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

April 12, 2010

Mr. Mark A. Schimmel
Site Vice President
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 ULTIMATE HEAT SINK INSPECTION REPORT 05000282/2010008; 05000306/2010008

Dear Mr. Schimmel:

On March 26, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an Ultimate Heat Sink inspection at your Prairie Island Nuclear Generating Plant, Units 1 and 2. The enclosed report documents the inspection findings, which were discussed on March 26, 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.

Based on the results of this inspection, one NRC-identified finding of very low safety-significance was identified. This finding involved a violation of NRC requirements. However, because of the very low safety-significance, and because the issue was entered into your corrective action program, the NRC is treating the issue as an Non-Cited Violation (NCV) in accordance with Section VI.A.1 of the NRC Enforcement Policy. If you contest the subject or severity of the NCV, 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 the 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, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure: Inspection Report 05000282/2010008; 05000306/2010008
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-282; 50-306
License Nos:	DPR-42; DPR-60
Report No:	05000282/2010008; 05000306/2010008
Licensee:	Northern States Power Company, Minnesota
Facility:	Prairie Island Nuclear Generating Plant, Units 1 and 2
Location:	Welch, MN
Dates:	June 15, 2009, through March 26, 2010
Inspectors:	G. O'Dwyer, Reactor Inspector A. Dunlop, Senior Reactor Inspector
Approved by:	A.M. Stone, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000282/2010008, 05000306/2010008; 06/15/2009 – 03/26/2010; Prairie Island Nuclear Generating Plant, Units 1 and 2; Ultimate Heat Sink.

This report covers an announced baseline inspection by regional inspectors. One Green finding associated with a Non-Cited Violation (NCV) was identified by the inspectors. 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 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. A finding of very low safety-significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the failure to determine the minimum cooling water system flow required after a design basis earthquake (DBE) to safely shutdown both reactors and to correctly translate these results into procedures. Specifically, the licensee failed to determine the cooling water flow rate necessary to shutdown both reactors after a DBE and ensure that this flow rate remained within the capacity of the emergency intake line. As a result, design bases were not correctly translated into procedures. The licensee confirmed through a preliminary calculation that the system remained operable.

  The finding was determined to be more than minor because the failure to determine the cooling water flow necessary to shutdown both reactors after a DBE could have provided incorrect guidance in the procedure and to the operators. This finding is of very low safety-significance (Green) because the design deficiency was confirmed not to result in loss of operability or functionality. This finding has a cross-cutting aspect in the area of problem identification and resolution because the licensee did not take appropriate corrective actions to address safety issues in a timely manner, commensurate with its safety-significance and complexity. [P.1(d)] (Section 1R07.1.b.(1))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R07 Heat Sink Performance (71111.07T)

.1 Triennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed completed surveillances, vendor manual information, associated calculations, performance test results, and cooler inspection results associated with the Unit 2 emergency diesel generator D2 lube oil cooler (034-021). This heat exchanger was chosen based on its risk significance in the licensee's probabilistic safety analysis, its important safety-related mitigating system support functions, and its relatively low margin.

The inspectors verified that testing, inspection, maintenance, and monitoring of biotic fouling and macrofouling programs were adequate to ensure proper heat transfer. This was accomplished by verifying the test method used was consistent with accepted industry practices or equivalent, the test conditions were consistent with the selected methodology and the test acceptance criteria were consistent with the design basis values. The inspectors also verified that the test results appropriately considered differences between testing conditions and design conditions and that the frequency of testing based on trending of test results was sufficient to detect degradation prior to loss of heat removal capabilities below design basis values.

The inspectors also reviewed the methods and results of heat exchanger performance inspections. The inspectors verified that the methods used to inspect and clean heat exchangers were consistent with as-found conditions identified and expected degradation trends and industry standards, the licensee's inspection and cleaning activities had established acceptance criteria consistent with industry standards, and the as-found results were recorded, evaluated, and appropriately dispositioned such that the as-left condition was acceptable.

In addition, the inspectors verified the condition and operation of the Unit 2 emergency diesel generator D2 lube oil cooler (034-021) was consistent with design assumptions in heat transfer calculations and as described in the final safety analysis report. This included verification that the number of plugged tubes was within pre-established limits based on capacity and heat transfer assumptions. The inspectors verified that the licensee evaluated the potential for water hammer and established adequate controls and operational limits to prevent heat exchanger degradation due to excessive flow induced vibration during operation. In addition, eddy current test reports and visual inspection records were reviewed to determine the structural integrity of the heat exchanger.

The inspectors verified the performance of the ultimate heat sink (UHS) by tests or other equivalent methods to ensure availability and accessibility to the inplant cooling water systems, which at Prairie Island Nuclear Generating Plant is designated the cooling water (CL) system. The focus was on attributes listed in IP 71111-07, Section 02.02.d.1, 2, and 4.

The inspectors verified that the licensee's inspection of the intake canal was thorough and of significant depth to identify degradation of the shoreline protection or loss of structural integrity. The inspectors performed walkdowns of accessible portions of the intake canal. This included verification that vegetation present along the slopes was trimmed, maintained and was not adversely impacting the embankment. In addition, the inspectors verified the licensee ensured sufficient reservoir capacity by trending and removing debris or sediment buildup in the UHS.

The inspectors reviewed the results of the licensee's inspection of the safety-related and non-safety-related intake bays. The inspectors verified that identified settlement or movement indicating loss of structural integrity and/or capacity was appropriately evaluated and dispositioned by the licensee. In addition, the inspectors verified that the licensee ensured sufficient reservoir capacity by trending and removing debris or sediment buildup in the UHS. The inspectors performed a walkdown of accessible portions of the plant screenhouse and the intake screenhouse.

The inspectors reviewed the licensee's operation of the service water system and the UHS. This included verification that instrumentation, which is relied upon for decision making, was available and functional. In addition, the inspectors verified that macrofouling was adequately monitored, trended, and controlled by the licensee to prevent clogging. The inspectors verified that the licensee's biocide treatments for biotic control were adequately conducted and that the results were monitored, trended, and evaluated. The inspectors also reviewed design changes to the service water system and the UHS. The inspectors reviewed the licensee's system configuration and flow assumptions during design basis earthquake conditions.

In addition, the inspectors reviewed corrective action program documents related to the heat exchangers/coolers and heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues and to evaluate the effectiveness of the corrective actions. The documents that were reviewed are included in the Attachment to this report.

These inspection activities associated with the Unit 2 emergency diesel generator D2 lube oil cooler and the UHS attributes constituted two heat sink inspection samples as defined in IP 71111.07-05.

b. Findings

(1) Failure to Determine the Minimum Cooling Water (CL) System Flow Required after a Design Basis Earthquake (DBE)

Introduction: A finding of very low safety-significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the failure to determine the minimum CL system flow required after a DBE to safely shutdown both reactors and to correctly translate these results into procedures.

Description: During this inspection, the inspectors noted that the licensee had submitted License Amendment Request (LAR) letter dated May 3, 2004, to change the methodology for demonstrating functionality of the CL system following a DBE. The revised methodology was to verify that the CL system flow to required components after a DBE would be adequate to safely shutdown both reactors and remain within the 11,600 gpm capacity limitation of the emergency intake line (EIL). In Section 2, the licensee stated the post-seismic event flow values, times, and UFSAR would be revised as necessary based on actual calculation results following approval of this new methodology. On May 10, 2005, the NRC issued a Safety Evaluation Report (SER) for Amendment Nos. 169 and 159. In Section 3.3.1 of the SER, the staff stated:

> "During a seismic event, it is expected that the intake canal will pump down, at which point CL system flow is limited to the capacity of the emergency intake line. The CL system flow rate during a seismic event (to be determined by the revised analysis) will determine when this limitation in flow capacity will occur (i.e., how long it takes for the intake canal to pump down) and when operators will be required to reduce CL system flow to within the capacity of the emergency intake line."

The inspectors noted that the calculation of record at the time the LAR was submitted was ENG-ME-310, "Evaluation of CL System Response following a Seismic Event with Reduced System Flow Rate," Revision 0, which was issued on March 17, 1997, prior to the LAR submittal. During this inspection the inspectors requested a copy of the new analysis or calculation performed in support of the SER and was informed that the analysis was not completed.

On September 13, 2006, the licensee initiated Action Request (AR) 01050047 to address an apparent discrepancy with Abnormal Operating Procedure AB-3, "Earthquakes"; calculation ENG-ME-310, Revision 0; and the plant's safe-shutdown list. As a result of lowering levels in the intake bay due to an earthquake (loss of the downstream dam), procedure AB-3 directed operators to reduce CL flow to within the capability of the EIL. The loss of the dam would result in lowering river and intake bay levels such that the EIL, which takes its suction from a low section in the river, would be the only source of water for the plants' ultimate heat sink (UHS). In AR 01050047, the licensee identified that AB-3 directed operators to reduce CL flow to a value within the capability of the EIL; however, this value was not supported by a calculation that determined the minimum required flow rate to safely shutdown both reactors. In addition, the licensee noted in the AR that the procedure directed operators to secure equipment that was credited for operation in the safe-shutdown list. The procedure also directed operators to secure all four containment fan coil units (FCUs), which was not consistent with the calculation that maintained two FCUs in operation. The licensee's initial review determined these issues did not affect the operability of the CL system. The preliminary analysis concluded that the required flow rate to support the shutdown of both units was within the 11,600 gallons per minute (gpm) capacity of the EIL. This analysis was never finalized, nor was procedure AB-3 updated to match the results of the analysis.

On February 7, 2007, the licensee initiated AR 01076107 as a result of a self-assessment prior to the NRC's component design basis inspection. This AR documented several concerns related to CL system calculation, including determining the impact on CL flow when a CL strainer was in backwash and the assumed river level at the start of an analyzed accident. The licensee performed a sensitivity analysis with the strainers in a backwash condition and determined that the flow rate required to support safety-related plant equipment remained below the EIL capacity limit.

On December 17, 2009, the inspectors identified that procedure AB-3 did not isolate the containment chiller system by closing valves MV-32144 and MV-32159. As a result, procedure AB-3 would not be able to reduce CL flow below the EIL capacity limit of 11,600 gpm and that other operator actions would be needed to reduce the CL flow to within the limitations of EIL. The licensee initiated AR 01211137, which concluded that other procedures that the operators would be performing during this time frame would close these valves or isolate their respective lines. These included procedures C35 AOP5, "Cooling Water Leakage Outside of Containment," and C35 AOP2, "Loss of Pumping Capacity or Supply Header Without SI," which would be entered based on flooding scenario from non-seismic CL piping or low CL system pressure. Although implementation of these additional procedures would have at some point during the event reduced CL flow to within the capacity of the EIL, it was not recognized that AB-3 would not have been able to be performed as written until these other actions were taken. The inspectors reviewed these scenarios and associated procedures and concluded that the procedures should have provided sufficient guidance to support AB-3 in reducing CL flow rate below the capacity of the EIL. On January 6, 2010, based on the inspectors' questions, the licensee found that the analysis only considered the case where one CL pump was operating after a DBE and failed to consider the more likely case of two CL pumps operating after a DBE. The licensee initiated AR 01212778 to document this omission. The licensee determined by an additional preliminary analysis that these issues did not affect the operability of the CL system.

At the end of the inspection period, the licensee had not yet approved the revision to calculation ENG-ME-310, nor revised procedure AB-3 to address the above deficiencies. Specifically, the licensee had not determined the minimum CL system flow rate required to safely shutdown both reactors after a DBE and therefore, could not verify this flow would be within the capacity of the EIL. In addition, the licensee needed to ensure that the results of the analysis did not conflict with the plant's licensing basis used by the NRC in its conclusions documented in the SERs associated with the CL system. Specifically, the NRC noted statements that the volume in the intake canal provided about a 4.8 hour supply of water or that the intake canal contained two to three times the minimum water volume for operator action needed to be reviewed. The licensee determined by preliminary analyses that these issues did not affect the operability of the CL system.

Analysis: The inspectors determined that the failure to ensure that the minimum CL flow being supplied by the EIL was sufficient to safely shutdown both reactors after a DBE and incorporate these results into plant procedures was a performance deficiency.

The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the availability, reliability and capability of safety-related equipment to respond to initiating events to prevent undesirable consequences. Specifically, the failure to determine the CL flow necessary to shutdown both reactors after a DBE could have provided incorrect guidance in the procedure. As a result, the operators may have reduced flow below the minimum required flow rates that could have caused the respective temperatures for the affected equipment to exceed their design or licensing limits.

The inspectors evaluated the finding using IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” SDP Phase 1 screening. The finding screened as very low safety-significance (Green) because the finding was a design deficiency confirmed not to result in loss of operability or functionality. Specifically, the licensee verified by preliminary analysis that the required CL flow necessary to shutdown both reactors after a DBE was within the capability of the EIL. As such, the required flow rates available to the affected equipment would be within their design and licensing limits. The inspectors performed an independent review of this analysis and had no further concerns.

This finding has a cross-cutting aspect in the area of problem identification and resolution, because the licensee did not take appropriate corrective actions to address safety issues in a timely manner, commensurate with their safety-significance and complexity. Specifically, the licensee initial evaluations were not adequate to fully address all of the concerns without further reviews by plant personnel and NRC inspectors. In addition, although the issue was first identified in 2006, the corrective actions to address this concern are untimely as the licensee still has not approved the required calculation, nor has the procedure been revised to reflect the calculations conclusions. [P.1(d)]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, as of February 26, 2010, the licensee failed to correctly translate applicable design basis into procedures. Specifically, calculation ENG-ME-310 was not revised to determine the CL flow rate necessary to shutdown both reactors after a DBE and that this flow rate remained within the capacity of the EIL. As a result, design bases were not correctly translated into procedure AB-3. Because this violation was of very low safety-significance and it was entered into the licensee’s corrective action program (ARs 01050047, 01076107, 01211137, and 01212778), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000282/2010006-01; 05000306/2010006-01).

4. OTHER ACTIVITIES

4OA6 Management Meetings

.1 Exit Meeting Summary

On March 26, 2010, the inspectors presented the inspection results to Mr. Kevin Ryan 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

K. Ryan, Plant Manager
J. Anderson, Regulatory Affairs Manager
S. Skoyen, Engineering Programs Manager
R. Murray, Plant Engineering Supervisor
S. Myers, Engineering Support Manager
J. Loeffler, Mechanical Design Engineer
C. Sansome, Mechanical Design Engineer (former GL 89-13 Program Owner)
J. Connors, Mechanical Design Engineering Supervisor, Fleet Design
K. Den Herder, GL 89-13 Program Owner
M. Davis, Regulatory Affairs Analyst
K. Mews, Licensing Engineer, Regulatory Affairs

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000282/2010008-01; 05000306/2010008-01	NCV	Failure to Determine the Minimum Cooling Water System Flow Required After a Design Basis Earthquake

Discussed

None

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.

CALCULATIONS

Number	Description or Title	Revision
ENG-ME-298	Intake Bay Water Capacity	0
ENG-ME-302	CL System Response to Seismic Event - Case 1.1	0
ENG-ME-310	Emergency Intake Line: Post Seismic Minimum Flow Requirement	0
ENG-ME-311	Evaluation of CL System Response Following a Seismic Event	0, 2
ENG-ME-347	Minimum Required Intake Bay Volume	1A
ENG-ME-355	Intake Canal Available Water Volume Comparison Analysis	1A
ENG-ME-479	Tube Plugging Criteria for Unit 1 EDG Heat Exchangers	1

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

Number	Description or Title	Date
01050047	Response to Seismic Event and AB-3	9/13/06
01074890	USAR Stated EIL Capacity 15,000 gpm Vice Actual 11,600 gpm	1/31/07
01076107	Low Level Issues With Cooling Water Calcs	2/7/07
01080474	Changes to USAR Section 10.4.1.2.2	3/5/07
01154003	Insp of Unit 2 Circ Bay had many Live Zebra Mussels	10/6/08
01164731	Snapshot Evaluation for Triennial Heat Sink Inspection	1/8/09
01172845	Extent-of-condition Review of D2 LO Hx	3/13/09
01184433	Prairie Island Nuclear Station Failed to Meet Some Commitments of GL 89-13	6/4/09
01185372	VTM XH-28-44-1 Contains 2 Lube Oil HX spec (d1)	6/12/09
01185566	ENG-ME-526 Post-LOCA Calc has Multiple Deficiencies	6/15/09
01185578	ENG-ME-479 U1 EDG HX Tube Plugging Limits Calculation has Multiple Deficiencies	6/15/09
01185644	WO 359034 Missing Signed Procedure	6/16/09
01185647	GL 89-13 Implementing Procedure Was Not Updated as Required	6/16/09
01185655	Pipe & HX Internal Inspection Forms Not Properly Reviewed Or Entered In Database	6/16/09
01185866	NRC Identified 2 Sheet Piling Tiebacks Were Bent on River Side of Intake Screenhouse	6/18/09
01185878	NRC Identified Step 7.7 of SP 1690 of WO 364811 Incorrectly Signed as Complete	6/17/09

CORRECTIVE ACTION PROGRAM DOCUMENTS REVIEWED

Number	Description or Title	Date
01185960	NRC Identified ECT Report Not in Records for EDG D2 LO Cooler in WO294380-11 iaw H53	6/18/09
01185963	NRC Identified Abnormal Operating Procedures C35 AOP1 & C35 AOP2 Appear to have Deficiencies	6/18/09
01186012	NRC Identified Unlabeled Wire Going Into Intake Canal	6/18/09
01186089	09UHS: NRC Identified SER for LAR 140 & 131 Incorrectly Stated Emergency Intake Line Capacity was 15,000 gpm	6/19/09
01186124	NRC identified Alarm Response Procedure C47041 AR26 had incorrect note	6/23/09
01186125	No Documentation for Predicted Post-DBE River Level Reduction in LA 128 & 120	6/23/09
01187199	NF-40315-1, Rev U had Incorrect Trip Setpoint For Non-safety CL Pumps	6/26/09
01188782	NRC Identified that Army Core of Engineers Lock & Dam 3 Report Not Reviewed	7/17/09
01195991	USAR 10.4.1.2.2 Needs to be Updated	9/1/09
01211137	CL System Response to a Seismic Event Actions	12/17/09
01212778	CL Response to Seismic Event	1/7/10

DRAWINGS

Number	Description or Title	Revision
NF-38600-1	Circulating Water System General Plan & Detail	76
NF-92718	Intake Screenhouse General Arrangement	K

MISCELLANEOUS DOCUMENTS

Number	Description or Title	Date or Revision
00-1089-1, 8945-X1	Unit 2 EDG Lube Oil Cooler Heat Exchanger Specification Sheet, 5/2/70	B
	Cooling Water System Health and Status Report	4/17/09
	NRC Letter to NMC: Prairie Island Nuclear Generating Plant, Units 1 And 2 - Issuance Of Amendments	5/10/05
	Response To Task Interface Agreement (TIA 2001-02) and Task Interface Agreement (TIA 2001-04) Regarding Evaluation Of Service Water System Design Basis Requirements At Prairie Island	8/29/02

PLANT PROCEDURES

Number	Description or Title	Revision
AB-3	Earthquakes	28
C35 AOP1	Loss of Pumping Capacity or Supply Header with SI	11
C35 AOP2	Loss of Pumping Capacity or Supply Header without SI	11
C35 AOP5	Cooling Water Leakage Outside Of Containment	6
C47041	Decreasing Level in the Intake Bay	0

WORK ORDERS

Number	Description or Title	Date
WR 47130	Electricians to Identify Wire Regional NRC Inspector Questioned Going Into Intake Canal	6/18/09
283254-02	U1 Circ Water Bay cleaning	3/4/08
283254-01	Unit 1 Circ Water Bay As-found Inspection	2/27/08
290177-01	Unit 2 Circ Water Bay As-Found Inspection on 10/5/08 per PINGP 1066	10/5/08
294380-01	D2 Lube Oil HX Inspection by PM# 7153-02	12/18/07
294380-01	D2 Lube Oil HX (DG 034-021) Internal Inspection	12/18/07
304022	P3108-2 CL Emergency Intake Line & Structure Inspection	4/20/04
362899-14	121 MDCLP Safeguards Bay Area As-Found & As-Left Inspections and Cleaning	3/25/09
362899-13	22 DDCLP Safeguards Bay Area As-Found & As-Left Inspections and Cleaning	4/3/09
362899-01	12 DDCLP Safeguards Bay Area As-Found & As-Left Inspections and Cleaning	4/4/09
0405815	D2 Lube Oil HX Inspection PM# 3001-2-D2	12/15/05
0405815	D2 Lube Oil HX (DG 034-021) Internal Inspection	12/15/05
9901268	Measure Flowrate Through the Emergency Intake Line	3/12/99

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AOP	Abnormal Operating Procedure
AR	Action Request
CFR	Code of Federal Regulations
CL	Cooling Water
DBE	Design Basis Earthquake
DDCLP	Diesel-Driven Cooling Water Pump
DRP	Division of Reactor Projects
FCU	Containment Fan Coil Unit
GL	Generic Letter
GPM	Gallons Per Minute
LAR	License Amendment Request
MC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PARS	Publicly Available Records
SDP	Significance Determination Process
SER	Safety Evaluation Report
UHS	Ultimate Heat Sink
USAR	Updated Safety Analysis Report
WO	Work Order

In addition, if you disagree with the characterization of the 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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Sincerely,
/RA/
Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure: Inspection Report 05000282/2010008; 05000306/2010008
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