U. S. ATOMIC ENERGY COMMISSION
REGION III
DIVISION OF COMPLIANCE

Report of Inspection

CO Report No. 263/70-20

Licensee:	Northern States Power Company Monticello License No. DPR-22 Category B
Dates of Inspection:	December 10 and 11, 1970
Dates of Previous Inspection:	November 30 and December 3, 1970
Inspected By: E. L. Jordan	Reactor Inspector 1-11-71
Reviewed By: for C. D. Feierabend	Responsible Reactor Inspector 1-12-71
for H. D. Thornburg	Senior Reactor Inspector 1-12-71
Proprietary Information:	None

SCOPE

Type:	Boiling Water Reactor
Power Level:	1670 Mwt (Low Power License: 5 Mwt)
Location:	Monticello, Minnesota
Type of Inspection:	Announced

The purpose of the inspection was to review preparations by the licensee for the initial critical test, to witness initial criticality, and the start of open vessel testing.

SUMMARY

Safety Items - None.

Noncompliance Items - None.

Unusual Occurrences - None.

Status of Previously Reported Problems - None.

Other Significant Items - The Monticello Reactor achieved initial criticality at 9:47 p.m. on December 10, 1970. Operations were found to have been conducted in accordance with established procedures. (Section F.)

Management Interview - Persons attending the management interview were:

C. Larson, Plant Superintendent
G. Jacobson, Plant Results Engineer
R. Hobson, GE Principal Test Design and Analysis Engineer
E. Jordan, CO:III Reactor Inspector

The inspector stated that preparations and conduct of the initial critical test appeared to be in accordance with procedures. The inspector stated that CO has an interest in the results of the core flow tests which have been proposed for Monticello. Mr. Hobson stated that the test results would be maintained in the startup test results file and would be available when the test is completed.

DETAILS

A. Persons Contacted

Northern States Power Company (NSP)

C. Larson, Plant Superintendent
M. Clarity, Assistant Plant Superintendent
G. Jacobson, Plant Results Engineer
D. Antony, Test Engineer
W. Hill, Test Engineer

General Electric Company (GE)

J. Miller, Operations Manager
G. Mattey, Operations Superintendent
R. Hobson, Principal Test Design and Analysis Engineer
R. Phillippi, Instrumentation Engineer (On loan from Dresden)

B. Administration and Organization

1. Review of Logs

The inspector reviewed the shift engineers log and the operations log for the week preceding December 10. The logs were

satisfactorily maintained and no problems were indicated that would affect beginning the initial criticality test.

2. Operations Committee

The Operations Committee was found to have reviewed the status of the standby gas treatment system (Section K) and the reactor manual control system (Section F) and determined that both systems were satisfactory for resumption of open vessel testing.

3. Startup Check Sheets

The inspector reviewed the startup check sheets and found them to be satisfactorily completed and signed off.

C. Operations

1. Equipment Malfunctions

The only equipment malfunction observed during the initial criticality testing was the inability to withdraw control rod 14-11 at 8:41 p.m. on December 10. The licensee vented the drive and satisfactory drive operation was obtained. The item is considered resolved.

2. Scrams

One inadvertent scram was received on December 11 due to unintentional over-notching of control rod 26-23 by the operator during a notch worth measurement test. The scram was initiated by IRM No. 15 (non-coincidence mode). The minimum period attained was approximately five seconds (from recorder chart). Maximum reactor power indicated by IRM was less than 60 kwt. A preliminary scram report by the licensee attributed the event to operator error. The licensee was found to have issued additional instructions to the operators in the shift log regarding the events. Satisfactory provisions exist for management review of scram reports.

F. Reactivity Control and Core Physics

1. Initial Criticality

The CO inspector witnessed initial criticality of the Monticello reactor. Criticality was achieved at 9:47 p.m. on December 10,

1970. Rod withdrawal was begun at 7 p.m. Control rod withdrawal was performed in accordance with rod sequence A and criticality was achieved with two groups of rods (30 rods) fully withdrawn. Moderator temperature was 71°F. The period attained on initial criticality was approximately 600 seconds. Withdrawal of the first rod in group 3 to notch 8 resulted in a 200 second period.

The licensee was found to have prepared a supplemental procedure to startup test No. 6. This provided additional test criteria including the predictions for initial criticality. The observed initial criticality, with 30 rods withdrawn, occurred well within the specified range (24 to 34 rods withdrawn). The licensee was found to have calculated the worths of control rods withdrawn in terms of a change in reactivity:

Change in Total Number of Rods Withdrawn	Resulting Change in % $\Delta k/k$
21 to 24	0.50
25 to 30	0.50
31 to 34	0.75
35 to 38	0.50

Criticality was achieved with a period of 176 seconds on rod sequence B with 29 rods (all of group 2) fully withdrawn and the first rod in group 3 at notch 10. The difference between sequence A and B is due to geometry of the core since the "open checkerboard" withdrawal pattern (groups 1 and 2) includes 30 and 29 rods, respectively, for sequences A and B. Criticality on sequence B is thus in agreement with sequence A.

2. Instrument Response

The licensee recorded flux response for each control rod withdrawn. Response was not detected for four rods in each sequence. According to Dr. Hobson, this was not unexpected because of the low worth of these rods. All were at or near the periphery. The response at critical was satisfactory.

Instrumentation response appeared to be satisfactory during the test. Signal to noise was found to be more than 100:1. SRM count rate with operational Sb-Be sources and SRM detectors is given below for three core conditions:

SRM Channel No. 11	All Rods Inserted (Shutdown)	Count Rate - cps: End of Group 1 (15 Rods Withdrawn)	Count Rate - cps: End of Group 2 (Initial Critical)
21	41	190	24×10^3
22	44	370	70×10^3
23	35	230	49×10^3
24	32	100	40×10^3

An increase in IRM indication was observed before the reactor was initially critical. IRM indications at a power of 2.7 $\times 10^{-4}$% or 4.5 kw (calculated from SRM indication and sensitivity) were adjusted as given below:

	Channel (Reading Range) 11	12	13	14	15	16	17	18
Initial Reading	48-1	10-2	16-2	62-1	65-1	10-2	13-2	53-1
Adjusted Reading	11.5-3	27-3	27-3	13-3	15-3	25-3	27-3	9-3

Sensitivity was sufficient to assure a trip well below the present license limit of 5 Mwt.

Trip settings of 120 to 125% would result in a maximum power level of from 20 to 60 kw based upon the sensitivity of adjusted IRM's. Differences between IRM channel indication similar to Dresden 2[1] were observed. Calibration of the IRM's is scheduled to be performed during heatup testing.

3. Reactor Manual Control Systems (RMCS)

Investigation by the licensee of the ability to select two adjacent switches (49-19 and 46-19) in the "refuel" mode[2] disclosed a bad relay in control rod 42-19 rod select circuit. The relay was made by L. P. Clark and Co., designated MR4MC-1023. The defect was in a normally open contact which was jammed closed by breakage of the encapsulating glass vial.

1/ CO Report No. 237/70-5, Section II.B.6.
2/ CO Report No. 263/70-19.

The licensee was found to have replaced the relay and the original trouble could not be duplicated.

During troubleshooting operations it was found possible to select two control rods by performing a series of abnormal switching operations. This included selecting two rods in rapid sequence in conjunction with operation of the power switch. Since this condition can only be performed by abnormal operation and only in the "prefuel" mode, the Operations Committee did not consider the matter to be of safety significance.

4. Core Flow Distribution Test - Core Flux Asymmetry

Discussions with Dr. Hobson and Mr. Jacobson disclosed that GE was proposing to perform an engineering test with the Monticello reactor to measure core flow distribution for the purpose of verifying GE assumptions concerning the cause of an apparent core flux distribution anomaly observed at Dresden 2. The test will be restricted to cold, depressurized, conditions with the reactor in the shutdown condition. The inspector stated that CO was interested in the results of the test.

K. Containment

Standby Gas Treatment System (SGTS)

The licensee was found to have started the SGTS daily since October 1970 except for one day during which modifications prevented testing. No failures were reported during this interval. In addition to the modifications identified as complete in a previous inspection report,[3/] the following modifications were found to have been completed:

1. Installation and checkout of two 12 kw electric room heaters and a 1.2 kw cubicle (B train) heater.

2. Modification of cooling air inlet to permit a mix of outside air and turbine building air.

3. Readjustment of low flow transfer setting to prevent accidental transfer due to flow signal noise. Transfer is now set to occur at a measured flow of 3300 cfm which corresponds to a $\Delta p \geq 0.25"$ of water between the reactor building and outdoors.

3/ CO Report No. 263/70-19.

The remainder of the planned modifications and the performance tests as described in CO Report No. 263/70-18 are scheduled to be completed before heatup testing is begun. Mr. Larson stated that the daily testing will be continued until the modifications are completed and the system performance has been demonstrated. The inspector reviewed the results of the December 10 surveillance test and found that the SGTS had satisfied the requirements of the test.

O. Fuel Handling

The inspector found that the change to operational Sb-Be sources, the necessary fuel transfer, and the change to operational source range monitors had been completed on December 9 in accordance with startup test procedure No. 7.