U. S. ATOMIC ENERGY COMMISSION
DIRECTORATE FOR REGULATORY OPERATIONS

REGION III

RO Inspection Report No. 050-263/73-05

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) 545 Mwe

Type of Inspection: Routine, Unannounced

Date of Inspection: May 23 - 25, 1973

Date of Previous Inspection: March 27 - 29 and April 4 - 5, 1973

Principal Inspector: P. H. Johnson (signed: E. Fiorelli for) 6/28/73 (Date)

Accompanying Inspector: None

Other Accompanying Personnel: None

Reviewed By: H. C. Dance, Senior Reactor Inspector (signed: E. Fiorelli for) 6/28/73 (Date)
BWR Operations

SUMMARY OF FINDINGS

Enforcement Action

A. The sodium pentaborate solution in the standby liquid control system storage tank was inadvertently diluted on May 24 to a concentration less than that required by the Technical Specifications. (Paragraph 7)

B. Changes made to a surveillance test during its performance were not approved by two individuals holding senior operator licenses. (Paragraph 4.n)

Licensee Action on Previously Identified Enforcement Matters

The licensee has posted warning signs on all drywell and pressure sensing taps to prevent inadvertent obstruction. Action on this item of noncompliance is considered to be completed. (Paragraph 11)

Corrective actions related to items 5.b and 5.c as noted during the May 1972 management inspection are considered to have been completed in view of clarifications to the pertinent requirements provided by a recent revision of the Technical Specifications. (Paragraph 5)

Warning labels have been applied to the torus manways to prevent their being opened without permission. (Paragraph 10)

Unusual Occurrences

A. The high pressure coolant injection system failed to operate during a test conducted on May 18, 1973. (Paragraph 12)

B. The standby liquid control tank was diluted below the minimum required concentration on May 24, 1973. (Paragraph 7)

C. The T-ring seal of a primary containment isolation valve was discovered on May 17 to be depressurized. (Paragraph 4.e)

Other Significant Findings

A. Current Findings: None

B. Status of Previously Reported Unresolved Items: 1/

1. A test was performed during the startup after the refueling outage to monitor safety valve reactions to relief valve and turbine bypass valve operations. (Paragraph 4.j)

1/ RO Inspection Report No. 050-263/72-06.

2. An inspection of reactor building to torus vacuum breaker T-ring seals was conducted during the refueling outage. (Paragraph 4.e)

Management Interview

An interview was conducted with Messrs. Larson (Plant Manager) and Clarity (Superintendent - Plant Engineering and Radiation Protection) at the conclusion of the inspection. The following matters were discussed.

A. The inspector stated that, based upon examination of the recent change to Section 6.0 of the Technical Specifications and the clarifications which it provided, corrective actions related to noncompliance items 5.b and 5.c of the October 19, 1972, enforcement letter were considered to have been completed. (Paragraph 5)

B. The inspector stated that the inspection had included a review of maintenance performed on torus-drywell vacuum breakers during the outage and subsequent differential pressure tests which verified satisfactory leak tightness. He asked whether consideration was being given to a periodic repeat of some form of differential pressure test in the future. The licensee responded that the matter had not yet been discussed, but that it would be given consideration. (Paragraph 4.d)

C. The inspector stated that inaccurate standby liquid control system storage tank level indications had been experienced by at least one other facility as a result of crystalization associated with a bubbler type level indicator. He noted that Monticello staff personnel currently appear to be giving sufficient attention to the level indicator to preclude an erroneous indication, but questioned the need for an improved indicator or a periodic operational check of the currently installed indicator to assure reliable indication. Facility representatives stated that the matter was currently under consideration. (Paragraph 8)

D. The inspector described the two items of apparent noncompliance which were noted during the inspection. (Paragraphs 4.n and 7)

REPORT DETAILS

1. Persons Contacted

Monticello Plant Staff

C. Larson, Plant Manager
M. Clarity, Superintendent - Plant Engineering and Radiation Protection
W. Anderson, Superintendent - Operations and Maintenance
G. Jacobson, Plant Engineer, Technical
M. Dinville, Plant Engineer, Operations
D. Antony, Engineer
L. Nolan, Engineer
M. Hammer, Engineer
H. McGilton, Engineer
L. Severson, Shift Supervisor
H. Seibel, Shift Supervisor

Other NSP Representatives

J. Meier, Quality Assurance Engineer
P. Krumpos, Quality Assurance Engineer
D. Musolf, Administrator, Nuclear Support Services

2. General

The plant was operating at full power at the time of the inspection, having started up during the previous week following a two and one-half month refueling outage. The off-gas release rate at the time of the inspection was approximately 3500 μCi/sec. A brief shutdown was scheduled for May 25 - 26 to shift to the A rod sequence.

3. Record Reviews

The following records were reviewed without comment during the inspection:

a. Operations Committee minutes for meetings conducted February 24, March 1 - 2, 6 - 7, 8, 9, 16, 21, 23, 27, and April 5, 6, 7, 13, and 25, 1973.

b. Shift Supervisor's Log, April 27 - May 5, 1973.

4. Refueling Outage Activities

A previous report[2/] described various activities planned, as of February 1973, to be conducted during the refueling outage. Several of these activities and others not discussed in the previous report were reviewed during this inspection, as follows:

2/ RO Inspection Report No. 50-263/73-02.

a. Vane Type Flow Switches. A licensee representative stated that both vane type flow switches have been removed from the reactor water cleanup system. As stated previously, the flow switch used in the HPCI cooling water line was removed since it was redundant to other indications. The original flow switch is still installed in the standby liquid control system, and the stubs of the original flow switch paddles, recalibrated for the reduced paddle area, remain in service in the residual heat removal system. Further action on replacement of the flow switches in the standby liquid control and residual heat removal systems was stated to be under consideration.

b. Diesel Generator Air Motors. The inspector reviewed, without comment, documents reporting (1) disassembly and cleaning of the four air starting motors on each diesel generator, including installation of new vanes and bearings where required; (2) cleaning of air piping, Y-strainer, air relay valves, and line lubricators associated with No. 12 diesel generator starting systems; (3) inspection of diesel-generator air boxes and top and lower decks, with no piston ring wear apparent; and (4) functional tests of both starting systems following maintenance.

c. MSIV Spool Valves. The former Numatics spool valves associated with the MSIV's were replaced during this outage with pneumatic operators manufactured by the Automatic Valve Company. The modification was reviewed and approved by the Operations Committee on April 7, 1973. The review noted that the modification did not constitute a change to the facility as described in the FSAR. The inspector also examined related documents describing design and qualification testing of the new assemblies and operational tests performed to verify proper MSIV operation after their installation. The previous carbon steel accumulators associated with the MSIV air systems were also replaced during the outage with others constructed of stainless steel. Proper MSIV operation using the new accumulators (with air supply line isolated) was also verified by testing conducted on May 11, 1973.

d. Torus-Drywell Vacuum Breakers. The previous inspection report[3/] described modifications made to improve operation of the torus-drywell vacuum breakers. The inspector examined results of drywell-torus differential pressure tests which indicated a pressure decrease from 0.4 to 0.1 psid in one hour. Curves provided by the reactor vendor indicated that this pressure drop would have occurred in 33 minutes with leakage equivalent to a one-inch orifice (a criterion established by the vendor as a conservative indication of acceptable leak-tightness). A test conducted at higher differential pressure gave a resulting pressure drop from 1.0 to 0.87 psid in 30 minutes, compared to a corresponding pressure drop in a period of 19 minutes for the equivalent of a one-inch orifice.

3/ RO Inspection Report No. 050-263/73-04.

e. Primary Containment Isolation Valves. The report[4] of the previous inspection described actions taken by the licensee to provide proper operation of the primary containment isolation valves which use a pneumatic seal. The inspector examined the results of local leak rate tests conducted after the repairs which verified proper valve tightness. Subsequent failure of one of the valves to close was described in a licensee report[5]. Discussion of the event with a facility representative and examination of photographs of the affected valve operator indicated the description of the event, its safety significance, and the corrective action taken to have been as described in the referenced report.

f. Relief Valves. Modifications recommended by the manufacturer and approved by the Operations Committee were installed in all four relief valves during the outage. The change summary document showed the modifications to include: (1) installation of monel pilot stems to eliminate differential thermal expansion between the pilot stem and the monel pilot bellows, (2) replacement of the stainless steel air operator gland with a silicone bronze gland having a larger clearance, to minimize the possibility of gland-stem seizure, (3) a reduction in main piston diameter (and installation of harder piston rings) to prevent rubbing between the main piston and cylinder, and (4) installation of a more positive locking mechanism on the second stage disc. Related documents indicated that all four valves were adjusted to a set pressure of 1068 psig using nitrogen, that all were tested under operating conditions on May 16, and that proper operation of associated pressure switches was verified. A facility representative stated that initial difficulty experienced with operation of the bellows leakage pressure switches caused them to be returned to the vendor, who attributed the cause to the unexplained presence of grease of a tacky consistency inside the pressure switch. The switches operated properly after cleaning. The representative stated that previous experience with the same type of switch indicated this condition to be an isolated case, but that new pressure switches of this type would be disassembled for inspection prior to use.

g. Safety Valves. The A and C safety valves were replaced by spare valves set with steam by the vendor. The replacement procedure was approved by the Operations Committee on February 27, 1973. Positions of blowdown rings on the A, B, and D safety valves were adjusted to agree with those on the C safety valve, which was set by the vendor in July 1972. The B and D safety valves were subsequently observed to be leaking slightly past the main seat during the post-refueling operational hydrostatic test. They were disassembled using a special procedure provided by the vendor which maintains main spring compression and setpoint of the valve. Proper leak tightness was observed following lapping and reassembly.

4/ Ibid.
5/ Letter, NSP to Directorat f Licensing, dated May 25, 1973.

h. Diesel Back-Up Start Relays. The inspector examined special test procedures which were approved by the Operations Committee and were subsequently conducted for both diesel generators on May 5, 1973. The tests included checks of individual relay operation and of proper logic circuitry functioning, and demonstrated starting system circuitry to operate as required. The test procedure indicated that the test would be repeated at a frequency of once per operating cycle.

i. Local Leak Rate Tests (LLRT's). Results of LLRT's (including MSIV leak tests) conducted during the outage indicated that several primary containment isolation valves did not initially satisfy the leak-tightness requirements. Five of the eight MSIV's exceeded the 11.5 SCFH allowed by Technical Specifications. Following stellite weld repair to one valve seat and lapping of the other valves, the maximum MSIV leak rate was determined to be less than 5 SCFH. Other valves which failed the initial leak rate test included HPCI and RCIC steam discharge check valves to torus, three of the four feedwater check valves, torus/drywell purge line valves, and RHR loop A containment spray isolation valves. Leak tests following repairs indicated satisfactory results in each case. A licensee representative stated that results of the LLRT's would be included in the report of the integrated primary containment leak rate test which will be submitted to the Directorate of Licensing.

j. HPCI Auto Isolation. A venturi flow indicator was installed in the HPCI steam line during the outage, in accordance with an installation procedure which was reviewed by the Operations Committee on March 20, 1973. The inspector examined Quality Assurance Documentation related to the installation which included material certifications, welder qualifications, correction of noted deficiencies, receipt inspection, shop hydrostatic test, welding procedure qualifications, nondestructive test results, and completed procedure signoffs. The inspector noted that weld repairs were not clearly described, and that the several socket welds associated with the venturi instrument piping were not individually identified in the procedure. Licensee representatives provided clarifying documents related to these items prior to the conclusion of the inspection. No other comments were made by the inspector.

k. Main Steam Line Testing. The licensee installed additional instrumentation on the main steam line during the outage, as described in a previous inspection report[6/], in accordance with a written procedure and 10 CFR 50.59 review which were approved by the Operations Committee on April 10, 1973. The safety review noted the instrumentation installed at existing blank flanges on the "A" steam line used a 1/4" hole, which would limit steam flow to 2340 lb/hr in the event of an instrument line break. The inspector reviewed Quality Assurance documents related to the instrument

6/ RO Inspection Report No. 050-263/73-02, page 14.

installation, including nondestructive testing of instrument line welds, materials certifications, use of the approved procedure for installation, and performance of a 10-minute hydrostatic test at 1563 psig.

A licensee representative stated that testing was performed following the outage to determine (1) the effect on the "A" safety valve of "A" relief valve operation, and (2) the effect of turbine bypass valve operation upon pressures in the main steam line. Although test data were still being reduced and analyzed, one preliminary observation during the testing was slight movement of the "A" safety valve (as indicated by an attached accelerometer) upon the first actuation of the "A" relief valve at 15 percent power with two main steam lines in service.

l. Valve Wall Thickness Verification[7/]. A licensee representative stated during the inspection that the wall thickness verification program had been completed, and that results indicated satisfactory wall thickness on all valves tested. Measurements were not taken on four valves, including the two reactor vessel drain valves, due to high radiation levels. He stated that a report summarizing the results of the inspection would be submitted in the near future.

m. Integrated Primary Containment Leak Rate Tests (IPCLRT). The inspector reviewed and discussed with a licensee representative the results of the IPCLRT conducted during the period May 4 - 7, 1973. A plot of hourly measurement indicated a consistent leak rate of 0.6418 %/day at the test pressure of 41 psig, compared to an allowable 0.9 %/day allowed by section 4.7 of the Technical Specifications. The results of the IPCLRT will be summarized in a report to the Directorate of Licensing, as required by Technical Specifications.

n. Surveillance Testing. The inspection included an examination of selected surveillance tests, particularly those required to be performed during each refueling outage. During review of surveillance test 0189, diesel automatic fast start initiation, a number of changes were noted to have been made to the procedure by the supervising test engineer after its approval by the Operations Committee without the approval of two senior licensed operators as required by Technical Specifications, paragraph 6.5.D. Changes included the placing of the diesel generator breaker in the pull-to-lock position instead of booting the relay, and the operation of switches or pulling of fuses not required by the initial procedures. The following additional surveillance tests, performed on the dates indicated, were examined without comment:

(1) MSIV Closure scram, May 12, 1973.

(2) 125 VDC battery discharge, April 18, 1973.

7/ RO Inspection Report No. 050-263/73-04.

(3) 250 VDC battery discharge, April 17, 1973.

(4) Diesel generator undervoltage and auxiliary power tests, May 2, 1973.

(5) Simulated auto initiation of LPCI, core spray, HPCI, automatic depressurization, RCIC, and diesel generators, May 9, 1973.

5. Management Inspection Items

Item 5, part b of the RO:HQ enforcement letter[8/] stated that changes to operating procedures (e.g., work request authorization forms and procedures) had not been properly reviewed and approved. The licensee's response[9/] stated that instructions covering the use of work request authorizations were considered to be administrative rather than operating procedures. Item 5, part c, of the RO:HQ letter stated that the Safety Audit Committee had not reviewed recommendations made by the Operations Committee relating to proposed procedures or changes thereto, or advised management concerning such recommendations. The licensee's response to this item stated that Technical Specifications were considered to require Safety Audit Committee review of changes when "matters of safety significance or potentially unreviewed safety question, or changes to matters contained in the FSAR" were involved. With reference to parts b and c, the licensee stated that a revised Section 6.0 of the Technical Specifications was being submitted which "should result in a clearer definition of such requirements ....." Change No. 6 to the Monticello Technical Specifications, issued April 3, 1973, did provide clearer definitions, the licensee's actions related to items 5.b and 5.c are considered to have been completed.

6. Critical Rod Position

A licensee representative stated during the inspection that reactor criticality was achieved on the fifteenth control rod at a temperature of 140°F during the initial startup after the refueling outage, compared to an NSP estimate of 31 rods. Upon determination of the apparent inconsistency, the reactor was shut down pending investigation, and General Electric Company was requested to calculate an estimated critical position. Core symmetry checks and verification of temperature coefficient gave expected results. After rechecking of the estimated critical position, it was determined that the NSP estimate was in error as a result of three factors: (1) underestimating the reactivity worth of the new fuel, (2) overestimating control rod worth by the technique of "smearing out" the B_4C reactivity worth over the entire cross sectional area of the control blade, and (3) overestimating the effect of fuel burnup on rod worth. A subsequent calculation provided a critical estimate of 14 rods on sequence B (the sequence in use at the time of startup) and 17 rods on sequence A. The estimate provided by General Electric predicted criticality on 13 or 14 rods. The subsequent startup proceeded as expected with a critical rod position in the predicted range.

8/ Letter, RO:HQ to NSP, dated October 19, 1972.

9/ Letter, NSP to RO:HQ, dated November 10, 1972.

7. Standby Liquid Control System Concentration

The inspector was informed during the final day of the inspection that an addition of water the previous evening (May 24) to the standby liquid control tank had resulted in dilution of the sodium pentaborate solution to a concentration less than that required by Technical Specifications. Review of the event with facility representatives indicated that the last sample, taken on May 12 after an addition of 100 gallons of water to the tank, had given a concentration of 11.9 weight percent of sodium pentaborate. On May 24, when system volume indicated a need for another water addition, the plant chemist incorrectly recalled the results of the previous analysis and authorized the addition of another 100 gallons of water. Subsequent analyses gave a result of 11.1 percent concentration, as compared to a minimum of 11.4 percent required by Technical Specifications during plant operation. Additional chemicals were added and a previously scheduled plant shutdown was conducted on May 25. The inspector stated that more attention should have been given to tank concentration prior to and during water addition, and indicated that the event was considered to be a violation of Technical Specification requirements. The matter was subsequently reported in a letter to the Directorate of Licensing on June 4, 1973.

8. Standby Liquid Control Tank Level Indication

The inspector examined standby liquid control tank indication during a tour of the plant and discussed its operation with an operator and a cognizant engineer. Tank level is monitored by the measurement of back pressure on a dip tube which continuously injects a stream of low pressure instrument air into the tank. Indicated levels before and after purging of the instrument were the same. The operator stated that most operators routinely purge the instrument before recording a reading, since the back pressure could increase due to the formation of sodium pentaborate crystals near the opening. The cognizant engineer stated that he had adopted the practice of verifying indicated level with a direct measurement of level inside the tank at intervals of approximately one to two weeks. The inspector noted during the interview at the conclusion of the inspection that instances of unreliable level indication with the bubbler type instrument had been experienced at at least one other facility and asked, although the level indicator appeared to be receiving sufficient attention at present, whether there were other plans to improve the reliability of the tank level indication. A licensee representative stated that other plans were being considered.

9. Reinstallation of Reactor Vessel Components

A licensee representative stated in a telephone conversation prior to the inspection that inventory following maintenance activities had revealed one vibration detector securing bolt to be missing. It could not be seen during a videocamera inspection of the vessel internals. Although the work related to the bolt removal was performed in the annulus area, and persons doing work felt confident that the missing bolt was still in the annulus, possibly behind a jet pump, the nuclear steam system supplier was asked to perform a safety analysis. This analysis noted that the diameter of the bolt and integral washer was slightly larger than the diameter of the flow orifices beneath the peripheral fuel bundles, and recommended inspection of these orifices to verify that the bolt had not fallen into one of them. The reactor vessel head had by this time been replaced, and was subsequently removed to permit inspection of the peripheral fuel bundle flow nozzles. The inspection verified the flow nozzles to be free of obstruction. A licensee representative stated that the event would be discussed in the fuel summary status report to the Directorate of Licensing.

The current inspection included an examination without comment of completed procedures shown below verifying reinstallation of reactor vessel head and internals, which were completed on the dates indicated (for components removed to permit reinspection of flow orifices, the date referred to the second reinstallation):

a. Removal of drywell rediation shield, April 20.
b. Installation of steam separator, April 27.
c. Latching of steam separator, April 27.
d. Dryer installation, April 27.
e. Installation of studs, April 23.
f. Installation of head, washers, and nuts, April 28.
g. Tensioning of reactor vessel closure studs, April 30.
h. Installation of reator vessel head insulation, April 30.
i. Installation of reactor pressure vessel head piping, May 1.
j. Completion of operational hydrostatic test, May 7.

10. Violation of Primary Containment

The previous report[10/] discussed the opening of a torus manway with the reactor at a temperature requiring primary containment integrity, and stated that caution signs were to be affixed by the licensee to prevent recurrence. The inspector examined the torus manway covers during the current inspection and noted a warning to have been painted on each cover requiring permission of the Shift Supervisor and Operations Supervisor prior to opening of the primary containment.

11. Drywell Pressure Sensing Taps

The licensee stated in his response[11/] to an item of noncompliance noted during a previous inspection[12/] that warning tags would be placed on drywell and torus sensing line taps to prevent their being inadvertently plugged or covered. The inspector examined a completed Work Request - authorization which indicated that these warning tags had been placed during the recent outage. The licensee's corrective actions for this item of noncompliance are considered to have been completed.

10/ RO Inspection Report No. 050-263/73-04.
11/ Letter, NSP to RO:III, dated August 22, 1972.
12/ RO Inspection Report No. 050-263/72-05.

12. HPCI Inoperability

A recent licensee report[13] discussed inoperability of the high pressure coolant injection system which was discovered during the startup following the refueling outage. Discussion of the occurrence with facility representatives indicated the event and corrective action to have been as described in the licensee's report. The failed gears were examined and appeared to show excessive wear, as might result from insufficient lubrication. No chips, broken teeth, or other indication of metallurgical failure were evident. The inspector observed a successful auto start on May 24, and examined the documentation of a satisfactory HPCI operability surveillance test performed later the same day.

13. LPCI Loop Selection Circuitry

The inspection included a review of the portion of LPCI loop selection circuitry associated with tripping of the recirculation pumps, in response to a circuitry error in this portion of the system reported by another BWR. Examination of circuit diagrams with a facility engineer and a subsequent telephone conversation on May 31 verified that the circuitry was properly connected and that it had been operationally tested during the startup testing program.

14. Modified Off-Gas System

A licensee representative stated in discussions during and subsequent to the inspection that the test program for the off-gas system is continuing. Delays were caused by difficulties with off-gas flow meters and the necessity to repair defective factory welds in the piping associated with the off-gas compressors. The licensee indicated that the system would be ready for plant tie-in by late summer.

15. Minnesota Pollution Control Agency Relationships

A previous report[14] discussed an alarm system which had been installed to provide the state Pollution Control Agency an indication of high release levels. A licensee representative stated during the inspection that the system is now operable. Pursuant to the agreement reached between the licensee and the state agency, the agency will call the plant upon receipt of an alarm to verify the alarm and determine its cause. Responsibility for action in the event of an emergency remains with the state health department.

13/ Letter, NSP to Directorate of Licensing, dated May 25, 1973.
14/ RO Inspection Report No. 050-263/72-02.

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A. RO Inspection Report No. 050-263/73-05

Transmittal Date : June 28, 1973

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C. Incident Notification From: ______
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