UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

August 5, 2009

EA-09-167

Mr. Michael D. Wadley
Prairie Island Nuclear Generating Plant
Northern States Power Company, Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 NRC INSPECTION REPORT 05000282/2009010; 05000306/2009010 PRELIMINARY WHITE FINDING

Dear Mr. Wadley:

On July 9, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Prairie Island Nuclear Generating Plant, Units 1 and 2. The enclosed report documents the inspection findings, which were discussed on July 9, 2009, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents one NRC-identified finding for Unit 1 of very low safety significance (Green). This finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance, and because the finding was entered into your corrective action program, the NRC is treating this finding as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC Enforcement Policy.

The enclosed inspection report also discusses a finding for Unit 2 that appears to have low to moderate safety significance (White). As documented in Section 4OA5 of this report, the Unit 2 component cooling water system was inadequately designed to ensure that the system would be protected from licensing basis events (such as high energy line breaks, seismic and tornado events) which could occur in the turbine building. The events in the turbine building could cause a loss of component cooling water inventory from both trains of equipment and a loss of safety function.

This finding was assessed based on the best available information, including influential assumptions, using the applicable Significance Determination Process (SDP). The preliminary safety significance of the finding was determined assuming that the design of the Unit 2 component cooling water system was inadequate for 1 year.

This finding was an immediate safety concern, and your staff declared the Unit 2 component cooling water system inoperable as required by your Technical Specifications. Within 2 hours, operations personnel manually closed component cooling water system isolation valves between the auxiliary building and the turbine building. This allowed the safety-related portion of the component cooling water system to be returned to an operable status and eliminated the immediate safety concern.

This finding is also an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the NRC Enforcement Policy. The current Enforcement Policy which can be found on the NRC's Web site at http://www.nrc.gov/reading-rm/doc-collections/enforcement.

In accordance with Inspection Manual Chapter (IMC) 0609, we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of this letter. The SDP encourages an open dialogue between the staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination.

Before the NRC makes its enforcement decision, we are providing you an opportunity to either: (1) present to the NRC your perspectives on the facts and assumptions used by the NRC to arrive at the finding and its significance at a Regulatory Conference, or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation at least 1 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. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of the receipt of this letter. If you decline to request a Regulatory Conference or to submit a written response, you relinquish your right to appeal the final SDP determination; in that, by not doing either you fail to meet the appeal requirements stated in the Prerequisite and Limitation Sections of Attachment 2 of IMC 0609.

Please contact John Giessner at (630) 829-9619 within 10 days of the date of this letter to notify the NRC of your intended response. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. You will be advised by a separate correspondence of the results of our deliberations on this matter.

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

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of

Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Prairie Island Nuclear Generating Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Prairie Island Nuclear Generating Plant. The information that you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure 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), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA by Gary L. Shear for/

K. Steven West, Director
Division of Reactor Projects

Docket Nos. 50-282; 50-306; 72-010
License Nos. DPR-42; DPR-60; SNM-2506

Enclosure: Inspection Report 05000282/2009010; 05000306/2009010
w/Attachment: Supplemental Information

cc w/encl: D. Koehl, Chief Nuclear Officer
G. Salamon, Regulatory Affairs Manager
P. Glass, Assistant General Counsel
Nuclear Asset Manager
J. Stine, State Liaison Officer, Minnesota Department of Health
Tribal Council, Prairie Island Indian Community
Administrator, Goodhue County Courthouse
Commissioner, Minnesota Department
of Commerce
Manager, Environmental Protection Division
Office of the Attorney General of Minnesota
Emergency Preparedness Coordinator, Dakota
County Law Enforcement Center

Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Prairie Island Nuclear Generating Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Prairie Island Nuclear Generating Plant. The information that you provide will be considered in accordance with Inspection Manual Chapter 0305.

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Emergency Preparedness Coordinator, Dakota County Law Enforcement Center

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Letter to M. Wadley from S. West dated August 5, 2009

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2
NRC INSPECTION REPORT 05000282/2009010; 05000306/2009010

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REGION III

Docket Nos:	50-282; 50-306; 72-010
License Nos:	DPR-42; DPR-60; SNM-2506
Report No:	05000282/2009010; 05000306/2009010
Licensee:	Northern States Power Company, Minnesota
Facility:	Prairie Island Nuclear Generating Plant, Units 1 and 2
Location:	Welch, MN
Dates:	June 15 through July 9, 2009
Inspectors:	K. Stoedter, Senior Resident Inspector P. Zurawski, Resident Inspector L. Kozak, Senior Reactor Analyst
Approved by:	J. Giessner, Chief Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000282/2009010, 05000306/2009010; 06/15/2009 – 7/9/2009; Prairie Island Nuclear Generating Plant, Units 1 and 2; Inspection of component cooling water system design deficiency.

This report covers an approximate 1-month period of inspection by the resident inspectors and a senior reactor analyst. One inspector-identified Green finding and one inspector-identified preliminary White finding were identified. The Green finding was considered a Non-Cited Violation of NRC requirements. One apparent violation (AV) was also identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Cross-cutting aspects were determined using IMC 0305, "Operating Reactor Assessment Program." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. An inspector identified Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified due to the licensee's failure to establish design control measures to ensure that the design basis for the Unit 1 component cooling water system (CCW) was correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to ensure that the safety-related function of the CCW system was maintained following a tornado/high winds induced failure of the CCW system piping to the 122 spent fuel pool heat exchanger. Corrective actions for this issue included providing procedural guidance to isolate the Unit 1 CCW system from the 122 spent fuel pool heat exchanger following the receipt of a tornado watch and evaluating the need for additional tornado missile protection for the CCW system piping to the 122 spent fuel pool heat exchanger.

  This finding was determined to be more than minor because it impacted the design control and external events aspects of the Mitigating Systems Cornerstone. The finding also impacted the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This finding was determined to be of very low safety significance due to the very low probability of the Prairie Island Nuclear Generating Plant experiencing a high wind condition that could generate a missile large enough to fail the Unit 1 CCW system piping to the 122 spent fuel pool heat exchanger. The cause of this finding was related to the cross-cutting element of Human Performance, Decision Making because the licensee failed to make safety-significant and risk-significant decisions using a systematic process to ensure that safety was maintained (H.1(a)). (Section 4OA5.1)

- Preliminary White. An inspector identified apparent violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified due to the licensee's failure to establish design control measures to ensure that the design basis for the Unit 2 CCW system was correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to ensure that the safety-related function of the CCW system was maintained following initiating events (such as high energy line break, seismic or tornado events) in the turbine building. This issue has been preliminarily determined to be of low to moderate safety significance (White). This issue was entered into the licensee's corrective action program as corrective action document 1145695. Upon identifying this issue, the licensee immediately declared the Unit 2 CCW system inoperable and entered Technical Specification 3.0.3. The Technical Specification was exited following the closure of several system isolation valves approximately 2 hours later. The closure of the isolation valves prevented the Unit 2 CCW system from being vulnerable to failure following events in the turbine building.

This finding was determined to be more than minor because it impacted the design control and external events aspects of the Mitigating Systems Cornerstone. The finding also impacted the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The initiating events in the turbine building could cause the CCW piping to fail. Loss of CCW inventory affects both trains of CCW based on the piping arrangement. The loss of both trains of CCW required a phase 3 significance determination. The results of the phase 3 assessment showed a delta core damage frequency of 3.2E-6, White. The cause of this finding was related to the cross-cutting element of Human Performance, Decision Making because the licensee failed to make safety-significant and risk-significant decisions using a systematic process to ensure that safety was maintained (H.1(a)). Since both the Unit 1 and Unit 2 cross-cutting aspects are from the same performance deficiency and are separated based on the risk determination, the aspect of H.1(a) counts as one cross-cutting aspect in this report. (Section 4OA5.1).

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000306/2008005-02: Component Cooling Water System Susceptible to High Energy Line Break Interaction

a. Inspection Scope

The inspectors reviewed the circumstances surrounding the licensee's failure to adequately design the component cooling water (CCW) system to ensure that the system was not vulnerable to failure following licensing basis events (such as high energy line break, seismic and tornado events) in the Unit 1 or Unit 2 turbine buildings. The turbine building events could cause a loss of CCW inventory and a loss of safety function.

b. Findings

Introduction: An inspector identified finding of very low safety significance and a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified for Unit 1 due to the licensee's failure to implement design control measures to ensure that the design of the Unit 1 CCW system was not vulnerable to failure during a tornado/high wind event.

In addition, an inspector identified apparent violation (AV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified for Unit 2 due to the licensee's failure to implement design control measures to ensure that the design of the Unit 2 CCW system was adequate to mitigate licensing basis events (such as high energy line break, seismic and tornado events) which could occur in the turbine building. The events in the turbine building could cause a loss of CCW inventory and a loss of safety function.

Description: On July 29, 2008, the licensee initiated corrective action program document (CAP) 1145695 to document that CCW piping located in the turbine building, and used to supply water to the chemistry cold lab, passed directly underneath high energy piping for the 15A and 15B feedwater heaters. As part of the CAP review, operations personnel requested that engineering personnel complete an operability review to evaluate the impact that a high energy line break (HELB) could have on the continued operability of the CCW system.

The current design of the CCW system, a safety related system, includes piping in the auxiliary building and piping in the turbine building. The piping in the turbine building supplies miscellaneous, non-safety related loads. Piping in the auxiliary building supplies safety related loads. Either unit can supply the miscellaneous loads. Based upon this information, the licensee initially determined that a failure of this piping would have no impact on the continued operability of the Unit 1 CCW system because the Unit 1 CCW system was not supplying water to the chemistry cold lab at the time this issue was identified. The licensee believed that there could be some impact on the Unit 2 CCW system because the 2A train of CCW normally supplied water to the chemistry cold lab. As a result, the licensee continued to review this issue.

On July 31, 2008, the licensee identified that a failure of a Unit 1 or a Unit 2 turbine building high energy line could impact the continued operability of the Unit 2 CCW system. The licensee conducted an additional operability review and determined that the Unit 2 CCW system was inoperable because initiating events could cause a complete loss of CCW inventory, if the CCW piping is severed. This would cause the CCW system to drain since there is no way to separate the safety related loads from the non-safety related loads. The licensee also determined that the operators' ability to bring Unit 2 to a cold shutdown condition following a HELB and a failure of the CCW system was impacted. Operations personnel immediately declared both trains of the Unit 2 CCW system inoperable and entered Technical Specification (TS) 3.0.3. While making preparations to shut down Unit 2, operations personnel identified and closed several CCW system manual isolation valves. This eliminated the immediate safety concern and allowed the safety-related portion of the CCW system to be returned to an operable status. The non-safety related CCW piping located in the turbine building remains isolated.

In mid-September 2008, the licensee completed an apparent cause evaluation (ACE) and determined that the interaction between the turbine building high energy piping and the CCW system had been identified in July 2006. However, no actions were initiated to assess the continued operability of the CCW system. The September 2008 ACE further stated that as part of a resolution to the July 2006 CAP, the licensee planned to perform a study to eliminate CCW as the cooling medium for the chemistry cold lab. While reviewing other CAPs associated with this issue, the inspectors found a draft June 2007 and the finalized January 2008 study report (attached to a CAP initiated in 2004). Both reports concluded that none of the CCW piping to or from the chemistry cold lab had been analyzed for susceptibility of failure due to a HELB or a failure from another system. In addition, the study report indicated that other piping near the CCW piping were high energy lines that could damage the CCW piping. The study concluded that if the HELB occurred near the CCW piping the CCW system would fail in approximately 6 minutes.

The inspectors reviewed several historical documents as part of this inspection. The inspectors determined that the licensee had identified potential design deficiencies with the CCW piping located in the turbine building multiple times. However, the licensee failed to properly prioritize the resolution of these deficiencies. During this inspection, the NRC also questioned the licensee regarding the need to include HELBs that could be induced from seismic and tornado/high wind events as part of their evaluation. After several discussions, the licensee agreed to include these external initiators as part of their evaluation.

On March 23, 2009, the licensee initiated CAP 1174370 to document that a portion of CCW piping that supplies water to the 122 spent fuel pool heat exchanger was not adequately protected from missiles generated during tornado/high wind events. The inspectors considered this an additional example of a failure to adequately design the CCW system to ensure that it was adequately protected from failure. This issue was specific to the Unit 1 CCW system as this was the system that was normally aligned to supply cooling water to the 122 spent fuel pool heat exchanger.

On April 15, 2009, CAP 1178236 was initiated to document that a turbine building HELB could result in internal flooding of safety-significant areas due to the release of feedwater/condensate from the pipe break, consequential failure of a cooling water (service water) line, and potential actuation of the fire protection system sprinklers/deluge system. This issue remained under licensee review at the conclusion of the inspection period.

Analysis: The Prairie Island licensing basis requires that the CCW safety related system is designed to ensure no loss of safety function occurs from natural phenomena such as earthquakes or high winds/tornados. In addition, the CCW safety function is required to be maintained for all licensing basis events including HELBs.

(1) Unit 1 Risk Analysis

The inspectors determined that the licensee's failure to design the Unit 1 CCW system such that it would be protected during licensing basis events (such as HELB, seismic and tornado events) was a performance deficiency that required evaluation using the SDP described in IMC 0609.

This finding was determined to be more than minor since it impacted the design control (initial design) and protection against external factors (seismic and weather) aspects of the Mitigating Systems Cornerstone. In addition, the finding impacted the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems equipment used to respond to events and prevent core damage. Since the failure of the CCW piping could potentially represent a loss of CCW system safety function following an external event, Table 4a of IMC 0609 required the completion of a phase 3 SDP evaluation. The senior reactor analyst (SRA) conducted a phase 3 evaluation to determine the risk contribution from a tornado/high wind event that could cause a failure of the Unit 1 CCW system. The SRA determined that the frequency of tornado/high wind events that could create missiles large enough to damage the Unit 1 CCW piping to the 122 spent fuel pool heat exchanger was less than 1E-6/yr. As a result, this issue was determined to be of very low safety significance (Green).

The cause of this finding was related to the cross-cutting element of Human Performance, Decision Making because the licensee failed to make safety-significant and risk-significant decisions regarding the design of the Unit 1 CCW system using a systematic process to ensure that safety was maintained (H.1(a)). Since both the Unit 1 and Unit 2 cross-cutting aspects are from the same performance deficiency and are separated based on the risk determination, the aspect of H.1(a) counts as one cross-cutting aspect in this report.

(2) Unit 2 Risk Analysis

The inspectors determined that the licensee's failure to design the Unit 2 CCW system such that it would be protected during licensing basis events (such as HELB, seismic and tornado events) was a performance deficiency that required evaluation using the SDP.

The finding was determined to be more than minor since it impacted the design control (initial design) and protection against external factors (seismic and weather) aspects of the Mitigating Systems Cornerstone. In addition, the finding impacted the cornerstone objective of ensuring the availability, reliability, and capability of mitigating systems equipment used to respond to events and prevent core damage. Since the failure of the CCW piping could be caused by an internal or an external event, and potentially represent a loss of CCW system safety function, a phase 2 SDP evaluation was required.

For the phase 2 SDP evaluation, the SRA used the transient with the loss of the power conversion system worksheet to represent the event which could result in a loss of CCW for Unit 2. The worksheets were solved assuming the condition existed for an exposure period of 1 year and that all the functions supported by the CCW system were unavailable. The CCW system provides cooling for all emergency core cooling system (ECCS) pumps, the CCW heat exchangers, and one method of reactor coolant pump (RCP) seal cooling. The unavailability of the CCW system affected the high pressure injection, high pressure recirculation, feed and bleed, and RCP seal cooling functions.

The result of the phase 2 evaluation was a Red finding using the counting rule. However, the SRA determined that the results were overly conservative because the worksheet did not account for the fact that the initiating events in the turbine building that would result in the loss of the CCW function were a subset of all transient with the loss of the power conversion system events. The SRA determined that a phase 3 SDP evaluation was necessary.

For the phase 3 SDP evaluation, the NRC staff determined that a number of initiating events could cause a break in the non-safety-related CCW piping in the turbine building and the subsequent failure of the Unit 2 CCW system. These initiating events included Unit 1 HELB events, Unit 2 HELB events, earthquakes, and tornadoes. The SRA primarily used the Risk Assessment of Operational Events (RASP) handbook and its references and the results from a licensee risk evaluation in the phase 3 SDP evaluation.

To estimate the frequency of HELB events that could result in the loss of the Unit 2 CCW system, the SRA used the generic pipe rupture frequency of 1.2E-10/ft-hr from Table 3A-2-1 of the RASP handbook for External Events. This value is taken from EGG-SSRE-9639, “Component External Leakage and Rupture Frequency Estimates”. The licensee provided the length of HELB piping that, upon rupture, could impact the CCW system. The total linear feet of Unit 1 high energy piping from the feedwater or condensate system that could rupture and impact the Unit 2 CCW system was 167 feet. The total length of Unit 2 HELB piping that could cause the same effect was 78 feet. Using this reference and the pipe lengths provided by the licensee, the initiating event frequencies for Unit 1 and Unit 2 were estimated to be 1.8E-4/yr and 8.6E-5/yr, respectively.

To estimate the conditional core damage probability (CCDP) given the HELB-induced loss of CCW, the SRA used the licensee’s risk evaluation results. The calculation assumed that either a Unit 1 or Unit 2 HELB impacted the CCW system piping to the chemistry cold lab resulting in a CCW system pipe break and draining of the Unit 2 CCW surge tank in approximately 6 minutes. This resulted in a failure of the Unit 2 CCW system that was not recoverable. The main feedwater/condensate, reactor water makeup, instrument air systems and auxiliary feedwater system crosstie were also assumed to be impacted by the initiating event and were considered to be unavailable in the CCDP calculation. The licensee calculated a CCDP for Unit 2 under these conditions of 1.2E-2. The dominant sequences involved a loss of CCW event, followed by the failure of the charging system, which resulted in the loss of all RCP seal cooling and a RCP seal loss of coolant accident (LOCA). If the seal LOCA was large enough (greater than 76 gallons per minute) it would not be able to be mitigated because all of the ECCS pumps were unavailable. This led to an unmitigated LOCA. When the initiating event frequency was combined with the CCDP, a total estimated change in core damage frequency (CDF) of 3.2E-6/yr was estimated. The SRA assumed that the baseline CDF was negligible compared to the CDF related to the finding. Therefore, the change in CDF was approximately equal to the CDF of the finding. The SRA assumed that if no design deficiency existed then the impact of the postulated non-safety-related failures would be limited to a transient with a loss of feedwater event rather than a transient with the loss of CCW function and as a result, baseline risk would be much lower than risk calculated given the plant condition created by the performance deficiency.

The SRA performed the same CCDP calculation using the Prairie Island Standardized Plant Analysis Risk Model (revision 3.47) and the same assumptions as the licensee. The Standardized Plant Analysis Risk Model produced similar results to the licensee's risk evaluation.

The postulated loss of Unit 2 CCW from a Unit 1 HELB during times when Unit 2 was shutdown and on shutdown cooling (SDC) was qualitatively evaluated. A loss of CCW under these conditions would ultimately result in a loss of SDC. Given that the exposure period of this condition was limited, decay heat was lower, and additional options for using the charging system for injection or recovering CCW were available, the risk contribution from a shutdown scenario was considered to be much less than the internal event at-power scenario.

The SRA also considered the risk contribution from earthquakes and tornadoes causing a failure of the non-safety-related CCW piping. The SRA determined that the frequency of seismic events or high wind events damaging the CCW line was less than the frequency of HELB events and as a result the risk contribution from external events was less than 1E-6/yr.

The impact of turbine building flooding as a result of the postulated HELB event was not explicitly evaluated in this phase 3 SDP. Recent information from the licensee indicated that a subset of the postulated HELB events that impacted the CCW system could also result in flooding the turbine building if a cooling water pipe was also ruptured and the fire deluge systems were initiated. If sufficient flooding occurred, the auxiliary feedwater, direct current (DC) power, and emergency diesel generator systems could also be impacted, in addition to the CCW system. Quantitative consideration of this impact would increase the estimated delta CDF of this finding. Note that if the turbine building roll-up doors were open (usually during summer months), then flooding would not impact these other systems.

The SRA quantitatively bounded the potential risk impact associated with the turbine building flooding by assuming that for 9 months of a year when the turbine building roll-up doors were closed (cold weather months) a HELB-induced loss of CCW for Unit 2 would also result in severe turbine building flooding 10 percent of the time. The result of this bounding evaluation which included the potential flooding effects was a delta CDF of less than 1.0E-5/yr. The SRA determined further quantitative evaluation of the internal flooding effects related to this finding was unnecessary because the result was not likely to change the overall conclusion of this SDP evaluation.

The large early release frequency (LERF) impact was negligible. Using the insights from IMC 0609, Appendix H, "Containment Integrity SDP," LERF was not impacted because the dominant core damage sequences did not involve an inter-system LOCA or a steam generator tube rupture event.

The conclusion of the phase 3 SDP was an estimated delta CDF of 3.2E-6/yr which represented a finding of low-to-moderate (White) safety significance. The dominant sequences involved a HELB-induced loss of CCW event, followed by the failure of the charging system, which results in the loss of all RCP seal cooling. If the seal LOCA was large enough it could not be mitigated because all ECCS were unavailable due to the loss of CCW.

The cause of this finding was related to the cross-cutting element of Human Performance, Decision Making because the licensee failed to make safety-significant and risk-significant decisions regarding the continued operability of the Unit 2 CCW system using a systematic process to ensure that safety was maintained (H.1(a)).

(3) Old Design Issue Review

NRC IMC 0305, "Operating Reactor Assessment Program," Section 4.11 defined an "old design issue" as an inspection finding involving a past design-related problem in the engineering calculations or analysis, associated operating procedure, or installation of plant equipment that does not reflect a performance deficiency associated with existing licensee programs, policy, or procedures. IMC 0305 stated that the NRC can refrain from considering safety significant inspection findings in the assessment program for a design-related finding as long as the following statements were true:

- The issue was licensee identified as a result of a voluntary initiative such as a design basis reconstitution;
- The performance issue was or will be corrected within a reasonable period of time following identification;
- The issue was not likely to have been previously identified by routine efforts such as normal surveillance or quality assurance activities; and
- The issue does not reflect a current performance deficiency associated with existing licensee programs, policy, or procedures.

With regards to Unit 1, the inspectors determined that this issue did not qualify as an old design issue due to being inspector identified. Specifically, the NRC's request that the licensee include tornado/high wind induced piping failures as part of their analysis resulted in identifying the design vulnerability associated with the Unit 1 CCW piping to the 122 spent fuel pool heat exchanger. In addition, this issue was not identified as part of a voluntary initiative. Instead, it was identified as part of the extent of condition review completed for CAP 1145695.

Although the Unit 2 CCW design deficiency was not likely to have been identified through routine efforts, licensee documentation indicated that CCW design issues associated with the CCW system piping to the chemistry cold lab piping was identified in the 1990's. Due to the long time that had elapsed since the initial identification of this issue, the inspectors concluded that the performance deficiency discussed above had not been corrected within a reasonable period of time due to the failure to properly prioritize the implementation of corrective actions. In addition, the failure to properly prioritize the implementation of corrective actions was reflective of a current performance deficiency associated with the licensee's corrective action program. Finally, there was operating experience including generic communication from the NRC which provided information which the licensee should have evaluated in the corrective action process. The licensee had also listed corrective action effectiveness as one of the six drivers of their recent performance decline. As a result, the inspectors concluded that the Unit 2 CCW design deficiency did not meet the criteria for being considered an old design issue.

Enforcement

(1) Unit 1 Enforcement

10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to assure that the design basis for safety-related functions of structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions.

The Prairie Island licensing basis requires that the CCW safety related system is designed to ensure no loss of safety function occurs from natural phenomena such as earthquakes or high winds/tornados. In addition, the CCW safety function is required to be maintained for all licensing basis events including HELBs.

Contrary to the above, as of March 23, 2009, the licensee had failed to establish measures to assure that the design basis for the Unit 1 CCW system had been correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to ensure that the Unit 1 CCW system was protected from failure following a tornado/high wind event. In addition, the events in the turbine building could have resulted in a loss of CCW inventory and a loss CCW safety function if the Unit 1 CCW system had been aligned to supply cooling water to the chemistry cold lab. However, because this violation was of very low safety significance (Green) and was entered into your CAP as CAP 1174370, it was treated as an NCV consistent with Section VI.A.1 of the Enforcement Policy **(NCV 05000282/2009010-01)**. Corrective actions for this issue included providing procedural guidance to isolate CCW from the 122 spent fuel pool heat exchanger following the receipt of a tornado watch and evaluating the need for additional tornado missile protection for the CCW system piping to the 122 spent fuel pool heat exchanger.

(2) Unit 2 Enforcement

10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to assure that the design basis for safety related functions of structures, systems, and components are correctly translated into specifications,

drawings, procedures, and instructions. Further, Criterion III requires that the design control measures shall provide for verifying or checking the adequacy of designs.

The Prairie Island licensing basis requires that the CCW safety related system is designed to ensure no loss of safety function occurs from natural phenomena such as earthquakes or high winds/tornados. In addition, the CCW safety function is required to be maintained for all licensing basis events including HELBs.

Contrary to the above, as of July 31, 2008, the licensee had failed to implement design control measures to ensure that the design basis for the Unit 2 CCW system was correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to ensure that the design of the Unit 2 CCW system was adequate to mitigate licensing basis events (such as high energy line break, seismic and tornado events) which could occur in the turbine building. The events in the turbine building could cause a loss of CCW inventory for both trains of CCW. This would result in a loss of the CCW safety function. This is an apparent violation of 10 CFR Part 50, Appendix B, Criterion III pending the completion of the final significance determination **(AV 05000306/2009010-02)**.

.2 (Closed) Licensee Event Report 05000282/2009-003-00: Component Cooling Water System Vulnerability to Tornado Missile Hazard

This licensee event report discusses the Unit 1 CCW vulnerability discussed above. No new information was provided in the report. The inspectors determined that this issue constituted a finding of very low safety significance and an NCV of 10 CFR Part 50, Appendix B. This licensee event report is closed.

4OA6 Management Meetings

.1 Exit Meeting Summary

On July 9, 2009, the inspectors presented the inspection results to M. Wadley and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

M. Wadley, Site Vice President
B. Sawatzke, Director Site Operations
K. Ryan, Plant Manager
J. Anderson, Regulatory Affairs Manager
L. Clewett, Business Support Manager
B. Flynn, Safety and Human Performance Manager
R. Hite, Radiation Protection and Chemistry Manager
D. Kettering, Site Engineering Director
J. Lash, Operations Manager
R. Madjerich, Production Planning Manager
J. Muth, Nuclear Oversight Manager
S. Northard, Performance Improvement Manager
M. Schmidt, Maintenance Manager
J. Sternisha, Training Manager

NRC

J. Giessner, Reactor Projects Branch 4 Chief
T. Wengert, Office of Nuclear Reactor Regulation Project Manager

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000282/2009010-01	NCV	Failure to Ensure Design Measures Were Appropriately Established for The Unit 1 Component Cooling Water System (Section 4OA5.1)
05000306/2009010-02	AV	Failure to Ensure Design Measures Were Appropriately Established for The Unit 2 Component Cooling Water System (Section 4OA5.1)

Closed

05000282/2009010-01	NCV	Failure to Ensure Design Measures Were Appropriately Established for The Unit 1 Component Cooling Water System
05000306/2008005-02	URI	Component Cooling Water System Susceptible to High Energy Line Break Interaction
05000282/2009-003-00	LER	Component Cooling Water System Vulnerability to Tornado Missile Hazard

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of 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.

<u>4OA5 Other Activities</u>

- CAP 1162511; Missed Opportunities to Identify HELB and CC System Interaction; December 15, 2008
- CAP 1145695; CC Piping Adjacent to HELB Location in Turbine Building; July 29, 2008
- ACE 1145695-04; September 4, 2009
- 478-AI-01; USAR Appendix I – Flooding; Revision 22
- 478-AI-03; USAR Appendix I - Pipe Stress and Pipe Whip; Revision 22
- CAP 737382; Non-Seismic Equipment in CC System Pressure Boundary; August 2, 2004
- NF-39297-1; Sampling System Units 1 and 2; Revision R
- FC-61-350; Component Cooling Water to H2 Generator as Built Print Record; Revision 2
- DC-496; Component Cooling Design Change; June 14, 1974
- FC-61-348; H2 Generator Cooling Return; Revision 2
- PI-233-39P23A; Pipe Stress Analysis – CC System, Part 2A; Revision 0
- PI-233-39P23A; Pipe Stress Analysis – CC System, Part 2B; Revision 0
- PI-233-39P23A; Pipe Stress Analysis – CC System, Part 2C; Revision 0
- PI-233-39P23A; Pipe Stress Analysis – CC System, Part 2D; Revision 0
- ENG-CS-278; Seismic Qualification of Components in CC System Pressure Boundary; Revision 1
- CAP 1163206; OBM for CAP 737382; December 19, 2008
- ACE 1163206;
- C1.1.14-1; Unit 1 Component Cooling System; Revision 24
- LER 2-08-01; Unanalyzed Condition Due to Both Trains of Component Cooling Being Susceptible to a Postulated High Energy Line Break; September 29, 2008
- NF 39297-3; Sampling Systems – Unit 1 & 2; Revision XX
- NF 39246-1; Unit 2 Component Cooling System Flow Diagram; Revision S
- NF-39246-2; Unit 2 Component Cooling System Flow Diagram; Revision G
- USAR Appendix I; Postulated Pipe Failure Analysis Outside of Containment; Revision 29
- SE 487-AI-01; USAR Appendix I Review – I.5.5 Flooding; Revision 1
- 2C14 AOP1; Loss of Component Cooling; Revision 16
- 1C14; Component Cooling System Unit 1; Revision 26
- C1.1.14-1; Unit 1 Component Cooling System; Revision 24
- 2C14; Component Cooling System Unit 2; Revision 27
- C1.1.14-2; Unit 2 Component Cooling System; Revision 29
- 2C12.1 AOP1; Loss of Reactor Coolant Pump Seal Injection; Revision 3
- 2C3 AOP2: Loss of Reactor Coolant Pump Seal Cooling; Revision 6
- 2E-0; Reactor Trip or Safety Injection; Revision 25
- 2E-1; Loss of Reactor or Secondary Coolant; Revision 22
- 2E-4; Core Cooling Following Loss of Residual Heat Removal Flow; Revision 11
- CAP 31550; Turbine Building HELB Analysis; February 18, 2002
- Sulzer Pump Letter dated February 5, 2009; SI Pump Loss of Cooling Water for Lube System Review
- CAP 34876; Turbine Building HELB Analysis; December 21, 1969

- CAP 53226; Delay HELB Analysis Corrective Action Due to Budget Does Not Comply with 5AWI3.15.5
- CAP 5530065; Evaluate Turbine Building HELB Analysis; November 23, 2003
- Operability Evaluation 1145695; Revision 0
- Project Review Group Meeting Minutes; September 12, 2008
- OPR000509; Non-Seismic Equipment in CC System Pressure Boundary; August 3, 2004
- Completion Notes for CAP 737382-15; No Date
- Review of Seismic Qualification of Components Identified in CAP 037749; No Date
- CE005702; Condition Evaluation for Non-Seismic Equipment in CC System Pressure Boundary; No Date
- Results of July 15, 2005, Meeting Held on Qualification of CC Piping 1-CC-79, 1-CC-80, and 1-CC-138 Routed to Sample Coolers in the Cold Lab and to 123 Nitrogen Compressor
- CAP 826114; Perform Seismic Analysis of 1-CC138 up to CC-71-1 and CC-71-2; March 29, 2005
- CAP 870304; Update USAR to Incorporate Turbine Building HELB Analysis; July 26, 2005
- CAP 1002268; HELB Project Cost Overruns; October 28, 2005
- CAP 1162511; Missed Opportunities to Identify HELB and CC System Interaction; December 15, 2008
- CAP 1143812; Turbine Building HELB Funding Delays Could Affect Project Success; July 10, 2008
- 03Q0418-C-002; Assessment of Turbine Building for N-S Tornado; Revision 1
- 03Q0418-C-003; Assessment of Turbine Building for E-W Tornado; Revision 1
- Component Cooling Piping Project Engineering Work Scope; no date
- NF-38500; Architectural Ground Floor Plan Elevation 695'-0"; Revision P-76
- NF-38501; Architectural Ground Floor Plan Elevation 715'-0"; Revision AE
- NF-38502; Architectural Ground Floor Plan Elevation 735'-0"; Revision P-76
- Prairie Island Response to Generic Letter 87-02; November 20, 1995
- SL-11973-014; Chemistry Lab Component Cooling Study; January 2008
- P9160S-001; Simulator Exercise Guide Scenario #1 (CC HELB Scenario); April 7, 2009
- RM-4; Job Performance Measure – Switch Reactor Makeup Tanks on Degas; March 20, 2009
- WO 379471-01; 22 Reactor Makeup Pump; March 18, 2009
- WO 379471 50.59 Screening – Measure Unit 2 RMU Flows to CC Surge Tank and VCT for PRA; Revision 0
- WO 379471 Reactivity Management Screening Checklist; No Date
- SI-3; Job Performance Measure – Switchover to Recirculation Per 2ES-1-2, Attachment K; June 16, 2009
- USAR Section 10.2.5; Reactor Makeup Water Deoxygenation System; Revision 25
- NF 39242; Flow Diagram Unit 1 and 2 Reactor Make-up and Demineralized Water Systems; Revision BI
- CAP 1174370; No Tornado Protection for Component Cooling Water Piping to the 122 Spent Fuel Pool Heat Exchanger; March 23, 2009
- OPR 1174370-01; Operability Review for CAP 1174370; March 24, 2009
- ACE 1174370-08; Apparent Cause Evaluation for CAP 1174370; June 11, 2009
- Abnormal Procedure AB-2; Tornadoes, Thunderstorms, and High Winds; Revision 33
- Temporary Procedure Change Request 033B; Add Steps to AB-2; May 22, 2009
- Procedure 1C14 AOP 1; Loss of Component Cooling Water; Revision 16
- CAP 1178236; No High Energy Line Break Flooding Calculation for the Turbine Building; April 15, 2009
- Reactor Coolant Pump Seal Loss of Coolant Accident Break Sizes and Loss of Coolant Accident Success Criteria; February 10, 2009
- Prairie Island Nuclear Generating Plant CC/HELB Summary Report; June 5, 2009

- Calculation PI-996-61-M01; Evaluation of the Effects of Jet Impingement on Component Cooling Water Piping in the Turbine Building; Revision 0
- Letter from A. Washburn, Sulzer Pumps, to S. Skoyen, Prairie Island Nuclear Plant; June 4, 2009
- MPR Report – Prairie Island Nuclear Generating Plant Safety Injection Pump Operability Evaluation; May 12, 2009
- Engineering Change 13762; Prairie Island Turbine Building HELB Sensitivity Study; February 11, 2009
- NF-39246-1; Unit 2 Component Cooling Water System Flow Diagram; Revision 76

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
ADAMS	Agencywide Document Access Management System
AV	Apparent Violation
CAP	Corrective Action Program Document
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CCW	Component Cooling Water
DC	Direct Current
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
HELB	High Energy Line Break
IMC	Inspection Manual Chapter
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records
RASP	Risk Assessment of Operational Events
RCP	Reactor Coolant Pump
SDC	Shutdown Cooling
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
TS	Technical Specifications
USAR	Updated Safety Analysis Report