U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-263/91005(DRP)

Docket No. 50-263

License No. DPR-22

Licensee: Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

Facility Name: Monticello Nuclear Generating Station

Inspection At: Monticello Site, Monticello, MN

Inspection Conducted: March 5 through April 16, 1991

Inspectors: S. Ray

P. Moore

D. Schrum

Approved By: R. W. DeFayette, Chief
Reactor Projects Section 2B

APR 2? 1991
Date

Inspection Summary

Inspection on March 5-April 16, 1991 (Report No. 50-263/91005(DRP)

Areas Inspected: A routine, unannounced inspection by the resident inspectors of operational safety verification; refueling preparations and outage activities; onsite followup of events; radiation controls; maintenance and surveillance activities; engineering and technical support; followup of previous inspection findings; licensee event reports; regional requests; and evaluation of licensee self-assessment capability and quality assurance program.

Results:

Plant Operations

The plant operated at various power levels until shutting down for a refueling outage on April 1, 1991. Significant activities planned for the outage included vessel weld examinations, replacement of feedwater flow venturies, valve replacements, refueling the core with GE-10 fuel bundles which includes new fuel channels designed to improve channel flow characteristics and mitigate channel bowing phenomena, and the decontamination of the reactor recirculation and residual heat removal piping.

Among the difficulties encountered in outage activities were a fuel channel dropped on a new fuel bundle that resulted in damage to both the channel and the fuel bundle, and local leak rate test failures that exceeded the maximum pathway leakage limit.

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Strengths were noted in outage planning and scheduling to maintain the plant in a low risk configuration.

Radiation Protection

Significant improvements and enhancements have been made in radiation protection programs. The use of electronic dosimetry, computer drawn survey maps, streamlined radiation work permits, simplified dressout requirements, remote video observation of the RWCU pumps, and increased utilization of the plant video disc walkthrough system for job planning were all instituted during this inspection period and were considered strengths in the radiation protection program. Decontamination efforts for reactor coolant piping systems appeared to be effective.

A normally locked door to a high radiation area was left blocked open and unattended for a short period of time resulting in a non-cited violation. No entries were made into the area while the door was open.

Maintenance and Surveillance Activities

Maintenance activities increased greatly with the onset of the outage. The majority of the activities were accomplished correctly and on schedule, except for the stripping of a reactor vessel head bolt by the de-tensioner device which caused some delay in activities.

An injury to a construction worker resulted in the transfer of a contaminated individual to an offsite medical facility and required a notification of unusual event.

Engineering and Technical Support

Modifications received a more thorough review by the licensee in response to an audit that was performed after several NRC findings involving weaknesses in the modifications.

Safety Assessment/Quality Verification

A detailed review of the licensee self-assessment capability and the quality assurance program indicated several program strengths. Reviews of events, procedures, modifications, and safety issues by the Operations Committee, Safety Audit Committee, Human Performance Task Force, and modification design and review teams were excellent.

A weakness was noted in that the minutes of the Operations Committee lacked sufficient detail.

DETAILS

1. Persons Contacted

D. Antony, General Manager, Monticello Nuclear Site
M. Clarity, Assistant to the Plant Manager
G. Crosby, Superintendent, Quality Services
B. Day, General Superintendent, Engineering
M. Hammer, General Superintendent, Maintenance
*W. Hill, Plant Manager
G. Hudson, Superintendent, Security
P. Johnson, Manager, Nuclear Projects
D. Nevinski, General Superintendent, Operations
T. Parker, Manager, Nuclear Support Services
R. Scheinost, General Superintendent, Site Services
J. Windschill, Superintendent, Radiation Protection
P. Kamman, Manager Nuclear Operations QA
G. Bart, Superintendent QA Monticello
D. Krech, Quality Supervisor
J. Rowan, Quality Supervisor
D. Musolf, General Manager Nuclear Support Group
G. Crosby, Superintendent Quality Services
M. Sleigh, Quality Engineer
S. Engelke, Superintendent Electrical and Instrumentation System Engineering
W. Brant, Site Superintendent

The inspectors also contacted other licensee employees including members of the technical and engineering staffs, and reactor and auxiliary operators.

*Denotes the licensee representatives attending the resident inspectors' exit meeting.

2. Plant Operations

After operating at various power levels during the first part of the inspection period, the plant commenced a shutdown for a refueling outage on March 31, 1991. The scheduled 56 day outage officially commenced on April 1 and continued through the rest of the inspection period.

a. Operational Safety Verification (71707)

The inspectors observed control room operations, reviewed applicable logs, and conducted discussions with control room operators during the inspection period. During these discussions and observations, the inspectors ascertained that the operators were alert, cognizant of plant conditions, attentive to changes in those conditions, and took prompt action when appropriate. The inspectors verified the operability of selected emergency systems, reviewed tagout records, and verified the proper return to service of affected components. Tours of the plant were conducted to observe equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations, and to verify that maintenance requests had been initiated for equipment in need of maintenance. The inspectors observed the reactor shutdown for refueling on March 31/April 1. Some minor problems were encountered and corrected by the operators during the shutdown as discussed below.

(1) While inserting control rods for the shutdown, the operators had to declare the rod block monitor (RBM) inoperable when they were unable to meet all of its procedural requirements. Step 14 in procedure #0212, "RBM Operability," Revision 12, indicated that it was required to be completed to meet the Technical Specifications operability requirements for the system. That step stated that the RWM should, upon selection of an out-of-sequence rod, indicate a withdrawal block, insert block, and a select error. Actually, Technical Specifications 4.3.B.3(a)(iii) only required that the RWM indicate a select error. The withdrawal error would only occur during startup and the insert error would only occur during shutdown. This procedure had been revised incorrectly on March 5, 1991. Since the operators were unable to generate the withdrawal block during the shutdown, they were not able to meet the requirement of the step and declared the RBM inoperable. They then proceeded to have all rod movements verified by a second operator as required by Technical Specifications until a determination was made by the nuclear engineer that the Technical Specifications had indeed been met.

(2) During the reactor shutdown as the operators were preparing to enter the range requiring the use of the intermediate range monitor (IRM) detectors, it was noted that IRM #13 was indicating significantly downscale of the other IRMs. All other IRMs were in range 8 and #13 could only be brought onscale by moving it to range 7. Operations staff made the decision that it would be preferable to declare IRM #13 inoperable and continue the shutdown without it. Since IRM #11 was also considered inoperable due to previously experienced electrical noise problems, and both IRMs #11 and #13 fed the same channel of the reactor protection system (RPS), a half scram was inserted on the RPS channel for the remainder of the shutdown.

Overall, the shutdown activities were well controlled and progressed smoothly in spite of the minor problems encountered.

b. Preparations for Refueling and Outage Operations (60705, 71707)

The inspectors observed the preparations for the refueling outage which included new fuel receipt, inspection, and channeling; scheduling; work area preparation; and other activities. The channels used on the new fuel were a new model designed to improve channel flow characteristics and mitigate the channel bowing phenomena.

(1) On March 7, 1991, while installing new fuel channels on new fuel bundles, a channel was dropped onto the fuel bundle causing

damage to both components. The incident occurred while operators were in the process of lowering the channel onto the bundle. After the channel had been lowered a few feet onto the bundle, the retaining nut on the channel handling tool came off of its eye-bolt causing the channel to be released from the hoist. The channel slid uncontrolled the rest of the way down the fuel bundle and the channel handling tool (still attached to the channel) impacted the bail handle on the top of the fuel bundle. Visual inspections showed that the bail handle was bowed, the upper tie plate was pushed down on the fuel rod expansion springs, and the support corners on the fuel channel were slightly bent. After discussions with the fuel vendor, the bundle was sent back to the vendor for inspection and repair. The repaired fuel bundle and a new channel were received back on site on April 16.

(2) In general, the inspectors observations confirmed that the outage activities were well planned. Activities were scheduled in detail including strategies to insure that systems required by technical specifications remained operational. In response to Information Notice 91-22, "Four Plant Outage Events Involving Loss of AC Power or Coolant Spills," and other recent industry events, the schedule was designed such that there would always be at least two sources of offsite power and redundant core cooling/flooding methods available to minimize the effects of single equipment failures. "System Windows" were planned to control maintenance activities and to help maintain the plant in a low risk configuration. A Plant Manager's newsletter was issued to increase personnel awareness of the potential for loss of power, loss of decay heat removal, and vessel draindown events. All supervisory personnel were kept aware of the current plant status and plans through a detailed briefing at a daily meeting.

(3) During the first part of the outage, one problem was obtaining parts in time to support the schedule. An example was a support clamp that was planned to be installed on a jet pump with a cracked riser bracket. At the end of the inspection period it appeared that the clamp may not have been available in time to install it. The licensee was evaluating alternate courses of action. Preliminary inspections indicated that the crack, which was discovered in the 1989 refueling outage, had not propagated and no new cracks were found. It appeared that the licensee's compensatory action for controlling the speed of recirculation pumps to avoid flow induced vibrations of the jet pumps was successful. Other planned projects such as installing new feedwater flow instrumentation and valve replacements appeared to be in jeopardy due to procurement delays. Additional management attention in this area may be necessary in the future.

c. Onsite Follow-up of Events (93702)

The inspectors performed onsite follow-up of events which occurred during the report period. These follow-ups included reviews of operating logs, procedures, Condition Reports, Licensee Event Reports (where available), and interviews with licensee personnel. For each event, the inspectors developed a chronology, reviewed the functioning of safety systems required by plant conditions, and reviewed licensee actions to verify consistency with procedures, license conditions, and the nature of the event. Additionally, the inspectors verified that the licensee's investigation had identified the root causes of equipment malfunctions and/or personnel errors and that the licensee had taken appropriate corrective actions prior to restarting the unit. Details of the events and the licensee's corrective actions developed through inspector follow-up are provided below.

(1) On March 14, 1991, operators noted what appeared to be a burned out light bulb on the main control board which indicates the breaker is closed for the "A" control rod drive (CRD) pump. When replacing the light bulb did not correct the problem, the operators wrote a work request to investigate and repair the condition. The next day, the system engineer was made aware of the situation and realized that the light monitored the trip coil circuit. Loss of continuity in the trip coil circuit would disable the electrical trips of the breaker. Operators were requested to remove the pump from service but were unable to trip the pump with the handswitch on the main control board or the local handswitch on the breaker. They tripped the breaker with the mechanical pushbutton.

During the licensee's subsequent investigation on March 15, the licensee determined that the condition was apparently due to the failure of an auxiliary relay on the breaker caused by a high resistance on one of the trip coil contacts. This resulted in inability to trip the breaker electrically. The licensee postulated that a fault on the CRD pump, causing an overcurrent condition on the breaker, could have resulted in the loss of the essential bus #15 which supplied power to the pump, because the CRD pump breaker would not have tripped to clear the fault. Thus a fault on the non safety-related CRD pump could have caused the loss of an entire safety-related bus.

The licensee reported the event to the NRC via the ENS and issued LER 91-004 on April 12, 1991. The condition had existed for about 13 hours before the CRD pump breaker was mechanically tripped. During that time the plant was susceptible to a loss of one of the redundant safety-related busses from a single failure. The condition could have disabled two of the four residual heat removal pumps as well as one of the redundant trains of core spray, the standby gas treatment system, the emergency filtration system, and the emergency service water. All division II safety-related equipment was available and capable of removing the worst case decay heat load.

As part of the corrective actions for the event, a procedure revision was written to provide direction to the operators on the actions to take for loss of an indicating light on a main control room 4KV breaker. Additionally, the auxiliary switch contacts in all safety-related 4KV breakers powering non safety-related loads will be cleaned during the present refueling outage. The preventative maintenance program for the 4KV breakers was expanded to clean the auxiliary switch contacts periodically. The inspectors had no additional concerns regarding this event and the LER is closed in section 6.b.(3) of this report.

(2) During the refueling outage, while performing a local leak rate test on containment isolation valves, several valves exhibited excessive leakage in excess of the maximum pathway containment leakage limits in the Technical Specifications. On April 1, 1991, the inboard isolation valves for the drywell floor and equipment drain sumps together were greater than 1240 scfh. This prompted the licensee to make the notification to the NRC that it had exceeded the maximum pathway leakage rate of 458 scfh. Three of the four outboard main steam isolation valves (MSIV), and one of the standby liquid control system check valves were also found to have leakage rates greater than specified. The redundant valves in the same lines as these valves were determined to have leakage rates within limits. However, since single failure of the redundant valves must be assumed in the worst case, the excessive leak rates will be reported in LER 91-005. Repairs to all the valves involved will be completed during the current outage.

(3) On April 10, 1991, a construction worker, on loan from the Prairie Island site, was injured in the radiation-controlled area while performing sandblasting activities. A metal plate that was part of a turbine valve that he was working on tipped over and fell on and broke the worker's leg. The plant fire brigade was called to respond to the medical emergency. After removing his outer layer of protective clothing, the protective clothing that he had on underneath indicated 400 cpm above background on a glove and between 100-200 cpm above background on the inner layer of protective clothing, which is normal for protective clothing. An Unusual Event was declared for the transport of a contaminated injured individual to an offsite medical facility. Licensee personnel were able to remove the glove and protective clothing enroute to the hospital, so that when the worker arrived he was not contaminated. The licensee was still investigating the circumstances of the injury.

No violations or deviations were identified.

3. <u>Radiation Controls</u> (71707)

a. The radiation protection department at Monticello has recently implemented some significant improvements designed to reduce exposure and achieve ALARA objectives.

Most apparent of these improvements was the implementation of electronic dosimetry. Electronic dosimeters were used in lieu of the standard pencil dosimeters and had a number of distinct advantages. They could be programmed to alarm at a set dose rate or accumulated dose. This alerts workers to either the presence of a high radiation field or the accumulation of a dose limit without having to rely on the repeated checking of a pencil dosimeter. In addition, the electronic dosimeters also functioned as portable dose rate meters allowing workers to verify that they were in a low dose area, whenever possible. The dosimeters also allowed for more accurate accounting of dose received for specific jobs or Radiation Work Permits (RWPs).

The RWPs at Monticello have also been improved. A general area RWP #1 now covers low rate areas such as the turbine building, intake structure, reactor building, and chemistry lab. Previously those areas were covered by four separate RWPs. Requirements for protective clothing have also been improved through the use of five classifications that cover the requirements from the basic booties and gloves through waterproof outers with plastic hoods. The classification numbers with dress requirements were posted at all dressout areas and were listed on each RWP located at the step-off pads, so personnal could easily identify and remember what the requirements were for each area.

The computer aided design system at Monticello was being used to generate survey maps and RWP guides. The survey maps were posted at all of the radiation areas and were clear, concise and easy to understand. Floor plans of all areas in the reactor and turbine building were located at the RWP signout and electronic dosimetry distribution area. These floor plans identified which RWPs applied to specific areas in the plant thus reducing the amount of assistance required from the radiation protection department. This was considered an excellent use of the licensee's computer aided design equipment.

The radiation protection department has also increased the utilization of the plant video disc (CEVUE) walkthrough system for job planning. Maintenance crews have been reviewing work sites and job planning while looking at the video picture of the area while radiation protection personnel described the characteristics of the radiation fields in the work area.

All of these improvements were phased in since the beginning of the year, so people were trained before the current refueling work. The benefits from these efforts have become obvious with the start of the outage. The influx of contractor personnel unaccustomed to Monticello's radiation protection practices on a daily basis have benefited the most from the simplification of the RWPs, survey maps, dressout requirements, and job planning with the video disc system. These practices were considered an improvement in the strength of the radiation protection program at Monticello.

b. On April 12, 1991, a licensee health physics technician discovered that a normally locked door to a high radiation area in the reactor water cleanup system pump room was blocked open and unattended. He immediately locked the door and began an investigation.

The investigation determined the the door had been blocked open by operators who were monitoring the pumps in the room for cavitation. As reactor vessel water level was lowered during the outage, operators were posted at the room to listen for evidence of pump cavitation. To avoid excessive exposure, one of the operators blocked the door open so that he could stand outside the high radiation area and listen through the open door. The operator did not inform or obtain permission from the control room or radiation protection to block the door. The door was left open but continuously attended through several operator relief periods. When the evolution was over and the last operator left the area, he forgot to remove the block and lock the door.

That operator was not the one who had originally blocked the door. When questioned the following day, the operator said he was aware that the door was blocked and should have been shut when he left but he had not thought about it at the time. The condition was discovered about 90 minutes later. The licensee did not believe that anyone entered the room while it was left unattended because a contamination barrier just outside the door was in place and dressout would have been required to enter.

The inspectors discussed the event with the regional specialist radiation protection inspector assigned to the Monticello plant. It was determined that the event was an isolated case of minor safety significance. Even with the door blocked open, it was unlikely anyone would enter without proper authorization because it was in a clearly marked contaminated area.

Technical Specification 6.5.B.1.c requires that for high radiation areas in which the intensity of radiation was greater than 1000 mrem/hr the doors shall be locked or attended to prevent unauthorized entry into those areas. The inspectors reviewed the radiation survey for the room and verified that some areas in the room had dose rates in excess of 1000 mrem/hr. Failure of the licensee to control the door to the high radiation area for a period of about 90 minutes on April 12, 1991, was considered a non-cited violation (263/91005-01(DRP)). The event was not cited because it met the requirements for exercising discretion in section V.G of the NRC Enforcement Criteria. The licensee's immediate corrective actions included tracking blocked open high radiation doors as limiting conditions for operation similar to fire doors. Permanent improvements to the control of high radiation barriers were still being evaluated.

c. One of the activities of the refueling outage was to conduct decontamination of several reactor coolant piping systems. Preliminary indications at the end of the inspection period indicated that the efforts to date were highly successful and would result in substantial dose savings for the remainder of the outage. Additional results of the efforts will be discussed in more detail in future inspection reports.

One non-cited violation was identified. This violation is closed and additional corrective actions will be followed by the LER.

4. Maintenance/Surveillance (61720, 61726 & 62703)

Station maintenance and surveillance activities of the safety-related systems and components listed below were observed or reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, and industry codes or standards, and in conformance with Technical Specifications.

The following items were considered during this review: the limiting conditions for operation were met while affected components or systems were removed from and restored to service; approvals were obtained prior to initiating work or testing; quality control records were maintained; parts and materials used were properly certified; radiological and fire prevention controls were accomplished in accordance with approved procedures; maintenance and testing were accomplished by qualified personnel; test instrumentation was within its calibration interval; functional testing and/or calibrations were performed prior to returning components or systems to service; test results conformed with Technical Specifications and procedural requirements and were reviewed by personnel other than the individual directing the test; any deficiencies identified during the testing were properly documented, reviewed, and resolved by appropriate management personnel; work requests were reviewed to determine the status of outstanding jobs and to assure that priority was assigned to safety-related equipment maintenance which may affect system performance. The following maintenance and surveillance activities were observed:

- WRA 91-1094, Replace Steam Separator Hold Down Bolts;
- WRA 91-1393, Repair MSIV Air Operator 2-86B, Failed Leak Rate Test;
- WRA 91-0400, Inspect Valve Internals and Associated Piping on MO-1749 Core Spray Test Return "A" Side;
- PM 4132, RBCCW Pump Motor;
- PM 4110, HPCI System General Inspection;
- PM 4115, HPCI Turbine Inspection;
- Test #0212 Rod Worth Minimizer Operability Test, Technical Specification 4.3.B.3.a;
- Test #0255-08-1D-2 RCIC Turbine Exhaust Vacuum Breaker Check Valve Leak Rate Test, Technical Specification 4.15;

- Test #0137-10, RCIC Exhaust Isolation Valve Leak Rate Test, Technical Specification 4.7.A.2;

- Procedure #9214, Install Carousel and De-tension RPV Head Bolts;

- Main Steam Line High Flow Group I Isolation Instrument Test and Calibration Procedure #0051, Rev. 12, Technical Specification 3.2 and 4.2;

- Test #0032, Calibration #0033, ECCS Pump Start Permissive Sensor.

a. Significant failures of the outboard MSIVs during the initial LLRT contributed to a reportable event that is further described in section 2.c.(2) of this report. The work performed to address that condition was reviewed by the inspectors.

b. Problems encountered during the reactor pressure vessel (RPV) head de-tensioning resulted in a slight delay in critical path outage activities. While de-tensioning one of the RPV bolts, the threads on the bolt and the de-tensioner attached to the bolt were apparently stripped. QA personnel observing the evolution indicated that a personnel error may have been involved in that the de-tensioner was not completely threaded onto the RPV bolt causing the threads to strip when an attempt was made to de-tension the bolt. The licensee is investigating.

No violations or deviations were identified.

5. Engineering and Technical Support (71707)

In response to past NRC concerns about the performance of modifications at Monticello, engineering management has implemented a technical review team to perform a detailed review of design output, post-modification testing plans, and plant interface during installation. The technical review team consists of senior members of the plant and Nuclear Projects Department engineering, maintenance, quality, and design standard groups. The team reviews the design documentation when it is necessary to confirm design adequacy and has been implemented for selected critical modifications scheduled to be installed during the refueling outage. This team was further discussed in section 6.d.(1) of this report.

No violations or deviations were identified.

6. Safety Assessment/Quality Verification

a. Followup of Previous Inspection Findings (92701)

(Closed) Open Item 91003-01: Lost TLD.

The licensee concluded its investigation of the lost TLD and determined that the individual did not receive any unmonitored dose for the period in question. The licensee also revised its TLD issuance report to emphasize that a lost TLD report must be initiated whenever a TLD is issued under circumstances other than the normal quarterly replacements.

b. Licensee Event Report (LER) Follow-up (90712 & 92700)

Through direct observation, discussions with licensee personnel, and review of records, the following LER was reviewed to determine that the reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished in accordance with Technical Specifications. Based on the inspectors' review, the following LERs are closed:

(1) (Closed) LER 89-029-01: Secondary Containment Inoperability.

The licensee approved the revised secondary containment capability test, procedure #0151, Rev. 11, on August 10, 1990. The results of this testing conducted on April 2, 1991 demonstrated that the secondary containment capability with the Standby Gas Treatment system fans running and the reactor building plenum exhaust fans turned off met the requirements of the Technical Specifications for operability. Specifically, the secondary containment was able to maintain between 0.35-0.40" water negative pressure with respect to the outside atmosphere. The requirement is for greater than 0.25" water.

(2) (Closed) LER 91-003: Reactor Scram During Startup due to Signal Spike on IRM Channel.

The inspectors observed the licensee's efforts and reviewed the calculations concerning this event. The subsequent startup activities were also evaluated. The cause of the scram was revised from being due to IRM spiking to a flux redistribution phenomena. The licensee revised its startup practice to pull the complete rod group notch by notch when approaching criticality instead of pulling several notches on a single rod within the group as was being performed when the scram occurred. The Safety Audit Committee committed a significant amount of time reviewing this event. The inspectors are satisfied with the licensee's review, analysis, and corrective actions for this event.

(3) (Closed) LER 91-004: Failed 4KV Breaker Could Have Disabled Low Pressure Coolant Injection.

This LER was discussed and closed in section 2.c.(1) of this report.

c. Follow-up of Regional Requests (71707)

In response to a finding that portable hydrogen recombiners were not being properly maintained and tested at another site, regional management requested that the resident inspectors reply to the following questions regarding the hydrogen recombiners:

(1) Is the use of a recombiner the primary or backup means of post accident containment hydrogen control?

Primary - (Only Method).

(2) Are the recombiners internal or external to the containment?

External.

(3) If external, are the containment penetrations dedicated to the recombiner or shared with the purge system.

Dedicated.

(4) Is the system from the containment to the connection point for the recombiner constructed safety grade? EQ?

Yes, it is both safety grade and EQ.

(5) Is the recombiner stored onsite or at a remote location?

Onsite - two units permanently installed.

(6) Is the recombiner shared with other sites?

No.

(7) Are procedures in place for obtaining, hook-up of, operation, and removal of the recombiner?

Procedures are in place for operating. Procedures are not needed for obtaining, hook-up, and removal because the units are permanent.

(8) Is there dedicated space to locate the recombiner and its associated equipment, such as air compressors, to facilitate hook-up?

N/A

(9) Is the equipment necessary for hook-up staged for its use?

N/A

(10) Is the hook-up equipment inventoried and checked for damage on a periodic basis?

N/A

(11) Are the containment penetrations and/or the hook-up locations readily accessible under accident conditions?

N/A

(12) Has the recombiner ever actually been transported to the site and hooked-up?

N/A

(13) Is the recombiner tested? How frequently.

Yes, The recombiners are tested per the Technical Specifications:

- Once per cycle - instrument calibration, heater ground checks, and visual inspection.
- Once per six months - reaction chamber temperature and current operability test.
- Type "A" leakage test performed along with containment type "A" leakage test.
- Other PMs per vendor recommendations.

(14) Is the recombiner inspected by the licensee periodically to ensure condition, ready for shipment, etc...

See #13.

(15) Do the Technical Specifications cover the recombiner?

Yes, Technical Specification 3.7.E.

(16) Does the FSAR include the recombiner?

Yes, Section 5.2.2.9.

d. <u>Evaluation of Licensee Self-Assessment Capability and Quality Assurance Program</u> (40500 & 35502)

(1) <u>Onsite Safety Review Committee</u>

The inspectors attended an Operations Committee (O.C.) meeting on March 15, 1991. All O.C. members were site managers: the plant manager, the assistant plant manager, general superintendent of operations, general superintendent of site services, general superintendent of maintenance, general superintendent of engineering, and superintendent of design standards. The use of managerial personnel indicated that the plant placed a high level of importance on the O.C. reviews and assigned the staff accordingly.

The members of the O.C. came to the meeting with appropriate concerns and questions, indicating that materials had been reviewed prior to the meeting. The reviews observed were conducted thoroughly by going through the document packages page by page. All O.C. members at the meeting contributed to the discussions and positive suggestions were made to improve the documents. Additional staff comments from personnel who were unable to attend the meeting were presented and discussed by the O.C. members.

The inspectors attended a full day (8 hour) O.C. meeting on March 28, 1991. The normal frequency of the full day sessions is bi-monthly which is twice as often as required by Technical

Specifications. Throughout the entire full day session, adequate time was taken to review the materials in detail and resolve any concerns that arose. The O.C. meeting was not dominated by any individuals and had good participation by all members. The O.C. chairman kept the meeting flowing but ensured that concerns were resolved prior to addressing new review items. The responsible individuals who wrote and assembled the packages that were reviewed by the O.C. were available at the meeting to address O.C. questions. In addition, plant personnel who were associated with specific review items attended the meeting to resolve O.C. questions or concerns.

A review of a painting procedure for the torus was a representative example of the level of discussions for safety concerns during the O.C. meeting. The O.C. members raised a concern that fume explosions could result if ventilation was lost. A concern was also raised that silica left in the torus following sand blasting would migrate into other systems. The O.C. held detailed discussions on these issues and required that the concerns be resolved prior to the approval of the procedure.

The inspectors reviewed the predistribution of review materials and concluded that the majority of documents to be reviewed were sent to O.C. members far enough in advance to be reviewed prior to the meeting. Only in isolated cases did the O.C. members not receive all of the materials for review.

All reviewed materials were either approved, approved with conditions, or rejected prior to moving on to new documents. Examples of conditional approval were the following: approval of procedures conditional on incorporation of O.C. comments, O.C. approval of a procedure with the confirmation that there would be a three party review outside of the O.C. meeting to resolve concerns of what materials should be included in a procedure, and O.C. approval contingent upon resolution of an Open Item Tracking System (OTS) open item. These conditional approval methods improved the efficiency of the O.C. by ensuring that packages needing only minor changes did not need to be resubmitted and reviewed by the entire O.C. again. Generally, one O.C. member was given the authority to make the final approval for the committee.

In addition, one recent initiative that greatly improved the efficiency and adequacy of O.C. reviews was the formation of a mod design team and a mod review team. The mod design team was a multidisciplinary group assembled to perform the design work on modifications. This greatly enhances the correctness and adequacy of plant modifications in that design problems are discovered and corrected early in the process.

The mod review team, which performs a comprehensive review at the completion of the design effort by the mod design team, is a multidisciplinary group composed of the following persons: general superintendent of engineering (plant), general superintendent of engineering (HPD), superintendent of electrical/civil/mechanical engineering, superintendent of quality services, superintendent of design standards, design change administrator, the system engineer for the system involved, the project engineering persons who perform mod package write-up, and the general superintendent of maintenance. The formation of these teams has greatly improved the quality of the mods submitted to O.C. for review, as evident by the reduced number of comments and rejections. These teams were found to supplement the O.C. review process and ensure the adequacy of the mods.

One weakness noted was that the O.C. meeting minutes were in a checklist format, indicating that a particular document was reviewed, but containing no written descriptions of safety issues, concerns, or differences of opinion among members at the meeting. As a result, the minutes were of minimal use for the Safety Audit Committee (SAC) to perform its oversight function of the O.C. The inspectors were informed that the checklist approach of tracking and documenting O.C. meetings had been used for two years.

In summary, the O.C. performed its required activities and contributed to the self-assessment of plant activities; however, additional attention should be given to O.C. meeting minutes.

(2) Safety Audit Committee (SAC)

The Safety Audit Committee (SAC) (offsite safety review committee) met quarterly to discuss plant conditions and events. The meetings were generally scheduled for one day at the Monticello Plant, one day at Northern States Power Corporate offices, and one day at the Prairie Island Plant. This was twice as often as required by Technical Specifications which was indicative of management's commitment to address safety issues, resolve problems, and maintain an awareness of plant events.

A review of SAC meeting minutes for the past year indicated that all SAC meetings were attended by a majority of SAC members. In addition, a very large number of non-members were in attendance at the meetings. The meeting minutes were in adequate detail to give a clear picture of the concerns expressed by SAC members, and how those concerns were addressed. SAC members generally submitted their concerns and questions, following the review of plant-provided materials, prior to the SAC meetings so that the plant had time to prepare a response for the SAC meeting. The selected plant staff then attended the SAC meeting to present relevant materials to address the SAC's concerns. Also, as additional questions come up, appropriate plant personnel were brought to the meeting to clarify or discuss the issues. By using this process, most questions were resolved at the meeting and very few open items had to be carried forward. There was a very good distribution of minutes with copies going to all relevant corporate and plant staff.

All significant plant materials were sent to all SAC members. The distribution of materials for review was tracked to maintain a history of what had been reviewed. The members decide which of the materials they would review based on their expertise. Any items that needed additional discussion or that represented a concern were added to the SAC meeting agenda.

The SAC membership is multidisciplinary with individual qualifications being tracked by a matrix to ensure a level of expertise in various functional areas. The inspectors reviewed the resumes of the SAC members and found a long history of nuclear experience.

Several SAC members had participated in Power Supply Quality Assurance (PSQA) audits during the last year. In addition, PSQA audits are reviewed by SAC members and recommendations were made for additional areas for review, or for improvements that could be made in the audits.

One weakness noted in SAC reviews was that, since Operations Committee (O.C.) meeting minutes were in a checklist format, SAC members could not review areas that may have been items of disagreement, areas of unreviewed safety concerns, or areas where additional discussion may have been needed. Also, a large number of O.C. meeting minutes mailed to SAC were stamp dated the same day indicating that O.C. minutes were not distributed in a timely manner to SAC for their review. SAC had not brought this item to O.C. attention.

The administration of the SAC process, which included the distribution of materials, setting up the agenda, ensuring SAC member participation, and issuing minutes, was performed in an excellent fashion.

The SAC contributed substantially to the licensee's self-assessment process.

(3) Power Supply Quality Assurance (PSQA)

The PSQA group is the licensee's audit group. The PSQA group had recently experienced several organizational changes. The basis for the reorganization was to make more efficient use of manpower, and better focus the efforts of the five PSQA individuals working at the Monticello site. The PSQA group performed required Technical Specification audits and an equal number of discretionary audits. The estimated number of each will be 30 for 1991 calendar year. This will represent a substantial increase in discretionary audits from 1990 when the plant was increasing the number of lead auditors.

The size and qualification of the PSQA staff were adequate to perform detailed audits. There has been a low turnover in the PSQA group during the last year. Seven full time lead auditors were available to perform audits, of which five individuals had engineering degrees. The lead auditors were, on average, performing 8-10 audits per year. With time off for training, vacations, etc, this equated to one audit per person per month. Approximately 5 to 10 percent of the QA staff's time was spent in training.

Reviewed audit reports were found to be fairly short (3 to 8 pages of written material) but indicated a detailed review of the subject area. A recent reformatting of reports provided for a new "management summary" at the beginning of the audit to give management a more instant indication of audit results. The new method of performing audits, where required audits were divided precisely by functional area, appeared to have worked well over the last year. Plant personnel and SAC members were requesting audits for areas of concern. An example was a requested audit of the licensee's corrective actions taken for the findings in the QA program as determined by a Stone and Webster audit. The majority of the Stone and Webster audit findings had been well addressed with no additional corrective actions needed. These two audits indicate that plant staff were proactive in determining PSQA problems and taking actions to correct the problems.

The technical expertise in the QA program was increased as a result of hiring additional engineers. All engineers but one were qualified as lead auditors. In addition, SAC member participation was requested depending on the member's expertise in a specific audit area. The areas inspected by each auditor were specified so that individuals gain expertise in specific areas. A review of audits indicated that the audits were not highly performance-based. But this slight deficiency was compensated for by the surveillance groups on-site who performed performance-based inspections. Approximately 150 surveillances were performed by the QA surveillance staff and approximately 150 surveillances were performed by the Nuclear Projects Department (NPD) surveillance staff.

The licensee allocated a large number of QA resources to the refueling outage. The full-time QA coverage for the outage was 3 to 5 persons. The audits and surveillances performed during the outage were very performance based. A recent organizational change aligns the five surveillance individuals under one QA manager who will be stationed at the site rather than at the corporate office. These individuals worked to a monthly plan of activities to be inspected, but they were allowed the freedom to pursue problem areas when one was found. The plant was timely in implementing corrective actions to resolve surveillance findings. In addition, the surveillance findings were reviewed and used for background information by the individuals performing QA audits.

The QA audits generally included many recommendations for work improvements. However, the QA group had no tracking program for those recommendations unless they rose to the level of findings. Because of that, plant management believed that there was some tendency for QA to elevate minor items to the level of findings simply so they could track their resolution. The plant did select and track corrective actions for important recommendations. The QA department was rejecting between 10 to 30 percent of the plants corrective actions for findings as inadequate. This indicates that the plant is not always thorough in its corrective actions. The other area of questionable effectiveness was that no formal root cause assessment systems existed in the QA program. The group does review audits to determine similar causes of problems, but this process is limited to the QA's staffs memory of similar events. Overall, PSQA has contributed to the licensee's quality verification and assessment of plant problems.

(4) Human Performance Task Force

The Human Performance Task Force (HPTF) was a 12 member group that met bimonthly to evaluate plant conditions and events for human performance problems. The HPTF consisted of a chairman and plant employees from Operations, Instrument and Control, Mechanical Maintenance, Electrical Maintenance, Engineering, Chemistry/Health Physics, Administration Services and Secretarial Support, Quality Engineering, Training, and Nuclear Projects Department (NPD). The NPD employee was added to the HPTF following organizational changes at the Monticello Plant. The group's recommendations, based on reviews and investigations, must be unanimous. The corrective actions recommended were implemented as low in the organizational structure as possible. A computerized tracking system was used to track recommendations. To date, all recommendations that have been made as a result of HPTF reviews have been implemented. This has been as a result of the site management's and corporate management's support of its activities.

One person per quarter was sent to Human Performance Evaluator training. Three individuals were to be sent to training in 1991 due to the resources required for the current refueling outage. All current members, except the member from NPD, have been to the Human Performance Evaluation System (HPES) training at least once. All of the plant's engineering staff were also trained in the HPES process.

Fourteen HPES root cause evaluations were performed during the past year. To ascertain the root cause of events once the investigation starts, no disciplinary action was allowed against employees so that they were open in discussing their involvement. One observation made by plant staff about the evaluation process was that it displaced many assumptions that were often held by a large number of staff personnel, but turned out not to be true. The outcome of an HPTF investigation was often greatly different from the assumed root cause for events. Monticello's unique approach to the HPES process using a HPTF contributed substantially to the licensee's self-assessment of human performance issues.

(5) Quality Services

The quality services group performed the onsite quality assurance/quality control (QA/QC) function. The main tasks of the onsite QA/QC group consist of normal QC functions, upkeep of the Administrative Control Directives and Administrative Work Instructions (ACD/AWI), surveillances, in-line review of modifications and documents issued by the engineering staff, receipt inspections, and analysis and trending of deficiencies and findings. Part of these duties were planned to be transferred to other site organizations to improve the efficiency of the work performed.

Improvement over the previous year was noted in the timeliness of the issuance of the quarterly summary report for this group. The trending of events and plant problems by Quality Services has been developed to follow many of the HPES categories for classification. One weakness was that the data used for trending was often not verified by an independent investigation. Thus, the actual root causes could be different than that indicated on deficiencies and could contribute to errors in the trend data and in the analysis of the trends.

The inspection of work activities by quality assurance inspectors was performed on a sampling basis. This allowed management flexibility to send the auditors to those jobs with the greatest importance. Jobs were also categorized by required expertise to perform the inspection, so that any individuals within the Q.C. staff could inspect certain jobs where a job did not require a great deal of technical expertise.

The quality services group had improved its work activities and contributed to the quality verification and self-assessment of the Monticello site.

No violations or deviations were identified.

7. Exit Interview

The inspectors met with the licensee representatives denoted in Paragraph 1 at the conclusion of the inspection on April 16, 1991. The inspectors summarized the purpose and scope of the inspection and the findings. The inspectors also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspectors during the inspection. The licensee did not identify any such documents or processes as proprietary.