UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

MAR 17 1992

Docket No. 50-306

Northern States Power Company
ATTN: Mr. L. R. Eliason
Vice President, Nuclear
Generation
414 Nicollet Mall
Minneapolis, MN 55401

Dear Mr. Eliason:

SUBJECT: NRC INSPECTION REPORT 50-306/92005

This refers to the special inspection conducted by the Nuclear Regulatory Commission Augmented Inspection Team (AIT) at your Prairie Island Nuclear Generating Plant during the period from February 21 through 25, 1992, concerning an interruption in decay heat removal during reduced inventory operations at Unit 2 which occurred on February 20, 1992. At the conclusion of the inspection, the findings were summarized at a public meeting attended by those members of your staff identified in the enclosed inspection report.

The enclosed copy of the AIT report identifies the areas examined during this inspection. Within these areas, the inspection consisted of selective examinations of plant hardware, procedures and other records, interviews with personnel, and observation of activities in progress.

The AIT concluded that management had made a number of changes in the process for establishing stable reduced inventory conditions in the reactor cooling system. Although intended as improvements, these changes were not all adequately evaluated, either individually or in the aggregate. As a consequence, a combination of factors, including inadequate supervision, level instrument design limitations, reduced engineering support, procedure ambiguities, and inadequate training led to a condition where the personnel who were draining water from the system believed they knew the current water level when, in fact, they did not. By proceeding despite questions about instrument and system behavior, operators did not exhibit an aggressive, questioning safety attitude. Water level went below that necessary for continued operation of the in-service cooling pump, making it necessary to shut off the pump and interrupt operation of the residual heat removal system.

A review of the inspection findings is continuing to determine whether the described activities violated NRC requirements. You will be advised by separate correspondence of the results of our review of this matter.

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MAR 17 1992

In accordance with Section 2.790 of the NRC's "Rules of Practice," a copy of this letter and the enclosure will be placed in the NRC Public Document Room.

Should you have any questions concerning this letter, please contact us.

Carl J Paperiello for
A. Bert Davis,
Regional Administrator

Enclosure:
NRC Inspection Report
50-306/92005

cc w/enclosure:
E. L. Watzl, Site Manager,
Prairie Island Site
M. Sellman, Plant Manager
DCD/DCB (RIDS)
OC/LFDCB
Resident Inspector, RIII Monticello
John W. Ferman, Ph.D.,
Nuclear Engineer, MPCA
State Liaison Officer, State
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Prairie Island, LPM, NRR
Robert M. Thompson, Administrator
Wisconsin Division of Emergency
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G. Holahan, NRR
W. D. Lanning, NRR
J. Zwolinski, NRR
E. Jordan, AEOD
G. Grant, EDO

MAR 1 7 1992

DCD (DCB)

Northern States
Power Company

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Original signed by
Carl J. Paperiello
A. Bert Davis,
Regional Administrator

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3/17/92	3/17/92				

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AUGMENTED INSPECTION TEAM REPORT

U.S. NUCLEAR REGULATORY COMMISSION

PRAIRIE ISLAND UNIT 2 LOSS OF DECAY HEAT REMOVAL

MARCH 17, 1992

INSPECTION REPORT NO. 50-306/92005

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EXECUTIVE SUMMARY

An NRC Augmented Inspection Team conducted an evaluation at the Prairie Island Plant site during the period from February 21 through 25, 1992, to review an event which occurred on February 20, 1992, involving interruption of operation of the residual heat removal system during reduced inventory operations at Unit 2. The Team reviewed records and interviewed personnel to aid in its evaluation.

The Team concluded that the proximate cause of the residual heat removal system interruption was the overdraining of the reactor coolant system while attempting to establish stable mid-loop operation conditions with level within the hot leg. This made it necessary to shut off the in-service residual heat removal pump and interrupt the heat removal process. The following were factors which directly contributed to the cause of the event:

1. The design of the electronic level measurement instruments was incompatible with the nitrogen pressure specified in the draindown procedure. The instruments were essentially unavailable during the entire draining process.

2. The draindown procedure did not adequately describe the required processes to achieve a reduced inventory condition.

3. The training and experience of the operators and support engineering were insufficient to perform the assigned tasks.

4. The operators and senior operators did not exhibit a questioning attitude with regards to safety. With two out of three channels of instrumentation inoperable and concerns over the behavior of the plant, the operators continued draining the reactor coolant system.

5. Management attention was inadequate in the areas of training, human factors, procedure and design reviews, and operator supervision.

The consequences of the event were minimal. The magnitude of the reactor coolant system heatup was a direct result of the early entry into mid-loop 2 days after the reactor was shut down. The safety significance of this event was also minimal given the large number of inventory addition capabilities and the availability of redundant heat removal mechanisms. There was no increase in radiation levels within the plant, no increase in coolant activity, and no release of radiation to the environment. No equipment damage resulted from the event.

The operators were effective in responding to this event using the appropriate procedures. No equipment malfunctions or procedural inadequacies were observed.

The Team questioned the Event Classification and timeliness of reporting to NRC. Subsequently, review of these issues was referred to the Region III Emergency Preparedness Section, who determined that the licensee's classification of the event at the Unusual Event level was in compliance with applicable requirements.

Data to support an independent quantitative risk assessment, using the Accident Sequence Precursor program, was collected and forwarded to Oak Ridge National Laboratory for their evaluation. Results of the risk assessment will be published separately.