**UNITED STATES**
**NUCLEAR REGULATORY COMMISSION**
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352
November 20, 2009

EA-09-152

Mr. Mark Schimmel
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company, Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: REVISED NON-CITED VIOLATION, PRAIRIE ISLAND STATION, UNITS 1 AND 2, INSPECTION REPORT 05000282/2009002; 05000306/2009002 (DRP)

Dear Mr. Schimmel:

By letter dated June 12, 2009, you contested Non-Cited Violation (NCV) 05000282/2009002-04, as documented in NRC Integrated Inspection Report 05000282/2009002; 05000306/2009002, dated May 14, 2009, which involved your staff's failure to maintain the Unit 1 reactor power level below the thermal power limit stated in the facility operating license. On July 1, 2009, the NRC acknowledged your letter and advised you that we would evaluate the information included in your June 12, 2009, letter and would inform you of the results of our evaluations. We have completed our review and have determined that a violation of NRC requirements occurred as outlined below.

In your June 12, 2009 letter, you indicated that:

- Only the 1 minute average power level peaks, resulting from the power oscillations, exceeded the nominal 100 percent reactor power level;
- The power level oscillations were due to automatic operation of the feedwater control system;
- The power level oscillations were not under direct control of a licensed reactor operator;
- The Nuclear Energy Institute (NEI) position statement, "Guidance to Licensees on Complying with the Licensed Power Limit," indicated that these oscillations were not considered intentional, and therefore, the average power, as measured by any means, including the shortest reasonable period (the period of one oscillation), never exceeded 100 percent reactor power; and,
- The maximum thermal power licensed limit was not exceeded because the short duration power level peaks were fluctuations inherent in the design of the controlling system and the average thermal power level was below the maximum thermal power licensed limit.

Because the bases for your contesting the NCV depends, in part, upon the guidance provided in the NEI Position Statement, the staff reviewed the NEI Position Statement and the NRC's safety evaluation and endorsement of the position statement, as documented in NRC Regulatory Information Summary (RIS) 2007-21, Revision 1, "Adherence to Licensed Power Limits," dated February 9, 2009. In reviewing the NEI Position Statement and the NRC RIS, the staff confirmed that:

- No actions were allowed that would intentionally raise reactor core thermal power above the licensed power limit for any period of time;

- Reactor power level changes were not considered intentional if they were small, short-term fluctuations in power that were not under the direct control of a licensed reactor operator (e.g., secondary-side control valve oscillations for pressurized water reactors);

- For preplanned evolutions expected to cause reactor power to increase, prudent action, based on prior performance, should be taken to reduce power prior to performing the evolution; and,

- The maximum thermal power licensed limit is not considered to be exceeded when the short duration peaks are a result of normal fluctuations inherent in the design of the controlling system as long as the average thermal power level is at or below the maximum thermal power licensed limit.

Given the above, the information you provided in your June 2009 letter and the information included in the May 2009 inspection report, the NRC determined that:

- In the weeks preceding the auxiliary feedwater pump test, the reactor operators administratively maintained Unit 1 steady state reactor power lower than rated thermal power. This action was necessary to account for abnormal oscillations in reactor power caused by a feedwater regulating valve material condition issue. These oscillations were not "inherent in the design of the controlling system," but rather were a result of a material condition that caused a degradation of the feedwater regulating valve.
- Prior to conducting the 11 turbine driven auxiliary feedwater pump test, the reactor operators noted that from past experience Unit 1 reactor power would increase during the test; however, the reactor operators took no action to lower the reactor power level prior to conducting the test.
- Procedure SWI O-50 provided conflicting and non-conservative direction to the reactor operators. Specifically, the licensee's reactivity management procedure, SWI O-50, stated that during transient conditions, such as secondary plant changes, reactor power shall not be allowed to knowingly exceed 100 percent. However, it did not direct operators to lower initial power prior to performing a test that could result in a power increase above the licensed limit. This resulted in the one-minute average exceeding 100.0 percent nine times for which the operators took no action.

Conclusion:

Based upon the above, the NRC determined that the reactor operators, following an inadequate reactivity control procedure, failed to take appropriate actions to ensure that reactor power would remain below the licensed power level limit. Prior to and during the 11 turbine driven auxiliary feedwater pump test, the reactor operators did not adjust reactor power to compensate for the expected power increase resulting from the test. Both the NEI Position Statement and the associated NRC RIS 2007-21, Revision 1, indicated that for preplanned evolutions expected to cause reactor power to increase, prudent action, based on prior performance, should be taken to reduce power prior to performing the evolution.

While the NRC recognizes that, in this case, operation of the reactor at the slightly increased power levels was of very low safety significance, since the maximum reactor power peak (using a 1 minute average) was about 100.1 percent, the NRC is concerned with the absence of adequate procedural guidance for conducting reactivity changes and with the apparent non-conservative decision-making demonstrated during this evolution.

Consistent with the above evaluation and determination, the previous NCV has been re-drafted in the enclosure to this letter to focus your attention and corrective actions so as to ensure that adequate procedural guidance is developed and implemented to properly control activities affecting reactivity management consistent with NRC RIS 2007-21.

In accordance with 10 CFR 2.390 of the NRC's “Rules of Practice,” a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html

Sincerely,

**/RA/**

Cynthia D. Pederson
Deputy Regional Administrator

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

Enclosure: Revised Sections of NRC Integrated Inspection Report 05000282/2009002; 0500306/2009002

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This is the revised Non-Cited Violation (NCV) 05000282/2009002-04 from NRC Integrated Inspection Report 05000282/2009002; 05000306/2009002. Replace finding number four with the finding below:

## SUMMARY OF FINDINGS

### A. NRC-Identified and Self-Revealed Findings

**Cornerstone: Barrier Integrity**

Green. A self-revealed finding and an NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" was identified due to the failure to have an adequate procedure for reactivity management. Specifically the guidance in SWI O-50, "Reactivity Management" was not adequate to ensure operators took prudent actions so as to maintain Unit 1 reactor power below the licensed power limit. Corrective actions for this issue included revising all associated operating procedures to ensure that operations personnel take action to lower reactor power if plant activities were expected to result in increases in power levels that exceed the licensed thermal power limitations.

The inspectors determined that this issue was more than minor because if left uncorrected the operation of the reactor beyond the limits specified in the operating license could become a more significant safety concern. The inspectors determined that this issue was of very low safety significance because the finding was only associated with the fuel aspect of the Barrier Integrity Cornerstone and no reactor safety limits were violated. The inspectors determined that this finding had a cross-cutting aspect in the Human Performance, Decision-Making area because the licensee failed to use conservative assumptions in making decisions concerning power level controls (H.1(a)).

## REPORT DETAILS

## 1. REACTOR SAFETY

**Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity**

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and Technical Specifications (TS) requirements:

- Unit 1 Turbine-Driven Auxiliary Feedwater Pump Monthly Test (Routine).

b. Findings

Introduction: A green self-revealed finding and a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" was identified due to the failure to have an adequate procedure for reactivity management. Specifically, the guidance in SWI O-50, "Reactivity Management" was not adequate to ensure operators took prudent actions, for preplanned evolutions expected to cause reactor power to increase or based on prior performance, to reduce reactor power prior to performing and during the evolution, so as to maintain Unit 1 reactor power below the licensed power limit.

Description: On January 2, 2009, operations personnel tested the 11 turbine-driven auxiliary feedwater (TDAFW) pump using SP 1102, "11 TDAFW Pump Monthly Test." For several weeks on Unit 1, the licensee had been maintaining reactor power at 99.7 percent due to material problems with a feedwater regulating valve. The material condition issue resulted in oscillations in reactor power, which were two to three times in magnitude to Unit 2's oscillations. During the pre-test brief, the operators discussed that reactor power would increase. However, the control room operators did not reduce reactor power, even though the testing was expected to result in some increase to reactor power levels. While performing this test, the control room received an alarm and identified that Unit 1 reactor thermal power had momentarily spiked above 100 percent based on the 1 minute average. Step 4 of Annunciator Response Procedure (ARP) 47013-0303 stated that the control room operators were only required to take action to reduce reactor thermal power if the last 5 minute reactor thermal power average exceeded 100 percent. Control room personnel checked the latest 5 minute average and determined that the average was not greater than 100 percent. As a result, no actions were taken to reduce Unit 1 reactor power.

Based on the 1 minute average, the Unit 1 reactor thermal power continued to momentarily spike above 100 percent approximately eight additional times during the TDAFW test, which was conducted over a 1 hour period. Operations personnel documented this condition in Corrective Action Program (CAP) 1164293. The inspectors reviewed the CAP and learned that the prior performances of SP 1102 were conducted with the main turbine operating in the valve position control mode. This mode of turbine operation allowed the position of the turbine control valves to remain relatively unchanged even though a portion of the steam flowing to the turbine was diverted to operate the 11 TDAFW pump. On January 2, 2009, operations personnel performed SP 1102 with the main turbine operating in first stage pressure mode. This mode of turbine operation allowed the control valves to move to maintain turbine first stage pressure constant while diverting steam to the 11 TDAFW pump. This mode of turbine operation, in conjunction with the feedwater regulating valve material condition issue, resulted in an increase in reactor thermal power. The highest reactor power level achieved was 100.1 percent based on the 1 minute average.

The inspectors reviewed ARP 47013-0303; Operating Procedure 1C1.4, "Unit 1 Power Operation;" Section Work Instruction (SWI) O-50, "Reactivity Management;" NRC Regulatory Issue Summary (RIS) 2007-21, "Adherence of Licensed Power Limits;" and RIS 2007-21, Revision 1. The inspectors reviewed the NEI "Position Statement on the Licensed Power Limit" dated June 23, 2008. Step 4.2.1 of the Position Statement reads as follows:

> "No actions are allowed that would intentionally raise core thermal power above the licensed power limit for any period of time. Small, short-term fluctuations in power that are not under the direct control of a licensed operator are not considered intentional."

In addition, Section 4.4 of the NEI Position Statement documented that the following actions constituted performance deficiencies:

- Intentional raising of reactor power above the licensed power limit, and
- Failure to take prudent action prior to a pre-planned evolution that could cause a power increase to exceed the licensed power level.

Based upon discussions with licensee personnel, a review of plant data and procedures, and the information provided above, the inspectors determined the performance of SP 1102 was an activity that was under the direct control of the licensed operators. In addition, the licensee failed to take prudent action to lower reactor power prior to performing SP 1102 even though there was a potential that the performance of this test could cause reactor power to exceed the licensed reactor power level.

Analysis: The inspectors determined that the failure to have an adequate procedure to reduce reactor power to properly manage anticipated changes in reactivity due to ongoing activities and changing plant conditions resulted in exceeding the licensed power limit and was a performance deficiency that required an evaluation using the NRC's Significance Determination Process (SDP) contained in Inspection Manual

Chapter 0609. The inspectors determined that this issue was more than minor because if left uncorrected the operation of the reactor beyond the limits specified in the operating license could become a more significant safety concern. The finding affected the Barrier Integrity Cornerstone for the fuel barrier. The inspectors determined that this issue was of very low safety significance (Green), because it only impacted the fuel aspect of the Barrier Integrity Cornerstone and no reactor safety limits were violated. The inspectors determined that this finding had a cross-cutting aspect in the Human Performance, Decision-Making area because the licensee failed to use conservative assumptions in making the decision not to adjust power prior to commencing the testing (H.1(a)). Specifically, the operators did not make conservative decisions when faced with a task that was expected to cause reactor power to rise.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed and accomplished by procedures appropriate for the circumstances. The licensee implemented the reactivity management procedure (an activity affecting quality) using Procedure SWI O-50, "Reactivity Management." SWI O-50, Revision 9 prescribed actions to ensure the licensed reactor power limit was not exceeded. Contrary to the above, on January 2, 2009, the licensee failed to have a procedure which was appropriate for the circumstances, in that, Procedure SWI O-50 contained guidance to reduce reactor power that was conflicting and not appropriate to properly manage anticipated changes in reactivity due to ongoing activities and changing plant conditions. However, because this violation is of very low safety significance and was entered into your corrective action program as CAP 1164293, it was treated as an NCV consistent with Section VI.A.1 of the Enforcement Policy **(NCV 05000282/2009002-04)**. Corrective actions for this issue included issuing operations guidance to ensure that actions were taken to lower reactor power if reactor power levels were expected to exceed the limit specified in the operating license, revising SWI O-50 to reflect that reactor power should be lowered prior to performing tests that could cause unacceptable increases in reactor power, and revising SP 1102 to provide guidance regarding potential impacts on reactor power during the performance of this test.

Conclusion:

Based upon the above, the NRC determined that the reactor operators, following an inadequate reactivity control procedure, failed to take appropriate actions to ensure that reactor power would remain below the licensed power level limit. Prior to and during the 11 turbine driven auxiliary feedwater pump test, the reactor operators did not adjust reactor power to compensate for the expected power increase resulting from the test. Both the NEI Position Statement and the associated NRC RIS 2007-21, Revision 1, indicated that for preplanned evolutions expected to cause reactor power to increase, prudent action, based on prior performance, should be taken to reduce power prior to performing the evolution.

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Sincerely,
**/RA/**
Cynthia D. Pederson
Deputy Regional Administrator

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

Enclosure: Revised Sections of NRC Integrated Inspection Report 05000282/2009002; 0500306/2009002
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Letter to M. Schimmel from C. Pederson dated November 20, 2009

SUBJECT: REVISED NON-CITED VIOLATION, PRAIRIE ISLAND STATION, UNITS 1 AND 2, INSPECTION REPORT 05000282/2009002; 05000306/2009002 (DRP)

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