October 19, 1999

EA 99-259

Mr. M. Wadley
President, Nuclear Generation
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

Dear Mr. Wadley:

SUBJECT: NRC INSPECTION REPORT 50-282/99014(DRS); 50-306/99014(DRS)

On September 22, 1999, the NRC completed the biennial heat sink performance inspection at your Prairie Island Nuclear Generating Plant. The results of this inspection were discussed on September 22, 1999, with Mr. T. Amundson and other members of your staff. The enclosed report presents the results of that inspection.

The inspection was an examination of activities under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, and interviews with personnel. Specifically, this inspection focused on the testing and maintenance of heat exchangers in high risk systems.

During this inspection, the NRC identified one issue of very low safety significance that has already been corrected and is discussed in the summary of findings and in the body of the attached inspection report. This issue was determined to involve a violation of NRC requirements, and because it potentially impacted the NRC's ability to oversee licensee activities, it could not be evaluated by the Significance Determination Process. This violation is being treated as a Non-Cited Violation consistent with Appendix C of the Enforcement Policy. If you contest this 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 Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Prairie Island Nuclear Generating Plant.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response to this letter, if you should choose to respond, will be placed in the NRC Public Document Room.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

/s/ J. M. Jacobson

John M. Jacobson, Chief
Mechanical Engineering Branch

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

Enclosure: Inspection Report 50-282/99014(DRS); 50-306/99014(DRS)

cc w/encl: Site General Manager, Prairie Island
Plant Manager, Prairie Island
S. Minn, Commissioner, Minnesota
Department of Public Service
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Dakota Community

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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:	50-282/99014(DRS); 50-306/99014(DRS)
Licensee:	Northern States Power Company
Facility:	Prairie Island Nuclear Generating Plant Units 1 & 2
Location:	1717 Wakonade Dr. East Welch, MN 55089
Dates:	September 20 - 22, 1999
Inspector:	James A. Gavula, Reactor Engineer
Approved by:	John M. Jacobson, Chief, Mechanical Engineering Branch Division of Reactor Safety

SUMMARY OF FINDINGS

Prairie Island Nuclear Generating Plant, Units 1 & 2
NRC Inspection Report 50-282/99014(DRS); 50-306/99014(DRS)

This report covers the pilot baseline inspection for the biennial review of heat sink performance. The heat sink performance inspection covers an inspectable area under the Initiating Events and Mitigating Systems cornerstones for which there is no performance indicator. Adequate or superior performance is not reported. Inspection findings were evaluated according to their potential significance for safety, using the NRC's Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent little effect on safety. WHITE findings indicate issues with some increased importance to safety, which may require additional NRC inspections. YELLOW findings are more serious issues with an even higher potential to affect safe performance and would require the NRC to take additional actions. RED findings represent an unacceptable loss of margin to safety and would result in the NRC taking significant actions that could include ordering the plant to shut down. Those findings that cannot be evaluated for a direct effect on safety with the Significance Determination Process, such as those findings that affect the NRC's ability to oversee licensees, are not assigned a color.

Cornerstone: Mitigating Systems

- Green: The inspector identified that a degraded heat exchanger tube on one of the emergency diesel generators had not been documented within the licensee's corrective action program. Although the licensee plugged the tube and corrected the specific deficiency, the lack of documentation within the corrective action program limited the licensee's ability to identify and trend a condition that had previously affected diesel generator design function.

- A non-cited violation of 10 CFR 50.59 was identified during closeout of an unresolved item from the 1997 System Operational Performance Inspection. It was determined that prior NRC approval should have been sought for the modification requiring manual actions to connect a nitrogen bottle on loss of instrument air. This issue had minimal impact on safety because corrective actions were taken when the issue was originally identified.

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R07 Heat Sink Performance

a. Inspection Scope

The inspector reviewed the surveillance procedure for the most recent thermal performance tests on the component cooling heat exchangers Nos. 11 and 12, the results of the periodic inspection and cleaning for the emergency diesel generator No. D2 heat exchangers, and the residual heat removal heat exchangers.

b. Observations and Findings

The inspector observed that the component cooling thermal performance calculation, performed under Work Order 9815409 for heat exchanger No. 12, used incorrect and nonconservative information. The calculation used 1491 active tubes instead of 1490, because the assumed number of plugged tubes was incorrect. This error had a very small impact on the calculated heat removal capability considering the existing fouling factors and did not challenge the approximate 25 percent tube plugging margin. However, the licensee considered the calculation to be part of the surveillance procedure and not a separate calculation, and therefore it was not checked and verified similar to other calculations. The licensee wrote General Action Item No. 19992760 to address this error.

The inspector observed that, for the residual heat removal heat exchangers, the licensee did not perform any testing, inspection, or maintenance to ensure proper heat transfer. The resolution of an industry-identified problem for baffle plate bypass leakage for these heat exchangers stated that internal inspection should not be done unless heat transfer degradation had been noted. The inspector questioned how heat transfer degradation could be noted if no specific activities were being performed in that regard.

4. OTHER ACTIVITIES

4OA1 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed several condition reports in the licensee's corrective action program that related to heat exchangers within the scope of the heat sink performance inspection to verify that the licensee adequately identified and resolved problems. In addition, the inspector reviewed Work Order 9713239, which contained the results of the last eddy current examination for the emergency diesel generator No. D2 heat exchangers.

b. Observations and Findings

The eddy current examination for emergency diesel generator No. D2 identified air cooler heat exchanger tube No. 16-2 as having greater than 90 percent wall loss, and recommended immediate plugging. The accompanying report noted that this tube had been identified for the same indication in 1989, with the same recommendation; however, the tube had not been plugged at that time. The inspector observed that the licensee had not documented this situation within their corrective action program, and that a past leak in a similar diesel generator heat exchanger, the lube oil cooler, had resulted in this diesel generator being unable to perform its design function. Although the licensee plugged the tube and corrected the specific deficiency, the lack of documentation within the corrective action program limited the licensee's ability to identify and trend a condition that had previously affected the diesel generator design function. Because the current heat exchanger tube degradation did not affect the diesel generator design function, this finding was determined to be of very low risk significance and was categorized as "Green." The licensee subsequently entered this issue into their corrective action program as Non-Conformance Report No. 19992904.

4OA4 Other

.1 (Closed) Violation 50-282/95014-01(DRS); 50-306/95014-01(DRS): Violation of Design Control Involving Waterhammer Analysis For Containment Fan Coolers. This issue was subsequently covered by Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," which is being resolved by the Office of Nuclear Reactor Regulation through TAC Nos. M96854 and M96855. This item is closed.

.2 (Closed) Unresolved Item 50-282/96008-10 (DRS): Operability Questions Regarding the Containment Fan Coolers. This issue was subsequently covered by Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," which is being resolved by the Office of Nuclear Reactor Regulation through TAC Nos. M96854 and M96855. This item is closed.

.3 (Closed) Violation 50-282/97290-01013(DRS); 50-306/97290-01013 (DRS): Violation of Test Control Involving Auxiliary Feedwater Acceptance Criteria. The licensee reviewed the inservice testing procedures for the safety-related pumps and verified that the acceptance criteria considered both the ASME Section XI criteria and the Updated Final Safety Analysis Report design requirements and specified the more limiting value. The procedures for the auxiliary feedwater pumps were revised appropriately. This item is closed.

.4 (Open) Violation 50-282/97290-01023(DRS); 50-306/97290-01023 (DRS): Violation of 50.71(e) Involving Failure to Update the Updated Final Safety Analysis Report Auxiliary Feedwater Accident Flow Rates. In their response to the Notice of Violation, dated November 14, 1997, the licensee stated that plans were to complete the remaining 14 items before the Updated Safety Analysis Report submittal of late 1998. This commitment was not met, and there are currently six items, where the activities necessary to allow the Updated Safety Analysis Report to be revised, have not been completed. Three of these items involve revisions to the high energy line break analysis, and the others require the submittal of a license amendment request for boron dilution. This is being tracked by licensee commitment tracking number 19980144.

.5 (Closed) Violation 50-282/97290-01033(DRS); 50-306/97290-01033 (DRS): Violation of Corrective Action Involving Failure to Review Auxiliary Feedwater Acceptance Criteria. The licensee took a number of corrective actions to improve its corrective action program, including establishment of a new corrective action system. All corrective actions associated with this violation are completed. The licensee's corrective action program was evaluated in August - September 1999, under the pilot inspection process for Problem Identification and Resolution. This inspection concluded that the licensee's corrective action process was acceptable. Therefore, this violation is closed.

.6 (Closed) Inspector Followup Item 50-282/97008-01(DRS); 50-306/97008-01(DRS): Review of Auxiliary Feedwater Flow Model. This licensee's flow model of the auxiliary feedwater system has been benchmarked and no adjustments to the results were made. This item is closed.

.7 (Closed) Unresolved Item 50-282/97008-09(DRS); 50-306/97008-09(DRS): Determination of Acceptability of Using Manual Action to Connect Nitrogen Bottle on Loss of Instrument Air. This item was forwarded to the Office of Nuclear Reactor Regulation where it was determined that the operator actions were sufficiently complex that errors of either commission or omission needed to be considered. Therefore, they determined that NRC approval should have been sought before this modification was implemented. The licensee took immediate corrective actions to remove the air bottle. The failure to recognize that prior NRC approval was required resulted in a violation of 10 CFR 50.59. This Severity Level IV violation is being treated as a **Non-Cited Violation (NCV 50-282/306-99014-01)** consistent with Appendix C of the NRC Enforcement Policy.

.8 (Closed) Unresolved Item 50-282/97008-10(DRS); 50-306/97008-10(DRS): Determination of Acceptability of Instrumentation Setpoint Uncertainties and of Administrative Control of Setpoints. This issue is under review by the Office of Nuclear Reactor Regulation as a generic concern, and when the review has been completed, the licensee will be informed of any necessary actions by separate correspondence. Therefore, this item is closed.

.9 (Closed) Violation 50-282/97008-13(DRS); 50-306/97008-13(DRS): Design Control Violation Involving Untimely Corrective Action on Cable Tray Separation Issue. The technical issues involved with the violation were reviewed and closed in inspection report 50-282/306-98005. The remaining corrective actions dealt with the corrective action program. As discussed above (Violation 282/306-97290-01033), the licensee took a number of steps to improve its corrective action program, including establishment of a new corrective action system. All corrective actions associated with this violation are completed. The licensee's corrective action program was evaluated in August - September 1999 under the pilot inspection process for Problem Identification and Resolution. This inspection concluded that the licensee's corrective action process was acceptable. Therefore, this violation is closed.

4OA5 Management Meetings

.1 Exit Meeting Summary

The inspector presented the inspection results to Mr. T. Amundson and other members of licensee management in an exit meeting on September 22, 1999. The licensee acknowledged the information and findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Amundson, General Superintendent Engineering
T. Breene, Superintendent, Nuclear Engineering
T. Downing, System Engineer
P. Hajovy, System Engineer
M. Heller, Superintendent, Mechanical Systems/Programs
S. Hiedeman, Superintendent, Mechanical Systems
T. Silverberg, General Superintendent, Operations, Acting Plant Manager

NRC

M. Kunowski, Project Engineer

INSPECTION PROCEDURE USED

IP 71111.07 (draft): Heat Sink Performance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

282/306/99014-01	NCV	Violation of 10 CFR 50.59 for Manual Actions During Loss of Instrument Air.

Closed

282/306/99014-01	NCV	Violation of 10 CFR 50.59 for Manual Actions During Loss of Instrument Air.
282/306/95014-01	VIO	Design Control for Containment Fan Cooler Waterhammer Analysis.
282/96008-10	URI	Operability of Containment Fan Coolers.
282/306/97290-01013	VIO	Violation of Test Control for Auxiliary Feedwater Acceptance Criteria.
282/306/97290-01033	VIO	Violation of Corrective Action Involving Failure to Review Auxiliary Feedwater Acceptance Criteria.
282/306/97008-01	IFI	Review of Auxiliary Feedwater Flow Model.
282/306/97008-09	URI	Determination of Acceptability of Using Manual Action to Connect Nitrogen Bottle on Loss of Instrument Air.
282/306/97008-10	URI	Determination of Acceptability of Instrumentation Setpoint Uncertainties and of Administrative Control of Setpoints.
282/306/97008-13	VIO	Design Control Violation Involving Untimely Corrective Action on Cable Tray Separation Issue.

Discussed

282/306/97290-01023	VIO	Violation of 10 CFR 50.71(e) Involving Failure to Update the Final Safety Analysis Report Auxiliary Feedwater Accident Flow Rates.

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
IFI	Inspection Followup Item
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
URI	Unresolved Item
VIO	Violation

LIST OF DOCUMENTS REVIEWED

Miscellaneous

Work Request 9815409, SP1304 U1 CC Hx Performance Test.
Work Request 9809590, SP 2304 U2 CC Hx Performance Test.
Work Request 9713239, D2 DG 18 Month Insp, 4/27/98, PINGP 1066 Form.
Work Request 9604984, D2 DG 18 Month Insp, 9/22/96, No PINGP 1066 Form Available.
Work Request 9407735, D2 DG 18 Month Insp, 3/14/95, PINGP 1066 Form.
Operating Experience Assessments: Westinghouse IG94002, “CCW HX Fouling and Containment Response.”
Generic Letter 89-13 Implementing Program, Section H.
NSP Materials and Special Processes Report, 5/1/98, Eddy Current Examination of D2 Air, Oil, and Jacket Cooler Heat Exchangers.

Condition Report Forms

Condition Report 19970880, Potentially Undersized EDG Oil Coolers.
Condition Report 19980144, IR 97008 Action 12.
Condition Report 19981123, Unit 2 AFWP Surveillance Procedures.
Condition Report 19983244, 22CC Hx Divider Plate Bending.

Procedures

Surveillance Procedure 1304 U1 Component Cooling Heat Exchanger Performance Test.
PM 3001-2-D2, Diesel Generator 18 Month Inspection, Rev. 11.