Reactor Facilities Branch

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

DEC 8 1975

Northern States Power Company
ATTN: Mr. Leo Wachter, Vice President
Power Production and System
Operation
414 Nicollet Mall
Minneapolis, Minnesota 55401

Docket No. 50-263

Gentlemen:

This refers to the inspection conducted by Mr. Choules of this office on November 14, 19-21, 1975, of activities at Monticello Nuclear Generating Plant authorized by NRC Operating License No. DPR-22 and to the discussion of our findings with Mr. Larson and others of your staff at the conclusion of the inspection.

A copy of our report of this inspection is enclosed and identifies the areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, interviews with plant personnel, and observations by the inspector.

During this inspection, it was found that certain of your activities appear to be in noncompliance with NRC requirements. The item and reference to the pertinent requirements are listed under Enforcement Action in the Summary of Findings section of the enclosed inspection report.

This notice is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Section 2.201 requires you to submit to this office within twenty days of your receipt of this notice, a written statement or explanation in reply, including: (1) corrective steps which have been taken by you, and the results achieved; (2) corrective steps which will be taken to avoid further items of noncompliance; and (3) the date when full compliance will be achieved.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room. If this report contains any information that you or your contractors believe to be proprietary, it is necessary that you make a written application to this office, within twenty days of your receipt of this letter, to withhold such information from public disclosure. Any such application must include a full statement of the reasons for which it is claimed that the information is proprietary, and should be prepared so the proprietary information identified in the application is contained in a separate part of the document. Unless we receive an application to withhold information or are otherwise contacted within the specified time period, the written material identified in this paragraph will be placed in the Public Document Room.

Should you have any questions concerning this inspection, we will be glad to discuss them with you.

Sincerely yours,

Gaston Fiorelli, Chief
Reactor Operations and
Nuclear Support Branch

Enclosure:
IE Inspection Rpt
No. 050-263/75-18

bcc w/encl:
PDR
Local PDR
NSIC
TIC
Anthony Roisman, Esq., Attorney

U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report of Operations Inspection

IE Inspection Report No. 050-263/75-18

Licensee: Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Monticello Nuclear Generating Plant
Monticello, Minnesota

License No. DPR-22
Category: C

Type of Licensee: BWR GE 1670 MWt

Type of Inspection: Routine, Announced

Dates of Inspection: November 14, 19-21, 1975

Principal Inspector: N. C. Choules — 12/5/75 (Date)

Accompanying Inspectors: None

Other Accompanying Personnel: None

Reviewed By: E. L. Jordan, Section Leader
Reactor Projects No. 2 — 12/5/75 (Date)

## SUMMARY OF FINDINGS

### Inspection Summary

Inspection on November 14, and 19-21, (75-18): Semi-Annual reports, plant operations, and nonrouting reports were reviewed. One item of noncompliance related to approval of a corrective maintenance procedure was identified.

### Enforcement Action

### Infraction

Contrary to Technical Specification 6.5, a corrective maintenance procedure, NPT-US-1, Nuclear Penetrant Testing Procedure, was used without Operations Committee review and approval by a member of plant management designated by the plant manager. (Report Details, Paragraph 2.f)

### Licensee Action on Previously Identified Enforcement Items

Not Applicable.

### Other Significant Items

A. Systems and Components

The licensee has completed the removal of cracks in the feedwater nozzles and replacement of the feedwater spargers.[1/]

B. Facility Items

The plant returned to power operation on November 19, 1975, after being shutdown for refueling and feedwater sparger replacement.

C. Managerial Items

None.

D. Noncompliance Identified and Corrected by Licensee

None.

E. Deviations

None.

1/ IE Inspection Report No. 050-263/75-16.

F. Status of Previously Reported Unresolved Items

Not Applicable.

Management Interview

The following persons were present at the management interview conducted at the close of the inspection.

Northern States Power Company (NSP)

C. E. Larson, Plant Manager
M. H. Clarity, Superintendent, Plant Engineering and Radiation Protection

A. Non-Routine Occurrences

1. AO 75-16

The inspector stated he was concerned with the failure of the relay coils which caused this occurrence and other occurrences. The licensee stated they were investigating circuit modifications of the under-voltage relay system controlled by this type coil that would prevent the controlled component from becoming inoperable when a coil failed. (Report Details, Paragraph 2)

2. AO 75-14, 75-15, 75-18, 75-19, 75-22

The inspector stated he had reviewed these occurrences and the licensee's corrective action appeared to be appropriate. (Report Details, Paragraph 2)

3. AO 75-20

The inspector stated he had reviewed the evaluations made of the feedwater nozzle cracks repairs and they appeared to be adequate. (Report Details, Paragraph 2)

4. AO 75-21

The inspector stated that the licensee's corrective action for this occurrence appeared to be appropriate. The inspector further stated that review of the maintenance package identified one noncompliance item in that the dye penetrant procedure used on this job was not reviewed and approved as required by the Technical Specifications. The licensee acknowledged the inspectors statement. (Report Details, Paragraph 2)

B. Semi-Annual Reports

The inspector stated he had reviewed the licensee's semi-annual report for the past year against the Technical Specification requirements and no discrepancies were noted. (Report Details, Paragraph 3)

C. Plant Operations Following Refueling Outage

The inspector stated he had reviewed licensee's actions related to returning systems disturbed during the outage to service, core reactivity testing, and surveillance testing performed during the outage and no discrepancies were noted. (Report Details, Paragraph 4)

D. Plant Startup

The inspector stated he had reviewed control room operations during portions of the plant startup which was in progress, and no discrepancies were noted. (Report Details, Paragraph 5)

E. Lost Parts in the Reactor Vessel

In a telephone conversation with the licensee on November 24 and 26, 1975, the inspector discussed the analysis of parts which had been lost in the reactor vessel during the current outage. The specific item for which a documented analysis was not available at the time of the inspection was a 1 1/2" x 3/16" x 6" aluminum gauge block. The licensee stated that they had reviewed the potential problems and had concluded that the block could not cover a fuel element orifice because of its length and would not plug the orifice if it entered the orifice width wise because of the small cross sectional area. The licensee further stated that they were quite certain that the gauge block fell on the jet pump support shelf and would probably stay in this location during operation. The inspector stated that this information should be documented in the licensee's files and an orifice flow blockage analysis should be performed for the gauge block. The licensee stated these would be done. (Report Details, Paragraph 5)

## REPORT DETAILS

1. Persons Contacted

Northern States Power Company

C. E. Larson, Plant Manager
M. H. Clarity, Superintendent, Plant Engineering and Radiation Protection
W. E. Anderson, Superintendent, Operation and Maintenance
D. D. Antony, Plant Engineer Operations
W. A. Shamla, Plant Engineer Technical
R. R. Rodger, Lead Plant Equipment Operational Reactor Operator
J. F. Heneage, Engineer
B. D. Day, Engineer
H. M. Kendall, Plant Office Supervisor
D. E. Nevinski, Engineer, Nuclear
M. F. Hammer, Engineer
P. A. Pochop, Quality Engineer
O. N. Iverson, Engineer
T. W. Grue, Engineer
R. A. Coranson, Engineer

Gherne Contracting Corporation

F. Mutter - Project Manager
R. Wagner, Project Superintendent

2. Non-Routine Reports

AO 75-14

The licensee informed the inspector by telephone on August 19, 1975, that a threaded nipple in a section of ½ inch seal water line between the No. 12 Reactor Water Cleanup Pump and the seal water heat-exchanger failed at the thread root. The details and corrective action for this occurrence are given in the licensee's report.[2/]

The licensee indicated in his report that they were investigating the possibility of replacing the pipe with tubing. At the time of this inspection, the piping on the No. 12 pump had been replaced with tubing and the licensee intends to replace the piping on the No. 11 pump with tubing wherever maintenance is performed requiring disassembly of the piping.

3. Non-Routine Occurrences

a. AO 75-15

2/ AO Rpt No. 050-263/75-14, NSP to DL, dtd 8/28/75.

The licensee informed the inspector by telephone on August 26, 1975, that a RHR torus cooling injection valve failed to close completely due to improper torque switch setting. The details and corrective action for this occurrence are given in the licensee's report.3/

The inspector reviewed this occurrence with the licensee's representative and determined that the torque switch was set at no flow conditions and did not take into account added forces due to flow. Setting the torque switch for flow condition corrected the closing problem.

b. AO 75-16

The licensee informed the inspector by telephone on August 27, 1975, that the HPCI Turbine Steam Supply Valve, NIO-2036, failed to operate due to the failure of the undervoltage relay coil. The details and corrective action for this occurrence are given in the licensee's report.4/

Review of this occurrence with licensee representatives, indicated that this same model relay coil had failed three other times during the lifetime of the plant. One failure was reported5/ and the other two failures were on nonsafety related equipment. The licensee representatives indicated that their electrical laboratory had inspected the failed coil and concluded that there was nothing abnormal about the failure and failures of this type could be expected periodically.

c. AO 75-18

The licensee informed the inspector by telephone on September 15, 1975, that a hydraulic snubber located inside the drywell on B loop LPCI line was low on oil. The details and corrective action for this occurrence are given in the licensee's report.6/

In review of this occurrence with the licensee's representative, it was indicated that the low oil resulted from an improperly installed seal as stated in the licensee's report.

d. AO 75-19

The licensee informed the inspector by telephone on September 22, 1975, that the count rate of Source Range Monitor 23 decreased below 3 CPS, the count rate required to consider the detector

3/ AO Rpt No. 050-263/75-15, NSP to DL, dtd 9/3/75.
4/ AO Rpt No. 050-263/75-16, NSP to DL, dtd 9/8/75.
5/ AO Rpt No. 050-263/75-08, NSP to DL, dtd 5/15/75.
6/ AO Rpt No. 050-263/75-18, NSP to DL, dtd 9/24/75.

operable by the Technical Specifications. The details and corrective action for this occurrence are given in the licensee's report.7/

The inspector reviewed the licensee's actions related to this occurrence. The decrease below 3 CPS occurred during or after removal of a fuel element from the core. No fuel was added to the core with the count rate less then 3 CPS. When the low count rate was discovered, the refueling crew contacted the Nuclear Engineer for directions and a change in the loading sequence was prepared and approved for use to bring and maintain the count rate above 3 CPS. The licensee's actions for this occurrence appear to be appropriate.

e. AO 75-20

The licensee informed the inspector by telephone on October 14, 1975, that cracks had been observed in the four feedwater nozzles. Details and corrective action for this occurrence are given in the licensee report8/ and a report which will be issued at a later date. The licensee's actions related to removal of these cracks were examined during a previous inspection.9/

The inspector reviewed documentation and evaluation of these craok repairs as follows:

(1) Monticello Feedwater Nozzle Cladding Crack Repair Report, NEDC-21120, prepared by the General Electric Company.

This report includes a stress evaluation, fatigue analysis, fracture mechanics evaluation, grind out weld repair evaluation, corrosion evaluation and a safety evaluation. The report concludes that the modified nozzle geometry still meets or exceeds the ASME Section III Criteria that was originally established for the vessel and that the repaired condition of the nozzles exceeds the requirements of ASME Section XI.

(2) Evaluation of the above report by NUTECH, Report No. NSP-01-004, and letter to NSP dated November 11, 1975.

In the evaluation and letter, NUTECH basically agreed with the conclusion of the General Electric Report.

(3) Statement of Review of the General Electric and NUTECH reports signed by the NSP Quality Control Engineer, and the Hartford Steam Boiler Inspection and Insurance Company's Authorized Inspection, in which they state to the best of

7/ AO Rpt No. 050-263/75-19, NSP to DL, dtd 9/30/75.
8/ AO Rpt No. 050-263/75-20, NSP to DL, dtd 10/22/75.
9/ IE Inspection Report No. 050-263/75-16.

their knowledge "corrective measures taken in the Monticello Feedwater Nozzle Repair Program conform to the rules of the ASME Code, Section XI."

f. AO 75-21

The licensee informed the inspector on October 14, 1975, that the No. 11 Emergency Diesel Generator failed to start due to a loose fitting on the No. 2 start system. The details and corrective action for this occurrence are given in the licensee's report.10/

Review of this occurrence indicated the diesel started on September 26, 1975, when started for Surveillance Test 1052. Maintenance was performed on August 6, 1975, and operability checks performed on August 7, 1975. The licensee concluded that the fitting was installed tight enough to hold air originally but was loose enough that vibration caused it to loosen further.

g. AO 75-22

The licensee informed the inspector by telephone on November 3, 1975, that a cracked weld had been discovered in the 1-inch pipe nipple to RV 2005, B Loop LPCI relief valve. The details and corrective actions for this occurrence are given in the licensee's report.11/

Review of this occurrence with the licensee's representative showed that they had rerouted the discharge piping from this relief valve and the relief valve for the redundant system to reduce the stress on the nipple at the relief valves.

Design change 75-MO-90 specified the changes to be made and the work was performed under a Work Request Authorization. Review of the work package indicated that dye penetrant checks had been performed on welds for these valves. The inspector inquired what procedure was used for the dye penetrant check. The licensee's representative showed the inspector a copy of NPT-US-1, Nuclear Penetrant Testing Procedure and stated it was used for the checks. The procedure was approved by a Level III Inspector but had not been reviewed by the Operations Committee and approved by a member of plant management. This is an apparent item of noncompliance. Technical Specification 6.5.C and 6.5.C.3 requires that procedures for preventive or corrective maintenance of plant equipment and systems that could have an effect on Nuclear safety be developed. Technical Specification 6.5 requires that these procedures shall be reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager.

10/ AO Rpt No. 050-263/75-21, NSP to DL, dtd 10/24/75.
11/ AO Rpt No. 050-263/75-22, NSP to DL, dtd 11/11/75.

4. Semi-Annual Reports

Subject reports for the periods of July 1, 1974, to December 31, 1974, and January 1, 1975, to June 30, 1975, were reviewed. Review of these reports indicated that information required by the Technical Specifications had been reported. Review of the control room log books indicated that three forced shutdowns during the reporting periods were as reported in the semi-annual reports.

5. Plant Operations Following Refueling OUtage

The inspector reviewed records and determined that the following were returned to service in accordance with approved procedures after being distributed during the outage.

a. Reactor recirculation 4-inch bypass line

These lines were replaced during this outage.[12/] Prior to returning the system to normal a 1000 psi hydro for four hours was performed and the welds examined. The hydro was signed off by the Hartford Steam Boiler Inspection and Insurance Company's Authorized Inspector as meeting the requirements of B 31.1-1967, paragraph 137.1.2.

b. Local Power Range Monitor (LPRM) Replacement

Several LPRMs were replaced during the outage. They were installed and checked in accordance with reactor maintenance procedure 9238, LPRM Replacement Without Spring Reel.

c. Control Rod Drive Changeout

Twenty control rod drive (CRD) assemblies were removed, inspected, rebuilt and reinstalled or replaced with spare CRD's. No stress corrosion cracks were observed in the collet housing areas of the CRD's. Surveillance tests performed on the CRD's included checks for flange leaks (ST-1072), CRD friction tests (ST-1064), normal drive times (ST-1054), cold and hot scrams (ST-00TI), CRD coupling check (ST-0075), and nuclear instrument response (ST-0076) were completed prior to resuming plant operations.

d. HPCI-9 Valve Inspection

(1) The subject valve was disassembled and a previous modification[13/] was inspected. A local leak rate test (0137-9) was performed on the valve prior to returning the valve to service. The leakage was 14.8 scfh which is less than

12/ IE Inspection Rpt 050-263/75-16.
13/ IE Inspection Rpt 050-263/75-05.

the Technical Specification 4.2.f.(2)(b) limit of 17.2 scfh.

(2) The inspector reviewed the following control rod withdrawal sequence and reactivity measurements which were performed prior to power range operation.

(a) The control rod withdrawal sequence was approved for startup and designated as Operations Manual Volume F, Memo. No. 437.

(b) Surveillance test P-0073, Shutdown Margin Requirement was performed. The shutdown margin with the most reactive rod fully withdrawn was -0.52% $\Delta$K which meets the requirements of Technical Specification 3.3.A.1 and 4.4.A.1.

(c) Measured critical rod position for the A.1 and B.1 rod configurations were within 0.5% $\Delta$K of the estimated critical rod positions.

(3) Review of licensee records indicated the following surveillance tests were performed during the outage as required by technical specifications:

0019, High Reactor Pressure Scram Calibration
0020, High Drywell Pressure Scram Calibration
0021, Condenser Low Vacuum Scram Calibration
0049, Source Range Channel Sensor Check
0066, Radiation Monitor Reactor Building Vent Sensor Check
0126, Reactor Coolant Drywell Leak Check
0173, Discharge Canal Sampler Operability
0205, Fuel Pool Level.

6. Plant Startup

The inspector witnessed portions of control room operations during power ascension testing from 25% to 50% power on November 19-21, 1975. The licensee obtained transient in-core flux traverses during the power ascension to evaluate the core flux distribution. No abnormal flux shapes were observed at these powers. No discrepancies were noted by the inspector during the observation of control room activities.

7. Lost Parts in the Reactor Vessel

The inspector reviewed the licensee's internal report covering the apparent loss of four parts in the pressure vessel. The items were as follows:

a. Two fuel channel clips approximately 4" x 2" x 2/8".

b. LPRM nose piece approximately 1/2" in diameter and 1" long.

c. An aluminimum gauge block approximately 1 1/2" x 6" x 3/16".

The licensee made an extensive search to find these missing parts but could not locate any of them. The inspector reviewed an analysis at the site which indicated that Items a and b did not have the potential to cause fuel bundle flow blockage and that Item c was believed to have the potential for flow blockage. A complete flow blockage analysis for the gauge block was not available at the time of the inspection.