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MONTICELLO NUCLEAR GENERATING PLANT

CYCLE 2 STARTUP REPORT

JULY, 1973

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA

5466

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MONTICELLO NUCLEAR GENERATING PLANT

CYCLE 2 STARTUP REPORT

I. Introduction

On March 2, 1973 the Monticello generator was taken off line and the reactor brought to a cold shutdown condition for the first refueling outage. A turbine overhaul and other maintenance projects were also completed during the outage. On May 19, 1973 the plant was returned to commercial operation following a short period of startup testing. References 1 and 2 (see Section IX) discussed the planned reload core; this report summarizes the final core loading and reports test results confirming calculations presented in those earlier references.

This startup report is submitted in compliance with Technical Specification 6.7.A.1. Specification 6.7.C.6 requires a "Summary Status of Fuel" report after the second refueling outage. While the latter report is not required at this time, the subject is discussed in this startup report since fuel inspection was a major part of the outage. Other pertinent outage activities are also briefly discussed in this report.

II. Summary

After refueling, the core consists of 459 fuel elements remaining from cycle 1, 5 reconstituted cycle 1 fuel assemblies and 20 new R-1 type fuel assemblies (as described in reference 1). Supplemental reactivity control is provided by 44 of the original 216 temporary control poison curtains. Startup testing demonstrated sufficient margin to maintain the reactor shutdown with the strongest control rod fully withdrawn from the core. This verified analytical calculations as did the observed cold and hot operating critical control rod patterns.

The core was sipped at the end of cycle 1. Inspection of the 25 fuel assemblies rejected because of leakage showed failures to be primarily due to the hydride mechanism. Removal of failed fuel improved the off-gas level significantly.

A bolt was suspected to have been lost in the reactor vessel; it was determined that the bolt did not lie in a location having potential safety implications. Plant changes included the installation of a venturi flow sensing element in the HPCI steam supply line and replacement of the spool valve operators on the main steamline isolation valves with an improved operator design.

Cycle 2 is designed for 3500 MWD/STU exposure; a coastdown in power is expected at the end of cycle to extend the cycle to meet power system requirements.

III. Fuel Performance

The fuel inspection program commenced immediately after the reactor shutdown. The full core was wet-sipped in the reactor vessel; 25 fuel assemblies were determined to have leaking pins. These assemblies were moved to the spent fuel pool where they were remotely disassembled and the individual pins non-destructively tested. A total of 163 individual fuel pins (0.7% of the total pins in the core) were rejected because of ultrasonic indication of water in the cladding or eddy current indication of cladding penetration greater than or equal to 35% of the cladding thickness. Evaluation of data collected in these tests indicated that the predominent failure mechanism was hydriding.

Of the 25 rejected fuel assemblies, 12 were reconstituted. Sound pins scavanged from the 13 remaining assemblies having similar exposures and enrichments were retested and used as replacements for failed pins in the assemblies being reconstituted. Five reconstituted assemblies are presently in the core while seven remain in the fuel pool for insertion in a later cycle. Success of the operation was demonstrated in that stack off-gas levels during equilibrium, full power operation dropped from a level of about 50,000 microcuries per second prior to the outage to about 10,000 immediately following the outage.

IV. Reactor Core Configuration

The Monticello reload submittal (reference 1) discussed a reference core in which 28 of the 484 initial core fuel assemblies were replaced with R-1 type reload assemblies and all but 64 of the original 216 temporary control curtains were removed. This reference core was based on calculations done prior to the outage assuming a preoutage core average exposure of 6800 MWD/T. As reported in reference 2, a preoutage exposure of 7115 MWD/T was achieved. Revised calculations at that exposure were done for a core with 464 original fuel assemblies, 20 R-1 reload assemblies and 44 curtains. That core, which represents the current Monticello core configuration, is shown in Figure 1. The average core exposure at the beginning of cycle 2 was 6802 MWD/T. The 172 curtains removed along with 13 rejected fuel assemblies are presently stored in the spent fuel pool awaiting shipment offsite; seven reconstituted assemblies are in the pool awaiting insertion in the core in a subsequent cycle.

Fuel was shuffled between symmetric core locations to facilitate fuel handling operations. An attempt was made to closely match the initial exposure when replacing the 25 leaking assemblies with the five reconstituted assemblies and the 20 assemblies in the designated R-1 locations. No attempt was made to shuffle fuel so as to alter the burnup distribution. With the exception of the 20 new R-1 assemblies, the exposure distribution was essentially unchanged.

Calculations for the shutdown margin reported in references 1 and 2 were based on the target end of cycle 1 exposure shape. During the outage, after determining the final core configuration, the shutdown margin was recalculated using the actual exposure shape existing at the beginning of cycle 2. It was found that at the most reactive time in the cycle (beginning of cycle with equilibrium samarium) the shutdown margin with the most reactive control rod fully withdrawn was .01 delta k which is the design target stated in the FSAR and used by General Electric. This is well in excess of the Technical Specification limit of .0025 delta k. Tests reported in the next section demonstrate sufficient shutdown margin for cycle 2.

V. Startup Testing

The objective of the startup testing was to verify that the core was reassembled properly and to confirm the calculations used in the design of the reload core. No attempt was made to duplicate the extensive dynamic startup testing program undertaken during the initial plant startup.

Core preparations for startup testing included certain calibration and verification steps. Following control rod drive maintenance and fuel movement, all control rods were stroked; scram timing was checked and adjusted as necessary in the cold condition. Six control rods were scrammed in the hot condition to verify the preestablished correlation between hot and cold scram times. After all in-core work was completed, the core was scanned by underwater television for proper fuel assembly orientation, curtain location, fuel assembly seating and general core condition. The individual fuel assembly serial numbers were checked to confirm fuel accountability records. The core was verified to be assembled as shown in Figure 1.

Startup testing was done to confirm core calculations which were made by both General Electric and Northern States Power. Shutdown margin demonstrations were done at cold conditions with the most reactive control rod withdrawn. Rod 06-31 was calculated to be the strongest rod; its worth, with all other rods inserted, was calculated to be .0334 delta k. Rods adjacent to the strongest rod had calculated worths of the order of .0315 delta k; control rods of this strength have very large individual notch worths over the range of interest. Therefore, a diagonally adjacent rod was withdrawn to demonstrate sufficient shutdown margin. The samarium concentration at the time of testing was greater than the normal operating level by an equivalent of .0024 delta k. This, in addition to the .0025 delta k shutdown margin required by Technical Specifications, meant that .0049 delta k shutdown margin had to be demonstrated. A total of .0081 delta k shutdown margin was demonstrated by fully withdrawing rod 06-31 and withdrawing rod 10-27

to notch position 08. The core was verified to be subcritical in this condition. Subcriticality was also verified for rods in symmetric locations where rod worths were essentially the same. Shutdown margin demonstrations were also successfully performed with rods 06-27 and 26-27 fully withdrawn; these were calculated to be the rods of next highest worth to the symmetric rods discussed above.

Criticality was achieved at 80°F on the 17th rod withdrawn in control rod sequence A and on the 14th rod withdrawn in sequence B. These rod patterns were well within the acceptance criteria of calculations. While critical, the critical rod configuration was rotated within constraints of the control rod sequence, demonstrating the symmetry of the core.

During the heatup, rods had to be continually withdrawn to increase power. This demonstrated a negative moderator temperature coefficient at all temperatures. Upon reaching rated power, the control rod inventory was in good agreement with that predicted by calculations.

VI. Related Refueling Outage Operations

During refueling the lower portion of the source holder of the most centrally located of the five neutron sources was bent. The section of the holder housing the antimony pins was not damaged. A safety evaluation determined that the source was not necessary; the shutdown count rate on startup instrumentation was far in excess of the minimum Technical Specification count rate requirement with the source removed.

During the outage vibration sensors, which had been semi-permanently installed for the initial startup test program vessel internal vibration monitoring, were removed. On completion, the inventory of parts showed one vibration instrument lead retainer bolt (5/8 inch diameter by 2 inches long) unaccounted for. It was suspected that if the bolt was in fact lost in the vessel, that it would have dropped onto the jet pump shelf between the shroud and the vessel wall. However, it could have dropped into the core region. A safety analysis showed that if the bolt had dropped into the core support piece of a tightly orificed fuel assembly, one particular orientation of the bolt might result in insufficient coolant flow and subsequent fuel failure; all other possible locations presented no safety problems. The bolt could only have entered such a location if the respective fuel assemblies were removed from the core at the time the bolt was dropped. Fuel assemblies were later removed from locations which were exposed during the time the bolt was handled; the fuel support pieces were then inspected. Other accessible potential resting places for the bolt were also inspected. The bolt was not found. It was concluded that if the bolt is in the reactor vessel that it is not in a location having potential safety implications.

On May 19, 1973 the ascent to rated power commenced.

VII. Major Plant Modifications

Among the numerous projects completed during the outage were two discussed in previous correspondence (references 3, 4 and 5).

A venturi was installed in the HPCI steam supply line between the existing elbow tap flow sensor and the inboard isolation valve, MO-2034. This venturi will be used in place of the elbow taps to initiate an isolation on high HPCI steam flow. As constructed, the flow through the B main steamline at greater than 50% or rated power previously affected the elbow tap flow signal in the conservative direction; that is, the sensor indicated excessive HPCI steam flow at high reactor power levels resulting in inadvertant HPCI isolation during surveillance tests. Startup testing verified proper operation and calibration of the HPCI system as modified. Additional information on this modification will be included in the Monticello Semi-Annual Operating Report.

Spool-type pilot valves on the main steamline isolation valve operators have been a source of problems due to their close tolerances and susceptibility to contaminant particles in supply air and ambient temperature effects. Spool valves in all MSIV pneumatic control systems were replaced with poppet-type air operated pilot valves which are expected to show superior performance. This modification is reported in detail in reference 5.

VIII. Expected Cycle 2 Operation

The design exposure increment for cycle 2 is 3500 MWD/T to the all-rods-out condition. Because of system load demand considerations, a power coastdown is planned to allow operation beyond that exposure while delaying the subsequent outage until late February, 1974. Previous correspondence (reference 6) discusses exposure effects on the scram reactivity curve used in the transient analysis. Recent calculations provided by General Electric indicate that the scram reactivity feedback available in cycle 2 will be greater than that used in reference 6 for the first 2250 MWD/T of cycle 2. At that time, however, withdrawal of additional control rods from the core will result in insufficient scram reactivity based on the most conservative assumed transient conditions. For this reason, it is anticipated that the end of cycle coastdown will commence before all rods are withdrawn from the core. Analyses show that after coasting down to a somewhat reduced power level, with the 2250 MWD/T rod pattern, additional control rods may be withdrawn until reaching the all-rods-out condition. After reaching the all-rods-out condition, continued coastdown is planned until the scheduled outage date.

IX. References

1. L O Mayer (NSP) to A Giambusso (USAEC), "Request for Authorization to Operate with Reload Fuel in the Core," dated February 20, 1973

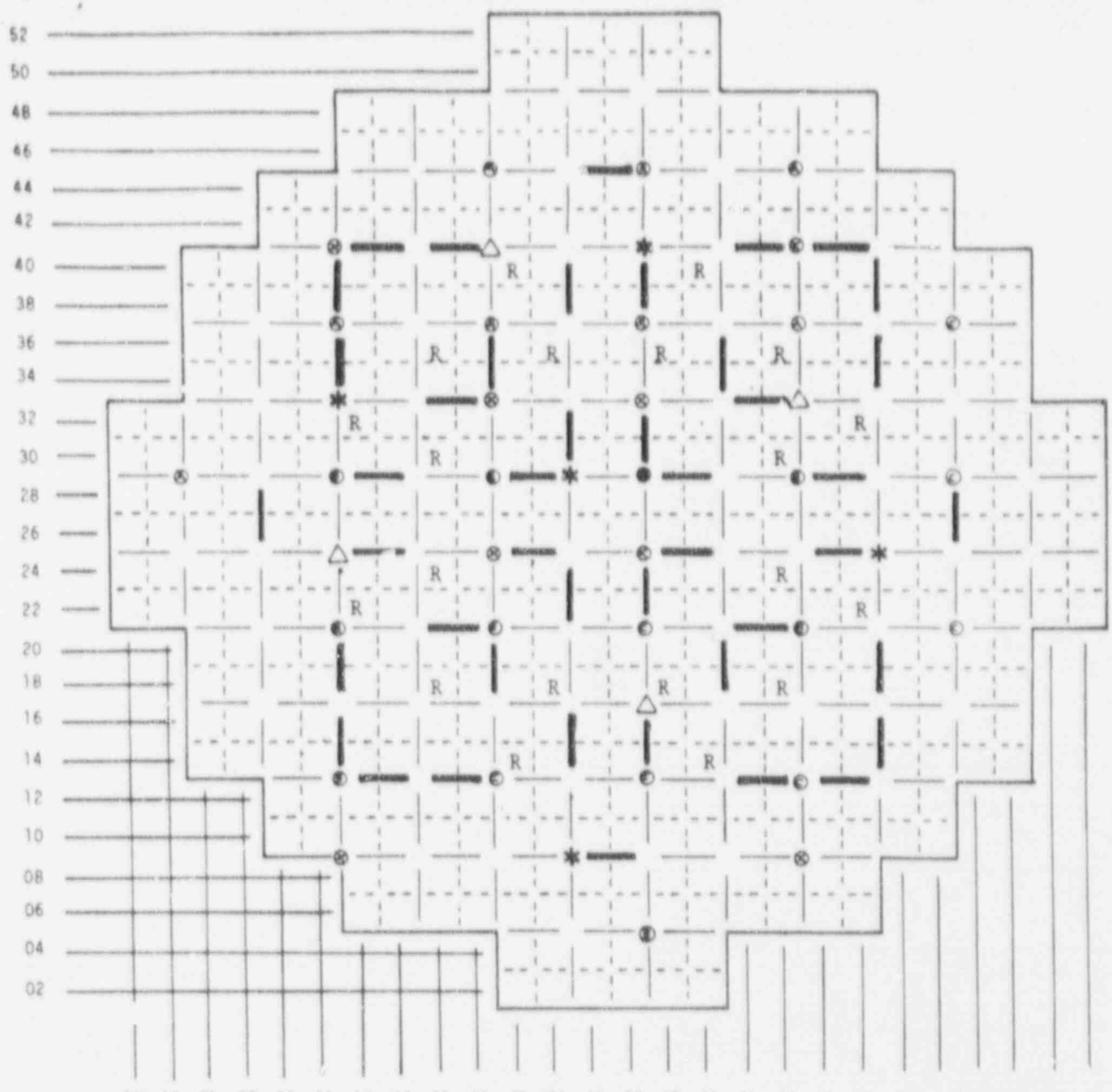
2. L O Mayer (NSP) to J F O'Leary (USAEC), "Supplementary Information Regarding the First Monticello Reload," dated April 13, 1973

3. L O Mayer (NSP) to Dr. P A Morris (USAEC), "Planned Modifications to the High Pressure Coolant Injection System Steam Line Flow Sensing Device," dated March 2, 1973

4. L O Mayer (NSP) to A Giambusso (USAEC), "Supplement to Report of Main Steam Isolation Valve Performance Dated December 28, 1972," dated January 22, 1973

5. L O Mayer (NSP) to D J Skovholt (USAEC), "Corrective Actions to Assure Main Steam Isolation Valve Reliability," dated June 27, 1973

6. L O Mayer (NSP) to A Giambusso (USAEC), "Supplemental Report of a Change in the Transient Analysis as Described in the FSAR," dated February 13, 1973



⊗ LPRM Location (Letter indicates TIP machine)

● LPRM Location (Common location for all TIP machines)

⊗ IRM Locations

△ SRM Locations

✱ Source Locations

▬ Remaining Curtain

R - Reload Bundle

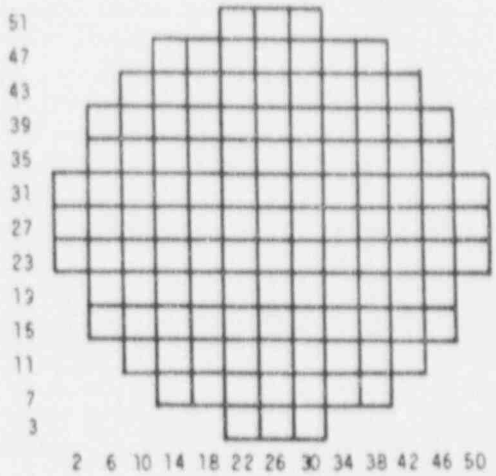


FIGURE 1
MONTICELLO CORE CONFIGURATION AT THE BEGINNING OF CYCLE 2

AEC DI' 'IBUTION FOR PART 50 DOCKET MAT 'AL
(TEMPORARY FORM)

CONTROL NO: 5466

FILE:

FROM:	DATE OF DOC	DATE REC'D	LTR	MEMO	RPT	OTHER
Northern States Power Company Minneapolis, Minnesota 55401 L. O. Mayer	7-12-73	7-16-73	x			

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J. F. O'Leary	1 signed	39		SENT AEC PDR X SENT LOCAL PDR x

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