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REGION I

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Licensee: Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Facility: Nine Mile Point, Units 1 and 2

Location: Scriba, New York

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Reactor Projects Section No. 1A
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__Inspection Summary__: This inspection report documents routine and reactive inspections of plant operations, radiological controls, maintenance, surveillance, and safety assessment/quality verification activities.

__Results__: See Executive Summary.

EXECUTIVE SUMMARY

Nine Mile Point Units 1 and 2
NRC Region I Inspection Report Nos. 50-220/92-29 & 50-410/92-34
12/13/92 - 01/23/93

Plant Operations

The control room staffs operated both units safely. Good integrated efforts in support of safe operation by the operators and the site support departments were observed. This included identification and evaluation of Unit 1 operation at 100.5% of rated thermal power for 12 hours, leak rate testing concerns, and reaction to the inadvertent opening of a switchgear normal supply breaker. In an isolated instance a station shift supervisor at Unit 1 was observed to be displaying inappropriate control room demeanor.

Radiological Controls

Routine tours indicated that site personnel used good radiological practices. Two minor weaknesses were observed, which if they occurred under different conditions could have resulted in exposures that were not as low as reasonably achievable or to unmonitored exposure. An unresolved item was opened on the controls implemented by NMPC for air system sampling programs.

Maintenance and Surveillance

The maintenance and surveillance activities observed were well conducted. Further, maintenance department personnel supported the safe operation of both units in their responses to day-to-day operational situations. Ultrasonic thickness measurements of the Unit 1 torus were consistent with the previous measurements.

Engineering and Technical Support

Testing of safety-related pump motor loads indicated that NMPC may not have used conservative values in their Unit 1 emergency diesel generator (EDG) load study. Specifically, the degradation of pump and motor combinations will cause increased motor load, which could impact the EDG load study. The EDG load study remained an unresolved item with NMPC reviewing the impact of degraded pump performance. Review of NMPC's secondary containment unit cooler engineering evaluation at Unit 2 continued. The testing and flushing of these coolers and the design assumptions for heat removal could not be directly related to the design accident service water configuration. This issue remained unresolved pending further review.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

1.1 Niagara Mohawk Power Corporation Activities

The Niagara Mohawk Power Corporation (NMPC) operated both Nine Mile Point Unit 1 (Unit 1) and Unit 2 (Unit 2) safely and essentially at full power during this period. On January 18, NMPC restructured the nuclear support organization when Mr. J. Firlit, who had been the Vice President - Nuclear Support, left the company. The groups that previously reported to Mr. Firlit divided to report to either the Vice President - Nuclear Generation or the Vice President - Nuclear Engineering.

1.2 NRC Activities

Resident inspectors conducted inspection activities during normal, backshift and weekend hours over this period. There were 16 hours of backshift (evening shift) and 11 hours of deep backshift (weekend, holiday, and midnight shift) inspection during this period.

2.0 PLANT OPERATIONS (71707,93702)

2.1 Routine Control Room Observations

Operators generally conducted routine control room activities well at both units. Operators efficiently conducted shift turnovers, including panel walkdowns for familiarization with current plant conditions and shift briefings. The quality and content of the logs was good. Normal control room communications continue to improve following removal of the SSS office walls at Unit 1 and remained strong at Unit 2. SSSs and assistant SSSs (ASSSs) provided good direction to the reactor operators for reactivity changes. This included reactor operators discussing and receiving approval from the senior operators before raising or lowering reactor power, to maintain rated thermal power.

Control room demeanor was generally maintained professional and productive. However, on December 24th, the inspector entered the Unit 1 control room and observed the on-shift SSS sitting at his desk in a posture that was not indicative of proper control room demeanor. The inspector approached the SSS and assessed that he was not asleep, although he did have his feet on the desk and his eyes closed. The inspector proceeded to conduct routine business with the SSS who was communicative and fully alert. The ASSS and other operators were fully alert to control room activities. The inspector discussed this observation with NMPC management who then took appropriate actions to counsel the individual involved and discuss control room demeanor with the operating crews.

2.2 Morning Meetings

NMPC began holding morning meetings, conducted by either the on-watch SSS (Unit 1) or the off-going SSS (Unit 2), to set the priorities for the day. NMPC attendance included the supervisors from the maintenance, operations, radiation protection, chemistry, and construction services departments, and others planning or with a need to understand the daily work. The inspector attended many of these meetings at both units and found that, while the format was different, they were successful in setting the goals and priorities needed by the operations staffs to support safe operation of the plants.

2.3 Exceeded Licensed Thermal Power Level

NMPC adequately responded to the January 11 determination by reactor engineering that for about 12 hours, Unit 1 exceeded its rated core thermal power of 1850 megawatts (MW); 1858 MW (100.5%). The reactor analyst identified that the reactor water cleanup (RWCU) flow was not being recognized by the process computer's core thermal power calculation. This resulted in the process computer not accounting for the approximately 8 MW thermal of reactor generated heat being removed by RWCU and a digital thermal power indication 8 MW less than actual. It appeared that the RWCU flow computer point had dropped out inadvertently, due to flow disturbances, when RWCU was placed in service the previous day. These flow disturbances apparently caused the computer to see the signal as a failed sensor at which point it was dropped from the computer scan. Thus when operators increased power following return of RWCU to service, the thermal power digital display indicated 1850 MW while power was actually 1858 MW or 100.5% of rated.

Immediate corrective actions included reducing recirculation flow to restore power below 1850 MW thermal, initiating a deficiency event report (DER) to learn the root cause of the event, and completion of a one-hour 10 CFR 50.72 notification for operation above licensed thermal power. Using stored data in the process computer, the reactor analyst was able to determine that thermal limits had not been exceeded during the time at 100.5% of rated thermal power. The inspector discussed the event and the preliminary findings of the root cause evaluation with the operations general supervisor. NMPC planned to complete several corrective actions to prevent recurrence, which included: completing necessary repairs to the RWCU system during the upcoming refuel outage to allow backwash and precoating RWCU filters without isolating the system; revision to the RWCU operating procedure to reduce RWCU system differential pressure when placing RWCU filters in service; and shift technical advisor (STA) performance of an OD-3, core thermal power calculation, once per shift and following events which may impact input to the core thermal power calculation. The inspector considered that the immediate actions taken were timely and adequate; this issue remained open pending review of the corrective actions to be documented in NMPC licensee event report (LER) 93-01.

2.4 Loss of Power to Safety-Related Switchgear

On January 5, Unit 2 operators, system engineers, and maintenance personnel responded well to a loss of power to the Division II emergency switchgear. Inadvertent opening of the normal supply breaker (103-04) at the emergency bus isolated the bus from reserve transformer B and off-site power line 6. The Division II EDG started and powered its bus on a loss of voltage signal. Operators and system engineers conducted a thorough investigation into the reason for the breaker opening; including an assessment of surveillance testing completed just before the breaker opened. NMPC decided that the surveillance testing on the under- and degraded-voltage relays had not caused the breaker to trip. The most probable cause was a fault in an optical isolator supplying the breaker control circuit with a trip signal from the non-safety related reserve transformer B protection circuit.

The inspector independently reviewed the under- and degraded-voltage relay surveillance test and determined that it had not caused the trip. This test tripped each under- or degraded-voltage relay one at a time and verified that the relay re-energized before going to the next relay. Since the under- or degraded-voltage trip requires two relays to see the low voltage condition, it could not have caused the trip.

Operators showed sensitivity to fuel oil build up in the EDG exhaust header (souping) with the machine running with very low loads and with the availability of fuel oil. The operations staff determined that due to the souping concern the EDG would be paralleled with the off-site system through breaker 103-04 to allow loading, and that the EDG would not be secured until resolution to the optical isolator problem was reached. Operators paralleled and loaded the EDG smoothly. Breaker 103-04 was reopened prior to I&C commencing work on the optical isolator to prevent an inadvertent trip. The SSS, ASSS, and STA evaluated fuel oil consumption and discussed plans for getting fuel oil on-site if the EDG needed to be run for an extended time.

I&C technicians and supervision effectively put together a work plan to replace the optical isolator. The inspector observed the disassembly of the optical isolator. When the electrical connection on the safety-related side of the device was removed there was a small fire in the connector. Based on this, a troubleshooting/repair plan was put together to replace the optical isolator and the damaged connector. Since the operations department was not sure how long this repair activity would take, they decided to repower the Division II from off-site line 6. This was done using an existing station procedure; by installing a breaker in the alternate breaker enclosure, which allowed alignment to the auxiliary boiler transformer and by realignment to the auxiliary boiler transformer to off-site line 6. The optical isolator work was completed on January 7 and the Division II switchgear was returned to its normal power supply from reserve transformer B.

2.5 Leak Rate Testing Issues

Unit 1 control room operators showed good safety perspective when they questioned the performance of two 10 CFR 50, Appendix J leak rate tests when the primary containment was required to be operable. The two penetrations were for systems that are no longer in use. The first was the torus makeup line initially installed to allow torus makeup from the condensate transfer system, through a normally closed outside containment isolation motor operated valve. Subsequently, a blank flange was installed downstream of the motor operated valve and a manual valve further downstream..was closed leaving the line unusable. The operators questioned if opening the local leak rate (LLRT) test connection, located between the blank flange and the closed manual isolation valve was a breach of containment.

The second was the reactor head spray line originally used to spray control rod drive water into the reactor vessel steam space through a normally closed outside containment motor operated isolation valve and an inside containment check valve. This line had been disconnected inside the drywell, with blank flanges installed on the vessel head and on the pipe downstream of the check valve. The operators questioned whether opening the LLRT connection between the check valve and the motor operate containment isolation valve was a breach of primary containment. The operator questioned these issues and did not allow the performance of the testing pending management resolution.

On January 15, the inspector attended a meeting between system engineering, operations, I&C supervision, licensing, and corporate engineering, where these issues and other Appendix J testing, to be completed before the scheduled February 19 reactor shutdown for refueling, were discussed. The system engineer charged with the Appendix J program conducted the meeting very well. Each case was reviewed in detail and plans for completion of necessary safety evaluations were discussed.

The inspectors found the safety evaluation performed to address these concerns had been well prepared and properly reviewed and approved. For the torus makeup line, the safety evaluation qualified the closed manual valve as a primary containment isolation valve. This was done by reviewing the previous Appendix J leak rate test performed while shutdown and by reviewing the physical characteristic of the valve. The evaluation for the head spray line determined that the blank flange installed inside the containment was the Appendix J boundary and that type B testing of the flange was required, rather than type C testing of the isolation valves. The inspector found that NMPC adequately determined that opening the test valves in either case would not be a breach of containment.

2.6 Power Distribution System Plant Walkdown

The inspector completed a general walkdown of the Unit 1 AC and DC power distribution system, using system drawings to determine expected breaker positions for various power boards. The inspector determined that the electrical distribution system was maintained in the proper configuration to support plant operation. General cleanliness and equipment labelling identification also appeared satisfactory.

2.7 (Closed) Unresolved Item (50-220/92-070-01): Inoperable Emergency Diesel Generator (EDG) Technical Specifications Issue

This unresolved item identified inconsistencies in the technical specification (TS) requirements and technical specification interpretation (TSI) #29, dealing with the operability of the EDGs in support of core spray (CS) system operability with Unit 1 in the cold shutdown condition. NMPC completed a revision to TSI #29, dated October 8, 1992, to clarify the TS operability requirements for EDGs and CS while in cold shutdown.

-- The identification of the TS 3.1.4.f requirement for operability of either the normal or emergency power source associated with the core spray subsystems whenever fuel is in the reactor vessel in the cold shutdown condition.

-- The identification that TS 3.4.4.a and 3.4.5.a require both EDGs to be operable whenever secondary containment integrity is required.

-- The identification that TS 3.6.3.a requires both EDGs and both 115 Kv lines operable to support the refueling condition.

The inspector concluded that the revision to TSI #29 was clear and consistent with the TS operability requirements. This item is closed.

3.0 RADIOLOGICAL AND CHEMISTRY CONTROLS (71707)

3.1 Routine Observations

The inspectors routinely toured radiation and contaminated areas at both units and observed personnel radiation protection practices and postings. Routine tours at both units identified minor radiological concerns, such as a catch containment that was not positioned to catch condensing water in the Unit 1 emergency cooling steam valve room. Radiation protection personnel quickly corrected these issues.

During preparation for surveillance testing on the low pressure core spray pump at Unit 2, the inspector observed personnel waiting in a posted radiation area with dose rates of about 2 mr/hr for the test to begin. The radiation area boundary was the pump room door and the individuals did not exit the radiation area, after installing test equipment, while waiting for the pump test

to commence. Also, during the surveillance testing discussed above, the inspector observed an individual take off his dosimeter and TLD several times, placing them on the floor, to allow easier reading of test equipment. Because of the low dose rates involved, these specific instances were not significant. However, in higher dose rate areas such instances could lead to radiation exposure that was not as low as reasonably achievable (ALARA) and to unmonitored exposure. The inspector discussed these observations with the radiation protection managers for both units, who were pursuing corrective actions.

3.2 Uncontrolled Sampling Rig

The identification of sampling equipment tied into the pneumatic operator supply line for the inside containment suppression chamber vent valve, without the knowledge of the operations crews, concerned the inspector. This equipment was attached to a manual drain valve upstream of the solenoid operated containment isolation valve and consisted of tubing, a pressure regulator, a pneumatic flow meter, and a filter paper housing. The 200 SCFH flow shown on the flow meter indicated that the manual valve was throttled open. The inspector reviewed plant drawings for the system and determined that, in that plant condition, nitrogen was being vented through the test equipment, and asked the SSS why the equipment had been installed. The SSS was not aware of the equipment and determined that it had been installed by the radiation protection department as part of routine sampling of the instrument air system to verify that radiological contamination had not occurred.

Review of the procedure that installed the equipment showed that there were no special controls over the equipment; the SSS did not need to be notified of its installation or the throttling of the plant equipment blocking valve. The intent of the procedure was sampling of the instrument air system monthly, at each unit. This was a commitment to NRC Bulletin 80-10: Contamination of Non-radioactive Systems and Resulting Potential Un-monitored, Uncontrolled Release to the Environment.

The administrative procedure for temporary modifications allows the installation of temporary equipment if it is controlled by a procedure, including the notification of the SSS and verification of configuration. However, there is a special case for sampling equipment that allows its installation and use, if controlled by procedure. It was not clear if this procedure needed to have the standard procedure controls for temporary equipment. Radiation protection department management stated that the procedure would be revised to prescribe the normal procedure controls for temporary equipment.

The procedure was deficient in that drawing a sample at the specified location, with the given plant conditions, did not allow sampling of the instrument air system. With the containment inerted the sample location would see nitrogen, not instrument air. This alignment is specified by operating procedures to prevent air leakage into the containment when it is inerted. The inspector considered this an unresolved item, pending review of the sampling procedures for other air systems and NMPC commitments to NRC Bulletin 80-10. **(220/410/92-29-01/92-34-01)**

4.0 MAINTENANCE (62703)

Through observations of safety-related maintenance activities, interviews, and review of records, the inspectors verified the: proper use of administrative authorizations, and tag outs, adequacy of procedures, use of certified parts and materials, calibration of measuring test equipment (M&TE), proper implementation of radiological control requirements, use of controlled system prints and wire removal documentation, and proper establishment of quality control hold points.

4.1 New Fuel Inspection and Storage

Unit 1 maintenance personnel properly received, inspected, and stored 172 new fuel bundles in preparation for the upcoming refueling outage. The inspectors observed various portions of the new fuel receipt process performed per N1-MMP-FHP-2, 3, 4, and 5 and interviewed several mechanics and operations department personnel involved with the evolution. Specific items observed included: radiological surveys of the metal shipping container; bundle removal and transfer to the inspection stand; new fuel bundle inspection; bundle channel inspection, cleaning, and mating; and fuel assembly transfer to the new fuel vault. Particular inspector attention was given to the bundle transfer to the inspection stand due to past problems with this part of the evolution (see section 4.6). While lifting the metal shipping container, which contained two new fuel bundles, the inspector observed independent verification of the installation of the bundle hold-down fixtures and safety straps, implementation of additional administrative controls over the evolution, and an overall high crew awareness with respect to procedural adherence and safety. Supervisory oversight was evident and good radiological and cleanliness controls were observed. All personnel interviewed were very knowledgeable of the procedure, inspection techniques, past or possible problems, and management expectations regarding the evolution. Several minor problems identified during the bundle inspections were properly dispositioned by NMPC with assistance from the vendor. In summary, this evolution was well controlled by knowledgeable mechanics who showed a proper safety perspective.

4.2 Emergency Diesel Generator Corrective Maintenance

On December 22, the inspector observed good coordination between Unit 2 operators and I&C technicians when the Division I EDG did not come up to speed within the required time during monthly surveillance testing. The cause for this was isolated to the non-emergency start air operated valve. Under emergency conditions this valve would not have been called upon to function. The valve was replaced and the EDG satisfactorily retested. The special report submitted by NMPC on January 21, 1993, adequately addressed this issue.

4.3 Emergency Condenser Thermocouple Replacement

I&C technicians properly replaced thermocouple 60-26 for emergency condenser (EC) 121 because of a broken ceramic insulator discovered during a biennial calibration. The thermocouple provides control room indication and alarms for the EC shell water temperature. Before and after the discovery of the cracked insulator, EC shell temperature indications were

normal and the system was considered operable. The inspector observed portions of the calibration for the comparable thermocouple on EC 122, replacement of the damaged thermocouple, and the associated post-maintenance calibration. The inspector also reviewed the calibration data for the other instruments calibrated per N1-IPM-060-002, EC miscellaneous instrument calibration, associated with EC system channel 12. Throughout the calibrations and the thermocouple replacement, the inspector observed good radiological control practices, communications, procedural adherence, and control of M&TE. Minor discrepancies identified by the I&C technicians were properly dispositioned. In summary, all aspects of the calibration and thermocouple replacement were considered satisfactory.

4.4 Recirculation Motor Generator Flow Stop Setting

On December 30, the inspector observed very good coordination between the operations crew, the maintenance department, and system engineering during establishment of recirculation pump motor generator set electrical and mechanical high flow stops. The pre-evolution briefing was well conducted and operators asked good questions, which clarified their understanding of the operation. The settings were conducted smoothly and professionally.

4.5. Off-site Power Line Outage

During the week of December 21, the inspector observed good coordination of construction activities, at Unit 2, which could have affected a Unit 1 off-site power supply line. Off-site line 4, which runs between Unit 1 and FitzPatrick was initially deenergized to allow construction work on a new storage building near the Unit 2 cooling tower. The proper TS limiting condition for operations (LCO) was entered. Following completion of this work the line was properly returned to service.

4.6 (Closed) Violation 50-410/91-24-01: Dropped New Fuel Bundles

On December 19, 1991, Unit 2 maintenance personnel failed to follow the new fuel inspection procedure, which resulted in damage to two new fuel bundles when they were dropped while being transported to the new fuel inspection stand, rendering them unusable. Poor worker practices, a lack of supervision for the specific job, and ineffective management actions to correct previous inattention-to-detail issues contributed to this event. The safety consequence of this event was low because the fuel bundles had not been irradiated and there were no radiological consequences. On February 6, 1992, during an enforcement conference between NRC staff and NMPC, NMPC admitted to the violation, and discussed its causes and comprehensive short- and long-term corrective actions. The corrective actions included disciplinary action against the foreman responsible for the event and the supervisory personnel who did not properly log and communicate a near miss precursor. A lessons learned transmittal was issued to alert others to the personnel errors made and the consequences of those actions. The new fuel inspection procedure was revised to include positive control measures designed to provide assurance that the procedure would be consulted frequently and to provide a measure of defense-in-depth. Following the enforcement conference, the NRC staff assessed that NMPC

took or planned to take appropriate corrective actions to address the issues. Subsequently, the inspectors reviewed various aspects of the corrective actions and observed satisfactory implementation of the procedural changes during Unit 1 new fuel inspections. This violation is closed.

5.0 SURVEILLANCE (61726, 61707)

Through observation of safety-related surveillance activities, interviews, and review of records, the inspectors verified: use of proper administrative approve, personnel adherence to procedure precautions and limitations, accurate and timely review of test data, conformance of surveillances to technical specifications, including required frequencies, and use of good radiological controls. Surveillance activities observed included those listed and discussed below:

N2-ESP-BYS-W675	125 Volts DC weekly battery surveillance
N2-EPM-GEN-V580	2GTS*MOV2A/3A breaker removal, inspection, and megger
N2-ISP-MSS-M002	Main steam line high flow instrument channel functional test
N2-OSP-RMC-W01	Control rod movement and position indication verification
N1-ITP-05	Thermography
N1-ISP-092-328	APRM #18 instrument channel calibration
N1-ST-06A	Containment spray loop 111 quarterly operability test

The above activities were effective with respect to meeting the safety objectives.

5.1 Assessment of Torus Wall Thickness Measurements

The inspector completed a general tour of the Unit 1 torus area and identified the locations for the torus ultrasonic test measurements. NMPC completed torus wall ultrasonic thickness measurements on January 11 for each of the twenty bays at the lower mid-bay inside and outside plates. A 13 by 5 grid divided each plate for a total of 130 measurements on each torus bay. The inspector reviewed the test data and determined the results were above the minimum wall thickness of 0.447 inches. The inspector found this data consistent with the previous measurements taken in August 1989. The inspector reviewed the RWP for the work and concluded the radiological controls were appropriate for the data collected.

5.2 Quarterly Reactor Recirculation Flow Loop 11 Calibration

Unit 1 I&C technicians properly performed N1-ISP-032-008, quarterly reactor recirculation flow loop calibration for channel 11. This surveillance satisfied the TS requirements and verified the operability and calibration of the reactor recirculation flow instrument and core differential pressure loop, which provide APRM flow and computer signals for input to flow biased scram setpoints. This surveillance required the insertion of a manual half-scram for approximately 8 hours. Because of the potential plant impact, close coordination between operations and the

I&C technicians was required and observed. The individuals performing the procedure were very knowledgeable and performed the surveillance expeditiously to reduce the time with a half-scram inserted. The M&TE used was properly calibrated and the surveillance met the acceptance criteria.

5.3 Anticipated Transient With Scram Channel Calibration Check

Unit 1 I&C technicians satisfactorily performed N1-ISP-036-010, the anticipated transient without scram and alternate rod insertion (ATWS/ARI) instrument calibration. This surveillance (performed once per cycle) verified operability of the instruments and trip circuitry of the ATWS ARI, and reactor recirculation pump trip. The inspector observed and reviewed parts of the surveillance related to channel 11 reactor vessel level transmitter LT 36-21A and interviewed several technicians and operators involved with the test. The technicians showed a thorough knowledge of the procedure and associated past problems. Good communications and radiological control practices were observed.

5.4 Monthly Loss of Off-Site Power/Loss of Coolant Accident Channel Functional Test

Unit 2 electrical maintenance personnel satisfactorily performed N2-ESP-ENS-M731, monthly off-site power/loss of coolant accident (LOOP/LOCA) channel functional test. This surveillance satisfied the requirements of TS 4.3.3 and verified the operability of the low pressure emergency core cooling system (ECCS) pump automatic start time delay relays under both normal and emergency power conditions. The inspector observed good coordination and communication between the electricians and the control room staff, which minimized the amount of time in a TS LCO. The electricians performing the surveillance demonstrated an excellent knowledge of the procedure and its overall plant impact. Supervisory oversight was evident and all M&TE was properly calibrated. The inspector reviewed the surveillance results and confirmed that the test verified operability of the low pressure ECCS pump automatic start time delay relays.

6.0 ENGINEERING AND TECHNICAL SUPPORT (71707, 92703, 37700)

6.1 Failure of a Hydraulic Control Unit High Point Vent to Fully Shut

On January 14, during Unit 1 control rod drive diagnostic testing, the high point vent valve (301-133) on hydraulic control unit (HCU) 10-31 would not fully close. This prevented disconnection of the diagnostic test equipment. NMPC questioned the operability of control rod 10-31 with the valve open and test equipment installed, inserted the control rod, and adjusted recirculation flow and other control rod positions to increase reactor power, within core thermal limits. NMPC completed a 10 CFR 50.59 safety evaluation for the configuration and determined that operation with the high point vent valve not fully closed and the test equipment installed was acceptable. In addition, the safety evaluation determined that rod 10-31 was operable and could be fully withdrawn. The inspector found that NMPC provided adequate technical basis for continued operation with the test equipment installed.

6.2 (Closed) Unresolved Item 50-220/91-80-10: Pump Curves to Verify Power Demands

This item dealt with the adequacy of the assumptions used in development of electrical loads placed on the Unit 1 EDGs by large safety-related pumps. The issue was that some manufacturers pump curves were not clear or did not include as-built pump horsepower and efficiency curves. NMPC completed the upgrading of the pump curves, which are maintained as controlled documents in mechanical design criteria (MDC) -11. This design information was used by NMPC engineering in establishing the EDG loading calculations discussed below. Based on this the inspector considered this item closed.

6.3 (Open) Unresolved Item 50-220/91-80-14: Emergency Diesel Generator Loading Calculations

This item dealt with the adequacy of the Unit 1 EDG load calculations. In response to Unresolved Item 91-80-10, discussed above, NMPC committed to perform actual electrical load measurements during quarterly pump surveillance testing on the large safety-related pumps, to allow updating or validation of the EDG loading calculations.

On January 6, one week before the scheduled commencement, the inspector met with NMPC engineering and licensing personnel to discuss the testing. NMPC engineering did not want to perform the testing because they were not sure that efficiencies of the pump motors could be developed based on measured data (i.e., voltage, kw, vars, power factor) and that the test may not be useful. The inspector asked if the overall efficiency of the pump-motor combination could be determined within the accuracy of the instrumentation used to gather the motor electrical parameters and the pump hydraulic parameters, and whether this total efficiency could be compared to the design efficiencies of the pump and motor combination. They stated that they thought that this was possible. Following this meeting, the inspector determined that NMPC engineering and licensing had not discussed the reasons for not wanting to conduct the testing with senior NMPC management, who had made the commitment, before approaching the inspector. The inspector discussed this issue with the Unit 1 Plant Manager and the Vice President - Nuclear Generation. They stated that a change to a written NRC commitment needed management approval, before approaching the NRC.

NMPC conducted the first tests on the 111 containment spray and 111 containment spray raw water pump-motor combinations on January 14. The inspector observed the installation of the electrical recording instrumentation at the pump motor breaker cabinets and found the technicians knowledgeable about the data to be taken and how the equipment functioned. The data (voltage, kw, amperage, vars) was taken while running the pump during the normal quarterly IST surveillance testing.

Following the testing the inspector received the electrical and pump performance data , the pump curves, and motor design information for these two pumps, and performed calculations to evaluate the performance of the pump-motor combinations. For the containment spray pump the efficiency and the motor electrical load were within the original design specifications and within the load specified in the EDG load study.

For the containment spray raw water pump, the motor drew more than its designed kw load for the given test condition. The design pump and motor rating is 500 shaft horsepower (373 kw) and the design efficiency of the motor at converting electrical energy to mechanical energy was 0.92, meaning that the motor should draw the about 405 kw when the motor is producing 500 shaft horsepower. For the tested condition of 2940 gpm, from the pump manufacturers' curve, the pump required about 495 shaft horsepower with a design efficiency of 0.84, meaning that the pump would be capable of producing 415 horsepower of pump work. This also meant that the design efficiency of the entire pump-motor combination should be approximately the product of the motor and pump efficiencies or 0.76. The measurements of electrical power to the motor showed that 430 kw was needed to produce about 375 horsepower (280 kw) of pump work. Thus the efficiency of the system as tested was 280/430 or 0.65.

Inspector review of the EDG load calculations showed that NMPC had assumed the 405 kw load for the pump based on rated conditions. Actual condition testing indicated that the pump-motor combination was not as efficient as designed, requiring about 430 kw. The inspector discussed this with the NMPC engineer reviewing the calculations, the engineer stated that this did take up some of the approximate 100 kw of margin in the EDG loading calculations.

Assuming that the motor efficiency did not change and that the only change occurred in the pump performance; the pump efficiency at the time of the test would be the tested pump-motor combined efficiency (0.65) divided by the design motor efficiency (0.92) or 0.70. This represents a reduction of 0.14 from the design valve of 0.84. Review of the hydraulic performance of the pump showed that pump degradation from the design condition could account for the observed reduction in efficiency. At the tested flow of 2940 gpm the developed head was measured at 202 psid, from the design pump curve the head should have been about 219 psid. This represented a degradation of 202/219 or 0.92. Based on this review and the fact that only design conditions and efficiency were used in the EDG load calculations, the inspector was concerned that NMPC had not considered the effects of pump degradation in the EDG loading study. Following discussions with the inspector, the Plant Manager directed that engineering review this concern. This item remained open.

6.4 <u>(Open) Unresolved Item 50-410/91-12-04: Buildup of Silt/Corrosion Products in Secondary Containment Unit Coolers</u>

During the current inspection period, NMPC cancelled a planned modification that would have doubled the capacity of each standby gas treatment system (SBGT) train at Unit 2. This modification would have: significantly increased the bulk air removal capability from the secondary containment and reduced the post-LOCA reliance on the heat removal capacity of the

secondary containment on the service water cooled unit coolers, in establishing the required 6 minute negative pressure draw down time. To evaluate the implications of this, upon the existing performance capability of the unit coolers under design basis conditions, the inspector reviewed the status of the ongoing program for testing and flushing. The testing effort has been in progress for over 18 months and has obtained thermal performance data on most unit coolers in the plant. Back flushing of the coolers has proceeded in parallel with testing as an attempt to reverse the gradual buildup of silt and flow blockage in the service water system that has caused a reduction in the heat removal capacity of the coolers.

NRC inspection report 50-410/92-20 examined the testing and flushing program in July 1992, and developed a concern over NMPC's of methodology of interpreting technical specification requirements and secondary containment operability when one or more coolers were out of service for maintenance or testing. At that time NMPC had not completed all testing, but had concluded that an average 30% degradation in the heat removal capacities of the unit coolers existed. Some general area coolers experienced degradation that approached 40%, based upon test results from May 1991.

During this inspection, the inspector reviewed the performance test results as of October 22, 1992, for 29 of the 33 unit coolers in the test program. Notable changes from the July 1992 results were documented in the heat removal capability of all unit coolers. The following table represents individual test results of the Division I general area coolers that can be removed from service without jeopardizing the operability status of other general area coolers as permitted by the licensee's technical specifications interpretation:

Division I - Secondary Containment General Area Coolers

Cooler ID	Calculated Design Capacity (BTU/hr)	Calculated Actual Capacity (BTU/hr)	Current Actual Capacity as percent of Design Capacity	July 1992 Actual Capacity as percent of Design Capacity
404A	121,000	110,000	90.9	Untested
404B	121,000	*110,000	Untested	Untested
407A	86,000	51,000	59.3	87.6
407B	86,000	62,000	72.1	Untested
407C	86,000	65,000	75.6	62.7
410A	79,000	64,000	81.0	78.1
411A	132,000	115,000	87.1	76.9
414A	140,000	109,000	77.9	Untested
Total	≈851,000	≈686,000	80.1	N/A

*Estimated

Based upon these results, it did not appear that back flushing created a significant improvement in the thermal capacities of the unit coolers over recent months. Some coolers experienced a reduction in capacity over the past few months. Other general area coolers recently tested in Division II showed degradation up to 40% of their design thermal capacities during accident conditions. Since July 1992, NMPC used a total average capacity degradation, for all unit coolers in the test program, of about 19%; as compared to the previously assumed 30% degradation.

For the cooler test program, and for operability considerations, NMPC engineering used a qualified commercial computer program ("Aircool") to recalculate the as-designed thermal capacity of each unit cooler. This program calculated individual thermal capacities up to ± 10% different from the original design specification indicated by the manufacturer. However, the total calculated Aircool capacity for all general area coolers was approximately 99% of the total original specification for the Division I equipment.

The inspector expressed concerns regarding the licensee's methodology for calculating the maximum thermal capacity available in each cooler after the service water system realigns following an accident signal. The analysis of service water flow used simplified assumptions that the accident flows could be calculated using a direct ratio from the original design full flow capacity to accident capacity and applying that ratio to the current tested capacity. The

inspector found the licensee's assumption of a direct ratio between service water full flow at original design to flow in an accident situation unsubstantiated because (1) the licensee doesn't have a documented integrated service water system test in the accident configuration, and (2) service water flow in each cooler has been decreasing because of silting, corrosion, and growth of micro organisms. Pending further NRC review of the licensee's test results and analysis of unit cooler thermal capacities during accident conditions, this item will remain open.

6.5 Review of 10 CFR 50.59 Evaluations and Plant Modifications in Unit 1 Reactor Building Ventilation System

During a recent Unit 1 outage in May 1992, NMPC performed a temporary modification in the normal reactor building ventilation system (RBVS), by removing a valve control time delay circuit which allowed the RBVS exhaust block valves to remain open longer than the supply block valves. The timer relay was originally intended to assure that a negative pressure would be maintained in the reactor building during the transition from normal to emergency ventilation, by keeping the exhaust valves open slightly longer while the RBVS fans coasted down following a trip on high radiation levels inside containment. The licensee performed an engineering analysis and a 10 CFR 50.59 evaluation for the temporary modification to address the ventilation system's capability following a refueling accident. The evaluation concluded that the reactor building negative pressure could be maintained without the time delay function, if the total stroke time of the exhaust valves did not exceed the stroke time of the supply valves by more than one second. Functional testing of the ventilation systems was performed during the outage to demonstrate that this could be achieved if the valves remained within their stroke time limits prescribed by ASME Section XI requirements.

Following the outage, the licensee converted this to a permanent modification. The quarterly IST test procedure for the block valves was amended to provide additional acceptance criteria that requires the block valves to stroke within one second of each other. The inspector reviewed the revised 10 CFR 50.59 evaluation, calculation S10-202-HVO5, "Reactor Building Emergency Ventilation," and the ventilation test procedures used to verify acceptability of the permanent modification. Several questions arose concerning the calculation because the analysis was not revised to reflect normal operating or accident conditions in the plant. The licensee acknowledged that this methodology probably did not accurately represent dynamic system behavior, but that the different case studies presented in the analysis allowed them to develop a range for the maximum allowable block valve stroke time difference and the proper valve sequencing.

Actual test results demonstrated that stroke time differences of less than 1 second prevented the reactor building pressure from exceeding atmospheric pressure as indicated by the control room instrument. The reactor building differential pressure (d/p) instrument provides an alarm function to alert plant operators if d/p approaches zero. The licensee also stated that their search

of historical plant operating and test data did not reveal any recorded instance at Unit 1 where the d/p ever exceeded the alarm point or achieved a positive value. Based upon these considerations, the inspector found that the permanent modification was adequately evaluated and implemented.

7.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (71707, 92700)

7.1 Review of Licensee Event Reports (LERs) and Special Reports

NMPC submitted a Unit 2 special report, dated October 12, 1992, discussing a non-valid test and failure of the Division II EDG that occurred on September 15, 1992. In preparation for a 115 kV off-site power line outage, the Division II EDG was manually started. However, the EDG output voltage could not be manually changed by the operator and the EDG was subsequently shutdown to investigate the problem. The investigation determined that a relay failed which caused the voltage regulator to operate in the emergency mode, vice the test mode (i.e., maintain a constant voltage of 4160 vice being able to vary to voltage). After replacement of the relay, the EDG was satisfactorily started and the ability to manually vary the voltage output was verified. The inspector considered that NMPC's conclusion that this event was not a valid test or failure was proper.

7.2 Site Operations Review Committee Meetings

The inspectors attended several site operations review committee (SORC) meetings during the period. The meetings were well conducted and focused on the safe operation of both units. Procedure changes being presented were generally well understood by the members before discussion began. In one instance, a procedure change to the preventive maintenance program required involvement by the members in their management capacity, rather than their SORC capacity, and was tabled after long discussion, to allow further management review outside the SORC.

8.0 MANAGEMENT MEETINGS

At periodic intervals and at the conclusion of the inspection, meetings were held with senior station management to discuss the scope and findings of this inspection. Based on the NRC Region I review of this report and discussions held with NMPC representatives, it was determined that this report does not contain safeguards or proprietary information.