MONTICELLO NUCLEAR GENERATING PLANT

Northern States Power Company

Minneapolis, Minnesota

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Cycle 3 Startup Report

and

Summary Status of Fuel Report

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I. Introduction

On March 15, 1974, the Monticello generator was taken off line and the reactor brought to a cold shutdown condition for the second refueling outage. A number of maintenance projects and plant modifications were also completed during the outage. On May 21, 1974, the plant was returned to power operation following a short period of startup testing. References 1 through 5 (see Section VII) present the analysis of the reload core; this report presents test results supporting the calculations for the reload core analysis.

This report is submitted in compliance with Technical Specification 6.7.A.1, "Startup Report," and 6.7.C.6, "Summary Status of Fuel Report." Since fuel inspection was an integral part of the refueling outage activities, these reports have been combined. Other pertinent outage activities are also briefly discussed in this report.

II. Refueling Activities and Core Performance

A. Core Configuration

The core loading following the outage is as shown in Figure 2-1 of references 1 and 5. A summary of the number of fuel assemblies of each type and the core average cycle exposure bounds is as follows:

	Cycle 1	Cycle 2	Cycle 3
Initial Core Fuel (7x7)	484	464	348
- as fabricated	(484)	(459)	(338)
- reconstit[illegible]	(0)	(5)	(10)
Reload 1 (7[illegible]ype B)	0	20	20
Reload 2 (8x8)	0	0	116
Curtains	216	44	0
BOC Exposure (MWD/STU)	0	6,802	7,806
EOC Exposure (MWD/STU)	7,115	10,779	Not Applicable

Reference 8 states that we will cooperate in an on-going fuel [illegible]rveillance program by inserting a pre-characterized 8x8 fuel assembly in the Monticello reactor. This assembly was inserted in core location 37-30, where it will serve as a lead exposure 8x8 assembly.

References 2 and 3 describe a Segmented Test Rod assembly designed for operation near the periphery of the core. This assembly was loaded in core location 23-04.

A total of 123 initial core fuel assemblies having an average exposure of 12,452 MWD/T were removed from the core. They were replaced by 116 new Reload 2 assemblies and 7 reconstituted initial core assemblies which had been stored in the spent fuel pool during the past cycle. The discharged fuel assemblies are presently stored in the spent fuel pool awaiting shipment offsite.

A significant number of fuel movements were made in the course of the refueling outage for the out-of-core sipping operation, curtain removal, the eddy current inspection of control rod blades discussed in reference 9, and the subsequent replacement of 6 control blades. The reloaded core was significantly shuffled, as planned, with respect to the pre-outage condition. The exposure distribution of the core was made flat and symmetric so as to give the lowest individual control rod worths, the greatest operational margin from thermal limitations on the fuel, and the most efficient burnup of the fuel.

Calculations for the shutdown margin reported in reference 1 were based on an assumed end of cycle 2 exposure distribution; it was determined that at the beginning of cycle 3, the most restrictive time in the cycle, 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. During the outage, the shutdown margin was recalculated to be .0133 delta k using the actual exposure distribution existing at the beginning of cycle 3. This is well in excess of that required by Technical Specifications. Tests summarized in Section II.C demonstrated sufficient shutdown margin for cycle 3.

B. Fuel Performance

At the time of the first refueling outage the Monticello core had a pre-outage exposure of 7115 MWD/T. The core consisted entirely of initial core 7x7 fuel. The pre-outage, full power, stack off-gas level was about 50,000 microcuries per second. The full core was wet-sipped in the reactor vessel; 25 fuel assemblies were determined to have leaking pins. These assemblies underwent thorough non-destructive testing which is reported in reference 6.

The rejected assemblies were replaced with 5 reconstituted assemblies and 20 Type B assemblies of an improved design as described in reference 7 (short, chamfered pellets, thicker cladding, lower moisture content, etc.).

During cycle 2 the stack off-gas level at full power was about 10,000 microcuries per second at the beginning of the cycle and tended to increase with time. While the Technical Specification limit is 270,000 microcuries per second, an administrative limitation was placed at 100,000 microcuries per second. Power was reduced to 74 percent of rated to meet the latter limitation at the end of cycle 2. The full core was wet-sipped in the reactor vessel immediately following the reactor shutdown; 56 leaking assemblies were identified. Any assemblies having questionable in-core sipping results and numerous other high exposure assemblies were sipped out-of-core to improve the sensitivity of the measurement. Of the 484 fuel assemblies in the core, 360 were sipped out-of-core, with 27 additional leaking assemblies identified. The total of 83 leaking assemblies were all from the initial core loading. Effectiveness of the sipping operation was demonstrated in that the stack off-gas level at 74 percent power was reduced from 100,000 microcuries per second prior to the outage to 6,800 microcuries per second immediately following the outage.

Fuel failures have been analyzed by disassembly of leaking assemblies and NDT testing of individual fuel pins. This was done at the first Monticello refueling outage and at other facilities having similar fuel types and operating conditions. The predominant failure mechanism identified in the relatively low exposure Cycle 1 was hydriding. A limited visual inspection of Cycle 2 leakers and experience at other facilities indicated that the pellet-clad interaction phenomena was the predominant failure

mechanism of Monticello fuel during cycle 2. This was expected in the higher exposure fuel. A correlation was observed between fuel failure and bundle exposure. Because of design improvements, it is expected that the 8x8, Reload 2 fuel and the Type B, Reload 1 fuel will demonstrate significantly improved performance over that of the initial core fuel.

On the immediate return to service following the outage, fuel performance was as expected. After having the generator on line for three days and ascending to 74 percent of rated reactor power, the stack off-gas level was 6,800 microcuries per second (without the modified off-gas treatment system in service). Over a short period of time the off-gas rate increased substantially. On June 5, 1974, the power level was raised to 97 percent of rated and off-gas increased to 66,000 microcuries per second. The previous administrative limit of 100,000 microcuries per second was replaced by an interim administrative limit of 65,000 in order to minimize the release until placing the modified off-gas treatment system in service and to trend fuel performance at that release rate. After one month of cycle 3 operation, power was restricted to 88 percent of rated to meet the 65,000 microcuries per second interim administrative limitation. This administrative limit was recently revised to 75,000 microcuries per second for routine operations. The corresponding power level at the time of this report is 90 percent of rated. Plans for premature replacement of the initial core fuel, which is believed to be the source of the high off-gas levels, are discussed in Section III.

C. 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. Dynamic tests were done to verify proper response of those systems on which modifications were made during the outage. No attempt was made to duplicate the extensive dynamic 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 friction tested and stroke timed. As a preventative maintenance procedure, 24 of the 121 control rod drives were overhauled or replaced with overhauled spares during the outage. Included were five drives which failed to insert the last six inches (beyond position 02) during a scram, a problem attributed to excessive seal leakage. All 24 overhauled control rod drives were scram-time tested in the cold condition. In addition, 5 rods were scrammed in the hot condition to verify the pre-established empirical 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, absence of curtains, 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 2-1 of reference 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 10-27 (and each of the 3 symmetric rods) was calculated to be the strongest rod; its worth, with all other rods inserted, was calculated to be .0277 delta k. Rods adjacent to the strongest rod had large individual notch worths over the range of interest. Therefore a diagonally adjacent rod was withdrawn to demonstrate sufficient shutdown margin. The samarium concentration in exposed fuel at the time of testing was greater than the normal operating level by an equivalent of .0015 delta k. Samarium poison in the new fuel was initially zero. As exposure accumulates the samarium in the 3/4 core of exposed fuel decreases while the samarium in the 1/4 core of new fuel increases to the equilibrium operating level producing a cancelling reactivity effect. To be conservative in determing required shutdown margin to be demonstrated, the absense of samarium in new fuel was neglected. Thus the required shutdown margin to be demonstrated included the .0025 delta k required by Tech Specs, the .0004 delta k value of R reported in reference 9 and the .0015 delta k samarium effect for a total of .0044 delta k. A total of .0114 delta k shutdown margin was demonstrated by fully withdrawing rod 10-27 and withdrawing rod 14-23 to notch position 10. 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 14-31 and 14-35 fully withdrawn; these were calculated to be the rods of next highest worth to the symmetric rods discussed above.

After making the reactor critical with the normal control rod withdrawal sequence, 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 near rated power, the control rod inventory was in good agreement with that predicted by calculations.

At various pressures during the heatup, relief valves were manually opened to verify the operability of all valves but in particular the operability of the four new safety/relief valves installed during the outage. (See Section IV.) Also, HPCI and RCIC operability was demonstrated at various pressures. Vibration in turbine exhaust lines to the torus was reduced substantially, as was slamming of check valves on those lines; this was anticipated after adding vacuum breakers and spargers on those lines as discussed in Section V.E.

III. Expected Cycle 3 Operation

The design exposure increment for cycle 3 is approximately 5300 MWD/T to the all-rods-out condition. This is sufficient to operate the reactor until the design end-of-cycle refueling outage in early March of 1975.

Reference 10 and many earlier submittals discuss exposure effects on the scram reactivity curve used in the transient analysis; a Prompt Relief Trip (PRT) system was proposed in reference 10 which will reduce the thermal duty on the fuel, resulting in acceptable performance with a broad band of conservatism for the worst scram reactivity curve anticipated in the life of the plant. References 12 and 13 state that the AEC Staff did not grant approval of the total installation of the proposed plant modifications and stated that six months are required for an evaluation of analytical methods before they can give full credit for the modifications. Final resolution of these matters is anticipated during the early months of Cycle 3.

For the interim period of time while awaiting the review and approval of PRT and analytical methods, a Cycle 3 transient analysis was prepared without credit for those topics under review. Reference 14 shows that, without additional refined calculations, all safety limits will be maintained during transients if the reactor is operated up to rated power until 750 MWD/T prior to the end of Cycle 3; at this exposure the control rod pattern must remain fixed until power ramps down to 95% of rated. Operation can continue at 95% of rated until all control rods are withdrawn from the core as discussed in reference 15. It is anticipated that such an unnecessary power reduction can be averted through the timely review and approval of the PRT system and analytical methods.

Another potential restriction on cycle 3 output, as discussed in Section II.B, is the current trend in off-gas levels. At the present time serious consideration is being given to sipping the core and replacing suspect assemblies at some time prior to the design end-of-cycle.

IV. Installation of Additional Safety/Relief Valves

On February 26, 1972 and July 14, 1972, main steam safety valves prematurely lifted on turbine trip transients. Neither incident resulted in a substantial introduction of steam into the drywell. However, owing to the nature of the events, it was judged by NSP, GE and the AEC, that these events should be avoided. Over the next year, the reliability of spring safety valves on BWR main steam systems was exhaustively studied by NSP. These studies included a dynamic analysis performed by Franklin Institute, inplace testing at Monticello, and cooperative efforts with GE. These studies indicated a number of deficiencies in the spring valve application which could not be straight-forwardly resolved. In addition to concern over valve application, it was noted by GE that the effective scram reactivity insertion rate of the control rod drive system decreased with increasing fuel exposure. This resulted in higher analyzed transient pressure and fuel heat flux during reactor transients. Analyses showed inadequate margin between the peak transient pressures and the safety valve setpoints. As an interim solution,

during late 1973, safety valve setpoints were increased from 1225 psig to 1240 psig.

In the evaluation of the data available in 1973, NSP concluded that as long as safety valves existed with discharges open to the drywell, containment pressurization would remain a possibility. In light of these concerns, the decision was made to eliminate the four spring safety valves and replace them with pilot operated safety/relief valves having discharges piped to the torus. It was shown by analysis that six combination valves set at 1080 psig provided sufficient capacity to meet ASME Section III core requirements. NSP elected to add two installed spares thus making an eight valve pressure relief and overpressure protection system. During the outage the four safety valves were replaced with safety/relief valves piped to discharge to the torus.

The relief valve discharge piping for the four new valves is fabricated and installed consistent with the existing installations. The valves and the piping from the main steam line down to the vent penetration is B.31.1-0-1967 and from the penetration to the rams head is ASME-3. The relief valve discharge lines are restrained to withstand a design basis earthquake utilizing the latest analytical techniques. For consistency, the existing discharge lines were reanalyzed for the same criteria. A total of 44 new hydraulic shock suppressors were installed in the drywell. A Certificate of Confromance was obtained by GE from Bergen Patterson which confirms the use of the proper (ethylene propylene) seal material. These snubbers will be examined on the same schedule as the existing 34 snubbers inside the drywell.

V. Miscellaneous Outage Activities

A number of inspections, repairs or modifications which have been specifically referenced in NSP-AEC correspondence to be performed during the 1974 refueling outage, were completed. These items are discussed below and are denoted by correspondence references.

A. Emergency Diesel Air Start System Modifications (Reference 16 and 17)

The air start system modifications included the installation of 2" air piping in the place of the 1" piping and the installation of a separate chiller drier for each of the four air start systems. The drier acts to cool and condense out water in the air before introducing it to the diesel compressed air tanks. The piping at the tank discharge was increased in size to minimize restriction of air flowing to the starting units.

Previously reported occurrences of start system malfunctions were attributed to inadequate air flow to the starting motor and to rust clogging the vent port on the air relay valve. As the new installed piping is carbon steel, special efforts were made to clean the piping of loose rust and foreign material prior to installation. The chiller-driers will maintain the air sufficiently dry to minimize the corrosion rate.

In the course of the design verification of the air start system modification, it was learned that seismic considerations had not been carried to the level of the air piping. It was decided that the modified piping would be properly restrained to withstand the design basis earthquake. All newly installed piping is rigidly restrained.

The diesel air starter modification testing included operational verification of the chiller-driers and air starters. Each starting system was verified to crank its respective diesel at ≧ 70 rpm. All new welds were visually inspected and checked for leakage.

B. HPCI and RCIC Turbine Control System Maintenance Inspection (Reference 18)

During startup after the 1973 refueling outage, while testing to verify the HPCI venturi installation, it was noted that adjustments to the HPCI turbine compensating flow controller might be more refined to control starting speed. As a result, it was decided to perform a detailed inspection of the HPCI and RCIC control systems during the next refueling outage. This inspection, which was done during the 1974 refueling outage, included the following items:

Calibration of the flow controller and flow transmitter
Inspection of the governor drive gears
Functional check of the governor
Inspection of the speed gear and governor drive gear for tightness
Inspection of the control valve linkage
Inspection of all electrical terminations

Along with the inspection, the HPCI and RCIC turbine control systems were modified by installing ramp generators to simulate startup flow increases and thus provide improved flow response during quick starts. Satisfactory performance was demonstrated following the maintenance inspection and modifications.

C. Replace Torque Switches in Main Turbine Seal System Limitorques (Reference 19)

The torque switch on turbine seal system valve MO-1045 was replaced during the outage.

D. Add Structural Facilities (References 20 and 21)

Pipe whip restraints were installed at the four high stress locations on the feedwater piping. Details of these restraints were submitted in reference 21. It was noted in that submittal that a reanalysis of the potential damage due to HPCI line whipping into the torus indicated that some deformation of the torus shell might result. Protective restraints have been installed on the HPCI steam line in the torus compartment which minimize the potential for this type of damage.

E. Install Spargers and Vacuum Breakers on HPCI and RCIC Exhaust Lines (Reference 22)

At Monticello and other BWR facilities utilizing light-bulb-torus containment systems, experience has shown interactions between torus and turbine discharge piping to be of significant magnitude. To minimize these interactions, spargers were added to the HPCI and RCIC discharge lines in the torus. It was also noted that following HPCI and RCIC operation, that the discharge piping would fill with water as the remaining steam condensed. As a result of this phenomena, significantly higher pressures would develop during initial operation as the steam pushed the water out of the pipe. This in turn, caused more significant vibrations on the piping. To reduce this effect, vacuum breaker lines were installed on the HPCI and RCIC turbine discharge piping which will vent air in to the lines as the steam condenses. This lighter fill medium will reduce pressure pulses and thus reduce impact on the torus.

All piping and equipment used in this installation were manufactured and installed in accordance with B.31.1.0-1967 and existing design specifications. Welds attaching the new spargers were radiographically examined. Welds attaching the new vacuum breaker piping were liquid penetrant examined and hydrostatically tested. All lines meet seismic class 1 criteria.

F. Inspection of RHR Piping (Reference 22)

An inspection was conducted on RHR system piping to look for evidence of damage resulting from water hammer. Ten welds on the "A" RHR shutdown return piping were inspected by ultrasonic examination. One stop weld on the MO-2014 safe-end to penetration weld exhibted a positive indication on UT. On radiographic examination the indication was determined to be an acceptable level of porosity; no repairs were necessary. All hangers were intact and showed no evidence of significant pipe movement.

G. Inspect and Modify MSIV Solenoids and Manifolds (Reference 23)

A surveillance procedure covering inspection and functional test requirements for main steam isolation valves was prepared by NSP Operations. MSIV inspections in accordance with this procedure were completed during this outage. The surveillance procedure included valve timing and yoke alignment. The Automatic Valve Corporation air operators were modified by adding a booster piston to the three-way test poppet valve as recommended by the vendor.

H. Vacuum Breaker Alarm System (Reference 24)

An alarm system has been installed to give control annunciation of torus-to-drywell vacuum breaker operation. The installation utilizes existing closed position limit switch wiring and detection capability. Relays in parallel with each indicator lamp at the local panel are arranged to energize a single control room annunciator. Similarly, relays are provided for the ten control room indicators to actuate a second control room annunciator. This provides two alarm circuits, each with separate position detector switches and so arranged that loss of power will result in annunciation (redundant and fail-safe to loss-of-power).

I. Removal of PEECO Flow Switch in SLCS (Reference 25)

A paddle-type PEECO flow switch was initially installed in the Standby Liquid Control System. The paddle of this type of flow switch was found broken off or bent in other applications. It was decided to remove the SLCS switch to preclude the possibility of a foreign piece or metal entering the reactor. It was determined that the switch was redundant to other means of sensing flow and therefore not required. Other indications are the boron storage tank draw-down rate, pump-running indication and decrease in neutron flux.

VI. References

1. L O Mayer (NSP) to J F O'Leary (USAEC), "Second Reload Submittal," dated November 19, 1973.

2. L O Mayer (NSP) to J F O'Leary (USAEC), "Submittal of NEDE-20179 Monticello Segmented Test Rod Bundle Submittal (Proprietary Information)," dated December 14, 1973.

3. L O Mayer (NSP) to J F O'Leary (USAEC), "Change to NEDE-20179 Monticello Segmented Test Rod Bundle Submittal (Proprietary Information)," dated January 15, 1974.

4. L O Mayer (NSP) to J F O'Leary (USAEC), "Supplemental Information to the Monticello Second Reload Submittal," dated February 8, 1974.

5. L O Mayer (NSP) to J F O'Leary (USAEC), "Supplemental Information to the Monticello Second Reload Submittal," dated April 1, 1974.

6. L O Mayer (NSP) to J F O'Leary (USAEC), "Submittal of Cycle 2 Startup Report," dated July 12, 1973.

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

8. L O Mayer (NSP) to J F O'Leary (USAEC), "Special Surveillance Program for 8x8 Fuel," dated February 14, 1974.

9. L O Mayer (NSP) to J F O'Leary (USAEC), "Inverted Poison Tubes in Control Blades," dated May 10, 1974.

10. L O Mayer (NSP) to J F O'Leary (USAEC), "Permanent Plant Changes to Accomodate Equilibrium Core Scram Reactivity Insertion Characteristics," dated January 23, 1974.

11. L O Mayer (NSP) to J F O'Leary (USAEC), "Change Request Dated March 1, 1974," dated March 1, 1974.

12. D J Skovholt (USAEC) to L O Mayer (NSP), "Approval of Plant Modification for Fuel Cycle 3," dated March 14, 1974.

13. D L Ziemann (USAEC) to L O Mayer (NSP), Question on Proposed PRT System Design, dated April 4, 1974.

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15. L O Mayer (NSP) to J F O'Leary (USAEC), "Planned Reactor Operation from 2000 MWD/T to End of Cycle 2," dated August 21, 1973.

16. L O Mayer (NSP) to J F O'Leary (USAEC), "Malfunction of #1 Starting System of #11 Emergency Diesel Generator," dated May 14, 1973.

17. L O Mayer (NSP) to J F O'Leary (USAEC), "Inoperability of No. 2 Starting System of No. 12 Diesel Generator," dated December 27, 1973.

18. L O Mayer (NSP) to J F O'Leary (USAEC), "High Pressure Coolant Injection System Inoperability," dated May 25, 1973.

19. L J Wachter (NSP) to B Grier (USAEC), "Correction of Response to Regulatory Operations Bulletin No. 72-3," dated June 22, 1973.

20. E C Ward (NSP) to A Giambusso (USAEC), "Postulated Pipe Failures Outside Containment," dated September 7, 1973.

21. E C Ward (NSP) to D L Ziemann (USAEC), "Postulated Pipe Failures Outside Containment," dated March 8, 1974.

22. L J Wachter (NSP) to J G Keppler (USAEC), Response to Information Request 74-1, dated February 7, 1974.

23. L O Mayer, (NSP) to J F O'Leary (USAEC), "Failure of Outboard Main Steam Isolation Valves AO-2-86B and AO-2-86C to Close," dated February 25, 1974.

24. L O Mayer, (NSP) to J F O'Leary (USAEC), "Further Vacuum Breaker Modifications," dated September 17, 1973.

25. L O Mayer, (NSP) to A Giambusso (USAEC), "Failure of PEECO Flow Switch Actuator Paddle," dated September 26, 1973.