U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket/Report Nos.:	50-220/98-06 50-410/98-06
License Nos.:	DPR-63 NPF-69
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:	May 24 - July 4, 1998
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EXECUTIVE SUMMARY

Nine Mile Point Units 1 and 2
50-220/98-06 & 50-410/98-06
May 24 - July 4, 1998

This NRC inspection report includes reviews of licensee activities in the functional areas of operations, engineering, maintenance, and plant support. The report covers a six-week period of inspections and reviews by the resident staff and regional specialists in the areas of effluents monitoring and Unit 2 inservice inspection.

PLANT OPERATIONS

During sustained Unit 1 control room observations, operators' attentiveness, procedure adherence, shift turnovers, log keeping, and control of activities were found to be acceptable. Supervisory oversight and communication were good, particularly during a control rod drive pump post-maintenance test and a feedwater pump swap. In-plant operators were knowledgeable of system and equipment functions. Material condition in the reactor building was acceptable.

Licensee response to the May 11, 1998 engineered safety feature actuation was appropriate. The cause of the event was poor work package and tagout development and a subsequent poor plant impact assessment by the Station Shift Supervisor prior to re-energizing the Division II trip unit power supplies.

During the Unit 1 planned shutdown on April 28, the licensee determined that the rod block function of the rod worth minimizer had not been properly tested since a 1974 Technical Specification change. This licensee identified and corrected violation of TS surveillance requirements was not cited.

MAINTENANCE/SURVEILLANCE

The second ten-year inservice inspection plan for Unit 2 was updated to reflect industry operating experience. The bases for selected relief requests were valid and accurate. Core shroud inspections were conducted in accordance with industry guidelines. NDE personnel were trained in accordance with the industry standards.

The Unit 2 post-refueling hydrostatic tes procedure was well written, and provided good instructions for control of activities. The inspections performed by NMPC during the test were comprehensive, and the licensee made the required repairs to reduce the total leakage to within specified acceptance criteria. The licensee took the necessary actions to request and obtain NRC approval for relief from the ASME Code requirements for noted leakage.

Executive Summary (cont'd)

ENGINEERING

The Unit 1 design deficiency involving the control room emergency ventilation system and interfacing auxiliary control room fire dampers (reference LER 98-12) was properly identified by the licensee and promptly corrected. Accordingly, this violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," was not cited.

Unit 1 engineering staff identified that since 1990, the reactor vessel level instrumentation could have been indicating as much as 6.5 inches higher than actual. This resulted in the low reactor water level trip settings being non-conservative and outside the allowable values provided in the TS. This licensee identified and corrected violation was not cited.

During the review of Unit 2 safety system logic testing per Generic Letter 96-01, NMPC identified that a number of logic circuits were not being tested as required by TS. Specifically, these circuits were not being properly test with the alternate offsite supply breaker supplying the divisional bus. Prompt and appropriate actions were taken to demonstrate logic system operability. This licensee identified and corrected surveillance testing deficiency was not cited.

During their Generic Letter 96-01 review of safety-system logic testing, NMPC identified that portions of the Unit 2 service water pump loss of offsite power (LOOP) automatic start sequencing and the LOOP/loss of coolant accident manual start interlock logic circuit were not being tested as required by TS. Prompt and appropriate actions were taken to demonstrate logic system operability. This licensee identified and corrected surveillance testing deficiency was not cited.

PLANT SUPPORT

The licensee established, implemented, and maintained an effective radiation monitoring system program with respect to electronic calibrations, radiological calibrations, system reliability, and tracking and trending.

The licensee established, implemented, and maintained an effective ventilation system surveillance program.

TABLE OF CONTENTS

Table of Contents (cont'd)

REPORT DETAILS

Nine Mile Point Units 1 and 2
50-220/98-05 & 50-410/98-05
May 24 - July 4, 1998

SUMMARY OF ACTIVITIES

Niagara Mohawk Power Corporation (NMPC) Activities

Unit 1

Nine Mile Point Unit 1 (Unit 1) started the inspection period shutdown for modifications to the control room emergency ventilation system (CREVS). These modifications were required to address design concerns (see inspection report (IR) 98-05, Section E8.5). On May 25 NMPC commenced a start-up of Unit 1, and full power was achieved on June 1. The unit remained at full power through the end of the inspection period.

Unit 2

Nine Mile Point Unit 2 (Unit 2) started the inspection period shutdown for its sixth refueling outage (RFO). Major work activities completed during the outage included, drywell flex hose removal/replacements, power range neutron monitoring system modification, emergency core cooling system (ECCS) suction strainer replacements, and core shroud inspections. On July 2, NMPC commenced a start-up of Unit 2. Power ascension and testing were in progress through the end 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. In addition, specialists from Region I conducted inspections in the areas of effluent monitoring and Unit 2 inservice inspection. The results of these inspection activities are integrated into 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). The inspectors verified that the UFSAR descriptions were consistent with the observed plant practices, procedures, and/or parameters. Exceptions noted were:

- missing fire protection material in the Unit 1 auxiliary control room (see Section E8.6); and,

- the Unit 1 CREVS being outside the design basis due the fire dampers closing following a loss of offsite power (see Section E8.7).

I. OPERATIONS

O1 Conduct of Operations

O1.1 General Comments (71707)[1]

Using NRC Inspection Procedure 71707, the resident inspectors conducted frequent reviews of ongoing plant operations. The reviews included tours of 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. Specialist inspectors in this area used other procedures during their reviews of operations activities; these inspection procedures are listed, as applicable, for the respective sections of the inspection report. In general, the conduct of operations was professional and safety-conscious; specific events and noteworthy observations are detailed in the sections below.

O1.2 Sustained Observation of Unit 1 Operations Activities

a. Inspection Scope (71715)

The inspectors observed the conduct of activities in the Unit 1 control room and plant, during day and evening shifts, to evaluate operator attentiveness, procedure adherence, supervisory oversight, shift turnovers, communication, log keeping, and control of activities.

b. Observations and Findings

The inspectors determined that operators' attentiveness, communications, and supervisory oversight were good, particularly during the performance of the post-maintenance test (PMT) of a control rod drive pump, and the swapping of feedwater pumps. The inspectors reviewed the operator and supervisor's logs and found them to be acceptable. TS entry and exit, major equipment status changes, and surveillance tests were properly recorded. Additionally, operators' response to overhead annunciators was prompt, and distractions in the control room were generally kept to a minimum.

The inspectors accompanied plant operators from three different crews on routine rounds of the reactor building to assess their knowledge of plant systems, ability to identify plant deficiencies, adherence to radiation and security requirements, and to assess housekeeping. The operators were knowledgeable of plant systems and equipment functions. Additionally, the in-plant operators notified the control room

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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 that was used as inspection guidance is listed for each applicable report section.

operators when deficiencies were identified, or when log readings were not as expected. The inspectors noted that radiological protection and security requirements were adhered to, and housekeeping issues were corrected when found. The inspectors considered the material condition of the reactor building to be acceptable.

c. Conclusion

During sustained Unit 1 control room observations, operators' attentiveness, procedure adherence, shift turnovers, log keeping, and control of activities were found to be acceptable. Supervisory oversight and communication were good, particularly during a control rod drive pump post-maintenance test and a feedwater pump swap. In-plant operators were knowledgeable of system and equipment functions. Material condition in the reactor building was acceptable.

O2 Operational Status of Facilities and Equipment

O2.1 Tours of Unit 2 During Refueling Outage (71707)

The inspectors conducted routine tours of the Unit 2 reactor and turbine buildings during RFO 6, focusing on areas that were normally inaccessible during power operation. Overall, the inspectors noted that equipment material condition was generally good. Housekeeping during the refueling outage was generally good, with the amount of debris generated and work-related equipment consistent with the ongoing level of maintenance and modification activity.

O8 Miscellaneous Operations Issues

O8.1 (Closed) LER 50-220/98-08: Rod Worth Minimizer TS Surveillance Requirement Not Met for Previous Shutdowns (71707, 90712, 92700)

On April 28, 1998, during a plant shutdown, the Unit 1 Reactor Engineering Supervisor identified that historically the control rod withdrawal sequence [rod-block] portion of the TS Surveillance Requirement (TSSR) for the rod worth minimizer (RWM) had not been performed. Specifically, TSSR 4.1.1.b(3)(a) lists four separate steps to verify the operability of the RWM; the fourth step checks the rod-block function of the RWM, verifying that an out-of-sequence control rod cannot be withdrawn beyond the rod-block setpoint. TSSR 4.1.1.(3)(b) requires the RWM be operable when the reactor is in the startup mode of operation or the run mode of operation below 20% power. Following this determination, the surveillance procedure was revised to test the rod-block function and it was performed satisfactorily prior to power being reduced below 20%.

The NMPC Surveillance Procedure N1-ST-V3, "Rod Worth Minimizer Operability Test & APRM/IRM [Average Power Range Monitor/Intermediate Range Monitor] Overlap Verification," which has been used to verify the operability of the RWM, stated that the check of the rod-block function was not required for a reactor shutdown. Further review by NMPC revealed that this surveillance had not been

performed for reactor shutdowns since 1974, when the surveillance requirement was added to the Unit 1 TS, but incorrectly interpreted by the NMPC staff. The failure to perform a surveillance of the rod-block function of the RWM is a violation of TSSR 4.1.1(3)(a)(iv). However, this non-repetitive, licensee identified, and corrected violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1 of the NRC Enforcement Policy. **(NCV 50-220/98-06-01)**

The inspectors monitored the performance of the surveillance test, reviewed the associated DER, and discussed the issue with the Reactor Engineering Supervisor. In addition, the inspectors conducted an in-office review of the LER and verified that it was completed in accordance with the requirements of 10CFR50.73. This LER is closed.

O8.2 (Closed) LER 50-410/98-13: Engineered Safety Feature Actuation Due to Personnel Error

a. Inspection Scope (71707, 92700)

The inspectors assessed the licensee response to the event, the root cause determination, and corrective actions. The assessment included a review of the applicable DER and LER, licensee procedures, work control documents, plant drawings and SSS's logs. The inspectors discussed the issue with the members of the Unit 2 system engineering and operations staffs. Also, the inspectors verified the completion of the LER in accordance with 10CFR50.73.

b. Observations and Findings

On May 11, 1998, while in refueling outage 6, Unit 2 experienced an inadvertent start of the Division II emergency diesel generator (EDG), as well as, realignment of residual heat removal (RHR) systems to their low pressure coolant injection (LPCI) mode. These engineered safety feature (ESF) initiations occurred upon reclosing the power supply breakers to a number of Division II trip units. At the time of the event, RHR pumps "B" and "C" were in pull-to-lock to prevent injection into a flooded reactor cavity. After the licensee determined the cause of the ESF actuation, the operators returned the EDG and RHR systems to standby. The inspectors discussed the event with the SSS and reviewed the applicable portions of the SSS's log and considered the licensee's actions taken in response to the inadvertent ESF actuation to have been appropriate.

The inspectors independently reviewed the work order associated with the RHR "B" heat exchanger level control valve, the applicable plant drawings and procedures, and agreed with the licensee's root cause determination that the ESF actuation was the result of poor work package and tagout preparation and a subsequently poor SSS plant impact assessment prior to re-energizing the Division II trip unit power supply.

The inspectors verified that the LER was completed in accordance with the requirements of 10CFR 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 response to the May 11, 1998 engineered safety feature actuation was appropriate. The cause of the event was poor work package and tagout development and a subsequent poor plant impact assessment by the Station Shift Supervisor prior to re-energizing the Division II trip unit power supplies.

O8.3 (Closed) URI 50-410/96-01-02: Overpressurization of the Unit 2 Reactor Water Cleanup System during Restoration (92901)

In January 1996, during the restoration of the Unit 2 reactor water cleanup (RWCU) system, a segment of piping briefly exceeded design pressure. The details of the event were reviewed and documented in IR 50-410/96-01. However, the item was unresolved pending the NRC review of the root cause and corrective actions, and a review of similar events associated with the Unit 2 RWCU system.

The inspectors reviewed the disposition of the associated DER, including the root cause and corrective actions and concluded they were appropriate. In addition, the inspectors discussed the DER with the Unit 2 Operations Manager. The DER also included a history of other RWCU system overpressurizations. The inspectors noted that the circumstances associated with the earlier events varied greatly from the 1996 event. The inspectors concluded that the January 1996 event did not result in a violation of regulatory requirements. Accordingly, this unresolved item is closed.

O8.4 (Closed) VIO 50-410/96-07-02: Inadequate Procedures for the Unit 2 Emergency Diesel Generator Duplex Strainers (92901)

During the NRC Integrated Performance Assessment Process (IPAP) Inspection, the team identified discrepancies with Procedure N2-OP-100A, "Standby Diesel Generators," Revision 5, associated with the alignment of the turbo lube oil duplex filter and the fuel oil duplex strainer. These discrepancies were considered violations of TS 6.8.1 regarding procedure adequacy. NMPC's letter dated November 15, 1996, provided the root cause and corrective actions for this violation. The inspectors reviewed this letter and related DERs, and concluded that the root cause and corrective actions were appropriate. The inspectors verified that Procedure N2-OP-100A and the associated alarm response procedure were revised, as necessary. Based on portions of the NMPC Procedure Writer's Guide reviewed and discussions with three procedure writers regarding the quality of the guidance, the inspectors considered the guidance appropriate to prevent recurrence. In addition, the inspectors discussed general procedure quality with four Unit 2 licensed operators, who indicated that the quality of procedures had improved.

Furthermore, inspectors have observed that Unit 2 operators routinely provide feedback to their management on procedural enhancements. This violation is closed.

II. MAINTENANCE [2]

M1 Conduct of Maintenance

M1.1 General Comments (61726, 62707)

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 10CFR50.65. Specialist inspectors used other procedures during their reviews of maintenance and surveillance activities; these inspection procedures are listed, as applicable, for the respective sections of the inspection report. 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-ST-Q19	Control Room HVAC [Heating, Ventilation and Air Conditioning] System Test
•	N1-IPM-Z09-006	Seismic Monitor Test
•	N1-ST-Q2	Control Rod Drive Pump 11 Post Maintenance Test
•	N1-ST-Q6B	Containment Spray System Loop 121 Test
•	N1-ST-Q1C	Core Spray Pump 112 Test
•	N1-ST-Q3	HPCI [High Pressure Coolant Injection] Pump and Check Valve Test
•	N1-ISP-201-476	High Drywell Pressure Instrument Trip Channel Test/Calibration
•	N1-ST-M9	CREVS [Control Room Emergency Ventilation System] Train Operability Test
•	N2-OSP-EGS-10Y@001	Simultaneous Start of Emergency Diesel Generators
•	N2-OSP-EGS-R003	Diesel Generator Loss of Offsite Power with no ECCS - Division I and II
•	N2-97-044	RCS [Reactor Coolant System] Vent and Drain Valve Installation and Flexible Metal Hose Replacement

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[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."

•	N1-ST-M4	EDGs / PB [Power Board] 102 and 103 Operability Test
•	N2-OSP-ICS-R002	RCIC [Reactor Core Isolation Cooling] System Flow Test
•	N2-OSP-SWP-R001	Service Water Actuation Test

M3 Maintenance Procedures and Documentation

M3.1 Unit 2 - Inservice Inspection

a. Inspection Scope (73753)

The inspectors reviewed the Unit 2 inservice inspection (ISI) activities that were part of RFO 6. The review encompassed observations of non-destructive examination (NDE) activities, and a review of the associated NDE procedures, and qualifications of the NDE personnel. Several relief requests were also examined to verify that the bases for the relief request was valid and accurate.

b. Observations and Findings

Unit 2 was in the first period of the second 10-year ISI plan interval, and had changed to the 1989 edition of Section XI of the American Society of Mechanical Engineers (ASME) code. The first 10-year inspection interval ended in April 1998. Despite having already entered the new inspection interval, Niagara Mohawk had not yet submitted the new inspection plan to the NRC for review. Niagara Mohawk attributed the delay to a loss of personnel, most notably the ISI Program Manager for Unit 2. The plan was eventually submitted on June 16, 1998.

The inspectors reviewed the ISI plan and observed that it reflected industry operating experience. Portions of two relief requests, which requested relaxation of an ASME Code requirement to perform a 100% surface examination on certain valve body welds and pump casings in the low pressure core spray (LPCS) and residual heat removal (RHR) systems, were walked down in the field to verify that the basis for the relief requests were valid.

Core Shroud: While conducting ultrasonic test (UT) examinations of the inside and outside diameter of the core shroud, NMPC identified numerous cracks in seven of the eight shroud horizontal welds. No cracks were noted in vertical welds. NMPC believed intergranular stress corrosion cracking (IGSCC) caused the shroud cracks; the location of the cracks (along the weld and within the heat affected zone) supported their conclusion. The results of the horizontal examinations are listed below:

Weld Identifier	Total Weld Length (in inches)	Weld Length Examined (in inches)	Cracked Length (in inches)
H1	691.1	387.2	17.1
H2	691.1	387.2	11.7
H3	650.7	459.2	9.8
H4	650.7	344.3	245.7
H5	651.0	332.0	243.0
H6	651.0	330.0	0.0
H7	630.7	500.0	204.3
H8	630.7	503.8	40.8

This was the first time the Unit 2 shroud was examined using UT equipment. Previous examinations, most recently performed during the 1995 refuel outage, were conducted visually using underwater cameras. The inspectors observed part of the inspections and verified that personnel were following procedures and properly recording indications and/or cracks.

Once shroud cracks were discovered, the inspectors verified that NMPC increased the scope of the inspection to include 100% of the accessible vertical and horizontal welds, as required by the industry's Boiling Water Reactor Vessel Inspection Program (BWRVIP) guidelines. NMPC's preliminary analysis of the shroud concluded that the cracking was not significant. In accordance with BWRVIP guidelines, repairs were not needed to support plant operation over the next operating cycle. In addition, NMPC is required to submit their final report to the NRC for review.

<u>Feedwater Nozzle Indication:</u> NMPC identified a crack during examination of feedwater nozzle weld 2RPV-KB20 using automated UT equipment; the crack was 5.3 inch long and 0.29 inches deep. In accordance with NRC Generic Letter 88-01 "NRC Position on IGSCC in BWR [Boiling Water Reactor] Austenitic Stainless Steel Piping," NMPC increased the weld inspection sample size to include the remaining three feedwater nozzle welds. No other cracks were identified.

Because the crack was located in the feedwater nozzle-to-safe end weld, where a material transition between stainless steel and inconel materials occurs, NMPC believed the crack was fabrication related vice service induced degradation, such as IGSCC or thermal fatigue. However, because this could not be conclusively determined, the crack was attributed to IGSCC for structural evaluation purposes.

There was evidence of an indication in the weld in the UT examinations conducted in 1990 and 1995; however, at that time, NDE personnel misinterpreted the signal and believed that material transition, which existed at the weld area, had caused a false signal to occur. Therefore, NMPC NDE personnel did not consider the

indication to be a crack. Following a 1997 discovery of a crack that was incorrectly interpreted at another nuclear facility, NMPC improved training in this area and instilled an increased sensitivity to indications in nozzle areas.

The inspectors reviewed a sample of the NDE training records and verified the personnel met the experience requirements outlined in SNT-TC-1A, "Recommended Practice, Personnel Qualification and Certification in Non-destructive Testing." Personnel interviewed were knowledgeable of NDE industry initiatives and events, including the aforementioned nozzle crack.

At the close of the inspection, NMPC was conducting a structural evaluation of the crack as required by Section XI of the ASME Code. Because the crack was classified to be caused by IGSCC, and since it did not meet the TS criteria for continued operation without evaluation, NRC review and acceptance of the evaluation was required before plant startup. This was granted by the NRC staff on June 25, 1998.

Oversight of NDE Activities: To ensure General Electric correctly interpreted the results from the nozzle weld examinations, a representative from the Electric Power Research Institute (EPRI) independently reviewed a sample of UT examinations of nozzle welds. The inspectors considered use of the EPRI representative to be a good initiative.

c. Conclusions

The second ten-year inservice inspection plan for Unit 2 was updated to reflect industry operating experience. The bases for selected relief requests were valid and accurate. Core shroud inspections were conducted in accordance with industry guidelines. NDE personnel were trained in accordance with the industry standards.

M7 Quality Assurance in Maintenance

M7.1 Unit 2 - Post-Refueling Hydrostatic Test of ASME Class 1 Systems

a. Inspection Scope (61726)

The inspectors reviewed the licensee's hydrostatic test procedure, witnessed portions of the hydrostatic test, and evaluated the licensee's actions to resolve deficiencies identified during the test.

b. Observations and Findings

The Unit 2 TSs and ASME Code Section XI (Article IWA-5000) require a hydrostatic pressure leak test (hydrotest) of the reactor pressure vessel, and all Class 1 boundary components, whenever the reactor vessel head has been removed or multiple Class 1 pressure boundary components have been opened. In addition, the Unit 2 Inservice Pressure Testing Program Plan requires that all Class 2 piping, unisolable from the Class 1 piping, be included in the hydrotest. During the

refueling outage, multiple Class 1 systems were opened for maintenance and the reactor vessel head was removed for refueling.

The licensee used Procedure N2-OSP-RPV-@002, "Reactor Pressure Vessel and All Class 1 Systems Leakage Test," to conduct the leakage test. The inspectors reviewed the procedure with the principal test engineer responsible for the test. The procedure specified that the test was to be conducted at a system pressure of 1020-1035 pounds per square inch (psi) and a maximum temperature of 202 degrees Fahrenheit (°F). The inspector noted that the procedure incorporated a reference to NRC Information Notice 98-13, "Post Refueling Outage Reactor Pressure Vessel Leak Testing Before Criticality," and the specific TS restriction which prohibits control rod withdrawal while in Mode 4, with reactor coolant temperature greater than 200°F. The inspectors considered the procedure to be comprehensive and it contained appropriate precautions and limitations for TS requirements, system operating limitations, heatup and cooldown restrictions, etc. Also, the procedure listed explicit prerequisites for completing all necessary work prior to the test, provided for the necessary quality control hold points, and contained the necessary attachments for documenting measuring and test equipment (M&TE) used. The procedure also explicitly prohibited any work (e.g., repairs) on the Class 1 systems while at full test pressure.

On June 22, 1998, the principal test engineer conducted a special evolution pre-test technical briefing on the conduct of the hydrostatic test with all personnel involved in the hydrotest. Before increasing the reactor coolant system to full hydrotest pressure, the licensee conducted an inspection at 500 psi. Valves with packing leaks were identified, and components with bolted flanges that were disturbed during the outage were tightened (after the system was depressurized) to prevent leakage at the higher hydrotest pressure. On June 24, 1998, the licensee used a test pump connected to the reactor coolant system at the head spray valve (2RHS*V64) to increase system pressure. Reactor coolant system (RCS) temperature was increased using the recirculation pumps and maintained at ≈170°F. System pressure was held at the 600 psi and 900 psi levels for preliminary leakage checks on the way to full hydrotest pressure.

The full pressure hydrotest required a visual (VT-2) inspection for leakage by quality assurance (QA) personnel. After full hydrotest pressure was achieved, the inspectors accompanied three QA inspectors during the inspection of primary system piping inside the drywell. The QA inspections were thorough and 32 components were identified with leakage, mostly from valve packing (non-pressure boundary leakage). However, three safety relief valves (SRVs) were identified with significant leakage (≈200 drops per minute (dpm)) and one local power range monitor (LPRM) was leaking ≈6dpm, all at their bolted flange connections. Although the ASME Code does not specify the total system leakage allowable for a successful hydrotest, the licensee's established acceptance criteria is contained in Procedure, M2-002, "ASME Section XI, System Pressure Testing Acceptance Criteria." NMPC's evaluation determined that one valve had a packing leak (≈200dpm) which was not acceptable. The packing on this valve was subsequently repaired, and the licensee declared the hydrotest satisfactory.

The ASME Code does not permit any pressure boundary leakage through Class 1 components, and the flange leakage from the SRVs and LPRM was not acceptable for plant operation. The 1989 edition of the ASME Code, Article IWA-5250, requires that the corrective measures for leakage at bolted connections include removal of the bolts, and a VT-3 visual examination for corrosion. The licensee removed and inspected two of the bolts from the LPRM flange and found no evidence of corrosion; the same bolts were reinstalled and torqued, the remaining two flange bolts were not removed, but were retorqued. Because the same bolts were used, a re-hydrotest was not required. The licensee considered that the leakage would not be stopped until the plant achieved full operating temperature. The Unit 2 TS require NMPC to obtain NRC approval for relief from the ASME Code before a satisfactory test could be declared.

On June 26 and 28, 1998, the licensee made formal verbal requests to the NRC for relief from the IWA-5250 requirements for the flange leakage inspections. The NRC granted relief based on 10CFR50.55a(a)(3)(ii). No unacceptable ASME Code conditions existed after the safety relief valves were inspected at full operating pressure and temperature.

c. <u>Conclusions</u>

The Unit 2 post-refueling hydrostatic test procedure was well written, and provided good instructions for control of activities. The inspections performed by NMPC during the test were comprehensive, and the licensee made the required repairs to reduce the total leakage to within specified acceptance criteria. The licensee took the necessary actions to request and obtain NRC approval for relief from the ASME Code requirements for noted leakage.

M8 Miscellaneous Maintenance Issues

M8.1 <u>(Closed) LER 50-220/98-11: Inadvertent Actuation of the Reactor Protection System Circuitry Due to Personnel Error (61726, 90712)</u>

On May 3, 1998, during a forced maintenance outage, Unit 1 experienced an invalid isolation of the shutdown cooling (SDC) system when an electrician inadvertently grounded the associated reactor protection system (RPS) logic in a motor-operated valve (MOV) isolation circuit. The SDC system had already been removed from service to support the MOV work. When the isolation signal was received, the control room operators contacted the field crew and had the work stopped; subsequently, the isolation valves were re-opened and the SDC system was restored to service. There was no violation of NRC requirements.

The inspectors reviewed the associated DER, and conducted an in-office of the LER and verified that it was completed in accordance with the requirements of 10CFR50.73. This LER is closed.

III. ENGINEERING

E1 Conduct of Engineering

E1.1 General Comments (37551)

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

E1.2 Evaluation of Unit 2 Reactor Reload Analysis

a. Inspection Scope (37551)

The inspectors reviewed the Unit 2 reload analysis, as summarized in the Unit 2 Core Operating Limits Report (COLR). The COLR was based on several design documents provided by the vendor, General Electric Nuclear Energy (GENE), and the Unit 2 TS. The inspectors reviewed the basis documents, the COLR, and the associated safety evaluation.

b. Observations and Findings

As required by the Unit 2 TS, Section 6.9.1.9, NMPC established the Unit 2 COLR for the upcoming cycle. The COLR is the plant specific document that provides core operating limits for the current reload cycle. Those limits are: (1) average planar linear heat generation rate, (2) average power range monitor flow-biased thermal power scram setpoint, (3) linear heat generation rate, (4) minimum critical power ratio, (5) core flow adjustment factor, and (6) rod-block instrumentation setpoint. The inspectors verified that the information in the basis documents provided by GENE, listed below, was accurately translated into the COLR.

- Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2, Reload 6, Cycle 7, J11-03211SRLR, Revision 0, Class I, dated March 1998

- Lattice Dependent MAPLHRG Report for Nine Mile Point Nuclear Station Unit 2, Reload 6, Cycle 7, J11-03211MAP, Revision 0, Class III, dated March 1998

The inspectors reviewed the Safety Evaluation (SE 98-048, Revision 0) and the associated Applicability Review (AR 27858). The SE was consistent with the GENE reference documents, and adequately addressed the reload design and related changes in the safety and thermal limits.

c. Conclusion

The inspectors considered the Unit 2 reload analysis, as submitted in the Core Operating Limits Report, to be acceptable and met the requirements of the Unit 2 TS.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) VIO 50-220/96-07-03: Unit 1 Final Safety Analysis Report Changed Without Required Safety Evaluation (92903)

During the NRC IPAP team inspection, the team identified a violation of 10 CFR 50.59 involving a revision to Unit 1, UFSAR, Figure X-6, that had been completed without a safety evaluation. NMPC's letter dated November 15, 1996, provided the root cause and corrective actions for this violation. The inspectors reviewed this violation response and concluded that the root cause and corrective actions provided were appropriate. The inspectors reviewed the implementing guidelines for the revised 10CFR50.59 safety evaluation and applicability review process and considered the changes to be appropriate to prevent recurrence. This violation is closed.

E8.2 (Closed) URI 50-220 & 50-410/97-01-02: Disparity Between the Results of an NRC Inspection and an NMPC Audit of the C&D Battery Vendor (92903)

In December 1996, the NRC inspected C&D Charter Power Systems, Inc. (C&D); C&D supplied lead-acid storage batteries to NMPC for safety-related applications. The NRC inspection discovered problems in C&D's implementation of its quality assurance (QA) program related to the dedication of battery cells after manufacture. Specifically, the vendor did not have an adequate dedication program or a basis for dedication for their products. In addition, the NRC noted that NMPC had last audited C&D in August of 1994. However, that audit did not identify the commercial grade dedication program implementation problems that the NRC found. The disparity between the NRC's and the licensee's findings suggested a possible problem with the effectiveness of NMPC's audit process, as implemented at the C&D facility. The significance of the disparity between the NMPC audit and the NRC inspection results was an unresolved item pending NRC assessment of the licensee review of the issue, necessary corrective actions, and a review of the extent of condition of any problems found. NMPC initiated DER 1-97-326 and 2-97-370 for Units 1 and 2, respectively.

The inspectors reviewed the DER disposition and determined that NMPC's explanation was reasonable. Specifically, the audit and inspection scopes were different. The NRC's inspections included all of C&D's facilities and a review of the C&D program for dedication of component parts which comprise battery cells. NMPC's audit scope was to assess the overall quality assurance of the manufacturing process, focusing on the final testing and acceptability of operation. The NMPC audit also included the results of prior audits conducted by other nuclear utilities as part of a joint utility program. No violations of regulatory requirements were identified. This unresolved item is closed.

E8.3 (Closed) URI 50-220 & 50-410/97-01-03: Operability of C&D Batteries Installed at Nine Mile (92903)

During the inspectors' review of the C&D battery issues (see Section E8.2 above), the NRC questioned the operability of the C&D batteries installed for Class 1E service. NMPC initiated DER 1-97-326 and 2-97-370 for Units 1 and 2, respectively. NMPC performed an Engineering Operability Supporting Analyses and determined that all of the C&D batteries were operable, based on the results of pre-installation battery capacity tests. These tests were performed in accordance with Institute of Electrical and Electronics Engineers (IEEE) Standard 450, "Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations." In addition, service tests were conducted post-installation.

The inspectors reviewed the DER dispositions and considered the conclusions reasonable and acceptable. No violations of regulatory requirements were identified. This unresolved item is closed.

E8.4 (Closed) URI 50-220 & 50-410/97-01-04: Ability of NMPC to Identify the Location and Use of Purchased Equipment (92903)

During the inspectors' initial review of the C&D battery issues (see Sections E8.2 and E8.3 above), NMPC stated that the only C&D batteries in use at Nine Mile were at Unit 1. Subsequently, the QA Manager informed the inspectors that at least one additional use of C&D batteries was at Unit 2, for the HPCS system. This caused the inspectors to challenge NMPC's ability to identify the location and use of purchased equipment. The QA department initiated DER C-97-0568.

The DER disposition revealed that a procurement engineer performed a cursory search for C&D batteries, using the Master Equipment List (MEL); this search only identified the batteries in use at Unit 1. NMPC investigation identified that the manufacturer and model fields in the MEL database were not filled in for all equipment, and it was not procedurally required to be completed. The cause for the apparent oversight was determined to be a lack of understanding on the part of the procurement engineer. The engineer searched only one database, but a better understanding of the MEL and further review would have identified C&D batteries in other databases, for example, the work control program (WCMOSSE).

The inspectors determined that NMPC's actions were acceptable, and that no violations of requirements existed. This unresolved item is closed.

E8.5 (Closed) VIO 50-410/97-12-05: Failure to Perform TSSR of Rod Sequence Control System (92903)

In November 1997, during a Unit 2 reactor start-up, a reactor operator (RO) recognized that the surveillance test to verify operability of the rod-block function of the rod sequence control system (RSCS) was inadequate. Specifically, the rod worth minimizer (RWM) also generated a rod-block that could mask the RSCS rod

block function being tested. TSSR 4.1.4.2.b requires that the RSCS be demonstrated operable by verifying that an inhibited control rod cannot be moved. NMPC determined that the root cause for this testing oversight was inadequate change management when the old RWM was replaced with a new design. NMPC also identified that the system engineer associated with the RWM modification recommended that procedure steps be added to bypass the RWM, but these steps were not incorporated. NMPC was unable to definitively conclude why the system engineer's recommendation was not incorporated. The inspectors reviewed the violation response and the revised surveillance test procedure and concluded they were acceptable. This violation is closed.

E8.6 (Closed) LER 50-220/98-09: Missing Fire Protection Material from Structural Steel (37551, 71750, 90712)

On April 28, 1998, during a structural engineering inspection, NMPC personnel discovered that fire proofing material was missing from a section of structural steel in the Unit 1 auxiliary control room (i.e., relay room). NMPC concluded that the missing fire proofing reduced the load carrying capability of the beam, if subjected to a fire, and as a result could have adversely affected the safe shutdown capability from the control room. NMPC determined that the material had been removed in 1986 during the installation of a modification to install a cable tray, and that the contractors had failed to re-install the fire proofing material. This condition was mitigated by the fire detection system and automatic and manual fire suppression systems, which could provide early detection and fire suppression to minimize the affects of a fire in the vicinity of the beam.

The inspectors' review of the LER and the associated DER concluded that the analysis of the event, the immediate compensatory actions and corrective actions (which included a verification that no similar conditions existed within the control room envelope), were acceptable. Notwithstanding, the failure to have adequate fire proofing installed is a condition contrary to 10 CFR 50, Appendix R, Safe Shutdown Analysis. However, this condition constitutes a violation of minor safety significance and is not subject to formal enforcement action. This LER is closed.

E8.7 (Closed) LER 50-220/98-12: Control Room Emergency Ventilation System Outside Design Basis Due to Fire Damper Closure Following a LOOP (92700)

As a result of deficiencies associated with the Unit 1 control room emergency ventilation system (CREVS) (see NRC IR 98-02, Section E7.1), NMPC contracted for an independent design review of the CREVS. That review determined that the fire dampers in the supply and return ducts to the auxiliary control room (i.e., relay room) would fail closed on a loss of offsite power. The auxiliary control room is part of the control room envelope, and the dampers are required to be open, if the CREVS is in service. NMPC determined the root cause to be an inadequate evaluation of the relationship between the dampers and the CREVS when the dampers were installed in 1981. The failure to perform an adequate review in 1981 is a violation of 10CFR50, Appendix B, Criterion III, "Design Control." However, 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 50-220/98-06-02) This LER is closed.

E8.8 (Closed) LER 50-220/98-10: Inaccurate Reactor Vessel Level Instruments Due to an Inaccurate Input Parameter

a. Inspection Scope (37551, 92700)

Unit 1 engineers determined that certain reactor vessel level instrumentation could have been indicating as much as 6.5 inches higher than actual. This resulted in the low reactor water level trips being non-conservative and outside the allowable values provided in the TS. The inspectors assessed the licensee's root cause analysis and corrective actions; including a review of the associated DERs, LER, engineering analysis, and plant modifications, applicable plant drawings, procedures, TS and UFSAR sections. The inspectors also discussed the issue with members of the Unit 1 staff, and performed visual inspections of the instrument reference legs. In addition, the inspectors verified the completion of the LER in accordance with 10CFR50.73.

b. Observations and Findings

On April 30, 1998, while Unit 1 was in cold shutdown, NMPC determined that certain reactor vessel level instrumentation could have been indicating as much as 6.5 inches higher than actual. These instruments provide the following functions:

- reactor water level high turbine trip
- reactor water low level scram
- reactor water level low-low level emergency core cooling system (ECCS) initiation
- reactor water level low-low level anticipated transient without scram/recirculation pump trip (ATWS/RPT) initiation
- reactor water level low-low level reactor coolant and primary containment isolation.

This higher-than-actual indicated water level resulted in the low reactor water level trips being non-conservative and outside the allowable values provided in the TS. NMPC evaluated the impact of the higher-than-actual reactor water level indications, and concluded that the consequences were negligible since the resulting delay in safety system actuation time would not have impacted the ability of the ECCS systems to perform their safety function. The inspectors reviewed this evaluation and considered it appropriate.

NMPC determined the root cause of the event to be inaccurate input parameters used during 1990 revisions to the reactor vessel level instrument calibration calculations. These instruments detect reactor vessel level by measuring pressure differential between two water columns: a variable leg (from the reactor) and a reference leg. Since the variable leg and the reference leg are at different temperatures, temperature compensation of the reference leg is required. Prior to

1990, the assumed reference leg temperature was per the value specified by the vendor. Following the 1990 calibration calculation revision, the reference leg average temperature was determined by averaging two installed thermocouples. In 1998, NMPC determined that this simple averaging of the readings was not appropriate because the thermocouples were all located on the lower half of the reference leg column. Therefore, the average temperature of the thermocouples was not representative of actual reference leg temperature conditions. The use of the lower average reference leg temperatures resulted in non-conservative trip settings from the reactor vessel level instrument outputs. The failure to adequately evaluate the 1990 change to the reactor vessel level instrument calibration calculation involving reference leg temperature assumptions is a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." However, 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 50-220/98-06-03)**

With respect to corrective actions, NMPC analyzed the instrument reference leg configuration, and determined a more accurate average temperature. This new average temperature was incorporated into the calibration calculation and the instruments were properly calibrated. Also, NMPC installed additional thermocouples on the top portion of the instrument reference leg column, and following plant restart, the reference leg temperatures were monitored to verify the calculation inputs.

The inspectors verified that the LER was completed in accordance with the requirements of 10CFR50.73. 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

Unit 1 engineering staff identified that since 1990, the reactor vessel level instrumentation could have been indicating as much as 6.5 inches higher than actual. This resulted in the low reactor water level trip settings being non-conservative and outside the allowable values provided in the TS. This licensee identified and corrected violation was not cited. **(NCV 50-220/98-06-03)**

E8.9 (Closed) LER 50-410/98-04 Supplement 1: Missed Technical Specification Required LSFT of Level 8 Trip of Main Turbine (92712)

The technical issues associated with this LER were previously documented in IR 50-410/98-02, Section E8.11. The inspectors completed an in-office review of the additional information provided in LER 50-410/98-04, Supplement 1, and found it acceptable. This LER is closed.

E8.10 (Closed) LER 50-410/98-10: Entry into TS 3.0.3 due to Incorrect Latching Mechanisms Installed in Emergency Switchgear (37551, 62707, 92700)

On April 22, 1998, during maintenance on the Unit 2 Division II emergency diesel generator (EDG), NMPC determined that incorrect latching mechanisms were installed on the EDG neutral breaker cubicle door. Subsequent investigation identified that eight of the seventeen breaker cubicle doors on the Division II emergency switchgear had the wrong size latches. NMPC determined that the incorrect latches could adversely affect the ability of the switchgear to properly function during a seismic event. The inspectors observed that appropriate TS actions were taken for each incorrectly sized latching device identified. This included entry into TS 3.0.3 when the SSS determined that the emergency core cooling system (ECCS) actuation instrumentation TS Section 3.3.3 limiting condition for operations (LCO) could not be met. TS 3.0.3 requires the initiation of a plant shutdown within one hour, however, repairs were made and the LCO was exited within twenty minutes.

NMPC determined the root cause to be personnel error during initial construction. The corrective actions included inspection of all emergency switchgear cubicle doors. One broken latch was identified on a Division I switchgear cubicle door and no discrepancies were found on Division III switchgear. NMPC's analysis of the event determined that the affect on the core damage frequency was relatively small. Nonetheless, due to the multiple ECCS components affected, Unit 2 was in a condition outside the design basis of the plant. The failure to ensure that the proper switchgear latches were installed is a violation of 10CFR50 Appendix B, Criterion III, "Design Control." However, the failure constitutes a violation of minor significance and is not subject to formal enforcement action. This LER is closed.

E8.11 (Closed) LER 50-410/98-11: Missed Technical Specification Surveillance Testing of Alternate Power Supply

a. Inspection Scope (37551, 92700).

The inspectors reviewed LER 50-410/98-11, associated DERs, and applicable TS. The inspectors discussed the issue with the members of the Unit 2 system engineering staff. Also, the inspectors verified the completion of the LER in accordance with 10CFR50.73.

b. Observations and Findings

During their Generic Letter (GL) 96-06, "Testing of Safety-Related Logic Circuits," review, NMPC determined that various surveillance test procedures for all three safety divisions failed to test the entire circuit when the alternate offsite supply breaker was powering the divisional bus. Specifically, the following TSSR had not been adequately completed for plant operations with the alternate offsite supply breaker supplying the divisional buses:

- TSSR.3.3.2. regarding logic system functional test (LSFT) for the Division I, II, and III response to an emergency core cooling system (ECCS) actuation.
- TSSR 4.8.1.1.2.e.11 regarding Division I, II, and III emergency diesel generator (EDG) response to an ECCS actuation signal, while operating the EDGs in the test mode connected to the bus.
- TSSR 4.8.1.1.2.e.10 regarding the Division I and II EDG capability to manually synchronize and transfer loads with offsite power upon restoration of offsite power.
- TSSR 4.8.1.1.2.e.4.b regarding the Division III load shed function during a loss of offsite power (LOOP).

Upon identification of the discrepancies, NMPC issued DERs to address the concerns. Subsequently, NMPC revised the applicable test procedures and successfully completed the required testing during RFO 6.

The inspectors reviewed the associated DERs, and discussed the issue with members of the Unit 2 system engineering staff. No additional concerns were identified. However, the failure to have properly tested a number of safety system actuation logic circuits was a violation of TS. 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 50-410/98-06-04)**

The inspectors verified that the LER was completed in accordance with the requirements of 10CFR50.73. 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

During the review of Unit 2 safety system logic testing per Generic Letter 96-01, NMPC identified that a number of logic circuits were not being tested as required by TS. Specifically, these circuits were not being properly test with the alternate offsite supply breaker supplying the divisional bus. Prompt and appropriate actions were taken to demonstrate logic system operability. This licensee identified and corrected surveillance testing deficiency was not cited. **(NCV 50-410/98-06-04)**

E8.12 (Closed) LER 50-410/98-12: Missed Technical Specification Logic System Functional Testing of Service Water Pump Circuitry

a. Inspection Scope (37551, 92700)

The inspectors reviewed LER 50-410/98-12, associated DERs, applicable TS, and surveillance tests. The inspectors discussed the issue with the members of the Unit 2 system engineering and operations support staffs. Also, the inspectors verified the completion of the LER in accordance with 10CFR50.73.

b. Observations and Findings

On May 7, 1998, during their Generic Letter (GL) 96-06, "Testing of Safety-Related Logic Circuits," review, NMPC determined that logic system functional test (LSFT) procedures for the Division I and II service water system (SWP) pumps failed to adequately test the entire circuit, as required by TS. Specifically, portions of the circuitry involving the SWP pump loss of offsite power (LOOP) automatic start sequencing and the LOOP/loss of coolant accident (LOCA) manual start interlock logic were not completely tested. TS SR 4.7.1.1.1.e.1 and 4.7.1.2.1.e.1 require that an LSFT of the SWP pump starting logic be performed for operating and shutdown conditions, respectively. Through discussions with the SSS and a review of the SSS's logs, the inspectors verified that the appropriate TS required actions were taken.

NMPC issued DERs to address the concerns, temporary changes to the surveillance procedures were implemented, and the remaining portions of the SWP pump starting circuitry were satisfactorily tested. The inspectors reviewed the associated DERs, the completed tests, and discussed the issue with members of the Unit 2 system engineering and operations support staffs with no additional concerns identified. The failure to have tested the entire SWP pump starting circuitry is a violation of TS. 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 50-410/98-06-05)**

The inspectors verified that the LER was completed in accordance with the requirements of 10CFR50.73. 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

During their Generic Letter 96-01 review of safety-system logic testing, NMPC identified that portions of the Unit 2 service water pump loss of offsite power (LOOP) automatic start sequencing and the LOOP/loss of coolant accident manual start interlock logic circuit were not being tested as required by TS. Prompt and appropriate actions were taken to demonstrate logic system operability. This licensee identified and corrected surveillance testing deficiency was not cited. **(NCV 50-410/98-06-05)**

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 appropriate 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

R2.1 Calibration of Effluent/Process/Area Radiation Monitors

a. Inspection Scope (84750)

The inspectors held discussions with system engineers, accompanied the system engineer during tours, and reviewed the most recent calibration results for the below listed effluent and process radiation monitoring systems (RMS) and associated flow meters to determine whether TS requirements and UFSAR commitments were properly implemented:

Unit 1: liquid radwaste effluent radiation monitor, service water effluent monitor, offgas radiation monitor, stack gaseous noble gas monitors (low and high ranges), and emergency condenser radiation monitors

Unit 2: Liquid Radwaste Effluent Radiation Monitor, Service Water Effluent Monitors, Cooling Tower Blowdown Effluent Monitor, Offgas Radiation Monitor, Stack Gaseous Noble Gas Monitors (Low and High Ranges), and Radwaste/Reactor Buildings Vent Monitor

b. Observations and Findings

At Unit 1, Instrumentation and Controls Department had the responsibility of performing electronic calibration and repair, while the Health Physics Department had the responsibility for radiological calibration. RMS reliability at Unit 1 was good. Calibration results were within established acceptance criteria. Linearity tests showed acceptable results. Tracking and trending efforts were good.

A dedicated Health Physics instrumentation group had responsibility for both electronic and radiological calibrations and for instrument repair at Unit 2. This has permitted the licensee to maintain a high level of oversight over the Unit 2 RMS system. Work orders pertaining to the RMS were typically initiated and completed within a single shift. RMS reliability was maintained at nearly 100% since the last inspection. Tracking and trending efforts were good. Calibration and linearity test results demonstrated system performance within established acceptance criteria.

c. Conclusion

The licensee established, implemented, and maintained an effective radiation monitoring system program with respect to electronic calibrations, radiological calibrations, system reliability, and tracking and trending.

R2.2 Surveillance Tests for Air Cleaning Systems

a. Inspection Scope (84750)

The inspection consisted of interviews with cognizant staff, tours, and reviews of high-efficiency particulate air (HEPA) filter and charcoal adsorption surveillance test results for the below listed systems.

Unit 1: Reactor Building Emergency Ventilation System (TS requirement), Turbine Building Ventilation System (UFSAR requirement), and Radwaste Building Ventilation System (UFSAR requirement)

Unit 2: Standby Gas Treatment System (TS requirement), and Turbine Building Ventilation System (UFSAR requirement)

The following procedures were reviewed:

- N1-TTP-GEN-V001 — Testing and Analysis of Radwaste Exhaust Ventilation System
- N1-ST-M8 — Reactor Building Emergency Ventilation System Operability Test
- N1-TSP-202-001 — Testing of Unit 1 Reactor Building Emergency Ventilation System
- N2-TSP-GTS-001 — Standby Gas Treatment System

b. Observations and Findings

No discrepancies were identified regarding charcoal adsorption surveillance tests or during tours for the Unit 1 reactor building emergency ventilation system, the Unit 1 turbine building ventilation system, the Unit 2 standby gas treatment system, and the Unit 2 turbine building ventilation system.

Minor discrepancies were noted regarding licensee Surveillance Procedure N1-TTP-GEN-V001 pertaining to the Unit 1 radwaste building exhaust ventilation system testing. In a July 2, 1998, telephone conference with NMPC personnel, the inspectors were informed that the Unit 1 radwaste building ventilation system HEPA filter was successfully retested and that improved acceptance criteria were added to Procedure N1-TPP-GEN-V001. The inspectors had no further questions regarding this matter.

c. Conclusions

The licensee established, implemented, and maintained an effective ventilation system surveillance program.

R7 Quality Assurance in RP&C Activities

R7.1 Review of QA Audit Related to Radiological Practices (84750)

The inspectors reviewed Audit No. 97015, "Environmental Protection, Radioactive Effluents, Radiological Material Processing, Transport, and Disposal," and discussed the audit findings and actions taken with the chemistry department staff. The licensee audit team identified a number of findings and strengths. None of the findings were assessed to have regulatory significance. Scope and depth of the audit was adequate. There were individuals with pertinent experience on the audit team. Responses to audit findings were timely and reasonable.

The licensee conducted an effective quality assurance audit of the radioactive effluent control program with respect to audit scope and depth, audit team experience, and response to audit findings.

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 exit meetings for specialist inspections were conducted upon completion of their onsite inspection:

- Effluents Monitoring — June 5, 1998
- Unit 2 Inservice Inspection — June 12, 1998

The final exit meeting occurred on July 24, 1998. During this meeting, the resident inspectors' 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.

ATTACHMENT 1

PARTIAL LIST OF NMPC PERSONS CONTACTED

Niagara Mohawk Power Corporation

R. Abbott	Vice President, Nuclear Engineering
D. Barcomb	Manager, Unit 2 Radiation Protection
D. Bosnic	Manager, Unit 2 Operations
J. Burton	Manager, Training
H. Christensen	Manager, Security
J. Conway	Vice President, Nuclear Generation
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
A. Julka	Director, ISEG
C. Merritt	Manager, Unit 2 Chemistry
P. Mezzafero	Manager, Unit 1 Technical Support
L. Pisano	Manager, Unit 2 Maintenance
N. Rademacher	Manager, Quality Assurance
R. Randall	Manager, Unit 1 Engineering
V. Schuman	Manager, Unit 1 Radiation Protection
R. Smith	Plant Manager, Unit 1
C. Terry	Vice President, Nuclear Safety Assessment & Support
D. Topley	Manager, Unit 1 Operations
K. Ward	Manager, Unit 2 Technical Support
D. Wolniak	Manager, Licensing

INSPECTION PROCEDURES USED

IP 37551	On-Site Engineering
IP 61726	Surveillance Observations
IP 62707	Maintenance Observations
IP 71707	Plant Operations
IP 71715	Sustained Control Room and Plant Observation
IP 71750	Plant Support
IP 73753	Inservice Inspection
IP 84750	Radioactive Waste Treatment, and Effluent and Environmental Monitoring
IP 90712	In-Office Review of Written Reports of Non-Routine Events at Power Reactor Facilities
IP 92700	Onsite Follow-up of Written Reports of Non-Routine Events at Power Reactor Facilities
IP 92901	Follow-up - Operations
IP 92903	Follow-up - Engineering
IP 93702	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND UPDATED

OPENED

50-220/98-06-01	NCV	Rod worth minimizer TS surveillance requirement not met for previous shutdowns
50-220/98-06-02	NCV	Control room emergency ventilation system outside design basis due to fire damper closure following a LOOP
50-220/98-06-03	NCV	Inaccurate reactor vessel level instruments due to an inaccurate parameter
50-410/98-06-04	NCV	Missed technical specification surveillance testing of alternate power supply
50-410/98-06-05	NCV	Missed technical specification logic system functional testing of service water pump circuitry

Attachment 1 (cont'd)

CLOSED

50-220/98-06-01	NCV	Rod worth minimizer TS surveillance requirement not met for previous shutdowns
50-220/98-06-02	NCV	Control room emergency ventilation system outside design basis due to fire damper closure following a LOOP
50-220/98-06-03	NCV	Inaccurate reactor vessel level instruments due to an inaccurate parameter
50-410/98-06-04	NCV	Missed technical specification surveillance testing of alternate power supply
50-410/98-06-05	NCV	Missed technical specification logic system functional testing of service water pump circuitry
50-220/96-07-03	VIO	Unit 1 final safety analysis report changed without required safety evaluation
50-220 & 50-410/97-01-02	URI	Disparity between the results of an NRC inspection and an NMPC audit of the C&D battery vendor
50-220 & 50-410/97-01-03	URI	Operability of C&D batteries installed at Nine Mile
50-220 & 50-410/97-01-04	URI	Ability of NMPC to identify the location and use of purchased equipment
50-410/96-07-02	VIO	Inadequate procedure for the Unit 2 EDG duplex strainers
50-410/97-12-05	VIO	Failure to perform TSSR of rod sequence control system
50-410/96-01-02	URI	Overpressurization of the Unit 2 RWCU system
50-220/98-07	LER	Senior reactor operator leaves the control room which is a TS violation
50-220/98-08	LER	Rod worth minimizer TS surveillance requirement not met for previous shutdowns
50-220/98-09	LER	Missing fire protection material from structural steel
50-220/98-10	LER	Inaccurate reactor vessel level instruments due to an inaccurate input parameter
50-220/98-11	LER	Inadvertent actuation of the reactor protection system circuitry due to personnel error
50-220/98-12	LER	Control room emergency ventilation system outside design basis due to fire damper closure following a LOOP

Attachment 1 (cont'd)

50-220/98-14	LER	Control room staffing in violation of TS due to unqualified SRO on shift
50-410/98-04-01	LER	Missed technical specification required LSFT of level 8 trip of main turbine
50-410/98-10	LER	Entry into TS 3.0.3 due to incorrect latching mechanisms installed in emergency switchgear
50-410/98-11	LER	Missed technical specification surveillance testing of alternate power supply
50-410/98-12	LER	Missed technical specification logic system functional testing of service water pump circuitry
50-410/98-13	LER	Engineered safety feature actuation due to personnel error

LIST OF ACRONYMS USED

APRM	Average Power Range Monitors
AR	Applicability Review
ASME	American Society of Mechanical Engineers
ASSS	Assistant Station Shift Supervisor
ATWS/RPT	Anticipated Transient without Scram/Recirculation Pump Trip
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel Inspection Program
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CREVS	Control Room Emergency Ventilation System
C&D	Charter Power Systems, Inc.
DER	Deviation/Event Report
dpm	drops per minute
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EEI	Escalated Enforcement Item
EPRI	Electric Power Research Institute
ESF	Engineered Safeguards Feature
°F	degrees Fahrenheit
GE	General Electric
GENE	General Electric Nuclear Energy
GL	Generic Letter
HEPA	High Efficiency Particulate Air
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilation and Air Conditioning
hydrotest	hydrostatic pressure leak test
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	Intergranular Stress Corrosion Cracking
IPAP	Integrated Performance Assessment Process
IR	Inspection Report
IRM	Intermediate Range Monitor
ISI	Inservice Inspection
I&C	Instrumentation and Controls
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
LSFT	Logic System Functional Test
MEL	Master Equipment List

Attachment 1 (cont'd)

MOV	Motor Operated Valve
M&TE	Measuring & Test Equipment
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NMPC	Nine Mile Point Corporation
NRC	Nuclear Regulatory Commission
PB	Power Board
PCT	Peak Cladding Temperature
PMT	Post-Maintenance Test
psi	pounds per square inch
QA	Quality Assurance
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RFO	Refueling Outage
RHR	Residual Heat Removal
RMS	Radiation Monitoring System
RPS	Reactor Protection System
RSCS	Rod Sequence Control System
RWM	Rod Worth Minimizer
SDC	Shutdown Cooling
SE	Safety Evaluation
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
SSS	Station Shift Supervisor
SWP	Service Water System
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
UT	Ultrasonic Test
WO	Work Order