U.S. NUCLEAR REGULATORY COMMISSION

REGION I

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Licensee:	Niagara Mohawk Power Corporation P. O. Box 63 Lycoming, NY 13093
Facility:	Nine Mile Point, Units 1 and 2
Location:	Scriba, New York
Dates:	November 9, 1997 - January 3, 1998
Inspectors:	B. S. Norris, Senior Resident Inspector T. A. Beltz, Resident Inspector R. A. Skokowski, Resident Inspector

Approved by: Lawrence T. Doerflein

Lawrence T. Doerflein, Chief
Projects Branch 1
Division of Reactor Projects

2/23/98
Date

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EXECUTIVE SUMMARY

Nine Mile Point Units 1 and 2
50-220/97-12& 50-410/97-12
November 9, 1997 - January 3, 1998

This NRC inspection report includes reviews of licensee activities in the functional areas of operations, engineering, maintenance, and plant support. The report covers an eight-week period of inspections and reviews by the resident staff.

PLANT OPERATIONS

The shift brief for the newly-installed emergency cooling (EC) condenser keepfull modification was synergistic and provided sufficient detail on the system hardware and operation. The conduct of control room activities during the Nine Mile Point Unit 1 (Unit 1) plant startup following repairs to the EC condensers was good and improved compared to previous startups. The overall reactor startup appeared to run smoother than some previous startups due to the improvement in control rod drive performance.

Both units provided adequate protection for cold weather conditions through the respective maintenance and operating procedures.

Niagara Mohawk Power Corporation (NMPC) identified that the Nine Mile Point Unit 2 (Unit 2) condensate storage tank (CST) building temperatures were not being maintained in accordance with the Updated Final Safety Analysis Report (UFSAR), and took appropriate corrective action to change the temperature control switches to the proper setpoint. (NCV) Additionally, NMPC identified that the capacity of the Unit 2 CST building heaters needed upgrading to effectively maintain desired temperature; this was appropriately evaluated and adequate compensatory actions were established until heater upgrades could be accomplished.

Upon identification that the safety relief valve (SRV) position indication at the Unit 2 remote shutdown panel (RSP) was unreliable during a control room fire due to a portion of the cabling and components being contained with the control room fire-zone, NMPC engineering staff recommended the incorporation of a caution in the remote shutdown procedure regarding the potential unavailability of the indication. Since the loss of SRV position indication could have been confusing to the operators during a plant shutdown from the RSP, the inspectors considered the scheduled revision date to be excessive, and after discussion with Unit 2 Operations and Plant Managers the caution was promptly incorporated.

Following the inspectors' identification of the Unit 1 hydrogen/oxygen (H_2O_2) analyzer cabinet doors being improperly secured, the licensee completed a technically sound and extensive analysis to determine that operation in this condition did not adversely impact the equipment operability. However, past operations with the H_2O_2 analyzer cabinet doors improperly secured indicated a poor questioning attitude on part of the Unit 1 operators, in that they failed to recognize the potential safety concern associated with the condition.

Executive Summary (cont'd)

The Unit 1 shutdown safety verification procedure was considered a valuable aid for the control room operators to assist in monitoring plant conditions and assuring that safety functions were sufficiently available during shutdown conditions. Periodic briefings of safety function status during work control meetings and shift turnover was good, in that it ensured personnel awareness of system status and allowed for feedback of any current or potential deviations.

Unit 2 licensed control room operators were not aware that the posted surveillance test data for standby liquid control (SLC) was out of date and that the surveillance was potentially overdue. A chemistry technician failed to post the SLC summary sheet after completion of the surveillance, as required by procedure. (NCV)

The Unit 1 operations and reactor engineering staffs' initiative to perform a procedure review prior to an infrequently performed evolution, (reactor shutdown by full control rod insertion), was appropriate. This review was good in that it identified the need for some procedural enhancements, and that on several occasions the mode switch was placed in REFUEL contrary to the technical specification (TS) requirements. (NCV)

MAINTENANCE

A Unit 1 EC condenser hydrostatic test pre-evolution brief was adequate. Communications during the test were good, in that formal three-way communications were consistently used. Operations and inservice testing supervision provided good oversight and assistance, which resulted in a well-coordinated evolution. However, the failure to vacate the test area of all nonessential personnel constituted a potential personnel safety hazard and a weakness in the licensee's control of the evolution.

Pre-evolution briefs for the Unit 1 EC condenser capacity test were detailed and safety-focused. Operators demonstrated a questioning attitude and the briefs were synergistic. The control room environment was very good and clear and formal three-part communications were consistently used. Radiation protection (RP) and security personnel controlled the outside areas appropriately, and samples and surveys by RP personnel appeared well-concerted. Test results received a timely and adequate supervisory review.

Due to inattention during a surveillance test, a Unit 2 technician inadvertently inserted a circuit card extender upside down, causing a reactor protection system half-scram signal. In addition, the surveillance test procedure did not contain a precautionary note which could have warned the technician of the potential plant impact if the card were incorrectly inserted.

Rigging and transfer of a Unit 1 EC condenser tube bundle were methodical and well controlled, due in part to good communication and coordination among all involved organizations.

Licensee's actions were appropriate in response to an unexpected isolation of the Unit 1 vent and purge system that occurred during radiation monitor troubleshooting. The

licensee's root cause of the event was reasonable and the Station Operating Review Committee's review of the event maintained the proper safety focus.

ENGINEERING

The Unit 1 modification of the EC keepfull system was well designed. The modification was installed according to the drawings, and adequately tested.

At Unit 2, NMPC's identification of a breach between an equipment qualification (EQ) classified harsh environment area and a mild environment area, an original construction deficiency, was considered good. (NCV) Particularly noteworthy was the recognition that in the event of a high energy line break, the breach could result in the potential loss of several safety-related systems. Once identified, the licensee took appropriate actions to repair the breach and to verify no other similar openings.

A Unit 2 reactor operator demonstrated a good questioning attitude in identifying that a TS required surveillance test for the rod sequence control system was inadequate. (VIO)

Prior to September 1996, NMPC failed to monitor the Unit 2 relay room temperature, as required by TS. (NCV) Furthermore, when the licensee identified this issue in 1996, they incorrectly dispositioned it, resulting in a failure to recognize that the condition was reportable, and missed an opportunity to identify other subsequently identified concerns related to the UFSAR description of the control room envelope.

The 1997 engineering review of the Unit 1 Safe Shutdown Analysis and Fire Protection Engineering Evaluation documents was good, in that it disclosed previous engineering deficiencies, particularly that emergency lighting required to support alternate shutdown of the plant was missing. (VIO) Earlier reviews of these documents were weak in that they failed to identify these deficiencies.

The licensee's review of an industry concern regarding possible communication between the drywell and the wetwell was appropriate, and their evaluation of other possible evolutions which created a drywell-to-wetwell flow path to be good. Actions taken at both units to address identified discrepancies were adequate.

NMPC receipt inspection of an Unit 1 EC condenser tube bundle was thorough.

PLANT SUPPORT

An inspection of normally inaccessible areas of the Unit 2 reactor water cleanup (RWCU) system found the material condition of the equipment to be satisfactory, with the condition of the equipment in the RWCU valve aisle to be particularly good. Housekeeping in the areas inspected was acceptable, and appropriate radiological controls were established.

An inadvertent automatic isolation of the Unit 1 drywell vent and purge lines, occurred due to personnel inattention-to-detail, particularly a failure to follow procedure. (VIO)

REPORT DETAILS

Nine Mile Point Units 1 and 2
50-220/97-12 & 50-410/97-12
November 9, 1997 - January 3, 1998

SUMMARY OF ACTIVITIES

Niagara Mohawk Power Corporation (NMPC) Activities

Unit 1

Nine Mile Point Unit 1 (Unit 1) started the inspection period in cold shutdown for repairs to the emergency cooling (EC) condensers. On December 7, 1997, the unit was started up, but was shutdown the next day due to high differential pressure on the core spray sprager. On December 9, the unit was restarted and achieved full power on December 12. The unit operated at full power throughout the remainder of the inspection period.

Unit 2

Nine Mile Point Unit 2 (Unit 2) started the inspection period in cold shutdown for repairs to a malfunctioning reactor recirculation flow control valve. On November 10, the unit was restarted and on November 14, the unit achieved 95% power. The unit was limited to 95% due to the moisture separator reheaters being isolated. This power level was essentially maintained throughout the remainder of the inspection period.

Nuclear Regulatory Commission (NRC) Staff Activities

Inspection Activities

The NRC resident inspectors conducted inspection activities during normal, backshift, and deep backshift hours. The results of the inspection activities are contained in the applicable sections of this report.

Updated Final Safety Analysis Report Reviews

While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the Updated Final Safety Analysis Report (UFSAR) related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters, with the exception of the following: the Unit 2 condensate storage tank (CST) building temperature switch setpoints (Section O2.2 of this report); and the Unit 2 definition of the control room envelope with respect to monitoring temperatures in the relay room (Section E8.3 of this report). In addition, the Unit 2 UFSAR describes that the equipment qualification (EQ) zones are controlled by other documents, a breach between the stairwell and the north auxiliary bay meant that the plant was not in accordance with those documents (Section E8.1 of this report).

I. OPERATIONS

O1 Conduct of Operations (71707)[1]

O1.1 General Comments

Using NRC Inspection Procedure 71707, the resident inspectors conducted frequent reviews of ongoing plant operations to verify that the units were operated safely and in accordance with licensee procedures and regulatory requirements. The reviews included tours of both accessible and normally inaccessible areas of both units, verification of engineered safeguards features (ESF) system operability, verification of adequate control room and shift staffing, verification that the units were operated in conformance with technical specifications, and verification that logs and records accurately identified equipment status or deficiencies. In general, the conduct of operations was professional and safety-conscious; specific events and noteworthy observations are detailed in the sections below.

O1.2 Unit 1 Plant Startup (71715)

a. Inspection Scope

At Unit 1, the inspectors observed operations supervision conduct shift briefs, monitored a reactor startup during both normal and backshifts, and evaluated operator adherence to applicable procedures.

b. Observations and Findings

Prior to Unit 1 reactor startup, the inspectors observed a shift brief conducted by Unit 1 Operations Department supervision. The brief adequately provided personnel with sufficient information regarding current plant conditions, evolutions completed or currently in progress, and evolutions anticipated during the upcoming shift, and allowed for discussion and questions. A brief was also held to discuss the newly-installed EC condenser keepfull modification system configuration and operation. The inspectors considered the modification brief to be good, in that the discussion was very synergistic and sufficient detail was provided on the system hardware and operation.

On December 7, 1997, Unit 1 startup commenced at 4:54 p.m. At 10:45 p.m., an auxiliary operator identified that the Loop 12 core spray (CS) system sparger break detection differential pressure (d/p) gage was indicating greater than expected. The gage indicated +0.5 pounds per square inch differential (psid) pressure, instead of the expected -0.2 psid. Reactor temperature at the time of discovery was approximately 200 degrees Fahrenheit (°F), and control room staff secured the

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[1] Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics. The NRC inspection manual procedure or temporary instruction (TI) that was used as inspection guidance is listed for each applicable report section.

heatup. A shutdown commenced at 2:40 a.m. on December 8, and was completed at 4:16 a.m. Based upon d/p gage indication and plant conditions at the time, the inspectors considered the licensee's decision to shutdown to be appropriate.

The licensee suspected that air entrapped in the CS system created the unanticipated indication. A system flush was performed and gage indications returned to expected values. An engineering evaluation was completed prior to reactor startup. The evaluation concluded that gage indications could fluctuate during plant startup, and that these fluctuations were acceptable provided that the differential pressure remained below the alarm setpoint of +1.5 psid. The unit was restarted on December 9 at 8:43 a.m.

The licensee continuously monitored the CS system sparger break detection d/p gages during the startup, and fluctuations were noted on both. The plant startup continued, with the highest noted gage reading approximately +1.2 psid, which was below the alarm setpoint. Once a steady-state power level of approximately 20% was reached, both gages indicated the normal expected values. The licensee informed the inspectors that these gage fluctuations may have occurred during previous reactor startups. Since these instruments provide local indication only and are recorded only during operator rounds, operator rounds and the noted fluctuations may never have coincided.

The inspectors noted that control rod manipulation during the startup was performed with less difficulty than during previous startups. Specifically, control rod double-notching and rod "sticking" (requiring operators to increase drive water pressure) were less prevalent. The inspectors discussed this issue with a reactor engineering supervisor and the system engineer. The improvement was attributed, in part, to driving all rods in during the recent reactor shutdown, vice "soft-scramming," in which the reactor is manually scrammed from approximately 20 percent power. When scrammed, the control rods seals are subjected to greater differential pressures which tends to increase wear. The inspectors considered that the overall reactor startup appeared to run smoother than some previous startups due to the improvement in control rod drive performance.

The licensee performed startup activities in accordance with the following Unit 1 procedures:

- N1-OP-43A, "Reactivity Control," Revision 04;
- N1-OP-43B, "Balance of Plant Startup and Shutdown," Revision 00;
- N1-OP-3, "Reactor Cleanup System," Revision 23;
- N1-OP-5, "Control Rod Drive System," Revision 31;
- N1-OP-16, "Feedwater System Booster Pump to Reactor," Revision 24; and
- N1-OP-31, "Tandem Compound Reheat Turbine," Revision 16.

The inspectors monitored licensee performance and noted procedural compliance during the startup. Additionally, conduct of control room activities continued to show improvement, in that (1) peer verification during switch manipulations within the control room was consistently utilized, (2) communications were good, and

(3) operators properly used alarm response and operating procedures.

c. Conclusions

The shift brief for the newly-installed emergency cooling (EC) condenser keepfull modification was synergistic and provided sufficient detail on the system hardware and operation. The conduct of control room activities during the Nine Mile Point Unit 1 plant startup following repairs to the EC condensers was good and improved compared to previous startups. The overall reactor startup appeared to run smoother than some previous startups due to the improvement in control rod drive performance.

O2 Operational Status of Facilities and Equipment (71707)

O2.1 Cold Weather Preparations (71714)

a. Inspection Scope

The inspectors reviewed NMPC's program for protection of safety-significant systems and equipment against extreme cold weather. The inspectors held discussions with operations, work control, and engineering staff.

b. Observations and Findings

At Unit 1, plant systems possibly affected by cold weather included circulating water and building ventilation. The operating procedures for both systems contained guidance for the abnormal condition of low ambient temperatures. Specifically, the circulating water system operating procedure, N1-OP-19, Rev. 21, Section H, discussed the indications of, and operator actions for, the formation of frazil ice at the intake structure. When lake temperature remains low, operators realign the circulating water intake gates for de-icing. With respect to reactor and turbine building ventilation systems, heaters would automatically energize and deenergize based upon system temperature. These heaters are located within the ventilation ducting and would not prevent snow and ice buildup on the intake louvers; however, control room annunciators would alarm for high system differential pressure if blockage occurred.

The inspectors verified the completion of Unit 2 maintenance activities for the cold weather conditions. The activities were in accordance with procedures N2-PM-A001, "Annual Draining and Refilling of ACUs [air conditioning units] and Cooling Coils," Revision 00, and N2-PM-A004, "Annual Removal and Installment of HVR [heating, ventilation, and refrigeration] Supply Prefilters," Revision 00. The operability of the service water heater system is verified during checks each shift (N2-OSP-LOG-S001) and periodically during the performance of Technical Specification (TS) 4.7.1.1.2 required surveillance. During the inspectors' review of N2 -OSP-LOG-S001, the inspectors noted that a recent change provided for verifying CST building temperatures during cold weather conditions (outside temperatures less than 15°F); a review of the issues that prompted this change is

provided in Section O2.2 of this report. The inspectors considered the cold weather preparations at Unit 2, and the controls in place to ensure the annual completion of these preparations, to be adequate.

c. Conclusions

Both units provided adequate protection for cold weather conditions through the respective maintenance and operating procedures.

O2.2 Unit 2 Condensate Storage Tank Building Minimum Temperatures not Controlled in Accordance with the UFSAR

a. Inspection Scope

In January 1997, NMPC identified that the CST building was not being maintained above the minimum temperature of 65°F as described in UFSAR Section 5.4.6.1.5. The inspectors reviewed the associated Deviation/Event Report (DER), and engineering supporting analysis, the calibration history of the applicable temperature instruments, and the relevant UFSAR sections and procedures. Additionally the inspectors discussed the issues with members of the licensee's operations, engineering, and licensing departments.

b. Observations and Findings

On January 13, 1997, NMPC identified that the Unit 2 CST Building was not being maintained above the minimum temperature of 65°F as described in UFSAR Section 5.4.6.1.5. On that day the ambient temperatures within the CST building ranged from 57°F to 66°F, but the temperature of the CST and reactor core isolation cooling (RCIC) system piping, as taken with a thermocouple thermometer, were above 65°F. Additionally, the NMPC determined that the temperature switches that controlled the CST building heaters had not been calibrated since February 1985. Furthermore, in May 1985 the setpoint to energize the CST building heaters, as documented on the applicable setpoint data-sheets, had been increased from 65°F to 70°F. NMPC documented these concerns in DER 2-97-0091. Upon receipt of the DER the Station Shift Supervisor (SSS) evaluated the operability of the CST, RCIC and high pressure core spray (CSH) system and determined, based on a review of the applicable UFSAR sections, that the systems were operable; however the SSS requested that the engineering department provided an engineering supporting analysis to verify the operability decision.

By January 15, 1997, the applicable heater control switches were recalibrated with a setpoint of 70°F. The Engineering Supporting Analysis (ESA) was completed the next day, and confirmed the operability of the affected systems. However, during the review, the Unit 2 engineering department identified that the actual minimum temperature for the CST building was 70°F as described in UFSAR Section 9.4.7. The 65°F minimum temperature described in UFSAR Section 5.4.6.1.5 was based on Preliminary Safety Analysis Report information, and this section of the UFSAR, and the actual heater control switch setpoints were not changed to reflect the

on Preliminary Safety Analysis Report information, and this section of the UFSAR, and the actual heater control switch setpoints were not changed to reflect the 70°F minimum.

After further evaluation by NMPC, the heater control switch setpoint was increased to 75°F to more effectively maintain CST building temperature, and the calibration of the switches every refueling outage was added to the licensee's preventive maintenance/surveillance testing scheduling data base. Also, during their evaluation, NMPC identified that the capacity of the heaters installed in the CST building should be increased to effectively maintain temperature, and as part of the corrective actions described in DER 2-97-0091, the heat capacity calculation was to be revised and the heaters were to be upgraded by March 31, 1998. As compensatory actions until the heater capacity is increased, Operations Department Log Procedures have been revised to monitor CST building temperatures, with detailed actions to take if outside temperatures should drop below 15°F.

The inspectors reviewed the applicable DER and discussed the issues with members of the NMPC design engineering staff, including the Engineering Manager, and considered their evaluation and the corrective actions taken, including the compensatory actions taken until the completion of the CST building heater upgrades, to be appropriate. However, the failure to change the actual setpoint of the CST building temperature switches to match the value specified within the UFSAR is a violation of 10 CFR 50 Appendix B, Criterion III, Design Control. This non-repetitive, licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. (NCV 50-220/97-12-01)

c. Conclusions

NMPC identified that the Unit 2 CST building temperatures were not being maintained in accordance with the UFSAR, and took appropriate corrective action to change the temperature control switches to the proper setpoint. (NCV) Additionally, NMPC identified that the capacity of the Unit 2 CST building heaters needed upgrading to effectively maintain desired temperature; this was appropriately evaluated and adequate compensatory actions were established until heater upgrades could be accomplished.

O2.3 Potential Unavailability of the Safety Relief Valve Position Indication at the Remote Shutdown Panel during a Control Room Fire

a. Inspection Scope

The inspectors reviewed the licensee's ESA associated the potential unavailability of the safety relief valve (SRV) position indication provided at the remote shutdown panel (RSP) during a control room fire.

b. Observations and Findings

On November 7, 1997, while Unit 2 was in a forced outage to repair a recirculation flow control valve, NMPC identified that a portion of the cabling and components that provide the SRV position indication at the RSP was contained inside the control/relay room fire zone, and therefore, the indication may not be available during a control room fire. This issue was documented in DER 2-97-3101 and the licensee considered the resolution of the issue to be a restart restraint. On November 8, NMPC considered the restart restraint resolved based on an ESA, and the plant was started up on November 10.

The ESA concluded that the SRV position indication at the RSP was not required in the event of a control room fire since the operation of these valves can be monitored with available process monitoring instrumentation, such as reactor vessel pressure and level, suppression pool temperature and level, and automatic depressurization system (ADS) accumulator pressure. Therefore, safe shutdown of the plant could be achieved and maintained without the SRV position indication at the RSP. However, the ESA also included a recommendation that the procedures related to the RSP should be revised to incorporate a caution to alert the operators that during a control room fire the SRV position indication may not be reliable, and that the process monitoring instrumentation should be used to verify SRV operation.

Subsequent to the plant restart, the inspectors reviewed the DER and associated ESA regarding the potential unavailability of the SRV position indication provided at the RSP during a control room fire. The inspectors considered the basis for not requiring the SRV position indication at the RSP during a control room fire to be consistent with the Unit 2 UFSAR and with Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50) Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." In addition, the inspectors considered the incorporation of the described caution into the procedures associated with the RSP to be appropriate. However, the inspector identified during the review of the DER disposition, that the caution recommended by the ESA was not incorporated prior to the plant restart, and that it was not scheduled to be incorporated until April 15, 1998.

The inspectors reviewed the applicable operating procedure, N2-OP-78, "Remote Shutdown System," Revision 10, and discussed the availability of the SRV position indication with three SSS-qualified individuals. The inspectors ascertained that the procedure did not explicitly describe the use of the SRV position indications at the RSP and the SSSs were aware of the process instrumentation available at the RSP to provide indication of SRV operation. However, the SSSs also indicated that a loss of valve position would add confusion to a plant shutdown performed from the RSP. Based on this review, the inspectors considered the six-month turn-around-time for the incorporation of the recommended caution to be untimely, and expressed this concern to the Unit 2 Operations and Plant Managers, who agreed with the inspectors concern. Subsequently, on December 18, 1997, a procedure change to N2-OP-78 was completed to incorporate the caution, and an operations

department night note was provided to inform the operators of the potential unavailability of the SRV position indication at the RSP during a control room fire.

c. Conclusion

Upon identification that the safety relief valve (SRV) position indication at the Unit 2 remote shutdown panel (RSP) was unreliable during a control room fire due to a portion of the cabling and components being contained with the control room fire-zone, NMPC engineering staff recommended the incorporation of a caution in the remote shutdown procedure regarding the potential unavailability of the indication. Since the loss of SRV position indication could have been confusing to the operators during a plant shutdown from the RSP, the inspectors considered the scheduled revision date to be excessive, and after discussion with Unit 2 Operations and Plant Managers the caution was promptly incorporated.

O2.4 Unit 1 Hydrogen/Oxygen Analyzer Cabinet Doors Improperly Secured

a. Inspection Scope

The inspectors reviewed the licensee's DER and associated ESA regarding operation with the Unit 1 hydrogen/oxygen (H_2O_2) analyzer cabinet doors improperly secured.

b. Observations and Findings

During a walkdown of Unit 1, the inspectors noted that the cabinet doors for both the #11 and #12 H_2O_2 analyzer cabinets were not closed, and that all but one of the cabinets door fasteners were missing. Additionally, the inspectors identified that the #12 H_2O_2 analyzer control cabinet was not closed. Although the unit was shutdown at the time this concern was identified, indications were that the doors had been open for an extended period preceding the plant shutdown. The inspectors were concerned that operation with the doors improperly secured could adversely impact the equipment. The licensee was unable to provide an immediate response to the inspectors' concern, and therefore, they documented the deviation in DER 1-97-2804.

During the licensee's evaluation of the concern, they identified the following:

- it was not readily possible to reinstall all the hardware needed to secure the doors due to missing parts, damaged connections and door alignment problems;
- tests used during the seismic qualification of the equipment was performed with all the mounting hardware in place and the doors secured; and
- procedure requires operations department personnel to access the cabinets during shift rounds; and operators were not aware of the potential operability concerns with the cabinet doors not fully secured.

The licensee performed an ESA regarding operation with the cabinet doors open. The analysis reviewed the concern from both a seismic and equipment qualification (EQ) perspective, and concluded that the cabinets were operable even with the

doors not secured. However, the licensee considered operation with the doors improperly secured was a nonconformance and repaired the fasteners and provided cautions in the applicable procedures regarding the need to secure the cabinet doors after entry.

The inspectors reviewed the ESA and considered it extensive and technically sound. Also, the inspectors considered the licensee's actions taken to address the nonconformance to be appropriate. However, the past operations with the analyzer cabinet doors improperly secured indicated a poor questioning attitude on part of the Unit 1 operators, in that the operators failed to recognize the potential safety concern associated with operating with the doors improperly secured.

c. Conclusion

Following the inspectors' identification of the Unit 1 hydrogen/oxygen (H_2O_2) analyzer cabinet doors being improperly secured, the licensee completed a technically sound and extensive analysis to determine that operation in this condition did not adversely impact the equipment operability. However, past operations with the H_2O_2 analyzer cabinet doors improperly secured indicated a poor questioning attitude on part of the Unit 1 operators, in that they failed to recognize the potential safety concern associated with the condition.

O3 Operations Procedures and Documentation (71707)

O3.1 Unit 1 Shutdown Safety Verification

a. Inspection Scope

Unit 1 was shutdown during the early portion of the inspection period to repair tube leaks identified in the EC condensers. The shutdown condition is an infrequent mode of plant operation; therefore, a specific licensee procedure is used for shutdown safety verification. The inspectors reviewed the Unit 1 procedure for adequacy and clarity, and evaluated the implementation of the procedure by the operations staff.

b. Observations and Findings

On September 15, 1997, Unit 1 was shutdown to repair EC condenser tube leaks. Unit 1 Procedure N1-ODG-11, "Shutdown Operations Protection Guidelines," Revision 9, was used by operators to assist in monitoring safety system status and assure that a minimum number of safety systems were available during changes in plant configuration. The procedure was designed to assist in monitoring the availability of the following safety functions:

- Decay heat removal (reactor and spent fuel pool)
- Inventory control
- Power availability
- Secondary containment control
- Reactivity control

The inspectors compared Procedure N1-ODG-11 to the NMPC higher tier Nuclear Interface Procedure (NIP) NIP-OUT-01, "Shutdown Safety," Revision 2. The inspectors verified that the systems and requirements of the NIP were translated into N1-ODG-11, including delineation of responsibilities for the Operations Department. The inspectors noted that operations staff were responsible for maintaining the unit in a safe condition by monitoring plant status and ensuring that equipment was maintained using a "defense-in-depth" configuration. The NIP charged the outage department with scheduling work activities such that redundant equipment within each safety function was not scheduled for maintenance concurrently.

The inspectors reviewed the management of the shutdown safety system status by the operations staff. Safety system availability was adequately monitored and current. The inspectors also noted that the current safety system status was provided to plant staff during work control meetings and control room shift turnover. The inspectors considered this periodic briefing of safety function status to be good, in that it established awareness of system status and allowed feedback from plant personnel of any current or potential deviations.

c. Conclusions

The Unit 1 shutdown safety verification procedure was considered a valuable aid for the control room operators to assist in monitoring plant conditions and assuring that safety functions were sufficiently available during shutdown conditions. Periodic briefings of safety function status during work control meetings and shift turnover was good, in that it ensured personnel awareness of system status and allowed for feedback of any current or potential deviations.

O4 Operator Knowledge and Performance (71707)

O4.1 Unit 2 Standby Liquid Control System Surveillance Test

a. Inspection Scope

While observing a Unit 2 reactor startup, an NRC inspector noted that the standby liquid control (SLC) surveillance summary sheet on the control room panels was out of date. The inspectors discussed the concern with the Station Shift Supervisor (SSS) relative to a mode change without all TS surveillance prerequisites being satisfied.

b. Observations and Findings

On November 12, 1997, while monitoring a Unit 2 reactor startup, an NRC inspector observed that the date on the SLC Summary sheet posted in the control room indicated that the associated surveillance test requirements were overdue. Unit 2 TS, Section 3.1.5, requires the SLC tank to be sampled every 31 days, the SLC summary sheet indicated that the surveillance test was last performed on October 7, 1997. Further review by the SSS revealed that the surveillance had been performed on November 3, and that the results were acceptable.

NMPC chemistry surveillance Procedure N2-CSP-SLS-M110, "Standby Liquid Control Monthly Surveillance," Revision 01, Step 8.4.9, requires the chemistry technician to post the SLC Summary sheet in the control room upon completion of the test. The failure to post the SLC Summary sheet in the control room is a violation of minor significance and is being treated as a Non-Cited Violation (NCV), consistent with Section IV of the NRC Enforcement Policy. **(NCV 50-410/97-12-02)**

Discussions with several SSSs and chief station operators (CSOs) confirmed that they had failed to recognize that the sheet was out-of-date, and that they should have noted the discrepancy. The Unit 2 Plant Manager stated that he expected the operators to be aware of all information on the control panels. The inspectors considered it a weakness on the part of the on-shift ROs and SROs in that, during control room panel walkdowns, they did not notice that the surveillance requirement appeared to be overdue.

c. Conclusions

Unit 2 licensed control room operators were not aware that the posted surveillance test data for standby liquid control (SLC) was out of date and that the surveillance was potentially overdue. A chemistry technician failed to post the SLC summary sheet after completion of the surveillance, as required by procedure. (NCV)

O7 Quality Assurance in Operations (71707)

O7.1 Review of INPO Evaluation

The inspectors reviewed the report from the Institute of Nuclear Power Operations (INPO) for the evaluation conducted from September 8 through 19, 1997. The evaluation examined the overall operation of the Nine Mile Point site, and was performed by peer evaluators from other nuclear facilities. The report identified no issues that the NRC was not already aware of, and no additional followup by the NRC was warranted.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) LER 50-220/97-11: Previous Plant Shutdown in Violation of Technical Specifications

a. Inspection Scope

The inspectors reviewed the details associated with licensee event report (LER) 50-220/97-11. The issues related to the event were discussed with the Unit 1 reactor engineering supervisor. The inspectors reviewed the documentation associated with the event. In addition, the inspectors reviewed the LER to verify completion in accordance with 10 CFR 50.73.

b. Observations and Findings

Prior to September 1997, most Unit 1 reactor shutdowns had been conducted by performing a "soft scram," in which the reactor was manually scrammed from approximately 20% reactor power. On September 12, 1997, operations and reactor engineering staff performed a review of operating procedures in preparation for a planned normal reactor shutdown by full control rod insertion. The inspectors considered the staffs' initiative to perform the procedure review was appropriate, particularly since the completion of a reactor shutdown by full control rod insertion had been an infrequently conducted evolution.

The inspectors considered the staff's review to be good, in that the need for some procedural enhancements were identified. Specifically, the staff identified that Unit 1 Procedures N1-OP-43, "Startup, Shutdown and Normal Operation," and N1-OP-43A, "Reactivity Control," did not provide specific guidance for when to place the reactor mode switch in the REFUEL position. The staff subsequently identified that, on several occasions, the mode switch had been placed in REFUEL in violation of TS Definition 1.1c, "Refuel."

The issuance of Unit 1 TS Amendment 99 on June 9, 1988, changed TS 1.1c and allowed the mode switch to be in REFUEL at temperatures less than 212°F, or for (1) vessel hydrostatic testing, (2) scram time testing, or (3) scram recovery operations. The licensee identified that on September 3, 1992, during a reactor shutdown using full control rod insertion, the mode switch was placed in REFUEL with reactor coolant temperature at 377°F. In this instance, none of the Unit 1 TS 1.1c conditions were met.

Prior to issuance of TS Amendment 99, TS 1.1c prohibited the mode switch to be in placed in REFUEL unless temperature was less than 212°F. Further licensee investigation revealed that prior to June 1988 the mode switch had been placed in REFUEL on several occasions with temperature greater than 212°F.

The inspectors discussed the issue with the reactor engineering supervisor who stated that placing the mode switch in REFUEL following a scram enabled the refuel one rod permit light, which is used for ALL RODS IN verification. The licensee concluded that no safety consequences occur by placing the mode switch in REFUEL above 212°F during a normal shutdown by control rod insertion. Notwithstanding, placing the mode switch in REFUEL on September 3, 1992, and on other previous occasions while above 212°F was a violation of Unit 1 TSs. This non-repetitive, licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. (NCV 50-220/97-12-03)

The inspectors verified that the LER was completed in accordance with the requirements of 10 CFR 50.73. Specifically, the description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. The root cause and corrective actions as described in the LER were reasonable. This LER is closed.

c. <u>Conclusions</u>

The Unit 1 operations and reactor engineering staffs' initiative to perform a procedure review prior to an infrequently performed evolution, (reactor shutdown by full control rod insertion), was appropriate. This review was good in that it identified the need for some procedural enhancements, and that on several occasions the mode switch was placed in REFUEL contrary to the technical specification (TS) requirements. (NCV)

O8.2 <u>(Closed) URI 50-220/96-01-04& 50-410/96-01-04: Change from an 8-Hour Shift to a 12-Hour Shift for Safety-Related Functions</u>

In early 1995, both units changed from a normal 8-hour shift to a 12-hour shift for facility staff who performed safety-related functions; e.g., licensed operators, auxiliary operators, health physicists, and key maintenance personnel. The change was implemented on a trial basis, evaluated after a year, and presented to the union for consideration/acceptance as a permanent change.

The TSs for both units, Section 6.2.2, states:

"Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the unit is operating ... Any deviation from the above guidelines shall be authorized by the Plant manager ... in accordance with the established procedures and with documentation of the basis for granting the deviation ... Routine deviation from the above guidelines is not authorized."

The inspectors' original concerns related to this issue were: (1) there was no documentation authorizing the exception to the above guideline, (2) a year-long trial period seemed excessive when considering the above statement that routine deviation was not authorized, and (3) whether the change was even allowed

without prior NRC approval. NMPC's position was that this section of the TS related only to the limitations associated with overtime, and that the 8-hour day/ 40-hour week was guidance only.

The inspectors discussed their concerns with NRC personnel in the Office of Nuclear Reactor Regulation and concluded that NMPC's position was acceptable. Specifically, the requirements were associated with ensuring that overtime was not excessive. Subsequently, in July 1996, NMPC submitted to the NRC requests to have the TS for each unit amended to reflect the change in work hours. On December 12, 1996, the NRC issued amendments to the applicable TSs, changing the staff work schedules allowing shifts as long as twelve hours. The inspectors had no further questions; this item is closed.

II. MAINTENANCE [2]

M1 Conduct of Maintenance (61726, 62707)

M1.1 General Comments

Using NRC Inspection Procedures 61726 and 62707, the resident inspectors periodically observed plant maintenance activities and the performance of various surveillance tests. As part of the observations, the inspectors evaluated the activities with respect to the requirements of the Maintenance Rule, as detailed in Title 10 of the Code of Federal Regulations, Part 50.65 (10 CFR 50.65). In general, maintenance and surveillance activities were conducted professionally, with the work orders (WOs) and necessary procedures in use at the work site, and with the appropriate focus on safety. Specific activities and noteworthy observations are detailed in the inspection report. The inspectors reviewed procedures and observed all or portions of the following maintenance/surveillance activities:

- N1-MFT-051B — Emergency Condenser Keep-Full System Modification Testing at Power
- N2-ISP-RCS-Q101 — Quarterly Channel Calibration of the APRM [average power range monitor] Recirculation Flow Unit
- N2-OSP-RMC-@004 — Rod Sequence Control System Operability Test
- N2-CSP-SLC-M110 — Standby Liquid Control Monthly Surveillance
- WO 97-05667-01 — Back Flush of the Core Spray Sparger Instrument Line
- WO 97-14457-00 — SLC Tank Temperature Indication and Control Switch Calibration
- N1-IST-HYD-007 — Emergency Condenser ASME [American Society of Mechanical Engineers] Hydrostatic Test
- N1-TSP-ECS-001 — Emergency Cooling System - Heat Removal Capability Test

[2] Surveillance activities are included under "Maintenance." For example, a section involving surveillance observations might be included as a separate sub-topic under M1, "Conduct of Maintenance."

M1.2 Unit 1 Emergency Cooling Condenser Hydrostatic Test

a. Inspection Scope

The inspectors observed Unit 1 staff perform an EC condenser hydrostatic test.

b. Observations and Findings

On December 1, 1997, Unit 1 staff performed a hydrostatic test on EC condenser Loop 11. The inspectors observed the test from the test pump (located on reactor building 281-foot elevation) and at the EC condensers (located on reactor building 340-foot elevation). The test was conducted in accordance with Unit 1 Procedure N1-IST-HYD-007, "Emergency Condenser ASME Hydrostatic Test," Revision 00.

Prior to performance of the test, the inspectors walked down the hydrostatic test pump and verified equipment setup. The inspectors noted that the test instrumentation (i.e. pressure gauge and relief valve) were within calibration periodicity and of proper range. The inspectors attended the pre-evolution brief and considered it adequate.

During the test, the inspectors noted that formal three-way communications were consistently used. Operations and inservice testing supervision were present at the hydrostatic test pump to monitor the evolution. Supervision provided good oversight and assistance during the ascent to, and while maintaining, test pressure. The inspectors considered that this oversight resulted in a well-coordinated evolution.

Prior to raising system pressure, control room staff made an announcement to vacate the test area, which included the area adjacent to the EC condensers. When test pressure was established, the inspectors observed the visual inspection activities, and identified no concerns. However, the inspectors did note that numerous maintenance personnel were still working adjacent to the EC condensers which were under test. The inspectors considered the failure to vacate the test area of all non-essential personnel constituted a potential personnel safety hazard and a weakness in the licensee's control of the evolution.

c. Conclusions

A Unit 1 EC condenser hydrostatic test pre-evolution brief was adequate. Communications during the test were good, in that formal three-way communications were consistently used. Operations and inservice testing supervision provided good oversight and assistance, which resulted in a well-coordinated evolution. However, the failure to vacate the test area of all nonessential personnel constituted a potential personnel safety hazard and a weakness in the licensee's control of the evolution.

M1.3 Unit 1 Emergency Cooling System Heat Removal Capability Test

a. Inspection Scope

The inspectors observed Unit 1 staff perform an EC system heat removal capability test.

b. Observations and Findings

On December 10, 1997, Unit 1 staff performed a heat removal capability surveillance test for both trains of the EC system to establish baseline thermal performance data on newly installed tube bundles. The test was performed in accordance with Procedure N1-TSP-ECS-001, "Emergency Cooling System - Heat Removal Capability Test."

The surveillance test was considered a special evolution as defined in licensee Procedure GAP-SAT-03, "Control of Special Evolutions." As such, the Unit 1 Technical Support Manager and the Unit 1 Reactor Engineering Supervisor were the Senior Manager and the Principal Test Engineer (PTE), respectively, for the evolution. GAP-SAT-03 states that the Senior Manager maintains overall responsibility and accountability for the conduct of the evolution, and that the PTE is responsible for supervising and conducting the evolution under the general supervision of the Senior Manager. The Senior Manager and the PTE conducted a management expectations brief and pre-evolution brief with the crew in accordance with GAP-SAT-03. Management's expectations for the conduct of operations were detailed and safety-focused. The pre-evolution brief emphasized any necessary precautions, defined personnel roles and responsibilities, and discussed the test abort criteria. The inspectors considered the brief to be sufficiently detailed and synergistic, with staff demonstrating a questioning attitude.

The inspectors observed the surveillance test from both the control room and outside the reactor building within the protected area. The control room environment was very good, in that only those personnel required for the test or for plant operations were present. Clear and formal three-part communication were consistently used. In the control room, data gathering was well-controlled by the PTE. Outside the reactor building, radiation protection (RP) and security personnel appropriately controlled personnel access to the area below the EC condenser vent lines. Post-discharge samples and surveys by RP personnel appeared well-concerted.

The inspectors reviewed the final surveillance test procedure and data results. The procedure received a timely and adequate supervisory review. The data results were complete and met established acceptance criteria.

c. Conclusions

Pre-evolution briefs for the Unit 1 EC condenser capacity test were detailed and safety-focused. Operators demonstrated a questioning attitude and the briefs were synergistic. The control room environment was very good and clear and formal three-part communications were consistently used. Radiation protection (RP) and security personnel controlled the outside areas appropriately, and samples and surveys by RP personnel appeared well-concerted. Test results received a timely and adequate supervisory review.

M3 Maintenance Procedures and Documentation (61726)

M3.1 Unexpected Half-Scram at Unit 2 During Surveillance Testing

a. Inspection Scope

During performance of a surveillance test at Unit 2 by instrument and control (I&C) technicians, a reactor protection system half-scram signal was unexpectedly received. The inspectors independently reviewed the surveillance procedure and associated DER, and discussed the event with the Unit 2 I&C General Supervisor and other involved personnel.

b. Observations and Findings

On December 18, 1997, during surveillance testing, a Unit 2 I&C technician inadvertently installed a test card extender incorrectly (upside-down), causing a half-scram signal. No other scram signals were present at the time and no control rod motion was experienced or expected. After discussion with the station shift supervisor (SSS), the technicians removed the test card extender and installed it properly, and continued testing without further incident. In addition, deviation/event report (DER) 2-97-3433 was initiated to track the root cause analysis and associated corrective action.

The surveillance test procedure, N2-ISP-RCS-Q101, "Quarterly Channel Calibration of the APRM [average power range monitor] Recirculation Flow Unit," Attachment 1, required the removal of circuit card "Z7" and attachment of an extender card. Step 7.2.5 then directs the technician to install the circuit card and extender in slot "Z7." The extender card was attached to the circuit card upside-down; thus, when the card was installed in the panel, a half-scram signal was generated.

The inspectors reviewed the surveillance test procedure and examined the circuit card and extender card. The circuit card has a meter and switch on the front of the card which makes the proper orientation easy to identify, and the circuit card was properly oriented during the surveillance test. However, the extender card can attach to the circuit card either right-side-up or upside-down; and there were no obvious markings to determine which way was correct. Discussions with another

I&C technician, who had performed this surveillance test previously, identified that there are small letters on one corner of the extender card. If the extender card is properly oriented, the letters will be in the upper corner. NMPC review of this event identified that the I&C technician was aware of the proper orientation of the extender card; but the technician's attention was improperly focused on the orientation of the attached cable vice the orientation of the extender card itself. In addition, the surveillance procedure did not contain a precautionary warning that there was a proper orientation for the extender card. The failure to include this information resulted in a weak procedure, and which allowed a challenge to the reactor protection system.

c. Conclusions

Due to inattention during a surveillance test, a Unit 2 I&C technician inadvertently inserted a circuit card extender upside down, causing a reactor protection system half-scram signal. In addition, the surveillance test procedure did not contain a precautionary note which could have warned the technician of the potential plant impact if the card were incorrectly inserted.

M4 Maintenance Staff Knowledge and Performance (61726)

M4.1 Transport of a Unit 1 Emergency Cooling Condenser Tube Bundle to the Refuel Floor

a. Inspection Scope

The inspectors observed the transfer of an EC condenser tube bundle from the Unit 1 reactor building track bay to the refuel floor. The inspectors assessed the movement with respect to applicable procedures and evaluated the coordination between the various licensee organizations.

b. Observations and Findings

On November 13, 1997, following receipt inspection (discussed in Section E8.8), the tube bundle for EC condenser 121 was transferred from the Unit 1 reactor building track bay to the refuel floor. The tube bundle was lifted from the delivery truck to the refuel floor (approximately 80 feet) using an overhead crane. The approximate weight of the bundle was 6100 pounds, and the crane was rated for 125 tons. The bundle was secured using wire slings, shackles and chain-falls. The rigging evolution appeared methodical and well-controlled. The movement of the bundle was performed under WO 97-04657-32, "Unload, Rig/Lift, Install New Tube Bundle in Em. Cond.," and controlled, in part, by Procedure N1-MMP-GEN-914, "Lifting of Miscellaneous Heavy Loads," Revision 02. The inspectors reviewed the applicable sections of the procedure and determined that licensee performance of the evolution was adequate.

Operations, Radiation Protection, Security, and Maintenance departments were some of the organizations directly involved in the tube bundle transfer. The inspectors observed that the transfer of the tube bundle was methodical and well-controlled, in part due to good communication and coordination between all of the organizations.

c. Conclusions

The rigging and transfer of a Unit 1 EC condenser tube bundle were methodical and well controlled, due in part to good communication and coordination among all involved organizations.

M8 Miscellaneous Maintenance Issues (92700, 92902)

M8.1 (Closed) LER 50-220/97-14: Vent and Purge System Isolation During Troubleshooting Due to Defective Equipment

a. Inspection Scope

The inspectors reviewed the details associated with LER 50-220/97-14, "Vent and Purge System isolation During Troubleshooting due to Defective Equipment," and the applicable sections of the UFSAR. The inspectors discussed issues related to the event with members of the NMPC staff, including the system engineer, and reviewed plant drawings of the related equipment. In addition, the inspectors observed the licensee's Station Operating Review Committee (SORC) meeting for the approval of the LER associated with this issue, and subsequently reviewed the LER to verify completion in accordance with Title 10 of the Code of Federal Regulations Part 50.73 (10 CFR 50.73).

b. Observations and Findings

On November 25, 1997, while performing troubleshooting on the Unit 1 stack gas radiation monitor (RAM-112-08A), an unexpected isolation of the vent and purge system occurred. At the time of the event Unit 1 was shutdown for repairs to the emergency cooling condensers. The operators verified that the stack radiation monitors were indicating normal radiation levels, with no abnormal conditions noted, reset the isolation signal and opened the vent and purge valves. The inspectors considered the licensee's actions taken in response to the event to have been appropriate.

Prior to the event stack radiation monitor RAM-112-08A had been experiencing a series of random downscale alarms; however, troubleshooting by the licensee failed to identify the cause of the problem. Therefore, the monitor was sent to the vendor for refurbishment; however, upon return from the vendor the problem persisted. The licensee determined that additional troubleshooting was need to locate the problem, and it was determined that the troubleshooting needed to be performed with the equipment energized. It was during this troubleshooting that the system isolation occurred.

Although RAM-112-08A only provides indication and alarm, it is bolted to a common chassis that contain other stack gas radiation monitors, particularly, RAM-RN10A, and RAM-RN10B, which provide the isolation signal to the vent and purge system valves. RAM-RN10A and 10B are powered from separate station batteries, but all the radiation monitors within this chassis share a common station ground for the station battery (DC) and 120 VAC.

The licensee's root cause analysis determined that the 24 VDC power supply for RAM-112-08 was defective due to an intermittent short to ground. During the troubleshooting, while the technician was attaching a test probe to the high voltage source an arc occurred between the high voltage source and the low voltage power supply. The arc traveled through the previously undetected failure in the low voltage power supply to the station ground. The subsequent high voltage potential on the ground was detected in RAM-RN10A and 10B, causing the trip relays to actuate which caused the system isolation. The inspectors discussed the root cause with the system engineer, including a review of the applicable plant drawings, and considered the root cause to be reasonable.

The inspectors observed the licensee's SORC meeting that approved the LER 50-220/97-14. The inspectors found the meeting to be completed with the proper safety focus and in accordance with the license's procedures.

During the SORC meeting, the technician involved with the event provided detailed information as to the part that the troubleshooting played in initiating the event. The licensee determined, that although enhancements to their controls for work with energized equipment could be made as a result of lessons learned from this event, the identified shortcomings during the troubleshooting, particularly that the troubleshooting initiated the arc between the high voltage source and the low voltage power supply, alone would not have caused the vent and purge system isolation without the ground within low voltage power supply. The particular improvements to the licensee's maintenance program for work on energized equipment would be addressed through the disposition of the associated DER. The inspectors considered the licensee's approach to be acceptable.

The inspectors verified that the LER was completed in accordance with the requirements of 10 CFR 50.73. Specifically, the description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. The root cause and corrective and preventive actions as described in the LER were reasonable. This LER is closed.

c. Conclusion

Licensee's actions were appropriate in response to an unexpected isolation of the Unit 1 vent and purge system that occurred during radiation monitor troubleshooting. The licensee's root cause of the event was reasonable and the Station Operating Review Committee's review of the event maintained the proper safety focus.

III. ENGINEERING

E1 Conduct of Engineering (37551)

E1.1 General Comments

Using NRC Inspection Procedure 37551, the resident inspectors frequently reviewed design and system engineering activities, including justifications for operability determinations, and the support by the engineering organizations to plant activities.

E2 Engineering Support of Facilities and Equipment (37551)

E2.1 Unit 1 Emergency Cooling Condenser Keepfull Modification

a. Inspection Scope

In September 1997, Unit 1 was shutdown due to tube leaks in the EC condensers. One of the corrective actions to prevent recurrence was the installation of a keepfull modification to maintain the water level in the EC condenser steam lines above the top of the tube sheet.

The inspectors reviewed the modification package, including the applicability review and safety evaluation, the licensing document change requests, the training summary, and the completed modification functional test procedure. In addition, the inspectors walked down the as-built modification and compared it to the design drawings.

b. Observations and Findings

The EC system is designed as a standby, natural circulation system. In September 1997, NMPC identified tube leaks on all four of the EC condensers, resulting in an 85 day forced outage. Investigation by NMPC determined that the leaks were due to thermal cycling of the upper-most tubes, caused by excessive leakage past the closed EC condensate return valves. The keepfull modification was designed to remove the thermal fatigue potential. In addition, maintenance was performed on the condensate return valves during the forced outage to minimize leakage.

The modification involved a tap-off from the high pressure discharge of the control rod drive (CRD) system, and tied into the EC condensate return line, to maintain the condensate level in the inlet side above the top of the tube bundle. The flow is via ½ inch stainless steel tubing, and includes filters, metering valves, flow meters, and relief valves. The keepfull system also includes thermocouples for detecting water level in the inlet piping. The inspectors walked down the installed keepfull system and identified no discrepancies between the as-built and the design drawings.

In addition, the inspectors reviewed the system summary used to train the shift operators before the system was placed in service. The training summary included a description of the system design, the expected response of the system, and the associated alarm response procedures for high and low water levels in the inlet piping. The inspectors monitored one of the shift briefs during which the training was provided, and considered the training adequate.

The inspectors reviewed the applicability review (AR 23527) and the 10 CFR 50.59 safety evaluation (SE No. 97-138). The inspectors also reviewed the licensing document change requests (LDCRs) for the applicable sections of the UFSAR (LDCR 1-97-UFS-086) and the inservice test program (LDCR 1-97-IST-010). The inspectors attended various Unit 1 SORC meetings related to the modification. Finally, the inspectors reviewed the modification functional test (N1-MFT-051B). After startup, the system performed as expected. The inspectors identified no discrepancies in the above engineering documents. The inspectors concluded the modification was well designed and installed.

c. Conclusions

The Unit 1 modification of the EC keepfull system was well designed. The modification was installed according to the drawings, and adequately tested.

E8 Miscellaneous Engineering Issues (90712, 92700, 93903)

E8.1 (Closed) LER 50-410/97-15: Opening Between Secondary Containment and Reactor Building Auxiliary Bay

a. Inspection Scope

The inspectors reviewed the details associated with LER 50-410/97-15, "Opening Between Secondary Containment and Reactor Building Auxiliary Bay," and applicable DERs. The inspectors visually inspected the affected portion of the auxiliary bay wall, and discussed issues related to the event with members of the NMPC staff. In addition, the inspectors reviewed the LER to verify completion in accordance with Title 10 of the Code of Federal Regulations Part 50.73 (10 CFR 50.73).

b. Observations and Findings

On November 14, 1997, while installing new emergency lights at Unit 2, a maintenance supervisor discovered a six inch by eight inch opening in the wall between the Unit 2 secondary containment stairwell and the north auxiliary bay (NAB). This wall forms a boundary between the environmental qualification (EQ) design basis harsh environment in the secondary containment stairwell and the mild environment in the NAB. Therefore, the opening provided a path for the harsh environment, which could occur following a loss of coolant accident (LOCA) or a high energy line break (HELB), to enter the NAB and adversely impact the equipment that had only been qualified for mild environment conditions. The NAB contains

safety-related Division I AC motor control centers (MCCs), and power panels that provide power to equipment for many systems, including RCIC, low pressure core spray (CSL), and the following Division I systems: SLC, service water (SW), residual heat removal (RHR), reactor building ventilation (HVR), standby gas treatment (GTS), and containment hydrogen recombiners (HCS). Also contained within the NAB is safety-related containment monitoring system (CMS) equipment and reactor recirculation system pump 'A' controls.

Upon discovery of the opening, the Unit 2 control room was notified. Based on the information, the licensed operators declared the equipment contained in the NAB inoperable, and the applicable TS were evaluated. The most restrictive limiting condition for operations (LCO) was associated with the Division I AC distribution system, which allowed eight hours of operations prior to entering the action statement requiring the plant to be in hot shutdown within the next twelve hours. Subsequently, NMPC notified the NRC of the condition in accordance 10 CFR 50.72. The licensee preformed a visual inspection, which included similar walls in the south auxiliary bay, to verify that no additional openings existed. Later that day, the opening was sealed and the equipment was returned to operable status within the time allowed by TS.

The inspectors reviewed the applicable portions of the SSS's logs and discussed the licensee's actions with an on-watch SSS, the Maintenance Manager, and the Maintenance Supervisor who discovered the opening. In addition, the inspectors visually inspected the repair to the opening, and noted that location was such that the opening was not readily obvious. The inspectors considered the identification of the opening to be good, and the recognition of the potential significance associated with the problem to be very good. The inspectors also considered NMPC's actions following the identification of the opening to be appropriate.

The LER described the root cause of the event as improper construction when the wall was formed during plant construction with a contributing cause of inadequate quality inspection at that time. However, at the time the LER was submitted, the licensee had yet to complete their analysis to determine the impact of the migration of the harsh environmental conditions into the NAB and the resultant impact on the non-qualified equipment. NMPC is expected to submit a supplemental report in July 1998, to convey the formal analysis results.

Although the UFSAR does not explicitly describe the EQ classifications of the various plant locations or equipment, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," describes that this information is controlled through documents outside the UFSAR. The failure to have adequately design and constructed the wall of the NAB, which resulted in the EQ concern for the safety-related equipment contained within the NAB, is violation of 10 CFR 50 Appendix B, Criterion III, Design Control. This original design and installation discrepancy was identified as part of a licensee's initiative, corrective actions were prompt and comprehensive, the violation was not likely to be identified by routine licensee efforts such as normal surveillance or quality assurance activities and the violation is not reasonably linked to current performance. Therefore, this apparent violation

of NRC requirements will not be cited in accordance with Section VII.B.3 of the NRC Enforcement Policy. (NCV 50-410/97-12-04)

The inspectors verified that the LER was completed in accordance with the requirements of 10 CFR 50.73. Specifically, the description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. However, throughout the LER, the licensee described the opening to be between secondary containment and the NAB. The inspectors considered the licensee's terminology to be potentially misleading since the NAB is considered inside secondary containment. The inspectors provided this concern to both the Unit 2 Engineering and Operations Managers, who acknowledged the inappropriate terminology. The root cause and corrective and preventive actions as described in the LER were reasonable. This LER is closed.

c. Conclusion

At Unit 2, NMPC's identification of a breach between an equipment qualification (EQ) classified harsh environment area and a mild environment area, an original construction deficiency, was considered good. (NCV) Particularly noteworthy was the recognition that in the event of a high energy line break, the breach could result in the potential loss of several safety-related systems. Once identified, the licensee took appropriate actions to repair the breach and to verify no other similar openings.

E8.2 (Closed) LER 50-410/97-14: Failure to Adequately Perform Technical Specification Surveillance on Rod Sequence Control System Due to Procedure Inadequacy

a. Inspection Scope

During a Unit 2 reactor startup, while performing a surveillance test to verify the operability of the rod block function of the rod sequence control system (RSCS), an operator realized that the rod worth minimizer (RWM) was also potentially generating a rod block signal. Since the RWM could block rod movement, the surveillance test, as written, could not verify that the RSCS was performing as required.

The inspectors monitored the performance of the surveillance test, discussed the concern with the operator and the SSS, and reviewed the corrected procedure and associated LER.

b. Observations and Findings

On November 10, 1997, during a Unit 2 reactor startup, a RO recognized that the surveillance test he was performing to verify operability of the rod block function of the RSCS was potentially inadequate. Specifically, the RWM could also generate a rod block. The surveillance test was stopped and the RO discussed his concern with shift supervision; subsequently, a procedure change was generated to bypass the RWM, and the surveillance test was completed successfully.

Unit 2 TS Surveillance Requirement (TSSR) 4.1.4.2.b requires the RSCS be demonstrated operable by verifying that an inhibited control rod cannot be moved. This TSSR is to be performed after the first control rod has been withdrawn for each reactor startup. The surveillance test procedure (N2-OSP-RMC-@004, "Rod Sequence Control System Operability Test," Rèvision 1) was intended to satisfy this TSSR. Because the RWM could mask the required rod block signal from the RSCS, previous surveillance tests back to 1990 failed to satisfy the TSSR.

NMPC determined the root cause for the missed surveillance was inadequate change management. Specifically, the RSCS procedure was not changed in 1990, when the existing RWM was replaced with a new design which imposed a rod block. In addition, there was an inadequate technical review of the RSCS procedure in 1991 when the procedure was last revised. The LER noted without explanation that the system engineer associated with the RWM modification recommended that procedure steps be added to bypass the RWM, but the steps were not incorporated.

The failure to adequately test the rod block feature of the RSCS is a violation of TS 3.1.4.2; in that the RSCS was not verified to be operable by performance of the associated test procedure. **(VIO 50-410/97-12-05)** Of concern to the inspector was the fact that the system engineer's recommendations in 1991, regarding changes to the RSCS procedure, were not incorporated.

The inspectors reviewed the LER and found it to be timely and to accurately describe the event. The immediate corrective actions were appropriate, the adequacy of the actions to prevent recurrence, including why the system engineer's recommendations were not incorporated during the procedure change, will be evaluated during the followup inspection of the violation. This LER is closed.

c. Conclusion

A Unit 2 reactor operator demonstrated a good questioning attitude in identifying that a TS required surveillance test for the rod sequence control system was inadequate. (VIO)

E8.3 (Closed) LER 50-410/97-12: Missed Technical Specification Surveillance of the Control Building Relay Room Temperature

a. Inspection Scope

The inspectors reviewed the details associated with LER 50-410/97-12, "Missed Technical Specification Surveillance of the Control Building Relay Room Temperature," and applicable DERs. They discussed the issues related to the event with the General Supervisor of Operations (GSO) and members of the operations support staff. In addition, the inspectors reviewed the LER to verify completion in accordance with Title 10 of the Code of Federal Regulations Part 50.73 (10 CFR 50.73).

b. Observations and Findings

On September 22, 1997, a member of the Unit 2 operations support staff recognized that prior to September 30, 1996, the surveillance test used to meet the TSSR 4.7.3.a, failed to include temperature verification of the control building relay room. Prior to September 30, 1996, NMPC had not periodically monitored the relay room temperature, and on that date NMPC initiated DER 2-96-2348 to determine if it was required to monitor relay room temperatures to satisfy TSSR 4.7.3.a, which requires shiftly verification that control room temperatures are maintained below 90°F. In response to DER 2-96-2348, NMPC incorrectly determined that monitoring the relay room temperature was not required by the TSSR; however, on September 30, 1996, NMPC revised Operations Surveillance Procedure N2-OSP-LOG-S001 to include the verification of the relay room temperature. The failure to perform the surveillance test prior to September 30, 1996, is a violation of TSSR 4.7.3.a. This failure constitutes a violation of minor significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy. **(NCV 50-410/97-12-06)**

On August 27, 1997, NMPC determined that they were not testing the control room outside special filter train system in accordance with the description of the control room envelope as described in the Unit 2 UFSAR. In particular, the control room envelope as described in the Unit 2 UFSAR included the relay room, and the relay room portion of the Control Room Outside Air Special Filter Train System (CRSFT) was not being tested. During the SORC review of the associated LER (LER 50-410/97-09, "Missed Technical Specification Surveillance of the Control Room Envelope," described in NRC IR 50-410/97-11), NMPC realized that the inclusion of the relay room as part of the control room for TSSRs meant that the disposition to DER 2-96-2348 was incorrect, and that although they were currently performing the TSSR for monitoring relay room temperature, that the failure to complete the TSSR prior to September 30, 1996, was a reportable event.

Although relay room temperatures were not being monitored prior to September 30, 1996, operators performed shiftly tours of the relay room providing the opportunity to identify excessive temperature conditions. In addition, control room operators would have been alerted to potential temperature concerns within the relay room by control room alarms associated with relay room unit cooler trips and relay room cooler return duct high temperature (80°F). Based on the above the licensee had no indication that the relay room exceeded the TS limit of 90°F.

The inspectors verified that the NMPC Procedure N2-OSP-LOG-S001 adequately included relay room temperature monitoring and that the instrument used to monitor the relay room temperature was calibrated and within the licensee's calibration program. The inspectors also reviewed DER 2-96-2348 and considered it a missed opportunity to have identified the concerns related to the UFSAR description of the control room envelope.

The inspectors verified that the LER was completed in accordance with the requirements of 10 CFR 50.73. Specifically, the description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. The root cause and corrective and preventive actions as described in the LER were reasonable. This LER is closed.

c. Conclusion

Prior to September 1996, NMPC failed to monitor the Unit 2 relay room temperature, as required by TS. (NCV) Furthermore, when the licensee identified this issue in 1996, they incorrectly dispositioned it, resulting in a failure to recognize that the condition was reportable, and missed an opportunity to identify other subsequently identified concerns related to the UFSAR description of the control room envelope.

E8.4 (Closed) LER 50-220/97-12: Additional 10 CFR 50 Appendix R Section III.J Lighting Deficiencies

a. Inspection Scope

The inspectors reviewed the details associated with LER 50-220/97-12. The issues related to the event were discussed with licensee staff and management. The inspectors reviewed the documentation associated with the event. In addition, the inspectors reviewed the LER to verify completion in accordance with 10 CFR 50.73.

b. Observations and Findings

On October 16, 1997, during a design bases review of the Unit 1 UFSAR, Section 10B, NMPC identified an area of Unit 1 that did not meet the requirements of 10 CFR 50, Appendix R, Section III.J, Emergency Lighting. Specifically, the Unit 1 Appendix R Safe Shutdown Analysis (SSA) for the control room fire, contained operator actions to locally verify EC condenser vent-to-torus isolation; however, these actions were not considered required because the SSA had credited the EC condenser vent-to-torus isolation resulting from the reactor protection system (RPS) vessel isolation signal. Therefore, no emergency lights were installed at the EC condenser vent-to-torus valve location. While preparing a 10CFR 50.59 safety evaluation to eliminate the operator actions for verifying the EC vent-to-torus isolation, an engineer determined that the RPS vessel isolation did not include the EC condenser vent-to-torus flow path. Therefore, not only were the operator actions to verify isolation of the EC condenser vent-to-torus required, emergency lights were required by 10 CFR 50, Appendix R, Section III.J, to be installed for the operators to perform the actions.

During NMPC's review of this issue, the licensee noted two missed opportunities to identify this discrepancy earlier. First, in January 1992, the developers of the Unit 1 SSA inappropriately credited the EC vent-to-torus isolation resulting from a reactor protection system vessel isolation signal. Second, an engineering review of the Fire Protection Engineering Evaluation (FPEE) 1-90-014, Revision 1, "Nine Mile

Point High/Low Pressure Inventory Loss Flow Path Analysis," which was a basis document for the SSA, failed to identify that the FPEE also incorrectly credited the RPS vessel isolation signal for EC condenser vent-to-torus isolation.

Licensee corrective actions included a review of the FPEE to ensure that all valves credited with closing on receipt of an RPS isolation signal actually operated as such. Emergency lights were installed at the EC condenser vent valves and this was visually confirmed by the inspectors. The next revision of the Unit 1 UFSAR will reflect the changes to the revised FPEE. The event was discussed with engineering personnel, and procedures associated with fire protection engineering preparation and review will be revised to ensure proper technical input, impact assessment, design reviews, and verification are in place.

The inspectors considered that inadequate engineering development and review of the SSA and FPEE in 1992 and 1996, respectively, reflected a weakness in engineering design review. The 1997 engineering review was good, in that it identified the previous deficiencies and recognized need for EC condenser manual isolation valve closure and the installation of emergency lights to support the evolution. Notwithstanding, the failure to provide emergency lighting for access to and closure of Unit 1 EC condenser vent manual isolation valves is a violation of 10 CFR 50, Appendix R, Section III.J. **(VIO 50-220/97-12-07)**

The inspectors verified that the LER was completed in accordance with the requirements of 10 CFR 50.73. Specifically, the description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. The root cause and corrective and preventive actions as described in the LER were reasonable. This LER is closed.

c. <u>Conclusions</u>

The 1997 engineering review of the Unit 1 Safe Shutdown Analysis and Fire Protection Engineering Evaluation documents was good, in that it disclosed previous engineering deficiencies, particularly that emergency lighting required to support alternate shutdown of the plant was missing. (VIO) Earlier reviews of these documents were weak in that they failed to identify these deficiencies.

E8.5 <u>(Closed) 10 CFR Part 21 Notification: Suppression Pool Bypass Leakage due to Postulated Standby Gas Treatment System Failure</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the details associated with 10 CFR Part 21 (Part 21) Notification, "Suppression Pool Bypass Leakage due to Postulated Standby Gas Treatment System Failure," and NMPC's evaluation of the applicability of the notification to both units. The inspectors reviewed the applicable DER for each unit and discussed the related issues with members of NMPC's engineering, licensing and operations staff.

b. Observations and Findings

On October 12, 1997, General Electric (GE) issued a 10 CFR Part 21 notification (SC 97-04) pertaining to a possible control cable shorting, which could cause spurious opening of the drywell (DW) and wetwell (i.e., the torus for Unit 1, and the suppression chamber for Unit 2) vent valves, thus creating bypass leakage and potentially reducing the pressure suppression capability of the wetwell. Both units evaluated the issue as described in the Part 21, and determined that it not to be a concern. However, NMPC engineering further evaluated the potential for DW-to-wetwell leakage particularly since other nuclear facilities had already identified evolutions which created a DW-to-wetwell flow path. The evaluation for Unit 1 resulted in an LER, and the review is provided in Section E8.6 of this report.

The licensee reviewed the Unit 2 procedures and on December 8, 1997, NMPC identified that during primary containment purging following forced and planned unit outages, a bypass pathway between the drywell and the suppression chamber was created during the high-flow inerting procedure when the suppression chamber and drywell air spaces were simultaneously purged to reduce containment oxygen levels. The same bypass lineup is utilized for the de-inerting process, which proceeded outages requiring primary containment access. NMPC determined that the last time they were in this line-up was November 12, 1997 following the startup from Forced Outage 97-03. Upon identification of this bypass pathway, NMPC verified that all the containment purge valves were closed and established administrative controls by hanging hold-out tags on the applicable equipment to preclude creation of the bypass pathway until a procedure change was completed. The inspectors verified that the hold-out tags were established to control the purge valves.

Also, upon identification of the concern, NMPC notified the NRC of the condition in accordance with 10 CFR 50.72. However, at the time the bypass pathway was identified, NMPC had yet to determine whether the condition would result in exceeding the maximum post accident design pressure rise within the containment. Upon completion of the their analysis, the licensee determined that the maximum post accident design pressure rise within the containment would not be exceeded, and on January 7, 1998, they retracted their earlier 10 CFR 50.72 notification.

The inspectors reviewed the Unit 2 DER associated with this Part 21 and considered the licensee's review of the issue as described in the Part 21 to be appropriate. Furthermore, the licensee's evaluation of other possible evolutions which created a DW-to-wetwell flow path to be good, and the actions taken to address identified discrepancies, were adequate.

c. Conclusion

The licensee's review of the 10 CFR Part 21 notification associated with possible communications between the drywell and the wetwell potentially reducing the pressure suppression capability of the wetwell was appropriate, and their evaluation

of other possible evolutions which created a drywell-to-wetwell flow path to be good. Actions taken at Unit 2 to address identified discrepancies, were adequate.

E8.6 (Closed) LER 50-220/97-15: Potential Bypass Leakage Path Between Drywell and Torus During Vent and Purge

a. Inspection Scope

The inspectors reviewed the details associated with licensee event report (LER) 50-220/97-15. The issues related to the event were discussed with the Unit 1 Plant Manager and Engineering Manager. The inspectors reviewed the applicable procedures and documentation associated with the event. In addition, the inspectors reviewed the LER to verify completion in accordance with 10 CFR 50.73.

b. Observations and Findings

On December 3, 1997, NMPC engineering staff determined that Unit 1 had operated in a configuration which could potentially impact the pressure suppression function of the torus. This condition was identified during an engineering review of General Electric (GE) 10 CFR Part 21, SC 97-04. The specifics of the Part 21 concerned control cable shorting which could cause spurious opening of drywell (DW) and torus vent valves, thus creating bypass leakage and potentially reducing the pressure suppression capability of the torus. This particular condition was evaluated by NMPC and determined not to be a concern at Unit 1.

The engineering staff further evaluated the potential for DW-to-torus interface conditions resulting from normal plant operations, since other nuclear facilities had already identified certain evolutions which created a DW-to-wetwell interface. NMPC subsequently determined that two Unit 1 operational lineups established this interface. During DW and torus inerting, deinerting, and primary containment pressure maintenance evolutions, the DW and torus vent valves were usually opened concurrently, establishing a DW-to-torus interface. Also, a DW-to-torus interface was established during periodic performance of suppression chamber (i.e. torus) to DW vacuum breaker surveillance testing. The inspectors considered the identification of the DW-to-wetwell interface conditions to be appropriate, considering similar industry events had been previously documented.

The licensee concluded that the vacuum breaker surveillance testing is performed to meet Unit 1 TS 4.3.6, Vacuum Relief. Technical Specification 3.3.6 states that when primary containment is required, all suppression chamber-to-drywell vacuum breakers shall be operable except during testing. NMPC subsequently retracted the portion of the 10 CFR 50.72 notification related to the vacuum breakers.

With respect to the DW-to-torus interface, the Operations Department issued a procedure change to prohibit concurrent opening of both the DW and torus vent and purge valves during primary containment venting, filling and makeup evolutions. The inspectors considered this procedure change to be prudent. The engineering staff also determined that the peak DW and torus pressures may increase slightly,

but should remain below their respective design pressures with the DW and torus interconnected. However, these results were preliminary, and a final analysis was to be completed by April 1998. This item will remain open pending the NRC review of (1) the licensee's root cause evaluation, and (2) the final engineering analysis of containment pressure within the DW and torus with the vent and purge valves open during a loss-of-coolant accident and the impact on emergency core cooling systems. (URI 50-220/97-12-08)

The inspectors verified that the LER was completed in accordance with the requirements of 10 CFR 50.73. Specifically, the description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. The corrective and preventive actions as described in the LER were reasonable. This LER is closed.

c. Conclusions

The identification of the drywell-to-wetwell interface during certain operational evolutions at Unit 1 was appropriate. A procedural change to preclude venting and purging the drywell and torus concurrently was prudent. A preliminary engineering evaluation of the peak containment pressure achieved by having a DW to torus interconnection resulted in pressures below design, with the final analysis to be completed in early Spring 1998. (URI)

E8.7 (Closed) LER 50-220/97-10: Technical Specification Required Shutdown Due to Emergency Cooling Condenser Tube Leak

The technical issues associated with this LER were described in NRC Inspection Report (IR) 50-220/97-07, Section O1.2, and NRC IR 50-220/97-11, Section M1.2. The inspectors verified that the LER was completed in accordance with the requirements of 10 CFR 50.73. Specifically, the description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. Although the licensee had yet to complete their root cause evaluation, their preliminary root cause evaluation indicated that the failure mechanism was thermal cycling as a result of leakage through the EC condenser return isolation valves. Due to piping configurations, this leakage allowed steam void formation in the upper portions of the inlet hemihead. Normally, the inlet hemihead should be completely filled with water. Therefore, this steam/water mixture in the upper tubes created the thermal cycling.

In addition to the tube bundle replacement in all four emergency cooling condensers, the licensee completed the following actions to prevent EC condenser tube leak recurrence: (1) repaired and tested the EC condenser discharge return isolation valves, (2) installed a keepfull modification (discussed in Section E2.1) to ensure that the inlet hemihead and tube bundle would remain covered with water during normal steady-state operation, and (3) installed thermocouples, and used thermography, to better trend and monitor EC condenser inlet piping water level.

The repairs to the EC condenser discharge return isolation valves appeared to be effective, in that the EC condenser inlet piping water level has been adequately maintained with minimal keepfull system flow. The inspectors considered the corrective and preventive actions as described in the LER were reasonable. The licensee will issue a supplement to this LER following a final root cause determination for the tube failures. This LER is closed.

E8.8 Receipt Inspection of a Unit 1 Emergency Cooling Condenser Tube Bundle

a. Inspection Scope

Using NRC Inspection Procedure 38702, "Receipt, Storage, and Handling of Equipment and Materials Program," the inspectors observed the receipt inspection of an EC condenser tube bundle. The inspectors reviewed the licensee procedure governing material receipt inspection, and discussed the inspection with the Procurement Inspector and Quality Assurance personnel.

b. Observations and Findings

On November 13, 1997, the second of four EC condenser tube bundles was delivered to Unit 1. The bundle was delivered to the Unit 1 reactor building track bay for inspection, off-loading, and transfer to the reactor building. The procurement inspector, using NMPC Procedure NPAP-INV-210, "Receipt, Test, Inspection and Processing of Materials, Parts and Services," Revision 08, performed an inspection of the tube bundle. The inspector looked for any visible damage that may have occurred during transport. Additionally, the procurement inspector ensured that the proper component had been shipped, in part, by ensuring that all applicable documentation was present, accurate, and corresponded to serial numbers and stamping present on the tube bundle. The procurement inspector identified no discrepancies, and the NRC inspectors considered the inspection to be thorough and in accordance with procedure.

The NRC inspectors reviewed the Delivery Inspection Checklist for completion and accuracy. The checklist and all required tags were complete and affixed to the tube bundle. The tube bundle was subsequently moved into the reactor building and transferred to the 340-foot elevation for placement within the EC 121 condenser shell (see Section M4.1).

c. Conclusions

NMPC receipt inspection of an Unit 1 EC condenser tube bundle was thorough.

E8.9 (Closed) URI 50-220/96-13-01: Analysis of Water In the Emergency Cooling Condenser Steam Lines During the Unit 1 Overfill Event

During the Unit 1 overfill event (November 1996), the inspectors questioned if the operability of the decay heat removal function of the EC system had been considered with respect to water acting as a blockage for the steam lines.

The inspectors discussed the unresolved item with Unit 1 engineering and operations personnel. NMPC confirmed that the EC condensers would be inoperable if the steam lines were filled with water. But it was determined that there were no credible accidents described in the Unit 1 UFSAR where a high reactor pressure would be coincident with a high reactor water level condition, and where the EC system would be expected to actuate. Pertinent accidents reviewed included:

- turbine trip without bypass
- turbine trip with partial bypass (full power)
- feedwater controller failure - maximum demand
- main steam line isolation valve closure
- safety valve actuation (over pressure analysis)
- loss of electrical load (generator trip)
- instrument air failure

Based on the above, the inspectors had no further questions and this item is closed.

E8.10 (Closed) LER 50-410/97-08: Potential Spurious Actuation of RWCU [reactor water cleanup] High Pressure/Low Pressure Interface Valves

The technical issues associated with this LER were described in NRC IR 50-410/97-09, Section E1.1. The inspectors performed an in office review of the LER and verified that it was completed in accordance with the requirements of 10 CFR 50.73. Specifically, the description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. The root cause and corrective and preventive actions as described in the LER were reasonable. This LER is closed.

E8.11 (Closed) LER 50-410/97-02 Supplement 3: Potential Inoperability of Emergency Diesel Generator Service Water Cooling Outlet Valves During a Control Room Fire

The technical issues associated with this LER were described in NRC IR 50-410/97-04, Section E8.4. The inspectors performed an in office review of the additional information provided in LER 50-410/97-02 Supplement 3, and found it acceptable. This LER is closed.

IV. PLANT SUPPORT

Using NRC Inspection Procedure 71750, the resident inspectors routinely monitored the performance of activities related to the areas of radiological controls, chemistry, emergency preparedness, security, and fire protection. Minor deficiencies were discussed with the responsible management, significant observations are detailed below. Specialist inspectors in the same areas used other procedures during their reviews of plant support activities; these inspection procedures are listed, as applicable, for the respective sections of the inspection report.

R2 Status of RP&C Facilities and Equipment (71750)

R2.1 Inspection of Normally Inaccessible Reactor Water Cleanup Areas at Unit 2

a. Inspection Scope

The inspectors assessed the material condition, housekeeping and radiological controls associated with the normally inaccessible reactor water cleanup (RWCU) valve aisle, and RWCU heat exchanger and RWCU "B" pump rooms.

b. Observations and Findings

During the planned outage of the Unit 2 RWCU system, the dose rates in the areas housing the system components were significantly reduced; the inspectors used this opportunity to tour these normally inaccessible areas.

The material condition of the equipment contained in the areas inspected was satisfactory, the material condition of the equipment within the RWCU valve aisle noted as particularly good. However, the inspectors noted that a cover for a conduit terminal point, which was labeled as containing safety-related cable, was only attached with one of the two fasteners. Upon informing the licensee, the cover was properly installed, and the licensee determined that the cable within that conduit had been removed as part of an earlier plant modification.

In general, housekeeping was acceptable, hoses were noted in both the heat exchanger and the "B" pump rooms. Discussion with the Assistant Station Shift Supervisor (ASSS) revealed that the hoses were needed for the filling and venting of the system during restoration, and that the hoses would be properly stored upon completion of that task. The inspectors also noted approximately four valve labels on the floor in the RWCU valve aisle. Following discussion with the licensee, the inspector ascertained that the labels were most likely removed during the maintenance in that area, the labels were subsequently reattach.

Radiological controls established for each area inspected were appropriate for the conditions.

c. Conclusion

An inspection of normally inaccessible areas of the Unit 2 reactor water cleanup (RWCU) system found the material condition of the equipment to be satisfactory, with the condition of the equipment in the RWCU valve aisle to be particularly good. Housekeeping in the areas inspected was acceptable, and appropriate radiological controls were established.

R8 Miscellaneous RP&C (92904)

R8.1 (Closed) LER 50-220/97-13: Engineered Safety Feature Actuation During Calibration Due to Personnel Error

a. Inspection Scope

The inspectors reviewed the details associated with LER 50-220/97-13. The issues related to the event were discussed with an Unit 1 chemistry supervisor. The inspectors reviewed the applicable procedures and documentation associated with the event. In addition, the inspectors reviewed the LER to verify completion in accordance with 10 CFR 50.73.

b. Observations and Findings

On November 10, 1997, a Unit 1 chemistry technician was performing a calibration of the stack gas radiation monitor (RN-10B). The technician was performing the evolution using Unit 1 Chemistry Surveillance Procedure N1-CSP-V325, "OGESMS [Off Gas Effluent Stack Monitoring System] Noble Gas Detector Primary Calibration," Revision 01, and work order (WO) 97-05326-00. N1-CSP-V325 required the technician to use a high energy gamma source as designated in the WO. The WO referenced a source designated "SA-1226"; however, the technician used a source designated "SA-296," which had an activity approximately ten times that of SA-1226. While positioning source SA-296 (the wrong source) on detector RN-10B, alarms on RN-10B and RN-10A occurred simultaneously. This caused an automatic isolation of the DW vent and purge lines, which is an engineered safeguards feature actuation.

At the time of the event, the plant was shutdown and two months into a forced outage. Prior to the event, the vent and purge valves were open to support DW ventilation. Following the ESF actuation, calibration procedure was secured and the actuation verified to be a result of the calibration. The stack radiation monitor signals were reset and the DW vent and purge valves reopened.

NMPC has recently strengthened their commitment to ensure that personnel use self-verification and peer-verification when performing all tasks, and especially during evolutions with potential safety impact on the plant. Notwithstanding, the inspectors considered this event to be a further example of failing to follow procedure due to personnel inattention-to-detail. This failure to implement procedures as written is a violation of Unit 1 Technical Specifications, Section 6.8.1. **(VIO 50-220/97-12-09)**

The inspectors verified that the LER was completed in accordance with the requirements of 10 CFR 50.73. Specifically, the description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. The root cause and corrective and preventive actions as described in the LER were reasonable. This LER is closed.

c. Conclusions

An inadvertent automatic isolation of the Unit 1 drywell vent and purge lines, occurred due to personnel inattention-to-detail, particularly a failure to follow procedure. (VIO)

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

At periodic intervals, and at the conclusion of the inspection period, meetings were held with senior station management to discuss the scope and findings of this inspection.

The final exit meeting occurred on January 23,1998. During this meeting, the resident inspector findings were presented. NMPC did not dispute any of the inspectors findings or conclusions. Based on the NRC Region I review of this report, and discussions with NMPC representatives, it was determined that this report does not contain safeguards or proprietary information.

X2 Pre-Decisional Enforcement Conference Summary

X2.1 Pre-Decisional Enforcement Conference Related to the Radioactive Waste and Transportation of Radioactive Materials Programs

On December 19, 1997, a pre-decisional enforcement conference was held at the NRC Region I office to discuss issues identified in IR 50-220 & 50-410/97-07. The issues were related to concerns related to the radioactive waste programs and the transportation of radioactive materials. Handouts used in the licensee's presentation at the conference are included as Attachment 2 to this report.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Niagara Mohawk Power Corporation

R. Abbott	Plant Manager, Unit 1 (Acting)
D. Barcomb	Manager, Unit 2 Radiation Protection
D. Bosnic	Manager, Unit 2 Operations
J. Burton	Manager, Quality Assurance
H. Christensen	Manager, Security
J. Conway	Vice President, Nuclear Engineering
G. Correll	Manager, Unit 1 Chemistry
R. Dean	Manager, Unit 2 Engineering
A. DeGracia	Manager, Unit 1 Work Control
S. Doty	Manager, Unit 1 Maintenance
K. Dahlberg	Plant Manager, Unit 2 (Acting)
G. Helker	Manager, Unit 2 Work Control
J. Kahn	Director, ISEG (Acting)
P. Mazzafero	Manager, Unit 1 Technical Support
L. Pisano	Manager, Unit 2 Maintenance
R. Randall	Manager, Unit 1 Engineering
V. Schuman	Manager, Unit 1 Radiation Protection
R. Smith	Manager, Unit 1 Operations
R. Tessier	Manager, Training
C. Terry	Vice President, Nuclear Safety Assessment & Support
K. Ward	Manager, Unit 2 Technical Support
C. Ware	Manager, Unit 2 Chemistry
D. Wolniak	Manager, Licensing

INSPECTION PROCEDURES USED

IP 37551	On-Site Engineering
IP 61726	Surveillance Observations
IP 62707	Maintenance Observation
IP 71707	Plant Operations
IP 71714	Cold Weather Preparations
IP 71715	Sustained Control Room and Plant Observation
IP 71750	Plant Support
IP 90712	In-Office Review of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92700	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901	Followup - Plant Operations
IP 92902	Followup - Maintenance
IP 92903	Followup - Engineering
IP 92904	Followup - Plant Support

Attachment 1

ITEMS OPENED, CLOSED, AND UPDATED

OPENED

50-410/97-12-01	NCV	CST Building Temperature Control Switch Setpoints not in accordance with the UFSAR
50-410/97-12-02	NCV	Failure to Post SLC Monthly Surveillance Data Sheet
50-220/97-12-03	NCV	Failure to met TS for Placing the Mode Switch in REFUEL
50-410/97-12-04	NCV	Breach Between Harsh and Mild EQ Areas
50-410/97-12-05	VIO	Failure to Adequately Perform TSSR on Rod Sequence Control System due to Procedure Inadequacy
50-410/97-12-06	NCV	Failure to Perform TSSR to Monitor Relay Room Temperature
50-220/97-12-07	VIO	Missing Appendix R Emergency Lights at the Emergency Cooling Condenser Valve Station
50-220/97-12-08	URI	Impact of Drywell/Wetwell Bypass on Containment Pressure
50-220/97-12-09	VIO	Failure to Follow Procedure Resulted in Inadvertent ESF Actuation

CLOSED

50-410/97-02-03	LER	Potential Inoperability of Emergency Diesel Generator Service Water Cooling Outlet Valves During a Control Room Fire
50-410/97-08	LER	Potential Spurious Actuation of RWCU High Pressure/Low Pressure Interface Valves
50-220/97-10	LER	Technical Specification Required Shutdown Due to Emergency Cooling Condenser Tube Leak
50-220/97-11	LER	Previous Plant Shutdown in Violation of Technical Specifications
50-220/97-12	LER	Additional 10 CFR 50 Appendix R Section III.J Lighting Deficiencies
50-410/97-12	LER	Missed Technical Specification Surveillance of the Control Building Relay Room Temperature
50-220/97-13	LER	Engineered Safety Feature Actuation During Calibration Due to Personnel Error
50-220/97-14	LER	Vent and Purge System Isolation During Troubleshooting Due to Defective Equipment

Attachment 1

50-410/97-14	LER	Failure to Adequately Perform Technical Specification Surveillance on Rod Sequence Control System Due to Procedure Inadequacy
50-220/97-15	LER	Potential Bypass Leakage Path Between Drywell and Torus During Vent and Purge
50-410/97-15	LER	Opening Between Secondary Containment and Reactor Building Auxiliary Bay
50-410/97-12-01	NCV	CST Building Temperature Control Switch Setpoints not in accordance with the UFSAR
50-410/97-12-02	NCV	Failure to Post SLC Monthly Surveillance Data Sheet
50-220/97-12-03	NCV	Failure to met TS for Placing the Mode Switch in REFUEL
50-410/97-12-04	NCV	Breach Between Harsh and Mild EQ Areas
50-410/97-12-06	NCV	Failure to Perform TSSR to Monitor Relay Room Temperature
50-220/96-01-04	URI	8-Hour vs 12-Hour Shifts
50-220/96-13-01	URI	Analysis of Water in the Emergency Cooling Condenser Steam Lines During Unit 1 Overfill Event
---	10CFR21	Suppression Pool Bypass Leakage due to Postulated Standby Gas Treatment System Failure

<u>UPDATED</u>

None

LIST OF ACRONYMS USED

ACU	Air Conditioning Unit
AOV	Air-Operated Valves
APRM	Average Power Range Monitor
AR	Applicability Review
ASME	American Society of Mechanical Engineers
ASSS	Assistant Station Shift Supervisor
CFR	Code of Federal Regulations
CMS	Containment Monitoring System
CRD	Control Rod Drive
CRSFT	Control Room Outside Air Special Filter Train
CS	Core Spray
CSH	High Pressure Core Spray
CSL	Low Pressure Core Spray
CSO	Chief Station Operator
DER	Deviation/Event Report
dp	Differential Pressure
DW	Drywell
EC	Emergency Cooling
ESA	Engineering Supporting Analysis
ESF	Engineered Safeguards Features
EQ	Equipment Qualification
FPEE	Fire Protection Engineering Evaluation
GE	General Electric
GSO	General Supervisor of Operations
HCS	Containment Hydrogen Recombiners
HELB	High Energy Line Break
HVR	Heating, Ventilation and Refrigeration
H_2O_2	Hydrogen/Oxygen
I&C	Instrumentation and Control
INPO	Institute of Nuclear Power Operations
IR	Inspection Report
LCO	Limiting Condition for Operation
LDCR	Licensing Document Change Request
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MCC	Motor Control Center
NAB	North Auxiliary Bay
NCV	Non-Cited Violation
NMPC	Niagara Mohawk Power Corporation
NRC	Nuclear Regulatory Commission
Part 21	Title 10 of the Code of Federal Regulations Part 21
PDR	Public Document Room
psid	pounds per square inch differential
PTE	Principal Test Engineer

Attachment 1

RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RO	Reactor Operator
RP	Radiation Protection
RSCS	Rod Sequence Control System
RPS	Reactor Protection System
RSP	Remote Shutdown Panel
RWCU	Reactor Water Cleanup
RWM	Rod Worth Minimizer
SGT	Standby Gas Treatment
SLC	Standby Liquid Control
SORC	Station Operating Review Committee
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
SSA	Safe Shutdown Analysis
SSS	Station Shift Supervisor
SW	Service Water
TS	Technical Specification
TSSR	Technical Specification Surveillance Requirement
UFSAR	Updated Final Safety Analysis Report
Unit 1	Nine Mile Point Unit 1
Unit 2	Nine Mile Point Unit 2
VIO	Violation
WO	Work Order

Attachment 2

NMPC Handouts Used During

Pre-Decisional Enforcement Conference

December 19, 1997

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT NUCLEAR STATION

NRC ENFORCEMENT CONFERENCE

DECEMBER 19, 1997

AGENDA

Opening Remarks	B. R. Sylvia
EEI 50-220&50-410/97-07-06	J. Torbitt
EEI 50-220/97-07-07	J. Torbitt
EEI 50-220/97-07-09	J. Torbitt
EEI 50-220&50-410/97-07-10	J. Torbitt
EEI 50-220&50-410/97-07-11	C. D. Terry
EEI 50-220&50-410/97-07-12	C. D. Terry
EEI 50-220&50-410/97-07-13	C. D. Terry
Management Perspectives	R. B. Abbott
Concluding Remarks	B. R. Sylvia

EEI 50-220/97-07-06

"The inspectors noted that the NMP1 PCP lacked any specifics on the way the unit processed wastes, and that it contained erroneous references to federal regulations, specifically outdated versions of 10 CFR 20 (significantly revised in 1994) and 49 CFR (revised in 1996). Failure to incorporate into the PCP the changes in 10 CFR 20 and 49 CFR indicate that the PCP was not periodically reviewed and revised. Failure to maintain the PCP is an apparent violation of NMP1 TS 6.8, which requires that procedures specified in Regulatory Guide 1.33 be written, maintained and adhered to for plant operations. Regulatory Guide 1.33 includes procedures for the processing of liquid and solid radwaste, for which the PCP is the core document."

EEI 50-410/97-07-06

"The inspectors identified that within this document, references to 49 CFR were out of date, following the revisions to 49 CFR in April 1996. Failure to maintain the PCP, is an apparent violation of Technical Specification 6.13."

EEI 50-220&50-410/97-07-06

- NMPC agrees that the cited references had not been updated
- NMPC believes that the PCP has appropriate specificity
- The License Document Owner (LDO) is responsible to maintain the PCP in response to changes in regulation and processes. There is no requirement to perform a periodic review of the PCP.

Root Cause

- Although implementing procedures were appropriately updated and content of the PCPs were not required to be changed, the LDOs failed to review references in the PCPs due to their inattention to detail

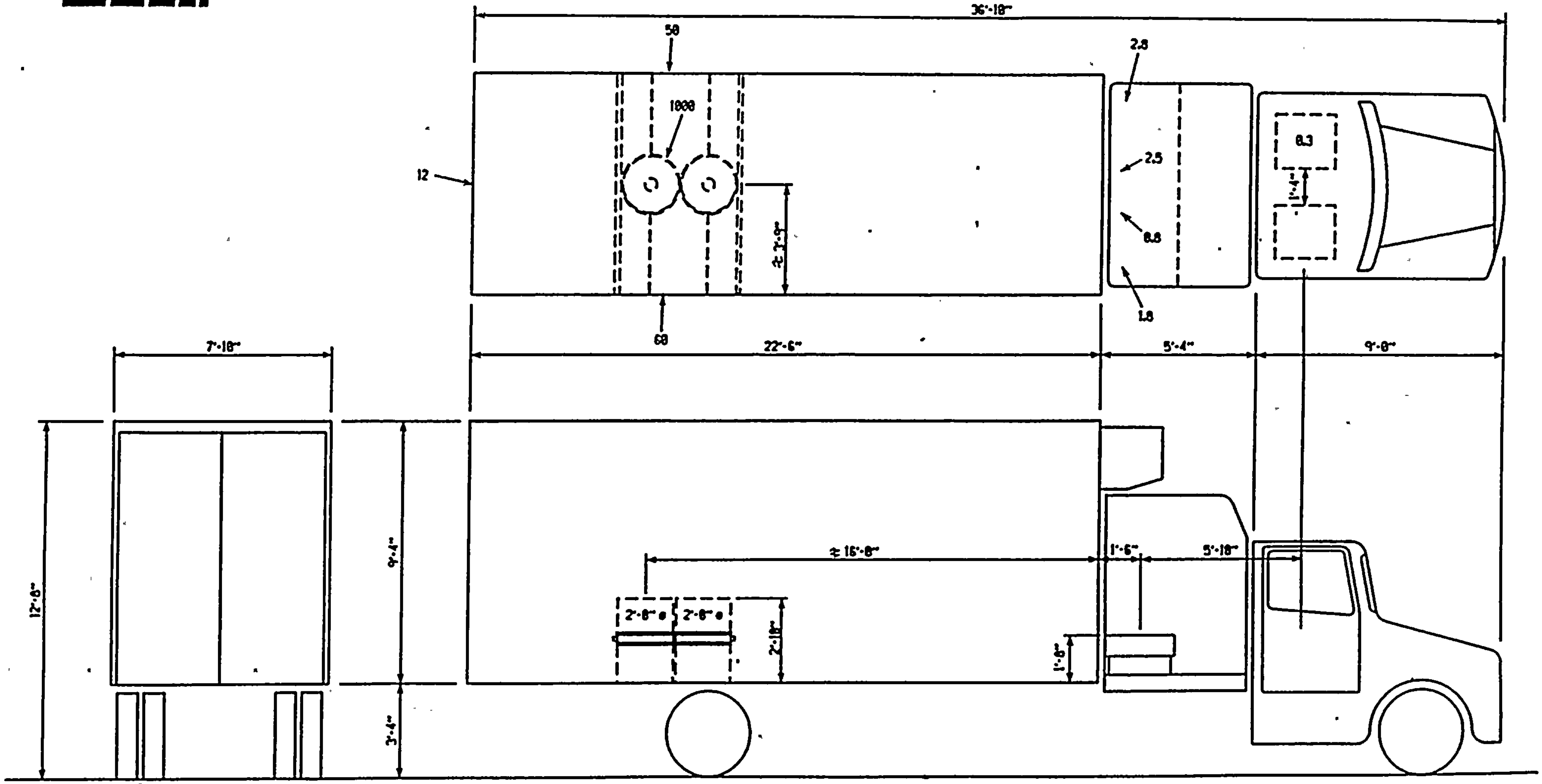
Corrective Actions

- PCPs reviewed and are being revised to update references
- Radwaste Supervisors were counseled with regard to their LDO accountability and ownership
- 49 CFR changes will now be implemented via the DER process

50-220/97-07-07

"On July 24, 1997, the licensee shipped two metal samples from the core shroud to BWX Technologies, Inc. Upon receipt, it was noted by BWX that radiation levels in an occupied portion of the vehicle were 2.8 milliRem per hour (mRem/hr), in excess of the regulatory limit of 2 mRem/hr, as specified in 49 CFR 173.441; this is an apparent violation."

TRUCK SURVEY AT BWX
(mr/hr)



EEI 50-220/97-07-07

- NMPC agrees with this violation

Root Cause

- Inadequate radiation survey prior to leaving Unit 1

Contributing Causes

- Inadequate technical review
- Non-conservative decisionmaking

Corrective Actions

- Radiation Protection Manager review
- Tailboard meetings with technicians
- Radiation Protection continuing training
- Personnel counseling and disciplinary action
- Radiation survey procedures are being revised.

EEI 50-220/97-07-09

"10 CFR 30.41 requires a licensee transferring radioactive material to verify that the transferee's license authorizes the receipt of the type, form, and quantity of byproduct material to be transferred. Contrary to the above, on two occasions, NMPC shipped radioactive material to an unlicensed facility. This is an apparent violation of NRC regulations."

"The inspectors considered the corrective actions for the 1995 event to be ineffective in preventing recurrence."

EEI 50-220/97-07-09

- NMPC agrees with this violation with the exception that we believe the 1995 corrective actions were appropriate.

<u>Root Cause</u>

- Cognitive error in that the responsible individual inadequately performed the required documentation review

<u>Corrective Actions</u>

- Radioactive material shipments are no longer made by warehouse personnel - now responsibility of Radwaste Operations Supervisor
- Responsible individual disciplined

EEI 50-220&50-410/97-07-10

"10 CFR 20, Appendix G, 'Requirements for Transfers of Low-Level radioactive Waste Intended for Disposal at Licensed Land Disposal Facilities and Manifests,' Section I.B, requires the shipper [NMPC] of radioactive waste to provide information regarding the shipment on the manifest, including the total radionuclide activity in the shipment. 10 CFR 71, 'Packaging and Transportation of Radioactive Material,' Paragraph 71.5, requires each licensee [NMPC] who delivers licensed material to a carrier for transport to comply with the requirements of 49 CFR 172. 49 CFR 172, Paragraph 172.203, requires for shipments of Class 7 (radioactive) material to include the activity contained in each package of the shipment. Contrary to the above, the shipping manifest did not accurately reflect the actual radionuclide activity of the shipment; this is an apparent violation of 10 CFR 20 and 49 CFR 172."

EEI 50-220&50-410/97-07-10

(continued)

"In addition, the inspectors determined that there were no procedures directly related to the loading of liners or the shipment of radwaste material."

EEI 50-220/97-07-10

- NMPC agrees with this violation

Root Cause

- Responsible individual failed to perform required verification

Contributing Cause

- Documentation related to liner location was in error (procedural adequacy was not a contributor)

Corrective Actions

- Responsible individual who signed manifest has been terminated
- Other personnel counseled and disciplinary action taken
- A verification of liner locations has been completed

EEI 50-220&50-410 97-07-11

"NMP1 TS 6.5.3.8 and NMP2 TS 6.5.3.8 require that an audit be conducted every 24 months of the respective unit Process Control Program (PCP). The inspectors reviewed the licensee's audit of the PCP (Audit 96002, dated December 19, 1996). This audit failed to identify out-of-date references to federal regulations contained in both the NMP1 and NMP2 PCPs. Additionally, while the audit did identify out-of-date references to training procedures in the PCPs, the DER issued to identify this finding was closed without the PCPs being revised to correct the defect. The inspectors noted that 10 CFR 50, Appendix B, Criterion XVI, 'Corrective Action,' requires that measures be established to assure that conditions adverse to quality are promptly identified and corrected. The inspectors noted that the failure to identify conditions adverse to quality, and the failure to ensure that such conditions are corrected, are apparent violations of 10 CFR, Appendix B, Criterion XVI."

EEI 50-220&50-410/97-07-11

- NMPC agrees that the DER from Audit 96-002 was closed without the PCPs being revised to correct the errors
- NMPC believes that the audit scope was appropriate
- Audit followup would have identified that the DER was inappropriately closed

Root Cause

- Inadequate communication between the responsible Branch Manager and the person who dispositioned the DER

Corrective Action

- PCPs are being revised to correct references (minor changes)
- The responsible Branch Manager has been counseled

EEI 50-220&50-410/97-07-12

"Regarding the use of vendor shipping casks, the inspectors noted that 10 CFR 71.12 allows the NRC to issue a general license to deliver for transport radioactive material in a package for which the NRC has issued a certificate of compliance. The general license requires that the licensee have in-place a QA program, approved by the NRC, that satisfies the provisions of 10 CFR 71, Subpart H (Quality Assurance). 10 CFR 71, Subpart H, requires, in part, that a comprehensive system of planned and periodic audits be conducted to verify compliance with all aspects of the QA program. The licensee transported radioactive material in a shipping container owned by SEG, Inc., for which the NRC has issued a certificate of compliance; however, the licensee had not conducted periodic audits of SEG to verify compliance with all aspects of the vendors NRC-approved quality assurance program. The inspectors noted that the failure to conduct audits of suppliers of NRC-certified shipping casks is an apparent violation of 10 CFR 71.12."

EEI 50-220&50-410/97-07-12

- NMPC disagrees with the violation
- The requirements apply to the shipper who is separately licensed and must have a 10 CFR 71 Subpart H QA program governing its activities, including audits

EEI 50-220&50-410/97-07-13

"Regarding the use of vendor provided radwaste processing systems, the inspectors noted that NMP1 TS 6.8 requires that procedures and administrative policies for activities listed in Appendix 'A' of NRC Regulatory Guide (RG) 1.33 be established, implemented and maintained. RG 1.33, Appendix 'A' lists procedures for the processing of radioactive waste. The NMP1 Radwaste PCP, paragraph 3.1.1, requires, in part, that radioactive waste may be processed using approved vendor equipment and procedures provided that the vendors have a QA program that meets NRC requirements. The inspectors noted that the failure to verify the QA program of radwaste processing vendors is an apparent violation of TS 6.8."

EEI 50-220/97-07-13

- NMPC disagrees with this violation

  "3.1.1 Waste Processing System

  The General Supervisor Radwaste shall ensure:

  a. Radioactive waste is processed using approved equipment with approved procedures.

  b. Radioactive waste may be processed using approved vendor equipment and procedures. Vendors must have QA programs that meet NRC requirements."

- Above requirements only apply to onsite processes. Vendors providing onsite waste processing services provide certification of QA program and are under the auspices of NMPC QA program and license
- In contrast, vendors providing offsite waste processing services are required by the NMPC contracting process to have the appropriate license and provide a certification of compliance with its QA program

MANAGEMENT PERSPECTIVE

- Events described of varying significance
- As a whole, NMPC believes that its waste processing and shipping programs are sound and have had a good performance record
- However, we recognize that the cited incidents have indicated that improvement is required
- These are human performance issues
- Radwaste Operations has not been a subject of self-assessment by responsible Branch Manager - prior results have been good

CORRECTIVE ACTIONS

- Job Performance corrective actions were conducted with the individuals responsible
- Senior Management reinforcement of ownership, accountability, and attention to detail with individuals and management/ supervision
- Responsible Branch Managers will include Radwaste Operations in future Branch self-assessment activities