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

May 27, 2010

EA-10-070

Mr. Mark A. Schimmel
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 05000282/2010010; 05000306/2010010
PRELIMINARY GREATER THAN GREEN FINDING

Dear Mr. Schimmel:

On May 3, 2010, 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 May 3, 2010, 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 discusses a finding that has preliminarily been determined to be Greater than Green – a finding of greater than very low safety significance resulting in the need for further evaluation to determine significance and therefore the need for additional NRC action. As documented in Section 4OA5 of this report, the emergency diesel generators, the auxiliary feedwater system, and the safety-related batteries were not protected from a loss of safety function following an internal flooding event, a licensing basis event, in the Unit 1 or Unit 2 turbine building. This finding was assessed based on the best available information, including influential assumptions, using the applicable Significance Determination Process (SDP). The results of the SDP were determined to be sensitive to several analysis assumptions which could be improved with additional information. Specifically, the NRC is interested in further refining (1) the population of high energy line break piping that can realistically interact with cooling water or fire protection piping, and (2) the likelihood of a consequential pipe failure given that a defined interaction occurs. For this second item, the NRC is seeking an engineering justification from you regarding the low or high likelihood of consequential pipe failure.

Upon identification of this issue, your staff performed an operability evaluation and determined that the operability of the safety-related equipment mentioned above could not be assured with the turbine building roll up doors in the closed position. As a result, a compensatory measure was established to ensure that the doors were maintained at least 16 inches open such that the flood waters could flow out the doors. Several other compensatory measures were also implemented. These compensatory measures would prevent a loss of safety function from occurring during a turbine building internal flooding event.

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 can be found at 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 the date 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 in writing 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. The final resolution of this matter will be conveyed in separate correspondence.

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.

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/

Steven West, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000282/2010010; 05000306/2010010
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-282; 50-306
License Nos:	DPR-42; DPR-60
Report No:	05000282/2010010; 05000306/2010010
Licensee:	Northern States Power Company, Minnesota
Facility:	Prairie Island Nuclear Generating Plant, Units 1 and 2
Location:	Welch, MN
Dates:	April 29 through May 3, 2010
Inspectors:	K. Stoedter, Senior Resident Inspector P. Zurawski, Resident Inspector L. Kozak, Senior Reactor Analyst
Approved by:	John B. Giessner, Chief Branch 4 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

Enclosure

SUMMARY OF FINDINGS

IR 05000282/2010010, 05000306/2010010; 04/29/10 – 05/03/10; Prairie Island Nuclear Generating Plant, Units 1 and 2; Inspection of turbine building internal flooding vulnerability.

This report covers the review of a design deficiency associated with the failure to protect several safety-related systems from a loss of safety function following a turbine building internal flooding event. The inspectors identified one apparent violation (AV) with a preliminary significance of Greater than Green. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). 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

- Preliminary Greater than Green. An apparent violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors due to the licensee's failure to establish measures to ensure that engineered safety features such as the emergency diesel generators, the auxiliary feedwater system, and the safety-related batteries were not adversely affected by events that cause turbine building flooding. As a result, flooding from these events would cause a loss of safety function for these systems. This issue was entered into the licensee's corrective action program (CAP) as CAP 1178236. Upon identifying this issue, the licensee implemented compensatory measures to ensure that the systems listed above were not adversely impacted following a turbine building internal flood.

  This finding was determined to be more than minor because it impacted the design control and external events attributes 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 inspectors performed a Phase 1 SDP evaluation and determined that a Phase 3 evaluation was required because the finding represented a loss of safety function of multiple mitigating systems. A Phase 2 SDP evaluation was not performed because the Phase 2 SDP worksheets do not apply to internal flooding events. The results of the Phase 3 SDP assessment showed that this finding was potentially Greater than Green. No cross-cutting aspect was assigned to this finding because licensee decisions made in regard to evaluating the susceptibility of mitigating systems equipment to turbine building internal flooding events were made more than 3 years ago and therefore, not reflective of current plant performance. (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 05000282/2009003-01; 05000306/2009003-01: Potential Turbine Building Flooding Issues

a. Inspection Scope

The inspectors reviewed the circumstances surrounding the licensee's failure to protect the emergency diesel generators, the auxiliary feedwater system and the safety-related batteries from a loss of safety function following an internal flooding event in the Unit 1 or Unit 2 turbine building.

b. Findings

Introduction: An apparent violation (AV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors due to the licensee's failure to establish measures to ensure that engineered safety features such as the emergency diesel generators (EDGs), the auxiliary feedwater system and the safety-related batteries were not adversely affected following a turbine building internal flooding event.

Description: In late 2009, the NRC issued a Green finding for Unit 1 and a White finding for Unit 2 due to the discovery that a high energy line break (HELB) in the turbine building could result in a loss of safety function for the component cooling water system. As part of the extent of condition review for this issue, the licensee identified that a turbine building HELB could result in the subsequent failure of cooling water piping and the actuation of fire protection sprinklers such that a large supply of water could be introduced into the turbine building. This large supply of water could have resulted in an internal flooding condition that impacted the safety function of the EDGs, the auxiliary feedwater pumps and the safety-related batteries. The licensee also identified that a turbine building internal flooding analysis had not been performed. These issues were documented in corrective action program document (CAP) 1178236 dated April 15, 2009. These issues were also documented as an unresolved item in NRC Inspection Report 05000282/2009003; 05000306/2009003.

Turbine Building Design

The Unit 1 and Unit 2 turbine buildings at Prairie Island consist of multiple elevations. The condenser pits (one per unit) were located below grade and had the ability to hold approximately 750,000 gallons of water. The first floor of the turbine building (commonly referred to as the 695-foot elevation) was located at grade level. The 695-foot elevation of the Unit 1 and Unit 2 turbine buildings also provided access to rooms/buildings containing the EDGs, the auxiliary feedwater system (for both units) and the safety-related batteries.

Design and Licensing Basis Information

The inspectors reviewed the licensee's licensing and design bases and found the following:

On August 3, 1972, the Atomic Energy Commission (AEC) (the former name of the Nuclear Regulatory Commission) issued a letter to Northern States Power Company (the licensee) requesting that the Prairie Island plant design be reviewed to determine whether the failure of any non-seismic-Class I equipment, particularly the circulating water system, could cause flooding sufficient to adversely affect the performance of engineered safety systems. The licensee was also required to determine whether the failure of any equipment could cause flooding such that common mode failure of redundant safety-related equipment would result.

To address this request, the licensee asked the plant's architect engineer, Pioneer Service and Engineering Company, to review the Prairie Island design. The results of this review were transmitted from the architect engineer to the licensee on September 21, 1972. The inspectors reviewed this letter and found that Pioneer's review was focused on the adequacy of the plant design to cope with a failure of the circulating water system, flooding of the condenser pit, and flooding the turbine building. The results of this review showed that a failure of the circulating water system would result in filling the condenser pit in 2.5 minutes. After filling the condenser pit, the cooling air ducts used to ventilate the D1 and D2 EDG rooms during engine operation would fill with water causing a loss of safety function. The rate of water level rise on the 695-foot elevation of the turbine building was estimated to be 1 inch every 15 seconds. Pioneer concluded that the 2.5 minutes available from the time the circulating water system failed until the water reached the EDG cooling air ducts was sufficient for the control room to dispatch a non-licensed operator to perform a visual inspection of the condenser pit, report back to the control room, and manually shut down the circulating water system.

On September 26, 1972, the licensee received an additional letter from the AEC. This letter requested that the licensee review the plant design to determine whether the failure of any non-seismic equipment, particularly in the circulating water and fire protection systems, could result in a condition, such as flooding or the release of chemicals, that might potentially adversely affect the performance of safety-related equipment required for safe shutdown of the facility or to limit the consequences of an accident. This letter became known as the DeYoung letter.

Northern States Power responded to the DeYoung letter on October 23, 1972. In this letter, the licensee concluded that "where the potential of flooding engineered safety features exist, the operator is provided with sufficient information and means to take corrective action in a timely manner." There was no discussion regarding the term "sufficient information" or how timely the actions needed to be. However, the inspectors believed that the time referred to in this letter was the 2.5 minutes discussed in the September 26, 1972, letter. The licensee further described that level switches located in the condenser pit would provide a control room alarm to alert operations personnel that water was accumulating inside the condenser pit. The licensee stated that "since all safety-related mechanical/electrical components are on or above floor elevation 695 when the condenser pit high-high-high water level annunciates, there will still be ample time for visual operator inspection of the situation and initiation of corrective actions.

There is no further danger of loss of safeguards by flooding after shut down of the circulating water pumps." The licensee concluded their response by stating that "any potential failure of non-Class I equipment does not pose a threat to the overall plant safety, either by impeding safeguard performance or by causing common mode failure of redundant safeguard related equipment." No safety evaluation report could be located to determine whether the AEC viewed the licensee's response as adequate. Conversely, no records could be located to indicate the response was inadequate.

On December 12, 1972, the licensee received a letter from the AEC regarding postulated steam pipe breaks outside of containment. This letter required that the Prairie Island Nuclear Generating Plant be designed such that the following statements were true:

- Failure of any structure, including seismic Class II or Class III structures, caused by the accident should not cause failure of any other structure, system or component in a manner to adversely affect the mitigation of the consequences of the accident and the capability to bring the units to a cold shutdown condition;

- Rupture of a pipe carrying high energy fluid, including a steam line rupture, should not directly or indirectly result in the loss of required redundancy in any portion of the protection system, Class 1E electric system, engineered safety feature equipment, cable penetration, or their interconnecting cables required to mitigate the consequences of that event and place the reactors in cold shutdown condition; and

- A discussion should be provided of the potential for flooding safety-related equipment in the event of failure of a feedwater line or any other high energy fluid line.

This letter became known as the Giambusso letter. The licensee's responses to this letter appeared to be focused on events occurring in the auxiliary building. There is currently no information contained in the Updated Safety Analysis Report (USAR) regarding HELBs in the turbine building.

On January 3, 1986, the licensee sent a letter to the NRC regarding the resolution of Generic Issue No. 77, "Flooding of Safety Equipment Compartments by Backflow Through Floor Drains." The letter contained the following statement about water flow in the turbine buildings:

> "Once the water goes above Elevation 695' - 0" the water storage capacity increases greatly such that, with the exception of the EDG room cooling air ducts, it would take about 3 more minutes to reach Elevation 695' - 10" whereby the flooding would affect safety-related equipment."

To address this statement, the licensee implemented a permanent modification to ensure that the circulating water pumps tripped due to high water level in the condenser pit. The inspectors determined that although the EDG room cooling air duct flooding vulnerability was identified in 1972, the modifications implemented to address this vulnerability were not installed until 1988 and 1989. The inspectors also found that the licensee had failed to consider other flooding scenarios (such as HELB induced flooding

or flooding due to random pipe breaks) as part of addressing any of the previous letters sent by the AEC or the NRC.

As discussed above, on April 15, 2009, the licensee initiated a CAP to document that a turbine building HELB could result in an internal flooding condition that impacted the safety function of the EDGs, the auxiliary feedwater pumps and the safety-related batteries. The licensee performed an operability review and determined that operability of these safety-related systems could not be assured. To remedy this immediate concern, the licensee opened both turbine building roll up doors to prevent water from accumulating on the 695-foot elevation of the turbine building if an internal flooding event occurred.

The same day, the inspectors reviewed the licensee's corrective action system to determine how the licensee had evaluated and addressed industry internal flooding operating experience (OE) from 2005. The inspectors found that the licensee had conducted an OE review, determined that the OE was applicable to Prairie Island, and assigned several actions to evaluate specific portions of the turbine building (including the battery rooms, the auxiliary feedwater pump room and the EDG rooms). However, no work had been performed on these reviews as of April 2009. In summary, the inspectors concluded that the failure to adequately protect the safety related components from the affects of license basis events was within their ability to foresee and correct, and is, therefore, a performance deficiency.

Since April 2009, the licensee has initiated additional CAPs and operability evaluations associated with internal flooding of the Unit 1 or Unit 2 turbine building. The licensee found that based upon the best available information, the pre-April 2009 plant configuration was not adequate to ensure that operations personnel could take appropriate actions following a turbine building internal flood to protect safety-related equipment prior to the equipment (both trains) being impacted by the flood water. The licensee took the following actions to ensure that safety-related equipment was protected:

- The turbine building roll up doors were opened (as the seasons permit) to allow the flood waters to exit the turbine building;
- The bottom of the roll up doors were modified to allow the doors to stay partially open during the fall and winter;
- Approximately 18-inch flood walls were constructed to protect the Unit 1 and Unit 2 EDGs;
- Valve access covers and metal plates on the floor of the auxiliary feedwater pump room were secured with fasteners to prevent additional water intrusion; and
- All doors leading from the turbine building into rooms housing safety-related equipment were inspected. Repairs were made to the door from the Unit 2 turbine building into the safety-related battery room to lessen the rate of water intrusion into the room.

Analysis: The inspectors determined that the licensee's failure to establish measures to ensure that the EDGs, the auxiliary feedwater system and the safety-related batteries were protected from a loss of safety function following an internal flood was a performance deficiency that required an evaluation using the Significance Determination Process (SDP) described in NRC Inspection Manual Chapter (IMC) 0609. The

inspectors also determined that this finding should be assigned to the Mitigating Systems cornerstone because it impacted systems used in short term and long term heat removal.

The inspectors performed a Phase 1 SDP analysis and concluded that the finding represented a loss of safety function of several mitigating systems including the EDGs, the auxiliary feedwater system and the safety-related batteries (direct current power). A Phase 2 SDP analysis was not performed because the Phase 2 process was not applicable for internal flooding scenarios. As a result, the inspectors requested that a regional senior reactor analyst (SRA) perform a preliminary SDP phase 3 analysis.

The SRA used spreadsheet calculations to estimate the risk from the internal flooding scenarios affected by this finding. The Standardized Plant Analysis Risk (SPAR) model for Prairie Island (Revision 3.45) and the Prairie Island Phase 2 SDP worksheets were used to determine the success criteria for loss of main feedwater events and to determine system functional requirements for auxiliary feedwater, feed and bleed, and high pressure recirculation.

The baseline core damage frequency (CDF) for the internal flooding scenarios related to the performance deficiency was assumed to be much lower than the CDF estimated for the plant in the pre-April 2009 configuration. Therefore, the CDF calculated in this SDP Phase 3 analysis was assumed to represent the delta CDF due to the performance deficiency.

As part of this SDP Phase 3 evaluation, the following three flooding scenarios were evaluated:

- High Energy Line Break-induced flooding and consequential failure of other piping;
- Random failure of a cooling water (CL) pipe; and
- Seismically-induced pipe failures.

<u>HELB-induced Flooding and Consequential Failure of Cooling Water Pipe and/or Fire Protection Piping</u>

The SRA conducted a plant tour to observe piping arrangements and the overall layout of the Unit 1 and Unit 2 turbine buildings. Plant general arrangement drawings and piping and instrumentation drawings were also reviewed. Based upon this information, the SRA assumed that the turbine building HELB could result in the consequential failure of CL or fire protection system piping such that an unlimited supply of water from the Mississippi River could be introduced into the turbine building causing an internal flood.

The SRA assumed that the postulated HELB resulted in a loss of main feedwater event. If the CL and/or fire protection (FP) piping also failed as a consequence of the HELB, and the CL and/or FP piping was not isolated by the operator, the auxiliary feedwater, instrument air, high pressure recirculation, and direct current power functions could be impacted. This would lead to core damage. Information provided by the licensee showed that if the HELB resulted in the consequential failure of the largest nonsafety-related CL pipe, the functions listed above could fail in approximately 1 hour due to the amount of water flowing into the turbine building. The amount of time available prior to

the function failing increased as the amount of water flowing into the turbine building dropped.

The inspectors and the SRA discussed the postulated HELB and turbine building flooding sequence of events with operations, engineering and licensee risk personnel. The inspectors also reviewed operating procedures and simulator training associated with HELBs and turbine building flooding. The SRA determined that isolating the flood sources required the operators to diagnose which system(s) were causing the flooding after the postulated HELB event. If the cause was diagnosed as a CL pipe failure, actions could be taken in the control room to stop the flow of water from the CL system using established procedures. If the diagnosis determined that the FP system was also introducing water into the turbine building due to a pipe break, sprinkler actuation or deluge system actuation, the operators would need to perform manual actions in the turbine building or plant screenhouse to stop the flow of water.

To simplify this portion of the SDP Phase 3 analysis, two categories of HELB-induced flooding events were analyzed. Category 1 was defined as HELB-induced flooding events that resulted in flow rates greater than 18,000 gallons per minute (gpm) (including any fire sprinkler flow). Category 2 was defined as HELB-induced flooding events resulting in flow rates between 10,000 gpm and 18,000 gpm for Unit 1 and between 7800 gpm and 18,000 gpm for Unit 2. These categories were selected after considering the results of Engineering Change EC 15656, “Evaluation of flooding times and flow rates associated with Unit 1 and Unit 2 TB [turbine building] for significance determination,” and Calculation ENG-ME-759, “Gothic Internal Flooding Calculation for the Turbine Building.” These documents showed that a loss of safety function would not occur if operator action was successful within 1 hour for 18,000 gpm floods and within 2 hours for Unit 1 10,000 gpm floods or Unit 2 7800 gpm floods. The length of HELB piping that could interact with the CL and FP piping was estimated by the licensee and resulting CL and/or FP flow rates were calculated. These assumptions were used in this preliminary SDP evaluation.

The SRA estimated the frequency of HELB events that interact with CL and/or FP piping by determining the pipe failure frequency for pressurized water reactor feedwater and condensate piping for “Major Flooding” using Electric Power Research Institute (EPRI) Document 1013141, Revision 1, “Pipe Rupture Frequencies for Internal Flooding PRAs,” Table A-51, and the pipe lengths provided by the licensee. The HELB pipe was assumed to interact with a target pipe (the non-safety-related CL and/or FP piping) and result in the failure of that target pipe.

Due to the complexity of responding to the postulated event, it was assumed that operator actions required at least 1 hour to detect and isolate the flood source. As a result, for the largest flood rates, the event could not be mitigated and core damage was assumed to occur. If the flooding flow rate resulted in the operators having between 1 and 2 hours to respond, a human error probability (HEP) for operator action was estimated using NUREG/CR-6883, “The SPAR-H Human Reliability Analysis Method.” The SRA calculated a HEP associated with failing to isolate flood sources of 0.33 assuming that the actions were performed under high stress, were of moderate complexity, and consisted of poor ergonomics for both diagnosis and action.

Random Failure of Non-safety-Related Cooling Water Piping

A failure of non-safety-related CL piping in the turbine building would initially result in filling up the condenser pit. When the water in the condenser pit was 5 feet deep, the circulating water pumps would trip resulting in a subsequent reactor trip. As water from the broken CL pipe continued to enter the turbine building, it would reach a level where it would impact the normal feedwater and condensate systems. Depending on the location and size of the pipe failure, operators may receive low cooling water pressure or high cooling water flow alarms in the control room. Similar to the HELB scenario, operators would need to diagnose the source of the flood and identify the failed pipe. After identification, the flood could be stopped by operator action within the control room.

Similar to the HELB-induced flooding information discussed previously, two categories of flooding were defined for this event. Category 1 was defined as random CL pipe breaks that resulted in flooding flow rates greater than 18,000 gpm. The second category was defined as CL pipe breaks that resulted in flow rates between 12,500 gpm and 18,000 gpm. Flood rates of less than 12,500 gpm would take greater than 1 hour to fill the condenser pit. Operator action to isolate the flood source after 1 hour was assumed to be reliable. For category 1 events, it was assumed that operators would not be able to take action to stop the water flow before both trains of mitigating equipment would be impacted. This was assumed to result in core damage. For the second category it was assumed that at least 1 hour was available for operator action.

The pipe failure frequency for Pressurized Water Reactor Service Water – River Water piping for “Major Flooding” from EPRI 1013141, Revision 1, “Pipe Rupture Frequencies for Internal Flooding PRAs,” Table A-20, was used with the turbine building pipe lengths of non-safety-related CL piping provided by the licensee to estimate the frequency of random CL pipe failures that could result in flooding impacts to mitigating systems.

Operator response to this postulated event was assumed to be less complicated than the response to a HELB-induced consequential failure of CL. For category 2 events, a HEP for operator action to isolate flooding before mitigating system damage was estimated to be 0.2. This estimate assumed high stress and poor ergonomics for diagnosis and high stress for action execution.

Seismically-Induced Failure of Non-safety- Related Turbine Building Piping

A seismic event can result in the failure of one or more non-safety-related pipes resulting in turbine building flooding. A loss of offsite power event may occur as a result of the seismic event. Since loss of offsite power was a consequence of the seismic event, the SRA concluded that the EDG function was required in the SDP evaluation. If flood sources were not isolated before the EDG function was lost (for Unit 1) or before other mitigating functions were lost, core damage was assumed.

Using guidance from the Risk Assessment of Operation Events (RASP) handbook, Volume 2 – External Events, only the “Bin 2” seismic events were assumed to represent a delta CDF. “Bin 2” was defined in the RASP handbook as seismic events with intensities greater then 0.3g but less than 0.5g. Earthquakes of lesser severity were unlikely to result in large pipe failures and earthquakes of a larger magnitude could result in major structural damage throughout the plant. The frequency of an earthquake in “Bin 2” was estimated to be 1.4E-5 per year. A high confidence low probability of failure

of 0.3g was assigned to the most susceptible component in the non-safety-related portion of the CL system based on preliminary information obtained by the licensee on the seismic fragility of CL system components. This value was used to estimate a pipe failure probability using an average bin acceleration of 0.38g. A 0.5 probability was then assigned using engineering judgment for the likelihood that the flood was large enough that operator action was not possible before damage to EDG, auxiliary feedwater, or direct current power systems occurred. This judgment was based on the conditional probability of large pipe rupture in large diameter piping in Table 3A-2-2 of the RASP Handbook, Volume 2.

Preliminary Significance Determination Process Phase 3 Conclusions

The total delta CDF calculated for Unit 1 and Unit 2 was greater than 1E-6 per year, which was determined to be greater than very low safety significance (Green). The dominant scenario was a HELB which interacted with non-safety-related CL and/or FP piping, causing the failure of that pipe and subsequent flooding. In this scenario, the operator fails to isolate the flooding prior to the flood damaging the auxiliary feedwater, instrument air and high pressure recirculation functions or prior to the flood damaging both trains of direct current power. If either of these flood impacts occurred, no mitigation was available and core damage was assumed.

The results of the SDP were determined to be sensitive to several analysis assumptions which could be improved with additional information. Specifically, the NRC is interested in further refining (1) the population of HELB piping that can realistically interact with CL and FP piping, and (2) the likelihood of a consequential pipe failure given that a defined interaction occurs. For this second item, the NRC is seeking engineering justification from the licensee for a low or high likelihood of consequential pipe failure in the identified pipe interactions rather than a probabilistic estimate of consequential pipe failure.

No cross-cutting aspect was assigned to this finding as the decision regarding the completion of reviews assigned as part of the OE evaluation were made greater than 3 years ago.

Old Design Issue Review

NRC IMC 0305, “Operating Reactor Assessment Program,” Section 04.11 defines an “old design issue” as an inspection finding involving a past design-related problem in the engineering calculations or analyses, the associated operating procedure, or installation of plant equipment that does not reflect a performance deficiency associated with existing licensee programs, policy, or procedures. Section 12.01(a) of IMC 0305 states that the NRC may refrain from considering safety significant inspection findings in the assessment program for a design-related finding in the engineering calculations or analysis, associated operating procedure, or installation of plant equipment if 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.

The inspectors determined that this issue did not qualify as an old design issue. Specifically, the issue was not licensee-identified as part of a voluntary initiative. It was identified as part of an extent of condition review for a previous NRC-identified issue. Had the licensee taken appropriate actions following their review of the 2005 OE, the issue would have likely been identified. Lastly, no actions had been taken after identifying that the plant was susceptible to turbine building flooding in 2005. As a result, the issue was not corrected within a reasonable period of time.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis, as defined in Section 50.2, and as specified in the license application, for those structures, systems and components to which this appendix applies 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.

Title 10 CFR 50.2 defines design basis as that information which identifies the specific functions to be performed by a structure, system, or component of a facility.

USAR Section 6.1.2.8 states, in part, that internal flooding which could be postulated to adversely affect the performance of engineered safety features was a part of the original plant design criteria. As such, the turbine building was designed to have the capacity to accommodate large internal floods since it takes time to increase the water levels to an elevation where nuclear safety-related equipment is located.

Section 6.1.1 of the USAR stated that the EDGs, the auxiliary feedwater system, and the safety-related batteries were engineered safety features of the Prairie Island Nuclear Generating Plant.

Contrary to the above, prior to January 29, 2010, the licensee failed to establish measures to assure that the applicable regulatory requirements and the design basis for the EDGs, the auxiliary feedwater system, and the safety-related batteries were correctly translated into specifications, drawings, procedures, and instructions. Specifically, the licensee failed to assure that a turbine building internal flooding event would not adversely affect the performance of multiple engineered safety features. This is an apparent violation of 10 CFR Part 50, Appendix B, Criterion III pending the completion of the final significance determination **(AV 05000282/2010010-01; 05000306/2010010-01, Failure to Ensure Design Measures Were Appropriately Established for the Emergency Diesel Generator, Auxiliary Feedwater, and Safety-Related Battery Systems)**.

4OA6 Management Meetings

.1 Exit Meeting Summary

On May 3, 2010, the inspectors presented the inspection results to M. Schimmel 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. Schimmel, Site Vice President
B. Sawatzke, Director Site Operations
K. Ryan, Plant Manager
J. Anderson, Regulatory Affairs Manager
C. England, Radiation Protection General Supervisor
D. Kettering, Site Engineering Director
J. Lash, Operations Manager
R. Madjerich, Production Planning Manager
M. Milly, Maintenance Manager
J. Muth, Nuclear Oversight Manager
S. Northard, Performance Improvement Manager
K. Peterson, Business Support Manager
J. Sternisha, Training Manager

Nuclear Regulatory Commission

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

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000282/2010010-01; 05000306/2010010-01	AV	Failure to Ensure Design Measures Were Appropriately Established for the Emergency Diesel Generator, Auxiliary Feedwater, and Safety-Related Battery Systems (Section 4OA5.1)

Closed

05000282/2009003-01; 05000306/2009003-01	URI	Potential Turbine Building Flooding Issues

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.

Section 4OA5

- 10 CFR 50.59 Screening 3188; OPR 1178239 Compensatory Measures; Revision 0
- 5AWI 8.9.0; Internal Flooding Drainage Control; Revision 4
- 5AWI 8.9.0; Internal Flooding Drainage Control; Revision 4
- Abnormal Operating Procedure 2C3 AOP2; Loss of RCP Seal Cooling; Revision 6
- Abnormal Operating Procedure 2C35 AOP2; Loss of Pumping Capacity or Supply Header Without SI; Revision 11
- C31 AOP1; Fire Protection Line Break; Revision 0
- C35 AOP2; Loss of Pumping Capacity or Supply Header Without SI; Revision 11
- C47022-0104; Turbine Building Steam Exclusion Actuated; Revision 45
- C47022-0307; Fire Header Lo Pressure; Revision 44
- C47520-0103; Loop A Cooling Water Hi Flow; Revision 32
- C47520-0201; 21 Cooling Water Pump Overload; Revision 32
- C47520-0202; 22 Cooling Water Pump Running; Revision 34
- C47520-0203; Loop A Cooling Water Lo Pressure; Revision 32
- CAP 1178236; No HELB Flooding Calculation for Turbine Building; April 15, 2009
- CAP 1179019; Actions from OEER 888906 Have Not Been Completed; April 21, 2009
- CAP 1179979; Unit 2 Turbine Roll Up Door Found at 14 Inches Open; April 28, 2009
- CAP 1192814; Turbine Building HELB Analysis Not Completed by INPO Date; August 7, 2009
- CAP 1199492; OPR 1174113 Conclusions Not Appropriate for CAP 1199165; September 24, 2009
- CAP 1202820; Potential Concern Raised for a HELB in the Turbine Building; October 16, 2009
- CAP 1203173; Potential Concern Raised Related to a HELB in the Turbine Building; October 19, 2009
- CAP 1203370; OPR 1178236-04 Did Not Consider Time to Enter AOP in Evaluation; October 20, 2009
- CAP 1206060; Potential CL System Unanalyzed Condition During HELB; November 6, 2009
- CAP 1208131; Insufficient Time for Operator Response in Certain Turbine Building HELBs; November 24, 2009
- CAP 1213357; Potential HELB Pipe Whip Impact on Doors 42 and 43; January 12, 2010
- CAP 1213638; Dumpster in Front of the West Turbine Roll Up Door; January 14, 2010
- CAP 1215137; Forklift Parked in Unit 1 Turbine Building Truck Aisle; January 25, 2010
- CAP 1218454; Cooldown to Cold-Shutdown After a HELB; February 16, 2010
- CAP 781440; Evaluate D1/D2 Compartments for Internal Flooding; November 20, 2004
- CAP 830732; Determine the Effects of Potential Flooding in the Turbine Building; April 8, 2005
- Internal Flooding – Accident Sequence Analysis for Turbine Building Floods; March 2010
- LER 2009-006-00; Unanalyzed Condition Due to Potential Safety System Susceptibility to Turbine Building Flooding Due to a Postulated High Energy Line Break; December 17, 2009
- Licensing and Design Bases for Prairie Island Nuclear Generating Plant Turbine Building Internal Flooding; January 29, 2010
- NRC Information Notice 2005-30; Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design; November 7, 2005

- OPR 1178236-04; Operability of Safety-Related Equipment Following Turbine Building Flooding; Multiple Revisions
- OPR 1203173-01; Impact of Turbine Building HELB on Auxiliary Feedwater Pump Room Heat Up; Revisions 0 and 1
- OPR 1206060-01; Evaluation of HELB Induced Flood, Loss of Offsite Power and Single Failure; Revisions 0 and 1
- PINGP Calculation ENG-ME-586; Effects of Flooding in the AFW Pump Room from a Postulated Pipe Rupture; Revision 0
- Plant Safety Procedure F9; High Energy Line Break/Leak; Revision 8
- Procedure H36; Plant Flooding; Revision 1
- Significance Determination Input Information for PINGP Turbine Building Internal Flooding; February 19, 2010
- Simulator Exercise Guide P9110SE-CLHELB-2; High Energy Line Break with Loss of Offsite Power – Cooling Water and Fire Protection Response; Revision 0
- Special Test Procedure TP 1398; Verify Physical Inputs to Internal Flooding Evaluations; Revision 1

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AEC	Atomic Energy Commission
AV	Apparent Violation
CAP	Correction Action Program
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CL	Cooling Water
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
FP	Fire Protection
gpm	Gallons Per Minute
HELB	High Energy Line Break
HEP	Human Error Probability
IMC	Inspection Manual Chapter
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PARS	Publicly Available Records System
RASP	Risk Assessment of Operational Events
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
SRA	Senior Reactor Analyst
USAR	Updated Safety Analysis Report

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/

Steven West, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000282/2010010; 05000306/2010010
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

DOCUMENT NAME: G:\PRAI\Prai 2010 010.doc
□ Publicly Available □ Non-Publicly Available □ Sensitive □ Non-Sensitive
To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII		RIII		RIII		NRR		OE		RIII	
NAME	JGiessner:dtp RLerch for on 5/20/2010		LKozak		SOrth *PPelke for		LJames*via Email from GGulla		GGulla via email		SWest *GLS for	
DATE	05/25/2010 JAG		05/20/2010		05/25/2010		05/25/2010		05/25/2010		05/26/2010	

OFFICIAL RECORD COPY

Letter to M. Schimmel from S. West dated May 27, 2010.

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2
NRC INSPECTION REPORT 05000282/2010010; 05000306/2010010
PRELIMINARY GREATER THAN GREEN FINDING

DISTRIBUTION:
Susan Bagley
RidsNrrPMPrairieIsland
RidsNrrDorlLpl3-1 Resource
RidsNrrDirsIrib Resource
Cynthia Pederson
Steven Orth
Jared Heck
Allan Barker
Carole Ariano
Linda Linn
DRPIII
DRSIII
Patricia Buckley
Tammy Tomczak
ROPreports Resource
ADAMS (PARS)
RidsSecyMailCenter Resource
OCADistribution
Bill Borchardt
Bruce Mallett
Roy Zimmerman
James Wiggins
Nick Hilton
Steve West
Gary Shear
Anne Boland
Kenneth OBrien
John Giessner
Rani Franovitch
Paul Bonnett
Jeff Circle
Laura Kozak
Jeff Mitman
Suni Weerakkody
Harral Logaras
Marvin Itzkowitz
Catherine Scott
Eric Leeds
Bruce Boger
Daniel Holody
Carolyn Evans

William Jones
Gerald Gulla
MaryAnn Ashley
Holly Harrington
Hubert Bell
Guy Caputo
Mona Williams
James Lynch
Viktoria Mitlyng
Prema Chandrathil
Paul Pelke
Sarah Bakhsh
Magdalena Gryglak
OEMAIL Resource
OEWEB Resource