generic Letter 88-14
NSP letter to NRC dated 1/29/90, Re: Response to Generic Letter 89-13
NRC letter to NSP dated 1/28/92, Re: Response to Generic Letter 89-13

Procedures

Abnormal Operating Procedure C35 AOP1, Loss of Pumping Capacity or Supply Header With SI, Revision 6
Abnormal Operating Procedure C35 AOP2, Loss of Pumping Capacity or Supply Header Without SI, Revision 5
Abnormal Operating Procedure C35 AOP3, Loss of Speed Circuit for 12[22] Cooling Water Pump, Revision 2
Abnormal Operating Procedure C35 AOP4, Cooling Water Leakage in Containment, Revision 7
Abnormal Operating Procedure C35 AOP5, Cooling Water Leakage Outside of Containment, Revision 3
Abnormal Operating Procedure C35 AOP6, Loss of Cooling Water Return Header Revision 4
Administrative Work Instruction 5AWI 3.15.5, Operability Determinations, Revision 4
Administrative Work Instruction 5AWI 6.5.0, Temporary Modifications, Revision 8

Alarm Response Procedure C47020, Location 470202-0405, Pumps 11 or 12 Cooling Water Strainer Hi ΔP, Revision 18
Alarm Response Procedure C47020, Location 47020-0504, 121 Safeguards Traveling Screen Failure to Wash. Revision 18
Alarm Response Procedure C47020, Location 47020-0404, 121 Safeguards Traveling Screen High ΔP, Revision 18
Integrated Checklist C1.1.35-1, Revision 7, Cooling Water System Unit 1, Revision 7
Maintenance Procedure D63, Installation Guidelines For Threaded Fasteners (Studs or Bolts), Revision 9
Maintenance Procedure PE 0027-02T 4.16kV Bus 27, Cubicle 2, Bus 27 Source to 121 Cooling Water Pump, Revision 0
Maintenance Procedure PM 3002-2-12 12 Diesel Cooling Water Pump Annual Electrical PM, Revision 18
Maintenance Procedure PM 3002-3-12 12 Diesel Cooling Water Pump Annual Electrical PM, Revision 5
Maintenance Procedure ICPM 1-012A 12 Diesel Cooling Water Pump Instrumentation Calibration - Part A, Revision 7
Maintenance Procedure ICPM 1-012B 12 Diesel Cooling Water Pump Instrumentation Calibration - Part B, Revision 3
Operating Procedure AB-3 Earthquakes, Revision 16
Operating Procedure AB-4 Floods, Revision 16
Operating Procedure 1C19.1 Containment System Integrity, Unit 1, Revision 10
Operating Procedure C1.1.38 Common Fuel System Status Checklist, Revision 12
Operating Procedure C37.3 Turbine Building Ventilation System, Revision 10
Operating Procedure C47519 Alarm Response Procedure Annunciator Location: 47519-0501 Fan Operating Procedure Cooling Water Return Flow Low, Revision 25
Surveillance procedure Pre-Op 16 Cooling Water System, Revision 1

SP 1106A	12 Diesel Cooling Water Pump Test, Revision 55
SP 1106B	22 Diesel Cooling Water Pump Test, Revision 54
SP 1106C	121 Cooling Water Pump Quarterly Test, Revision 14
SP 1144	Safety Injection Relay SI-24X Contact Verification Refueling Outage Test, Revision 8
SP 1146	Safety Injection Relay SI-14X Contact Verification Refueling Outage Test, Revision 7
SP 1151	Cooling Water System Quarterly Test, Revision 19
SP 1293	Flood Preparation Flood Control Panel Inspection/Installation, Revision 7
SP 2083	Unit 2 Integrated SI Test with a Simulated Loss of Offsite Power 23
SP 2144	Safety Injection Relay 2SI-24X Contact Verification Refueling Outage Test 11
SP 2146	Safety Injection Relay 2SI-14X Contact Verification Refueling Outage Test 11

Modifications

76L-287	Install CL Pump Bearing Lubricating Water Filter, dated 3/1/79
78L447	Overhaul/Upgrade 121 CL Pump dated 6/6/78
79L513	12 Cooling Water Diesel Temperature Sensors and Indicators dated 3/8/79
79L535	Elimination of Cooling Water Flow Switches on #121 and #122 Control Room Chillers and Administration Building Chiller, dated 10/27/80
80L580	Add Duplex Filter to CL Pump Bearing Lubricating Water, dated 11/16/81.
81Y195	Replace CL Butterfly Valves With Flow Control Valves, dated 6/8/82.

82Y280	Modify Cooling Water/Chilled Water Supply and Return Lines to Containment FCUs, dated 8/23/83
82Y729	Replace Disk Nut on Unit 2 CFCU Cooling Water Isolation Valves, dated 3/26/84
86Y675	Add Well to Supply CL Pump Bearing Water, dated 12/2/87
89L125	Delete CL Discharge Header Pressure Switch Manifold, dated 6/5/92
88A0021	Upgrade Vertical Cooling Water Pumps, dated 8/4/95
89L147	Project Description/Safety Evaluation: Cooling Water Supply Valves Logic Change
90L201	Install a Rotary Screw Vacuum Pump for Eductor System, dated 7/18/90
92Y175	Corrosion Monitoring Improvements
92L369	Install Loop Seals and Check Valves on CL to AFW Isolation Valves, dated 2/8/94
94L482	Project Description/Safety Evaluation: 121 Cooling Water Pump Safeguard Ventilation (Screenhouse Safeguards Ventilation Control Circuit), dated 05/30/95
95CL04	CC Heat Exchanger Outlet Cooling Water Cvs, dated 12/15/95
95L517	Cooling Water Auto Closure Logic Modification
168	Change the CC Hx Low CL Flow Alarm to a High Outlet Temp. Alarm, dated 7/31/78
317	Change Diesel Cooling Water Pumps' Start Times From 30 Seconds to 15 Seconds, dated 7/12/73
318	Change CL Pump Start Timers From 30 Seconds to 15 Seconds, dated 7/12/73

Temporary Modification 95T047, 9/15/95, Add Backup Air Supply to CL Strainer Backwash Valves, plus air supply calculation

Safety Evaluations

88A0021	Modification of Cooling Water Pump Bearings and Shaft Supports, dated 3/83
307	Diesel Generators and Diesel Cooling Water Pumps Fuel Oil Piping Design Issues, Revision 0
SE 477	Opening Selected CC And CL Breakers For Appendix R Concerns, Revision 0

Condition Reports

Condition Reports Generated To Address Inspection Issues

CR 20004776	Vertical CL Pump Bearing Lubricating Supply Downgrade to Non Q -- May Require Additional Documentation
CR 20004876	Low Pressure Sensing Line for 22 DDCLP Routed in Sleeve, which Doesn’t Meet Criteria
CR 20004897	Concerns with CL Vertical Seismic Analyses
CR 20004922	Calculation E-H6-1, Revision 5, Contains a Summary Table which was Not Updated at the Last Revision dated 11/02/00

Other Condition Reports Reviewed

CR 20004924	Evaluate Need to Protect Fuel Oil Supply Line for 22 DCLP in 12 DCLP Room
CR 20001584	DCLP Placed in Local to Start During SP1106 and Manual to Start during C35 bypassing Auto-Starts Momentarily without LCO Logging
CR 19980645	DDCLP FO Transfer Pumps Local Motor Starters Could Be Flooded in Screenhouse Basement at 67' if Pipe Break Exceeds Sump Pump
CR 19940739	Corrosion of William Powell Gate Valve Disc Holders
CR 19960396	DDCLP Right Angle Drive Heat Exchanger Tube Failure
CR 19970677	Leak on 12 DDCLP Angle Drive Heat Exchanger
CR 19970737	Missed Surveillance and Lack of Pressure Switch Separation
CR 19980222	11 MDCLP 8-Year Tear Down Found Impeller Worn - Cavitation
CR 19981032	Error Report for Proto-Power Software Proto-HX
CR 19990530	Check Valve CL-43-2 Showed Erosion on Body Near Seat
CR 19993394	12 and 22 DDCLP Jacket Heaters Are Not Interlocked with the Lube Oil Pump as Described in the USAR
CR 20003765	MV-32371 and 32472 Will Fail before 2034
CR 20003868	Cooling Water Leak in Turbine Oil Cooler Supply Line

Drawings

1NE-40005-43	4160V Switchgear Normal Unit 1 Schematic Diagram, (11 Cooling Water Pump Bus 13 Cubicle 8), Revision PP
CE-1.0205 Sht2	Control Engineering & Design Standards
DEN-25550	Elevation 20QL-26 Double Suction Pump
NE-40007-116	480V Switchgear and Aux Unit 1 Schematic Diagram, Revision ST
NE-40008-28	Motor Control Center 1AB Bus 1 Schematic Diagram, Revision Z
NE-40008-30	Motor Control Center 1AB Bus 1 Schematic Diagram, Revision S
NE-40008-95	Motor Control Center 1AB Bus 2 Schematic Diagram, Revision J
NE-40009-70	12 Diesel Cooling Water Pump Schematic Diagram, Revision CZ
NE-40009-71	12 Diesel Cooling Water Pump Schematic Diagram, Revision TV
NE-40009-71.1	22 Diesel Cooling Water Pump Schematic Diagram Revision CZ
NE-40009-71.2	22 Diesel Cooling Water Pump Schematic Diagram Revision RT
NE-40012-44	11 Circulating Water Pump Bay Level Schematic Diagram Revision G
NE-40012-46	12 Circulating Water Pump Bay Level Schematic Diagram Revision G
NE-40012-48	Interconnection Wiring Diagram: 121 Diesel Cooling Water Pump Oil Storage Tank Level and 122 Diesel Cooling Water Pump Oil Storage Tank Level, Revision D
NE-40013-5	Motor Control Center 1AB Bus 1, Motor Control Center 1K Bus 2, Revision L
NE-40013-6	Motor Control Center 1AB Bus 2 Revision W
NE-40013-43	240/208/120V AC Distribution Panel136 Train A Schematic Diagram: Diesel Cooling - Water Pumps 121 Oil Storage Tank Pump, Revision M
NE-40013-46	240/208/120V AC Distribution Panel137 Train B Schematic Diagram: Diesel Cooling Water Pumps 122 Oil Storage Tank Pump, Revision K.
NE-40405-34	21 Cooling Water Pump Bus 25 Cubicle 4 Schematic Diagram, Revision AF.

NE-40407-91	480V Switchgear and Aux Unit 2 Schematic Diagram for 480V Bus 260 (Motor Control Center 2EB Bus 1: 21 Traveling Water Screen, 23 Traveling Water Screen, 121 Old Screenhouse Well), Revision BU
NE-116785-12	Bus 25 Cubicle 1 FDR. To Bus 27 Cubicle 2 Schematic Diagram, Revision B
NE-116786-31	Bus 26 Cubicle 17 FDR. To Bus 27 Cubicle 1 Schematic Diagram, Revision A6-A
NE-116786-34	121 Cooling Water Pump Cubicle 2, Revision B
NE-116786-35	121 Cooling Water Pump Bus 27 Cubicle 1 and 2, Revision B
NE-116786-36	121 Cooling Water Pump Bus 27 Cubicle 2, Revision C
NF-38276	Powerhouse Steel Flood Protection Panels, Revision D
NF-38350 - 3J	Screenhouse Concrete and Reinforcement Plan at Elev. 685 ft.
NF-38350 - 3L	Screenhouse Concrete and Reinforcement Sections and Details
NF-38350-28	Screenhouse Miscellaneous Steel Details, Revision G
NF-38350 - 28H	Screenhouse Miscellaneous Steel Details
NF-38500	Architectural Ground Floor Plan at Elevation 695'-0", Revision T
NF-38607-1	Circulating Water System, Emergency Cooling Water Intake Pipe, Plan & Profile, Revision K
NF-36607- 2B	Circulating Water System, Emergency Cooling Water Intake Crib Details, Revision B
NF-38607-3	Circulating Water System, Emergency Cooling Water Intake Pipe Excavation & Installation, Revision F
NF-38607-4B	Circulating Water System, Emergency Cooling Water Intake Crib Pilecap Details, Revision B
NF-39216-1	Flow Diagram Units 1 and 2 Cooling Water - Screenhouse Revision X
NF-39216-2	Flow Diagram Unit 1 Cooling Water - Turbine Bldg. Revision U
NF-39216-3	Flow Diagram Unit 1 Cooling Water - Aux. Bldg. Revision R
NF-39216-4	Flow Diagram Unit 1 Cooling Water - Containment Revision F
NF-39217-1	Flow Diagram Unit 2 Cooling Water - Turbine Bldg. Revision W
NF-39217-2	Flow Diagram Unit 2 Cooling Water - Aux. Bldg. Revision R
NF-39217-3	Flow Diagram Unit 2 Cooling Water - Containment Revision G
NF-39222	Unit 1 Feedwater System, Revision AX
NF-39223	Unit 2] Feedwater System, Revision AX
NF-39232	Flow Diagram Fuel and Diesel Oil System Units 1 and 2 Revision AD
NF-39260-1	Screenhouse General Arrangement, Revision K
NF-39263 - 2L	Screenhouse Cooling Water Piping
NF-39312-5	Leveltrol, Alarm and Gage Glass Piping Unit No. 1 Revision D
NF-40022-1	Circuit Diagram - 4kV and 480V Safeguard Busses Unit 1 Revision E
NF-40022-2	Circuit Diagram - 4kV and 480V Safeguard Busses Unit 2 Revision D
NF-40026	480V Motor Control Center 1A, 1AA, 1AB Circuit Diagram Revision V
NF-40038	480V Circuit Diagram Motor Control Center 1M, 1TA Revision X
NF-40119-1	Cable Tray System Unit 1 & 2 Screenhouse Ground Floor Revision Q
NF-40153-8	Wiring Diagram 4.16kV Switchgear Bus 13 Cubicle 8 Revision F
NF-40161	External Connections Cooling Water Pumps Revision H
NF-40193-1	Wiring Diagram Bus - 1 Motor Control Center 1AB Revision T
NF-40193-2	Wiring Diagram Bus - 2 Motor Control Center 1AB Revision J
NF-40216-1	Wiring Diagram Bus - 1 Motor Control Center 1TA Revision Q
NF-40216-2	Wiring Diagram Bus - 2 Motor Control Center 1TA Revision S
NF-40242-4	External Connections Motor Control Center 1AB Revision J

NF-40243-8	External Connections Motor Control Center 1K Revision G
NF-40277-17	Wiring Diagram Terminal Cabinets 1233 Revision K
NF-40283-2	Wiring Diagram 230Vac Distr. Panels - A Train Revision R
NF-40283-3	Wiring Diagram 230Vac Distr. Panels - B Train Revision Q
NF-40295-2	External Wiring Diagram 12 Misc. Systems Relay Rack Instruments Revision P
NF-40295-7	External Wiring Diagram 12 Misc. Systems Relay Rack Instruments Revision D
NF-40306	Wiring Diagram Terminal Cabinets 1301, 1302, 1750, & 1751, Revision P.
NF-40315-1	Interlock Logic Diagram Cooling Water System - Unit 1 and 2 Revision S
NF-40315-2	Interlock Logic Diagram Cooling Water System - Units 1 and 2 Revision N
NF-40315-3	Interlock Logic Diagram Cooling Water System - Units 1 and 2 Revision F
NF-40315-4	Interlock Logic Diagram Cooling Water System - Units 1 and 2 Revision F
NF-40315-5	Interlock Logic Diagram Cooling Water System - Units 1 and 2 Revision E
NF-40315-6	Interlock Logic Diagram Cooling Water System - Units 1 and 2 Revision L
NF-40315-7	Interlock Logic Diagram Cooling Water System - Units 1 and 2 Revision L
NF-40315-8	Interlock Logic Diagram Cooling Water System - Units 1 and 2 Revision P
NF-40315-9	Interlock Logic Diagram Cooling Water System - Units 1 and 2 Revision E
NF-40315-10	Interlock Logic Diagram Cooling Water System - Units 1 and 2 Revision A
NF-40315-11	Interlock Logic Diagram 121 Cooling Water Pump Revision B
NF-40323-2	Interlock Logic Diagram Fuel & Diesel Oil System - Units 1 and 2 Revision G
NF-40553-4	Wiring Diagram 4.16kV Switchgear Bus 23 Cubicle 4, Revision E
NF-40908	Front View & Wiring Diagram 22 Diesel Cooling Water Pump Oil Day Tank Control Panel, Revision B
NF-92880	Wiring Diagram 120/240 Vac Panel 145 & 119 D1 and D2 Emer Gen & Equip (MCC 1TA Bus ½), Revision J
NF-116752	Unit 1 - Unit 2 Safeguards Consolidated Circuit Diagram, Revision A
NF-117013	D5/D6 Bldg. - Struct. Steel Misc. Plans, Sect. & Dets, Revision B
NF-173000	Flood Protection Key Plan and Details, Revision A
X-HIAW-1-235	Logic Diagrams - Index and Symbols Revision C
X-HIAW-48-27	Schematic Diagram of Air Piping Between Engine and Receiver Tanks
X-HIAW-48-28	Air Receiver Tanks for Air Starting System, dated 12/03/71
X-HIAW-48-72	20 QL-26 Typical Section Double Suction Pump, Revision C
X-HIAW-106-292	Screenhouse Cooling Water A6737," Revision F
X-HIAW-432-2	Wiring Diagram for Model 55D-2 Submerged Pump, dated 10/19/71
X-HIAW-432-3	Wiring Diagram for Model 55D-2 Submerged Pump, dated 10/19/71
X-HIAW-432-4	Alternate Wiring Diagram Model No. 55D-2, dated 10/19/71

X-HIAW-432-5	Typical Selector Switch Wiring Diagram Model No. 55D-2, dated 10/19/71
X-HIAW-432-6	Wiring Diagram Recommended Installation for Model No. 55D-2 10/19/71
X-HIAW-432-7	Typical Wiring Diagram Model No. 55D-2, dated 10/19/71
X-HIAW-444-9	Interconnection Wiring Diagram, Revision B

Calculations

DPT-35105	Seismic Design for Prairie Island Generating Plant Worthington Order Number DPT-35105, dated 10/26/71
E-385-EA-001	5kV Cable Sizing - 3E-385-EA-009 Relay Settings and Coordination, Revision 2
E-385-EA-021	480V Switchgear Branch Breaker Settings, Revision 1
E-415-EA-3	Degraded Voltage Relay Drop-Out, Revision 1
E-415-EA-12	Grid Voltage at Degraded Voltage Relay Maximum Tolerance, Revision 0
ENG-EE-024	Exide and C&D Battery Hydrogen Generation Calculation, Revision 0
ENG-EE-061	Unit 1 4kV Bus Minimum Voltage, Revision 0
ENG-EE-121	Cable Sizing Calculation for MV-32159 for Project 98EB02, Revision 0
E-H6-1	Motor Operated Valve Terminal Voltage Calculations, Revision 5
ENG-ME-020	D1/D2 and DDCLP Fuel Oil Storage Capacity, Revision 1
ENG-ME-072	14-inch CW Tie In to Chilled Water System, Revision 0
ENG-ME-162	Seismic Analysis of Vertical CL Pumps, March 1, 1971
ENG-ME-178	Screenhouse Ventilation Evaluation, Revision 0
ENG-ME-190	Verify Capability of Screenhouse Sump Pumps in Flood Conditions, dated 5/15/95
ENG-ME-200	Upsurge in Pump Chamber emergency Intake System, dated 11/22/95
ENG-ME-203	Evaluation of Screenhouse Internal Flooding Due to NSR Line Failure, dated 7/27/95
ENG-ME-218	Single Failure Analysis for the Cooling Water System
ENG-ME-219	Safeguards CL Pump NPSHR Static Head Equivalent
ENG-ME-242	Two-Phase Flow Momentum Effects in FCU #24
ENG-ME-244	CC Heat Exchange TCV Position Range, dated 11/17/95
ENG-ME-254	Addenda 0 & 1, 4/6/99, Safeguard Bay Supply Capacity Analysis
ENG-ME-261	14 FCU Piping Modification (94L442 Add 2), 1/10/96
ENG-ME-265	FCU Manway Bypass Flow, dated 1/27/96
ENG-ME-292	Determination of Possible Flow Rate in CL to AFWP Piping with Gate Valve Half Open to Verify Design Flow Shall Pass, Revision 2
ENG-ME-294	Safeguards Bay Sluice Gate Seismic Analysis, Revision 0
ENG-ME-298	Intake Bay Water Capacity, dated 1/18/97
ENG-ME-302	CL System Response to Seismic Event; Case 11, dated 2/7/97
ENG-ME-310	Emergency Intake Line; Post-Seismic Minimum Flow Requirement, dated 3/17/97
ENG-ME-338	Analysis of ZH Chiller Back-Up Air Supply
ENG-ME-340	Evaluation of Two-Phase Flow Through 22 & 24 FCU with the DDCLPs At 93%, dated 10/31/97
ENG-ME-347	Minimum Required Intake Bay Volume dated 2/6/98

ENG-ME-355	Intake Canal Available Water Volume Comparison Analysis dated 3/17/98
ENG-ME-404	Loss of Off Site Power with One Cooling Water Pump, Revision 0
ENG-ME-409	Unit 1 Emergency Diesel Generator Heat Exchanger Performance with Reduced Cooling Water Flow, Revision 0
ENR-ME-339	9/30/97, CL System with Degraded Pumps (93%)
SPC-EA-006	4160 Volt Safeguards Degraded Bus Voltage Setpoint, Revision 1
SPC-EA-007	4160 Volt Safeguards Bus Undervoltage and Loss-of-Voltage Setpoints, Revision 1
SPCCL033	Loop A Diesel Pump 12 Start Pressure Switch, dated 10/17/00
SPCCL045	1 Turbine Loop "A" Cooling Water Header Low Pressure, dated 11/5/97

TIN No. 95-1200, 8/20/95, Containment Fan Coil Unit 21 Thermal Performance Test, Data Evaluation and Uncertainty Analysis.

Unnumbered Calculation Starting Air Low Pressure Preliminary Evaluation

“Calculations for Air Receiver and Fuel Oil Day Tank,” Patten Industries Letter, dated September 8, 1972

“Screenhouse, Steel Support for Fuel Oil Tank,” Book 59-1, Sheets 43-55, December 11, 1970

Work Orders/Work Requests

9405014	121 SFGDS Traveling Screen Diff. Pres. Switch/Gage, dated 7/29/94
9403773	CL Discharge to U1 TB Standpipe Corrosion Monitor Isol, dated 5/26/94
9509889	Repeat CC HX Flow Measurement Test, dated 10/26/95
9604166	MV-32159 Didn't Close 1st Time with CS-46144, dated 4/24/96
9604411	Inspect and Clean Emergency Intake Line, dated 6/14/96
9705105	Rework 121 Cooling Water Pump AMP Indication (Replace Current XDCR for 121 CLP at BKR 27-2), dated 07/1997
9711858	Relief Lifting Starting Air Compressors, dated10/14/97
9800851	121 SFGDS Traveling Screen D/P Bubbler Air Leak, dated 2/19/98
0013303	Work Order: Calibrate Omega Temperature Instruments, dated 11/02/2000

SAFETY EVALUATIONS

88A0021	Modification of Cooling Water Pump Bearings and Shaft Supports 3/83
	307 Diesel Generators and Diesel Cooling Water Pumps Fuel Oil Piping Design Issues 0

Miscellaneous

Design Bases Document - DBD SYS-35, Revision 4, 6/11/99, Cooling Water System

DBD Ref. 7.6.62 - NSP Document “Cooling Water System Report,” dated 10/24/88

DBD-SYS-20.09 - Design Bases Document for the DC Auxiliaries System, Revision 3

DBD-SYS-38A - Design Bases Document for the Emergency Diesel Generator System, Revision 2

DBD TOP-03 - Design Bases Document for the Environmental Qualification of Electrical Equipment, Revision 3

DBD TOP-16 - Design Bases Document for the Electrical Design Issues Topic, Revision 1
Design Standard - (No number), “Engineering Design, Fabrication, and Installation Summary for Single Failure Criterion,” Section 2.4.3, Revision 1
Fluor Specification AD-M 455, “Apparatus Data Diesel Generator ”
Vendor Manual - X-HIAW-145-13, “Safeguards Traveling Screens,” Revision 0
Vendor Manual - Worthington Vertical Suction Pump (Type QL) 12 QL and Larger, Revision 8
Vendor Manual - XH-432-8, Diesel Fuel Oil Transfer Pumps, Revision 3
DTP-35105 and DTP-35106, Seismic Reports, Pioneer Service and Engineering Co., October 26, 1971, and November 15, 1971
SS 496 -- Standard Specification for Vertical Pumps, dated 9/69
Project 892103 Single Failure Analysis for the Cooling Water System at the Prairie Island Nuclear Generating Plant Units 1 and 2, dated 5/90
FOI A0862 - CL Single Failure Analysis Open Issues
CE 1.0204 Control Engineering & Design Standards
CE 1.0205 Control Engineering & Design Standards
Safe Shutdown Circuit Analysis for 11 Cooling Water Pump Located in Fire Area FA-41B (Unit 1 Side)

LIST OF ACRONYMS USED

AFW	Auxiliary Feed Water
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CL	Cooling Water
CR	Condition Report
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
FSAR	Final Safety Analysis Report
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
NCV	Non-Cited Violation
NOED	Notice of Enforcement Discretion
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
psig	pounds per square inch gauge
PVC	polyvinyl chloride
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
TS	Technical Specifications
TVC	temperature control valve
URI	Unresolved Item