U. S. ATOMIC ENERGY COMMISSION
DIRECTORATE OF REGULATORY OPERATIONS

REGION III

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

Licensee: Northern States Power Company
419 Nicollet Avenue
Minneapolis, Minnesota 53401

Monticello Generating Plant
Monticello, Minnesota

License No. DPR-22
Category: C

Type of Licensee: BWR (GE) 545 Mwe

Type of Inspection: Routine, Unannounced

Dates of Inspection: October 3 - 5, 1972

Date of Previous Inspection: July 12, 1972

Principal Inspector: P. H. Johnson — 11/1/72 (Date)

Accompanying Inspectors: None

Other Accompanying Personnel: None

Reviewed By: G. Fiorelli, Chief
Reactor Operations Branch — 11/1/72 (Date)

SUMMARY OF FINDINGS

Enforcement Action: None

Licensee Action on Previously Identified Enforcement Matters

All drywell pressure sensing and sampling lines have been inspected for flow restrictions. Warning tags will be placed on the drywell pressure sensing taps at the first opportunity during a future outage. (Paragraph 20)

Unusual Occurrences

A. An RCIC isolation switch did not trip at the required trip point during a surveillance test on March 1, 1972. (Paragraph 25)

B. The outboard HPCI steam isolation valve did not close during a routine surveillance test on April 3, 1972. (Paragraph 24)

C Insertion of control rod 22-31 terminated six inches from the fully inserted position during a scram on May 23, 1972, and on two subsequent occasions. (Paragraph 10)

D. On June 14, 1972, both pumps in the redundant "A" loop of the Residual Heat Removal Service Water system did not meet Technical Specification head-flow requirements. (Paragraph 26)

E. Stack activity release rate increased from 27,000 to 69,000 uCi/sec on August 17-18, 1972. (Paragraph 31)

F. Two vane type flow switches were observed during an inspection on August 31, 1972, to have failed. (Paragraph 16)

G. A relief valve failed to operate properly following reactor scrams on July 10 and 21, 1972. (Paragraph 5)

Other Significant Findings

A. Current Findings

Contrary to Section 3.8.A of the Technical Specifications which states that continuous monitoring of the radioactive gases released from the stack is a required condition for operation, timely restoration of stack gas monitor flow was not made on June 20, 1972. Actions taken by the licensee to prevent recurrence were reviewed during the inspection. (Paragraph 30)

B. Status of Previously Reported Unresolved Items

1. Investigation into operation of safety valves is continuing. (Paragraph 6)

2. Inspection of vane type flow switches by the licensee will continue until completed. (Paragraph 16)

3. Inspection of reactor building to torus vacuum breaker valve seats is planned during the refueling outage. (Paragraph 18)

Management Interview

A management interview was conducted at the conclusion of the inspection with the following Monticello plant staff personnel in attendance:

M. Clarity, Superintendent - Plant Engineering and Radiation Protection
W. Anderson, Superintendent - Operation and Maintenance
L. Eliason, Radiation Protection Engineer
G. Jacobson, Plant Engineer, Technical
M. Dinville, Plant Engineer, Operation

The following matters were discussed:

A. The inspector indicated that an item of apparent noncompliance had been noted in that timely restoration of the stack gas monitor flow had not been made on June 20, 1972. (Paragraph 30)

B. Testing of diesel generator backup starting circuitry and relays was discussed. Plant management stated that plans had been made to test these components at least during each refueling outage commencing in 1973, and indicated that the tests would be performed more frequently if determined to be advisable and feasible. (Paragraph 28)

C. The inspector noted that (1) the minutes for several Operations Committee meetings conducted since June had not yet been written, (2) certified final reports of abnormal occurrences at least a month previous were not yet available for review, and (3) the Quality Assurance program was in an intermediate stage of preparation. (Paragraph 34) The inspector commented on the increased amount of plant-related activities which would transpire in the ensuing months, the added work load and responsibilities of which would require additional efforts and present an added challenge to the plant staff.

D. In response to a question from the inspector, management representatives stated that the rubber seats on the torus to reactor building vacuum breakers would be inspected during the refueling outage. (Paragraph 18)

E. The inspector noted that a favorable analysis of relief valve reaction forces had been reported by Bechtel Corporation. (Paragraph 7)

F. The following additional items were briefly discussed:

1. Uninterruptible power system (Paragraph 19).

2. HPCI auto-isolation (Paragraph 22).

3. Revision of Emergency Plan (Paragraph 4).

4. Protection of reactor vessel studs (Paragraph 9).

5. Operation of Limitorque valves (Paragraph 24).

6. Continuing investigation of safety valve performance (Paragraph 6).

REPORT DETAILS

1. Persons Contacted

M. Clarity, Superintendent - Plant Engineering and Radiation Protection
W. Anderson, Superintendent - Operation and Maintenance
G. Jacobson, Plant Engineer, Technical
L. Eliason, Radiation Protection Engineer
M. Dinville, Plant Engineer, Operations
W. Shamla, Engineer, Instruments
L. Nolan, Engineer
D. Antony, Engineer
J. Pasch, Engineer
M. Hammer, Engineer
R. Jacobson, Chemist
S. Pearson, Shift Supervisor
R. Kmitch, Shift Supervisor
R. Tigue, Reactor Operator
J. Carstens, Reactor Operator
W. Boehme, Reactor Operator
M. Brant, Reactor Operator
E. Earney, Reactor Operator

2. Log and Record Reviews

The following logs and records were reviewed without comment:

a. Reactor and Control Room Log, June 1, 1972, through July 31, 1972

b. 10 CFR 50.59 Change File

c. Unusual Occurrence Reports

d. Significant Operating Event Reports

3. Plant Personnel and Staffing

Discussions with plant management representatives indicated that Mr. M. Dinville, Plant Engineer, Operations, has accepted a position on the staff of NSP's new Cherbourne County plant, although he is expected to remain with the Monticello plant staff through the completion of the 1973 refueling and maintenance outage. The inspector was also informed that General Electric Company has been contracted to provide program management services for the outage, and that a significant amount of outside help will be employed.

4. Revision to Monticello Emergency Plan

Discussion with a licensee representative indicated that comments received from the Radiological Assistance Team of the Chicago Operations Office in a letter dated June 13, 1972, are presently being incorporated into a revision of the Emergency Plan.

5. Relief Valves

The failure of the D relief valve to open following a scram on July 10 and the licensee's investigation and followup corrective action are discussed in a previous report.[1/] A recurrence of the operating difficulty on July 21 was discussed in Region III[2/] and licensee[3/] reports. As described in the licensee's report and verified through discussions with plant staff personnel and review of the abnormal occurrence report, investigation into the cause of relief valve maloperation resulted in the discovery of a small bellows leak. A pressure switch is installed to detect such leaks. The solenoid-operated valves which are included to provide a means of testing the pressure switch were found to have been designed for tight shutoff only when pressure is applied from the side away from the pressure switch and bellows chamber. Tests summarized in the licensee's report indicated that the A and D leak detection systems could not have detected a small leak. This situation was corrected by replacing the original pressure switches (setpoints ranging from 61 to 78 psig) with new pressure switches set to trip at 5 psig. The bellows assemblies of the remaining relief valves were leak tested with satisfactory results.

During the previous inspection,[4/] the licensee had indicated an intention to manually operate each relief valve after approximately one month's operation and to inspect the second stage pistons during a future outage. Discussions with plant staff personnel indicated that all relief valves were test operated in September 1972 with satisfactory results. A licensee representative indicated that tentative plans call for the complete inspection of two relief valves during the refueling outage, and that the second stage pistons of the remaining relief valves will also be inspected at that time, since this provides a good check on possible leakage of the pilot valve.

1/ RO Inspection Report No. 050-263/72-5
2/ RO Inquiry Report No. 050-263/72-12
3/ Letter, NSP to Directorate of Licensing, dated July 28, 1972
4/ RO Inspection Report No. 050-263/72-5

4. Revision to Monticello Emergency Plan

Discussion with a licensee representative indicated that comments received from the Radiological Assistance Team of the Chicago Operations Office in a letter dated June 13, 1972, are presently being incorporated into a revision of the Emergency Plan.

5. Relief Valves

The failure of the D relief valve to open following a scram on July 15 and the licensee's investigation and followup corrective action are discussed in a previous report.[1/] A recurrence of the operating difficulty on July 21 was discussed in Region III[2/] and licensee[3/] reports. As described in the licensee's report and verified through discussions with plant staff personnel and review of the abnormal occurrence report, investigation into the cause of relief valve maloperation resulted in the discovery of a small bellows leak. A pressure switch is installed to detect such leaks. The solenoid-operated valves which are included to provide a means of testing the pressure switch were found to have been designed for tight shutoff only when pressure is applied from the side away from the pressure switch and bellows chamber. Tests summarized in the licensee's report indicated that the A and D leak detection systems could not have detected a small leak. This situation was corrected by replacing the original pressure switches (setpoints ranging from 61 to 78 psig) with new pressure switches set to trip at 5 psig. The bellows assemblies of the remaining relief valves were leak tested with satisfactory results.

During the previous inspection,[4/] the licensee had indicated an intention to manually operate each relief valve after approximately one month's operation and to inspect the second stage pistons during a future outage. Discussions with plant staff personnel indicated that all relief valves were test operated in September 1972 with satisfactory results. A licensee representative indicated that tentative plans call for the complete inspection of two relief valves during the refueling outage, and that the second stage pistons of the remaining relief valves will also be inspected at that time, since this provides a good check on possible leakage of the pilot valve.

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6. Safety Valve Operation

The previous inspection report[5/] provided a description of the brief operation of the A safety valve following a reactor scram on July 10, 1972, and the investigation and corrective action undertaken by the licensee at that time. The safety valve which lifted momentarily following a scram on February 26, 1972, was replaced on June 4, 1972, with a valve recently received from the manufacturer. The valve which was removed was shipped to Dresser Industries for inspection and testing. Review of the results of the tests by Dresser, correspondence from a private consultant retained by NSP to assist in the investigation of safety valve operation, and other documents available at the site provided the following additional information:

a. During the tests at Dresser, the safety valve lifted below its original setpoint due to seat leakage caused by previous valve chatter. The valve chatter was associated with adjustment of the upper and lower blowdown rings. Conversation with the licensee during the previous inspection indicated that safety valves were installed as received from Dresser; i.e., that no adjustments or setpoint checks had been performed at the site. The Dresser tests also indicated a tendency for the safety valves to lift below their setpoint during rapid pressure transients.

b. Investigation by an NSP subcontractor has apparently resulted in the discovery of a 15 hz resonance due probably to a rocking of the disc on the seat. The testing which continues includes the superimposition of pressure oscillations (up to 30 hz and 100 psi amplitudes) on a slowly increasing pressure to observe their effects on safety valve operation.

c. Investigation in these and other areas is continuing.

The licensee indicated that plans were being made to install additional instrumentation on the steam lines during the refueling outage to permit further evaluation of possible pressure waves. The adjustment of the blowdown rings will be checked during the refueling or a previous extended outage. It was also indicated that a portable safety valve test assembly is being developed which will permit the setting of safety valves onsite using steam prior to their reinstallation.

5/ Ibid.

7. Safety and Relief Valve Reaction Forces

A recent letter[6/] to the licensee relayed information concerning the failure of relief valve installations at another facility. Review of a letter[7/] during the inspection indicated that the stresses due to dynamic and pressure forces in the relief valve assemblies were found to be within the allowable limit of 18,000 psi, assuming that seismic forces and forces due to relief valve operation do not occur simultaneously. It was noted that the letter addressed itself to relief valves, and not to safety valves. Subsequent to the inspection, a licensee representative indicated by phone that Bechtel had confirmed the satisfactory analysis to have also included safety valves, and that a letter to NSP would provide documentation.

8. Recirculation Pump Control

Operating performance of the recirculating pump control system was discussed with licensee representatives in view of two operating events associated with the system during the Spring of 1972. No further operating difficulties have been encountered, apparently owing primarily to replacement of an amplifier card having an intermittent open circuit which was located with the use of a recorder. A loose wire had also been found in the MG set exciter control circuit. Plant staff personnel considered the recent operation of the recirculation pumps to be satisfactory.

9. Protection of Reactor Vessel Head Studs

The inspector discussed with cognizant plant staff personnel the problems of reactor vessel head stud deterioration encountered by two other facilities within the past year. The plant staff representative stated that this matter would be further discussed with the inspector at a later date following further investigation and consultation with General Electric Company.

10. Control Rod Scram Performance

Region III[8/] and licensee[9/] reports discussed the performance of control rod 22-31 during a reactor scram on May 23, 1972. On this

6/ Letter, RO:III to NSP, dated April 28, 1972
7/ Letter, Bechtel Corporation to NSP, dated January 24, 1972
8/ RO Inquiry Report No. 050-263/72-08
9/ Letter, NSP to Directorate of Licensing, dated June 19, 1972

occasion and during subsequent scrams on July 10 and 21, rod travel stopped at the "02" position (six inches from fully inserted). The scram time recorder indicated the rod to have inserted to 90 percent of its travel in 2.87 seconds on May 23. During the scrams on July 10 and 21, 90 percent insertion times of 3.08 and 3.01 seconds, respectively, were observed. Technical Specifications allow a maximum 90 percent insertion time of 5.0 seconds. Review at the site of the Significant Operating Event Report prepared by the plant staff indicated the observed performance to be due to higher than normal buffer piston seal leakage as determined by "stall flow" measurements. General Electric had indicated similar performance to have been observed at other facilities, caused by higher than normal leakage across the stop piston seals, such that the last buffer hole (0.040" diameter) could not pass all the water, thereby forming a hydraulic lock. The licensee has indicated[10/] his intention to replace this rod drive during the next major cold shutdown outage. A review of the rod drive system conducted by the Region III office in June 1972, did not indicate significant trends in overall rod drive system performance.

11. Scram Discharge Volume Modification

This modification was reported by the licensee in Semiannual Report No. 2 (period ending 12/31/71). The scram discharge volume was enlarged in 1971 to provide a volume of 3.34 gallons per drive mechanism. This allows sufficient space for collection of the water discharged from above the stop piston during a scram and maximum expected seal leakage flow. This modification was performed based on a recommendation from GE which resulted from excessive seal leakage observed at another facility in early 1971. A safety analysis was performed, and the modification was reviewed and approved by the Operations Committee. The affected portions of the scram discharge volume were dye-checked, radiographed, and hydrostatically tested following the modification.

12. High Steam Flow Switch Sensing Line Snubbers

The installation of snubbers in the sensing line for high steam flow differential pressure switches was described in the licensee's Semiannual Report No. 2 (period ending 12/31/71) as having been performed as a plant modification pursuant to 10 CFR 50.59. General Electric Company recommended installation of the snubbers (and also provided a recommended size) following the observation of spurious

10/ Ibid.

differential pressure variations which resulted in a reactor scram. Following a safety evaluation, the modification was approved by the Operations Committee in June 1971, with the recommendation that periodic response time checks be made to ensure against plugging of the snubbers. Operational checks were performed following installation by the application of a differential pressure to the sensing lines, with switch response times observed to be less than one second (to switch set point equivalent to 144 percent of the full power differential pressure). At the request of the Safety Audit Committee, a procedure for the measurement of response time was also developed. Performance of this procedure in the workshop with a spare differential pressure switch and associated snubbers yielded a response with a time constant of 1.28 seconds as compared to an allowed time constant of 2.5 seconds (corresponds to a trip response time of 0.4 seconds, as assumed by the FSAR).

13. Failure of Main Steam Line Drain Valves to Close

A licensee report[11/] discussed the failure of a main steam line drain valve to close on July 28, 1972. Discussions with plant staff personnel indicated the event to have been as described in the subject report. As stated in the report, a followup inspection revealed 12 contactors to have loose screws. Licensee personnel concluded that the screws had apparently not been sufficiently tightened during initial manufacture. All loose screws were tightened as necessary, which should prevent recurrence, since lockwashers are provided.

14. MSIV Reset Switches

During a previous inspection [12/] the licensee indicated an intention to install a second MSIV reset switch during an outage in May 1972, such that one switch would reset the inboard and one switch would reset the outboard MSIV's. The inspector verified by observation of the control panel that the intended installation had been completed.

15. Fuel Pool Siphoning

The potential for inadvertent siphoning of the fuel pool was discussed with a plant staff representative during the inspection and a subsequent telephone conversation. Discussion with the licensee representative and review of a piping diagram of the fuel pool system indicated

11/ Letter, NSP to Directorate of Licensing, dated 8/28/72
12/ RO Inspection Report No. 050-263/72-03

that pump suction line configuration precludes the possibility of inadvertent siphoning of the fuel pool.

16. Vane Type Flow Switches

Licensee representatives indicated plans during a previous inspection[13/] to inspect vane type flow switches installed in the Monticello plant. An internal memorandum viewed during the inspection and discussions with plant staff personnel indicated that the flow switches installed in the Monticello plant are manufactured by Power Engineering and Equipment Company, Inc. A total of 20 vane type flow switches are in use in the Monticello plant, only eight of which are installed on lines which discharge either directly or indirectly to the reactor vessel. Various actions are being considered by plant personnel which will result in the removal of all vane type flow switches and their replacement with alternate means of flow indication. First priority was being given to those switches installed in lines which directly or indirectly discharge to the reactor:

a. RHR Pump Minimum Flow Control. A recent licensee report[14/] provided details of an inspection of the four vane type flow switches which provide minimum flow control for the four RHR pumps (a pump protection feature). This report stated that two of the four switches had encountered failure. Following the inspection, one-inch triangular shaped paddle pieces were left on all four of the switches, with the switches recalibrated to perform the intended function using the smaller paddle piece. A safety evaluation of the occurrence was also provided in the licensee's report.

   An internal memorandum indicated that General Electric has recommended modification of the system to utilize flow signals derived from the LPCI loop flow elements. These elements indicate total loop flow (from two pumps), and GE estimated that the minimum reliable setpoint would be approximately 20 percent of the total flow of both pumps (1600 gpm). In the containment cooling mode, no flow monitoring from this instrument would be available, and the recirculation valves would be left open to provide low flow protection. The memorandum indicated that consideration was also being given

13/ RO Inspection Report No. 050-263/72-02
14/ Letter, NSP to Directorate of Licensing, dated 9/26/72

by plant personnel to the use of a flow element and an associated differential pressure switch for each pump, such that each pump would be provided with individual minimum flow protection. This would permit the minimum flow setting to be approximately 600 gpm, which would permit finer control of cooldown rates during the shutdown cooling mode of RHR system operation.

b. HPCI Cooling Water Low Flow Alarm. The licensee's report[15/] also stated that the flow switch associated with the HPCI cooling water low flow alarm had been inspected, showed no signs of failure, and had been temporarily reinstalled. The licensee concurs with a GE recommendation that it be removed from the system since the alarm is redundant to (a) the gland seal condenser high pressure alarm, and (B) the turbine lube oil cooler effluent high temperature alarm.

c. Reactor Water Cleanup System Low Flow Pump Trip. One vane type flow switch is provided in this system for the protection of each of the two cleanup system pumps (causes the trip to pump in the event of a low flow condition). An internal memorandum indicated that both of these switches are installed upstream of the cleanup system demineralizer, such that any pieces which may have resulted from a vane failure would not pass beyond the demineralizer inlet. A bypass line around the demineralizer has not been used since operation commenced, and will not be used until integrity of these switches is verified. Cognizant staff personnel concur with a GE recommendation that the flow switches be replaced with Venturi flow elements and differential pressure switches. This will necessitate an increase of the minimum flow setpoint from 30 to 50 gpm, although this should pose no problem in view of the normal 80 gpm pump output.

d. Standby Liquid Control System Flow Alarm. This alarm is installed to indicate actuation of the standby liquid control system. The associated flow switch will be inspected and replaced with an alternative mode of indication during the Spring 1973 refueling outage, since the system must be made inoperable. (This switch is welded in place.) A safety analysis viewed by the inspector considered failure of this switch highly unlikely since it has not been subjected to flow conditions. Although GE has not yet presented a

15/ Ibid.

recommendation for a replacement switch, a flow element with differential pressure switch was considered by cognizant plant staff personnel to be suitable.

17. Drywell Instrument Air Supply Modifications

The licensee had indicated during discussions prior to the inspection an intention to modify the portion of the instrument air system supplying control air to main steam isolation valves and other components within the drywell, since the small air leakage (2-3 cfm) of instrument air into the drywell necessitates periodic purging of the drywell and torus with nitrogen to keep the oxygen concentration within Technical Specification limits of 5 percent. Two different modifications are planned. Discussions with plant staff personnel during the inspection indicated the status of the modifications to be as follows:

a. An interim modification will provide a supply of pressurized nitrogen from the nitrogen makeup system to the drywell instrument air piping. The original instrument air system will serve as a backup supply in the event of improper nitrogen system pressure. The 10 CFR 50.59 change documentation included a safety evaluation which concluded that the change was not an unreviewed safety question, since reliability of the instrument air supply to the drywell would be significantly improved. The modification has been approved by the Operations Committee, and all materials required for its completion are onsite. Work on portions of the system up to the point of tying in to the existing instrument air piping is in progress. A plant staff representative stated that the modified system would be connected during a future outage of two - three day's duration (may be the refueling outage).

b. A recirculating compressor system will be installed as a normal drywell instrument air supply. A design study for the installation of this system has been completed by Bechtel Corporation. Licensee representatives stated that the system will include two compressors (one operating and one backup) which will draw gases from the primary containment atmosphere and discharge them through suitable filter equipment to the drywell instrument air piping. Licensee representatives expressed an intent to install the system during the refueling outage.

18. Vacuum Breaker Rubber Seats

The licensee indicated plans during a previous inspection[16/] to inspect the rubber seats on the primary containment-to-reactor building vacuum breakers during a future outage. Plant staff personnel contacted during the inspection indicated that this inspection would be performed during the 1973 refueling outage.

19. Uninterruptible AC Power System

An uninterruptible power system installed in the Monticello plant provides a power source for several important plant instrument and control functions. Power to the uninterruptible system is obtained from three sources: (1) a motor control center fed* by No. 11 diesel generator, (2) a motor control center fed* by No. 12 diesel generator, and (3) an AC generator normally driven by an AC motor, with a 250 V DC motor provided as a backup. The AC-DC motor-generator supplies power through a manual transfer switch to distribution panel Y10, which supplies several control functions, the most significant of which are considered to be the HPCI and RCIC flow controllers. At the time of the inspection, the AC-DC motor-generator was out of service, with panel Y10 receiving electrical power from its alternate source, a motor control center. In the event of a loss of offsite power, this configuration makes HPCI and RCIC operation dependent upon starting of No. 11 diesel-generator.

Discussions with plant staff personnel indicated that the motor-generator unit had been placed out of service because of DC motor speed control problems. Corrective action taken by the licensee included rerouting a tachometer cable which had been previously located in the same cable tray as several power cables and making other minor adjustments to the speed control system.

The existing operating configuration of the uninterruptible power system is not in violation of Technical Specifications, and potential loss of power to the system is treated in the FSAR (Section VIII-7.2).

The inspector indicated that routine operation of the plant with this electrical configuration makes HPCI operation contingent upon operation of No. 11 diesel generator in the event of a loss of offsite power. Licensee representatives agreed, stated that the AC-DC motor-generator unit was now considered to be operable and would be placed into service during the next outage of a few days' duration.

* In the event of loss of off-site power

16/ RO Inspection Report No. 050-263/72-02

20. Tape on Drywell Pressure Sensing Taps

Following the previous inspection, the licensee was informed[17/] that he was considered to have been in noncompliance with Technical Specifications in that two of the four drywell pressure sensing taps were found to have been covered with tape. The licensee's response[18/] stated that all drywell and torus pressure sensing and sampling lines had been inspected for flow restrictions, and that warning tags would be placed on the sensing line taps in the torus and drywell during a future outage. Discussions with plant staff personnel during this inspection indicated that these warning tags had not yet been placed, since the drywell has not been de-inerted since the July 21 scram occurrence. It was indicated that these tags would be fabricated in the near future and installed the next time the primary containment is de-inerted.

21. Secondary Containment Integrity

The inspector inquired about the possibility of failure of reactor building ventilation system isolation dampers to isolate when a fan breaker is racked out. Monticello staff personnel had recently reviewed this question based upon reports received from the AEC clearing house, and stated that the system arrangement at Monticello is such that:

a. Each supply fan damper is controlled by "a" contacts on its associated ventilation fan motor such that the damper opens and closes automatically when the fan is started and stopped, and,

b. The dampers fail closed on loss of either electrical power or air pressure. It was, therefore, concluded that a similar occurrence would not be expected at Monticello.

22. HPCI Auto Isolation

Previous performance of the HPCI auto-isolation system was discussed in an inpsection report[19/] and a licensee report to DRL.[20/] A representative of the licensee's corporate office stated that plans

17/ Letter, RO:III to NSP, dated 7/31/72
18/ Letter, NSP to RO:III, dated 8/22/72
19/ CO Inspection Report No. 050-263/71-13
20/ Letter, NSP to DRL, dated 9/28/71

are underway to install a Venturi flow indicator in the steam line during the Spring 1973 plant outage, although this is dependent upon hydrostatic testing requirements.

23. Torus Suction Ring Header

An inpsection of the torus performed on May 23, 1972, subsequent to a reactor scram revealed that a nut had backed off one of the torus suction header vertical support bolts and that the bolt was bowed, with crushed threads in the load bearing area. The entire 20" torus suction ring header support structure was subsequently inspected on June 5-8, 1972, and results were reported[21/] to Region III. Plant staff personnel stated during the inspection that their evaluation and corrective action had recently been completed and that a followup report should be forthcoming in the near future.

24. Failure of HPCI Isolation Valve to Close

Region III[22/] and licensee[23/] reports discuss the failure of the HPCI outboard steam isolation valve (MO-2035) to close during a routine surveillance test conducted on April 3, 1972. The licensee's report indicated that the valve operator had failed to engage after the valve had been manually backseated to prevent steam leakage. This report also stated that manual operations of the valve will disengage the motor gears, although they should automatically re-engage when the motor is energized. Discussions with a member of the plant staff and review of available documents indicated that: (1) the valve was repacked during a brief outage in May 1972, (2) efforts to reproduce the occurrence following "very hard" manual backseating during this outage were unsuccessful, (3) a memorandum has been issued to plant operators prohibiting manual backseating and requiring electrical operation of any motor operated valve after its having been manually operated, and (4) inspection of the valve operator is planned during the 1973 refueling outage. A letter received by NSP from the valve manufacturer (Limitorque Corporation) stated that a helical friction coil within the operator (designed to re-engage the motor following a manual operation) may occasionally fail to function due to the presence of dirt, although this is uncommon. To prevent occurrence of this difficutly with other

21/ Letter, NSP to RO:III, dated 7/5/72
22/ RO Inquiry Report No. 050-263/72-05
23/ Letter, NSP to Directorate of Licensing, dated 4/25/72

similar valves, plant management representatives indicated an intention to incorporate the valve operating instructions (prohibiting manual backseating and requiring electrical operation following manual operation) into the facility's operating procedures.

25. RCIC Isolation Switch, Improper Trip Point

The inspector reviewed the licensee's Significant Operating Event Report relating to the discovery of an RCIC isolation switch trip point drift during a surveillance test on March 1, 1972. Review of the condition, which also included discussions with a licensee representative and viewing of the internals of a similar switch, indicated the condition to have been as described in Region III[24/] and licensee[25/] reports. The licensee representative stated that other switches of similar construction which were susceptible to the same difficulty had been inspected and found satisfactory.

26. RHR Service Water Pump Performance

On June 14, 1972, both service water pumps in loop "A" of the residual heat removal service water (RHRSW) system delivered a head of 530 feet at the rated flow of 3500 gpm. A 550 foot head is required by Technical Specifications. The licensee's investigation and corrective action were reported[26/] to the Directorate of Licensing.

To obtain additional information on RHRSW pump performance, No. 13 RHRSW pump was subsequently removed from the system and returned to the manufacturer (Worthington) for inspection and testing. Results of the tests agreed well with the performance data originally provided with the pumps, although higher shutoff head and lower maximum capacity had been observed from the pump during operation at Monticello. Following reinstallation of No. 13 RHRSW pump, system checks were performed which indicated a strong dependence of pump shutoff head upon bowl to impeller clearance. All four pumps were adjusted to a clearance of 1/8", at which point the 13 RHRSW pump shutoff head agreed well with the manufacturer's performance data, although heads of the other three pumps were 45 to 100 feet higher. An internal plant memorandum stated, however, that test results have

24/ CO Inquiry Report No. 050-263/72-03
25/ Letter, NSP to DRL, dated 3/15/72
26/ Letter, NSP to Directorate of Licensing, dated 7/3/72

shown the bowl-to-impeller clearance adjustment to have little effect on the pump performance at rated flow. It indicated that the following additional steps are contemplated to eliminate any remaining uncertainty in the presently installed flow metering assembly:

a. Installation of a filler ring downstream of the No. [illegible] RHRSW flow orifice similar to that installed[27] upstream of the orifice.

b. Drilling of a 0.25 inch hole through the orifice plate flush with the inside upper surface of the piping to assure that air is not being trapped on the upstream side of the orifice. Such a hole is prescribed by ASME Code for orifices located in non-vertical piping sections.

c. Utilization of a second set of orifice differential pressure sensing taps in addition to the presently used taps to eliminate possible effects of swirling action of the water upon the flow measurements.

Consideration is also being given to the procurement of a calibrated orifice from the University of Minnesota for use in verifying the accuracy of the installed flow metering system. The licensee has also submitted a Technical Specification change request[28] which would allow additional operating margin for the RHRSW pumps while still maintaining the required differential pressure between the tube and shell sides of the RHR system heat exchanger.

27. Slow Diesel Generator Start

A plant staff representative reported during the inspection that a "start failure" alarm had been received during a routine surveillance test of No. 12 diesel generator on September 15, although the diesel had actually started. Investigation indicated the cause to be a partially plugged metering orifice in the air relay which opens a valve to admit air to the air motors after the starting pinions on the No. 1 starting system are engaged. The No. 2 starting system functioned normally. The No. 1 starting system was tested satisfactorily following inspection and cleaning of the metering orifices. Orifices in other starting systems were inspected with satisfactory results. The licensee will make a 30-day report to the Directorate of Licensing.

27/ Ibid.
28/ Letter, NSP to Directorate of Licensing, dated 7/24/72

28. Diesel Generator Backup Start Relays

Review of diesel generator start circuitry with a licensee representative indicated that, in response to information obtained from the AEC clearing house, plant staff personnel had conducted a review of the Monticello diesel generator starting system. Review by the inspector of an internal memorandum indicated that the Monticello diesel generator starting circuits use Square D Class 9050 pneumatic timing relays. Each of the Monticello diesels has 13 Class 9050 relays in its starting logic. Eight of these on each diesel provide protection for various modes of engine start failure by supplying a start signal to the second starting system; i.e., their operation is not required during a normal start. The remaining five relays operate alarms or perform non-essential functions. The memorandum further stated that operation of the second starting system had occurred on occasions because of failure of the first starter motor pinion to engage the flywheel ring gear, and that in all cases proper operation of the backup starting system had occurred.

A plant staff representative stated that a review of the diesel starting system is presently in progress to provide a procedure for testing of the logic and all relays in the diesel starting circuitry, and to determine the estimated outage time required. A licensee representative stated that the complete starting system would be checked for proper operation during the Spring 1973 refueling outage and at least during each refueling outage thereafter; it was indicated that more frequent tests would be conducted if results of the initial tests and the amount of diesel outage required for their conduct indicate a shorter frequency to be desirable and feas.

29. Airborne Activity in Drywell

Review of significant operating event reports indicated that on May 14, 1972, a small increase had been noted in airborne radioactivity levels within the drywell while three maintenance workers were performing maintenance without the benefit of respiratory protection equipment. The report stated that personnel monitoring in the form of smears, urine samples, and thyroid counts was conducted, resulting in a determination that the three individuals had received an estimated 2.6, 4.4, and 6.9 percent of their maximum permissible body burdens. It was also concluded by the licensee that the maximum activity to which the individuals were exposed was approximately 1.2×10^{-9} uCi/cc, and that none of the individuals was exposed to airborne radioactivity levels in excess of 10 CFR 20 limits.

The licensee concluded that the activity had resulted from the head vent being left open on the reactor vessel while the reactor water was slowly heating up, forcing noncondensible radioactive gases through the head vent into the drywell.

The occurrence investigator recommended that the drywell head vent be kept closed whenever personnel are working within the drywell unless approved procedures provide for opening the primary system while the drywell is occupied. The licensee plans to incorporate such a requirement into plant operating procedures.

30. Loss of Stack Sampling Flow

The loss of stack monitoring sample flow which occurred on June 20, 1972, was described in Region III[29/] and licensee[30/] reports. Review of the Abnormal Occurrence Report prepared by the plant staff and discussion with licensee representatives verified the account of the occurrence to be as provided in these reports, summarized as follows:

a. While operating at rated power at approximately 0430 on June 20, 1972, the stack gas monitor was observed to show a gradual decrease in indicated release rate. Purging of the monitoring chamber and performance of a source check from the control room indicated a loss of sampling flow (subsequent review of the recorder traces showed the sample pump to have tripped at about 0315 without initiating the low flow alarm).

b. At approximately 0730 an operator went to the stack to investigate and found the sample pump to have tripped. Sampling flow was restored at this time by restarting the pump.

c. Failure of the low flow annunciator circuit resulted from a poor connection between the annunciator card and the annunciator panel.

d. No change was observed in air ejector offgas release rate during the period.

Section 3.8.A of the Monticello Technical Specifications states as a limiting condition for operation that "Radioactive gases released

29/ RO Inquiry Report No. 050-263/72-10
30/ Letter, NSP to Directorate of Licensing, dated 6/30/72

from the stack . . . shall be continuously monitored." The licensee was considered to be in violation of this Technical Specification in that timely restoration of sampling flow was not made.

Discussion with plant staff personnel indicated that the stack monitor sample pumps are of a carbon vane construction. Review of the occurrence by plant personnel indicated that the operating pump may have stopped as a result of overheating due to close tolerances and poor lubrication within the pump. Plans are being made to improve the lubrication system for the two sample pumps during the 1973 refueling outage. Clearances within the pump have also been increased slightly to reduce the possibility of overheating (proper sampling flow rate was verified following the increase in pump tolerances). The loose connection in the low flow annunciator circuit was corrected, and a new surveillance procedure has been established which provides for alteration of operating pumps on a weekly interval, with an accompanying operational test of the low flow annunciator circuit. A plant staff representative also stated that the event had been discussed with operating personnel.

31. Gaseous Activity Release Rate

The Region III office was informed[31/] by telephone on August 18, 1972, that the stack activity release rate had increased from a nominal level of 27,000 uCi/sec the previous day to a stable release rate of approximately 69,000 uCi/sec. The increased activity release rate was attributed by General Electric to fuel clad degradation caused by hydriding. Review of release records and discussions with plant staff personnel during the inspection indicated that the weekly average release rate had increased to a maximum of 60,000 uCi/sec, followed by a gradual decrease to approximately 40,000 uCi/sec near the end of September. The rod drive control system was returned to the "A" rod withdrawal sequence following a reactor shutdown in late September. Release rates as observed during the inspection and reported in subsequent telephone conversations with staff personnel indicated the activity release rate with the "A" sequence to be reasonably stable within the range of 45,000 - 50,000 uCi/sec. This is considerably below the annual average release limit established by Technical Specifications, Section 3.8.A.

31/ RO Inquiry Report No. 050-263/72-14

32. Off-gas System Holdup Time

An examination of observed off-gas system holdup times was made during the inspection. Review of testing records indicated that the off-gas holdup time was last determined on May 18, 1972, with a resulting holdup time of 71.5 minutes (as compared to a minimum of 30 minutes indicated by the FSAR). Staff personnel indicated that condenser air in-leakage tests are performed monthly and have shown no noticeable variation since that time. The holdup time determination was made by counting Kr^{87} at the air ejector and at the stack while using Xe^{133} at both locations for standardization of count rates.

33. Minnesota Pollution Control Agency Relationships

Discussion with licensee representatives indicated that NSP and the PCA have discussed provisions for three monitor points to be covered by the proposed alarm system. Consideration is being given to the following: (1) stack activity release rate, (2) liquid radwaste effluent monitor, and (3) discharge canal monitor. A signal from each of the existing monitors at these locations would be transmitted to an indicating trip unit located onsite. Signal transmittal and alarm function designs are being finalized.

The licensee representative also stated that the possibility of an onsite PCA inspector is still active, although no definite plans have been made.

34. Quality Assurance Manual

Preparation of the Operating Quality Assurance Program for the Monticello plant is continuing. A representative of the Nuclear Services Corporation, which is assisting in preparation of the program, was onsite during the inspection. Discussions with a licensee representative and review of portions of the program which are currently being prepared indicated that the completed program will include three levels of documents:

a. Operational Quality Assurance Program. This is a broad and general policy document concerning controls to be applied in the various areas associated with plant operation. Among other things, it defines applicability and scope of the program, and was considered by the licensee's representative to be in nearly final form.

b. Operational Quality Assurance Manual. This is a collection of documents which govern those activities of personnel which effect quality. They provide more specific guidelines in the implementation of the Quality Assurance program, and were considered to be in an intermediate stage of preparation.

c. Instructions. These will be the working level documents which implement the guidance provided by the program document and the QA manual. Examples are the Operations Manual, a recently written instruction on the preparation of a safety analysis, surveillance testing procedures, and maintenance procedures. The QA Manual is being written in such a manner that requirements for major revisions of existing instructions will be minimized.

Licensee representatives are giving priority effort to document the complete program as soon as possible.