U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket/Report Nos.:	50-220/96-14 50-410/96-14
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:	December 1, 1996 - January 11, 1997
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EXECUTIVE SUMMARY

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This integrated inspection report includes reviews of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection.

PLANT OPERATIONS

The December 5, 1996, power reduction at Unit 2 to exchange operating feedwater pumps was completed in accordance with approved procedures. The pre-evolution briefing contained an appropriate level of detail for the plant conditions, and discussions by the operators indicated a thorough understanding of the upcoming evolution.

The licensee demonstrated good safety perspective by shutting down Unit 2 upon identification of the missed Updated Final Safety Analysis Report (UFSAR) required control rod drive (CRD) housing support gap verifications. The licensee's inspection revealed out-of-tolerance gaps on 19 of 185 CRDs. Their engineering supporting analysis, which determined that the out-of-tolerance gaps did not affect the operability of the system, was considered technically sound by the inspectors. Subsequent changes to the applicable procedure appear adequate to ensure the future gap verification. However, the failure to completed the UFSAR-required gap verifications was classified as a Non-Cited Violation.

The inspectors identified an open polyethylene bag adjacent to the Unit 1 spent fuel pool that was not properly controlled with respect to foreign material exclusion accountability due to inattention by personnel. This was classified as a Non-Cited Violation.

At Unit 1, the inspector's identified several examples of improperly stored ladders. The recurring failure to appropriately store and/or secure ladders located near safety-related and other important equipment was considered a weakness.

MAINTENANCE

Unit 1 Operations Department conducted a liquid poison system quarterly surveillance in a controlled manner. Coordination and communications between Operations and Inservice Testing (IST) personnel were very good. Operations and IST personnel knowledge regarding the evolution was good. The test data was complete and received a timely review.

Unit 2 appropriately included the non-safety-related electrical switchgears within the scope of the Maintenance Rule, and evaluated the risk associated with on-line maintenance for the switchgear.

Executive Summary (cont'd)

ENGINEERING

Niagara Mohawk Power Corporation (NMPC) notified the Nuclear Regulatory Commission (NRC) that Unit 2 may have operated outside its design basis due to the potential for a 10 CFR 50, Appendix R, fire-induced hot short condition that could result in damage to three shutdown cooling valves. The licensee took appropriate immediate corrective actions to address the concern.

NMPC appropriately notified the NRC of potential conditions outside design basis identified during their review of Generic Letter (GL) 96-06. The licensee's review identified several drywell penetrations at both units that could potentially exceed the design pressure during an accident due to thermal expansion of entrapped water. The operability determinations were adequate and in accordance with the guidance provided in GL 96-06; the operability determinations for the Unit 1 core spray high point vent and post-accident sampling lines were particularly appropriate and conservative.

The licensee issued a voluntary Licensee Event Report concerning potential overstressed pipe supports for the reactor building closed-loop cooling system. The supports could become overstressed due to thermally induced longitudinal expansion. The root cause evaluation and corrective actions were appropriate.

NMPC evaluation and corrective actions to NRC-identified emergency cooling system discrepancies appeared appropriate. However, the licensee's failure to implement the existing technical guidance to ensure adequate valve packing gland nut thread engagement was classified as a Non-Cited Violation.

The failure to follow plant procedures resulted in the installation of temporary scaffolding around the Unit 2 standby liquid control (SLC) system tank for an extended period without proper engineering analysis. This was classified as a Non-Cited Violation.

PLANT SUPPORT

On two occasions, the inspectors identified the same high radiation area access gate to be unlocked. The corrective actions to the first occurrence were ineffective, and the inspectors considered this a recurring failure of procedural adherence. **(VIO 96-14-03)**

REPORT DETAILS

Nine Mile Point Units 1 and 2
50-220/96-14 & 50-410/96-14
December 1, 1996 - January 11, 1997

SUMMARY OF ACTIVITIES

Niagara Mohawk Power Corporation (NMPC) Activities

Unit 1

Nine Mile Point Unit 1 (Unit 1) started the inspection period at full power. On December 24, Unit 1 experienced a failure of #11 circulating water pump. This resulted in a reduction of power to 78% for approximately 48 hours. Full power operation resumed and continued to the end of the report period.

Unit 2

Nine Mile Point Unit 2 (Unit 2) started the inspection period at full power. On December 5, power at Unit 2 was reduced to approximately 55% to support a feed water pump (FWP) exchange. Unit 2 was returned to full power 38 hours later. On December 19, Unit 2 conducted a reactor shutdown to perform control rod drive (CRD) housing support gap inspections. The reactor was started up on December 23, and the unit achieved full power on December 26. The unit maintained essentially full power for the remainder of the inspection period.

Organizational Changes

On December 1, 1996, Messrs. Martin McCormick and Carl Terry exchanged responsibilities within the Nine Mile Point nuclear organization. Mr. McCormick became the Vice President - Nuclear Engineering and Mr. Terry became the Vice President - Nuclear Safety Assessment and Support.

Nuclear Regulatory Commission (NRC) Staff Activities

Inspection Activities

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

Updated Final Safety Analysis Report (UFSAR) Reviews

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for additional verification that licensees were complying with UFSAR commitments. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR related to the areas inspected. The inspectors verified that, with the exception of the Unit 2 CRD housing support gap inspections described in Section O1.3, the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters.

I. OPERATIONS

O1 Conduct of Operations (71707)[1]

O1.1 General Comments

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. 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 2 Power Reduction for Feed Water Pump Exchange

a. Inspection Scope

On December 5, 1996, a planned power reduction of the Unit 2 reactor was completed to allow the B FWP to be removed from service due to excessive seal leakage. The inspectors observed portions of the power reduction because the B reactor recirculation system (RCS) pump was experiencing higher than normal vibration that could have been aggravated during the evolution. In addition to observing the power reduction, the inspectors observed the pre-evolution briefing, and reviewed applicable plant procedures.

b. Observations and Findings

The inspectors observed portions of the Unit 2 power reduction completed on December 5, 1996. Power was lowered to remove the B FWP from service because of excessive seal leakage. Control rods were inserted until power was reduced to 90%, then reactor recirculation flow was decreased until power reached 55%. The power reduction was preformed in accordance with approved Procedure N2-OP-101D, "Power Changes," Revision 3.

The inspectors observed the reactivity manipulation pre-evolution briefing provided to the crew performing the reactor recirculation flow decrease. This briefing was completed in accordance with Procedure N2-ODP-OPS-0110, "Reactivity Management Program," Revision 7. Since the B RCS pump was experiencing higher than normal vibration, which could be aggravated during the flow decrease, the briefing included a review of the emergency downpower actions. During the briefing, discussions by the operators indicated their understanding of the upcoming evolution. Before the power reduction, the RCS pump vibrations were higher than normal but within the manufacturer's limits. Before and during the power reduction, NMPC monitored and trended additional parameters for indication of pump degradation; no indications of RCS pump degradation were observed.

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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.

During the evolution, management oversight was provided by the Operations Manager, who was present in the control room during a majority of the power reduction. The power reduction was completed without incident, and Unit 2 was returned to full power on December 7.

c. Conclusion

The December 5, 1996, power reduction at Unit 2 for the removal of the B FWP was completed in accordance with approved procedures. The pre-evolution briefing was thorough, with an appropriate level of detail for the plant conditions; discussion by the operators indicated a thorough understanding of the upcoming evolution.

O1.3 Unit 2 Missed UFSAR-Required CRD Housing Support Gap Inspections

a. Inspection Scope

On December 19, the licensee commenced a shutdown of Unit 2 from 100% power, as required by technical specification (TS) 3.1.3.8, due to a failure to perform an UFSAR-required gap inspection of the CRD housing supports (commonly referred to as the "shoot-out steel"). Unit 2 was shutdown and the required inspections were completed; adjustments were made to restore the CRD housing support gap back within specification; and Unit 2 was subsequently restarted.

The inspectors reviewed the applicable deviation event/reports (DERs), engineering supporting analysis, UFSAR sections, TS and plant procedures. The inspectors also observed portions of the plant shutdown; visually inspected the CRD housing support; and reviewed the gap inspection results.

b. Observations and Findings

As part of the corrective actions associated with a previous event, NMPC was conducting a review of the UFSAR to validate that necessary programs and procedures were in place The review identified that the UFSAR requirement to inspect the CRD housing support after reinstallation, with particular attention to maintaining the correct gap between the CRD flange and the housing support, was not contained in the Unit 2 procedures; and thus, had not been performed since initial startup.

The CRD housing supports prevent a significant nuclear transient in the event a drive housing breaks or separates from the bottom of the reactor vessel. Unit 2 TS 4.1.3.8, "Control Rod Drive Housing Support," requires a visual inspection to ensure that the CRD housing support is in place following disassembly. However, UFSAR Section 4.6.3.2 states: "When the support structure is reinstalled, it is inspected for correct assembly with particular attention to maintaining the correct gap between the CRD flange lower contact surface and the grid." The gap, as described in the UFSAR, Section 4.6.1.2.3, is approximately 1-inch between the contact surface of the CRD flange and the support grid. This gap allows the CRD housing support to accomplish its intended function, while providing sufficient

clearance to prevent contact stresses caused by thermal expansion. Although the TS-required visual inspections of the CRD support housing were being performed, the gap was not verified to be acceptable.

On December 19, 1996, while Unit 2 was operating at 100% power, NMPC notified the NRC in accordance with Title 10 of the Code of Federal Regulations (CFR), Part 50.72, that a shutdown of Unit 2 had commenced in accordance with TS 3.1.3.8, due to a failure to perform a UFSAR-required inspection. The failure to perform the required gap inspections placed the unit in a potentially unanalyzed condition. The inspectors observed portions of the plant shutdown and determined it to be performed in accordance with Procedure N2-OP-101C, "Plant Shutdown," Revision 11.

While shutdown, the licensee inspected the CRD housing support gap in accordance with Work Order (WO) 96-16814-00. They identified that 19 of the 185 CRDs had gaps that were slightly outside the vendor recommended tolerance of 1 inch ± 0.12 inches. The smallest measured gap was 0.828 inches, and the largest gap was 1.190 inches. The gaps on all CRDs were adjusted back within tolerance. Unit 2 was restarted on December 23 and the unit achieved full power on December 26. Subsequent to the plant restart, the licensee completed an engineering analysis and determined that the CRD housing support had always been operable. This analysis was supported by General Electric (GE) analysis, and evaluated both extremes of the as-found inspection results. Specifically, the largest gap would not have resulted in an impact load on the housing support that would exceed the allowable stress. Therefore, the support would have prevented any significant nuclear transient in the event a drive housing broke or separated from the bottom of the reactor vessel. Additionally, the smallest gap still provided sufficient clearance to prevent contact stresses caused by thermal expansion. The licensee changed Procedure N2-MMP-RDS-670, "CRD Support Steel Removal & Installation," Revision 02, to ensure proper gap verification would be performed in the future.

The inspectors performed an independent visual inspection of the CRD housing support. The inspector also reviewed the method used by the licensee to verify the gap between the CRD and the housing support. No concerns were identified. The inspectors also reviewed the NMPC engineering analysis and the GE supporting analysis and considered them technically sound. However, the failure to include the UFSAR-required CRD housing support gap verification in plant procedures is a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." This licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy.

The Unit 1 UFSAR also refers to a 1-inch gap for the CRD support housing; however, no verification was required. Previous maintenance activities at Unit 1 had not required adjustment of the CRD housing support nuts (the nuts used to set the gap); therefore, NMPC determined that the original gap was maintained. After the end of this inspection period, Unit 1 inspected the CRD housing support gaps during Forced Outage 97-01 (January 17 through 19, 1997). All gaps were found

within tolerance. Unit 1 was still evaluating changes to their procedures to ensure future gap verification.

c. Conclusion

Upon identification of the missed UFSAR-required CRD housing support gap verification, NMPC shut down Unit 2. The licensee demonstrated a good safety perspective by shutting down the unit instead of attempting to perform an engineering evaluation to justify continued operation. The licensee's inspection revealed out-of-tolerance gaps on 19 of 185 CRDs. Their engineering analysis, which determined that the out-of-tolerance gaps did not affect the operability of the system, was considered technically sound. Subsequent changes to the applicable procedure were adequate to ensure the future gap verification.

O7 Quality Assurance in Operations (71707, 40500)

O7.1 Unit 1 Housekeeping

a. Inspection Scope

On November 29, 1996, the inspectors toured Unit 1 reactor and turbine buildings. Several housekeeping discrepancies were identified. The inspectors reviewed the applicable plant procedures and discussed the issues with the Station Shift Supervisor (SSS) and Unit 1 management.

b. Observations and Findings

Inadequate Control of Foreign Material Exclusion (FME) Areas on the Refuel Floor

During a tour of the Unit 1 refuel floor, the inspectors noted a large yellow polyethylene (poly) bag located approximately one foot from the edge of the spent fuel pool (SFP). FME controls were in effect surrounding the SFP. The bag was open and contained smaller yellow bags, tags, and trash. There was no indication that any contents of the bag fell into the SFP. No activity was in progress on the refuel floor. However, recent activities included new fuel inspection and movement of new fuel bundles into the SFP.

At the time, the area surrounding the SFP was being controlled as a Level 2 cleanliness local work zone and posted as such. An FME Material Accountability Log was present on the refuel floor. The inspectors reviewed the log for the area surrounding the SFP; however, no log entry for the bag was identified. The inspectors informed an operator performing rounds of the poly bag, who then notified radiation protection (RP) personnel and the SSS. RP directed the operator to remove the bag and contents from within the FME boundary.

The inspectors discussed the concern with the Reactor Engineering Supervisor. The supervisor stated that the bag originated from work on the refueling bridge. The bag had been appropriately recorded in the Material Accountability Log for the

refueling bridge, and logged out upon removal. The inspectors confirmed the log entries. The technician who removed the bag from the refueling bridge, however, placed it within the FME boundary for the SFP area, without logging the bag into that area. Although the area surrounding the SFP was conspicuously posted, the technician stated he was unaware of this area being FME controlled.

NMPC Procedure GAP-HSC-02, "Local Work Zones and System Cleanliness Controls," Revision 05, establishes administrative controls for maintaining local work zones. Section 3.4.3 of GAP-HSC-02 requires a Material Accountability Log to ensure material accountability within Level 1, 2 and 3 work zones. Plant management re-emphasized the importance of FME controls with the staff. However, the failure to adhere to Procedure GAP-HSC-02 is a violation of TS 6.8.1 regarding procedures. 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.

Repetitive Occurrence of Unsecured Ladders Adjacent to Safety-Related Equipment

Unsecured ladders have been occasionally identified in both units by NRC inspectors. In Unit 1, the inspectors discovered unsecured 10-foot A-frame ladders adjacent to safety-related components, such as hydraulic control units and scram discharge volume vent and drain valves (on August 28, 1996) and containment spray pump #121 (on October 2, 1996). On November 14, 1996, the inspectors identified an unsecured ladder adjacent to the operating feedwater pump. Although not safety-related, the feedwater pumps are used for high pressure coolant injection. The ladders were not in use at the time of discovery, nor did it appear that any activity had recently taken place.

On November 29, 1996, the inspectors observed a 10-foot A-frame ladder adjacent to the #15 reactor recirculation pump (RRP) motor generator (MG) set. Although the RRP MG set is not safety-related, damage to the MG set would potentially result in an unanticipated plant transient.

The inspectors discussed the issue with the SSS and Plant Manager. In each case, actions were taken by the licensee to secure the ladders.

c. Conclusions

Due to personnel inattention to postings, material accountability controls were violated on the Unit 1 refuel floor in that an open poly bag was left inside the FME area around the SFP. Also, after use, ladders were occasionally left unsecured in the vicinity of safety-related and other important equipment. The ladders falling could potentially render necessary emergency equipment unavailable or cause a plant transient. The licensee took appropriate corrective action for the discrepancies identified during this inspection.

The above examples were identified by the NRC. The recurring failure to properly store material is indicative of weak management oversight with respect to housekeeping.

O8 Miscellaneous Operations Issues (90712)

O8.1 (Open) LER 50-220/96-11: Reactor Scram Caused by the Main Generator Lockout Relay Trip

The inspectors reviewed the subject Licensee Event Report (LER) and determined that it satisfactorily described the event. However, the assessment of the root cause analysis and corrective actions will remain open pending the enforcement conference related to the reactor overfill event associated with the reactor scram. A detailed review of the issues detailed in this LER is contained in NRC Inspection Report 50-220/96-13, Section O2.3.

II. MAINTENANCE [2]

M1 Conduct of Maintenance (61726, 62707)

M1.1 General Comments

Using NRC Inspection Procedures 61726 and 62707, the inspectors periodically observed the licensee perform plant maintenance activities and conduct various surveillance tests. In general, maintenance and surveillance activities were conducted professionally, with the WOs and necessary procedures in use at the work site, and with the appropriate focus on safety. Specific activities and observations are detailed below. The inspectors reviewed procedures and/or observed portions of the following maintenance/surveillance activities:

• WO-93-3119-00	Repair parallel interlock
• N2-EPM-GEN-5Y555	GE 13.8 kV [kilo-Volt] Magna-Blast Breaker and Associated Motors
• N1-ST-Q8	Liquid Poison Pump and Check Valve Operability Test
• N1-OP-12	Liquid Poison System
• N1-ST-M1	Liquid Poison Pumps Operability Test
• N1-ODP-IIT-0101	Establishment of IST [Inservice Testing] Pump and Valve Acceptance Criteria
• N1-ODP-IIT-0102	Analysis and Trending of IST Results
• N1-ITP-01	Ultrasonic Flow Test
• N1-ITP-02	Vibration Measurement
• N2-MMP-RDS-670	CRD Support Steel Removal & Installation

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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."

M1.2 Unit 1 Liquid Poison Pump and Check Valve Operability Surveillance Test

a. Inspection Scope

On December 30, 1996, Unit 1 Operations and Inservice Test (IST) personnel performed a quarterly surveillance test of the liquid poison system to verify the operability of the pumps and associated discharge check valves. The inspectors reviewed applicable licensee procedures and TSs; observed equipment setup and testing for one train of the system; reviewed completed surveillance tests conducted within the last year; and discussed the surveillance test results with the Assistant SSS (ASSS) and IST Supervisor.

b. Observations and Findings

The inspectors observed the surveillance test locally in the reactor building. Face-to-face and remote communications were very good and the transfer of information went well. The inspectors noted good control of valve manipulations, and independent verifications were adequately performed. Operator knowledge regarding certain procedural steps and anticipated system response appeared adequate. The IST technician appeared very knowledgeable with respect to test equipment installation and operation. All test equipment was within current calibration cycle. The inspectors verified proper system restoration upon completion of the surveillance test.

The inspectors reviewed the completed surveillance test procedure. The test results received a timely review and evaluation by the ASSS. The recorded data was complete and within allowable specification. However, the licensee identified that one vibrational data point on pump #12 was in the "Alert Range." Although the pump was still operable, an increased surveillance frequency was instituted, in accordance with NMPC Procedure N1-ODP-IIT-0101, based upon American Society of Mechanical Engineers Code, Section XI requirements.

The inspectors reviewed prior surveillances and noted that liquid poison pump #12 had exhibited similar vibration in July 1996. Since July however, the pump vibrational data had been in the acceptable range and was taken off an increased surveillance frequency in November 1996, in accordance with the plant procedure. The inspectors discussed the #12 liquid porson pump vibration issue with the IST supervisor, who stated that a definitive cause for the vibration was not known at this time.

c. Conclusions

The inspectors observed a quarterly surveillance test of the Unit 1 liquid poison system; overall, the inspectors determined that the test was conducted in a well controlled manner. Coordination and communications between Operations and IST personnel were very good. Personnel knowledge regarding the surveillance was also good. The surveillance test data received a timely review. Higher than normal

vibration identified on one pump was appropriately trended; although a definitive cause had not yet been determined.

M1.3 Maintenance Rule Evaluation of Unit 2 NonSafety-Related Switchgear-003

a. Inspection Scope

The inspectors observed Unit 2 operators transfer a nonsafety-related switchgear (2NPS-SWG003) to the alternate power source in preparation for preventive and corrective maintenance. The inspectors evaluated the on-line maintenance activities with respect to the Unit 2 Maintenance Rule Procedures. The inspectors reviewed portions of applicable plant procedures, the WO and holdouts. (A holdout is a component tagging process used to provide protection for personnel and/or equipment during operation, maintenance and modification activities, which is commonly referred to as tagout within the industry.)

b. Observations and Findings

On January 3, 1997, Unit 2 performed preventive and corrective maintenance on circuit breaker 2NPS-SWG003-14, the 13.8 kilovolt (kV) normal feed to nonsafety-related switchgear 2NPS-SWG003. Switchgear SWG003 provides power to large balance-of-plant loads, including one FWP, three circulating water pumps, a condensate pump and a booster pump, and several nonsafety-related load centers. Loss of this equipment could potentially result in a plant scram.

The inspectors observed the pre-evolution brief, and considered the review of operator actions in the event of a loss of switchgear SWG003 to be appropriate. The inspectors observed the transfer of switchgear SWG003 to its alternate power source. The transfer was completed in accordance with Procedure N2-OP-71A, "13.8 kV AC [Alternating Current] Power Distribution," Revision 2. The WO and holdout were developed and approved in accordance with the appropriate procedures. After maintenance was completed, switchgear SWG003 was returned to its normal power source.

Discussions with the SSS indicated appropriate precautions were taken during the preparation of the holdouts. The inspectors verified that switchgear SWG003 was appropriately contained in the scope of Maintenance Rule (10 CFR 50.65) and controlled by Unit 2 Procedure N2-MRM-REL-0104, "Maintenance Rule Scope," Revision 00. Discussions with the SSS indicated that Unit 2 appropriately evaluated the risk associated with the transfer for switchgear SWG003.

c. Conclusion

Unit 2 appropriately included nonsafety-related switchgear SWG003 within the scope of the Maintenance Rule, and evaluated the risk associated with on-line maintenance for the switchgear.

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment (37551, 40500)

E2.1 Hot Shorts Vulnerability of Unit 2 Shutdown Cooling Valves

a. Inspection Scope

On December 17, 1996, NMPC notified the NRC that Unit 2 was potentially outside the design basis for 10 CFR Part 50, Appendix R, with respect to shutdown cooling. A hot short condition could have caused motor operated valves (MOVs) in the residual heat removal (RHR) system to be driven closed and mechanically damaged, preventing remote shutdown capability. The inspectors reviewed the applicable DER, engineering supporting analysis, and NRC Information Notice 92-18, "Potential for Loss of Remote Shutdown Capability during a Control Room Fire," and discussed the issue with the Unit 2 design engineering and operations staff. The inspectors also assessed the adequacy of the licensee's short term corrective actions.

b. Observations and Findings

During review of a Unit 1 issue related to Appendix R fire-induced hot shorts in MOVs, NMPC design engineering staff discovered a similar concern at Unit 2. In particular, several MOVs were susceptible to mechanical damage if the hot short bypassed the torque switch causing spurious operation and a valve stall condition. The concern was applicable to 32 of 45 MOVs that would be controlled at the remote shutdown panel during a control room fire, Appendix R shutdown. Of the 32 susceptible MOVs, 28 were part of redundant trains. Thus, failure of one redundant MOV would not prevent remote shutdown capability. However, four MOVs were determined not to have adequate redundancy.

The four valves (2RHS*MOV112, 113, 142, and 149) determined to be susceptible were documented on DER 2-96-3379, and the licensee made a 1-hour notification to the NRC in accordance with 10 CFR 50.72. As part of their immediate corrective actions, NMPC closed three of the valves (RHS*MOV112, 142 and 149) and disconnected the valves from the power supply such that a control room fire could not damage these valves and disable the safe shutdown function. The fourth valve (RHS*MOV113) is normally closed and de-energized to preclude a fire-induced loss-of-coolant accident (LOCA) at the high/low pressure interface location.

MOVs 112 and 113 are isolation valves in the RHR suction path for the shutdown cooling mode, and are required for plant cold shutdown from the control room or the remote shutdown panel. MOVs 142 and 149, RHR discharge to radiological waste isolation valves, are required to be operable from the remote shutdown panel prior to and after initiation of RHR system shutdown cooling mode to flush the stagnant water in the shutdown cooling piping.

Valves RHS*MOV112, 142, and 149 are normally closed. Procedural controls were established, through the use of a holdout, to preclude spurious operation of the

valves and possible mechanical damage. The inspectors verified that the valves were deenergized, and that the appropriate procedural controls were in place. This remains an unresolved item pending the completion of the licensee's analysis and subsequent NRC review. **(URI 50-410/96-14-01)**

c. Conclusions

The licensee's identification and immediate corrective actions to address three shutdown cooling valves potentially susceptible to damage during a 10 CFR 50 Appendix R fire-induced hot short condition were appropriate.

E2.2 Potential Overpressurization Concerns Relative to NRC Generic Letter 96-06

a. Inspection Scope

NMPC engineering staff's preliminary evaluations of NRC Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions," identified specific piping that could potentially be overpressurized during a design-basis LOCA. The overpressure conditions could result in piping exceeding the allowable stresses in several systems; the following drywell penetrations have been identified by NMPC:

- core spray high point vent lines (Unit 1)
- post-accident sampling line (Unit 1)
- shutdown cooling system lines (Unit 1)
- drywell equipment and floor drain lines (Unit 1)
- drywell unit cooler lines (Unit 2)
- reactor recirculation pump seal cooler lines (Unit 2)

As a result, NMPC determined that both units were potentially outside the design basis and notified the NRC in accordance with 10 CFR 50.72.. To determine the adequacy of the licensee's immediate corrective actions, the inspectors reviewed the applicable DERs, engineering supporting analyses and GL 96-06, and discussed the concerns with NMPC engineering and operations personnel.

b. Observations and Findings

Unit 1

Unit 1 engineering staff determined that the core spray (CS) high point vent lines and the post-accident sampling (PAS) line could potentially overpressurize and exceed allowable stresses during a design basis LOCA. During a design basis LOCA, the potential existed for water trapped between the containment isolation valves (CIVs) to heat up and thermally expand, to the point where piping integrity was not assured.

On December 13, 1996, NMPC notified the NRC, in accordance with 10 CFR 50.72, that Unit 1 was potentially in a condition outside the design basis. NMPC

additionally issued DER 1-96-3350 to internally document the concern. NMPC considered both the CS and PAS systems operable, through use of administrative controls. Specially, the piping between the CIVs for both systems was drained every 24 hours to maintain an adequate expansion volume.

NMPC determined that following a postulated design basis LOCA, the thermal expansion of the fluid trapped between the normally closed shutdown cooling (SDC) CIVs could potentially result in internal pressure exceeding allowable stresses. The fluid temperatures within the SDC piping penetrations were "hot" as a result of valve seat and packing leakage. NMPC's preliminary engineering evaluation determined that the already elevated fluid temperature would reduce the net thermal expansion during a design basis LOCA, such that the peak pressure was within the design basis rating for the pipe.

NMPC design engineering analyses identified potential overpressure concerns associated with the lines for the drywell equipment drains (DWED) and drywell floor drains (DWFD). The DWED and DWFD inside and outside CIVs were normally open and the piping not normally filled with water. However, the piping would contain water during pump-down operations. During the design basis LOCA, the CIVs would automatically close and could potentially trap water between the CIVs. Thermal expansion of the trapped fluid could potentially exceed allowable internal stresses for the pipes. Based on the design of the DWED and DWFD outboard CIVs, NMPC preliminarily determined that the valve disc would unseat at approximately 150 pounds per square inch (psi), thus providing pressure relief to maintain the piping within design basis pressure.

On December 20, 1996, NMPC notified the NRC, in accordance with 10 CFR 50.72, that Unit 1 was potentially in a condition outside the design basis. NMPC issued DER 1-96-3419 to internally document the concern. Both the SDC and DWED/DWFD systems were considered operable based upon present system configuration.

The potential overpressure conditions identified at Unit 1 during the licensee's review of GL 96-06 remains an unresolved item pending the completion of the NMPC's evaluation to determine if this condition was outside the design basis and subsequent NRC review. **(URI 50-220/96-14-02)**

<u>Unit 2</u>

NMPC identified four penetrations at Unit 2 that could potentially exceed the design pressure during an accident due to thermal expansion of entrapped water between the inboard and outboard CIVs. The penetrations allow reactor building closed loop cooling (RBCLC) water to flow into and out of the drywell for drywell unit coolers and reactor recirculation pump seal coolers. NMPC documented the concern in DER 2-96-3427. On December 20, 1996, NMPC notified the NRC on the condition in accordance with 10 CFR 50.72.

The engineering supporting analysis associated with the DER 2-96-3427 based continued operability of the equipment on projected valve leakage. With leakage considered, the calculated maximum pressure expected during thermal expansion would not exceed the allowed pressures. The inspectors reviewed the engineering analysis and deemed the basis for operability to be consistent with guidance provided in GL 96-06. However, this item remains unresolved pending the completion of NMPC's evaluation to determine if this condition was outside the design basis and subsequent NRC review. **(URI 50-410/96-14-02)**

c. Conclusions

NMPC appropriately notified the NRC in accordance with 10 CFR 50.72 of potential conditions outside design basis identified during their review of GL 96-06. The licensee's review identified several drywell penetrations at both units that could potentially exceed the design pressure during an accident due to thermal expansion of entrapped water. The inspectors determined that the operability determinations were adequate and in accordance with the guidance provided in GL 96-06. The operability determinations for the Unit 1 CS high point vent and PAS lines were particularly appropriate and conservative.

E8 Miscellaneous Engineering Issues (90712, 92700, 92903)

E8.1 (Closed) LER 50-220/96-09: Potential Overstressed Pipe Supports Caused by Design Deficiency

a. Inspection Scope

The inspectors reviewed the Unit 1 voluntary LER related to the potential for reactor building closed loop cooling (RBCLC) system pipe supports within containment to be overstressed in the event of a LOCA coincident with a loss of offsite power (LOOP). The inspectors discussed the issue and conclusions with Unit 1 Engineering management and staff, and evaluated the licensee's root cause determination and corrective actions.

b. Observations and Findings

On October 21, 1996, during the review of an engineering analysis related to the RBCLC system pipe supports, Unit 1 plant management determined that the supports within containment could be overstressed in the event of a LOCA coincident with a LOOP. The deficiency was identified as the result of an engineering evaluation performed in response to similar industry operating experience (a LER issued by Haddam Neck Nuclear Station on July 22, 1996). The performance of the engineering evaluation appeared prudent.

The licensee determined that, in the event of a LOCA with a LOOP, the temperature of the RBCLC water and piping within containment could increase before power and flow were restored, resulting in thermally induced longitudinal expansion and potential overstressing and failure of some U-bolt supports. However, the licensee

determined that this deficiency did not result in a significant safety concern as the piping would remain intact and operable even considering a subsequent seismic event. The licensee also determined that the plant was not in a condition outside the design basis. Particularly, the engineering evaluation determined that the design basis load combinations for RBCLC, as described in the UFSAR, did not address accident loads because the system was not safety-related. However, good engineering practice would have included the accident loads, particularly thermal stresses resulting from accident conditions.

The licensee evaluated the RBCLC piping inside containment and identified thirteen 8-inch piping supports and thirteen 4-inch piping supports requiring modification. Work requests for repair and/or modification were generated and added to the work scope for Refueling Outage (RFO) 14, scheduled for March 1997.

The inspectors discussed the LER with the Engineering Manager and a structural engineer. NMPC issued the voluntary LER due to the generic implications of the situation, which was similar to that discussed in GL 96-06. The LER satisfactorily described the event. The causes and the corrective actions are detailed and appropriate. The inspectors had no further questions.

c. <u>Conclusions</u>

The licensee performed an engineering evaluation of Unit 1 in response to industry operating experience and determined that RBCLC system pipe supports within containment could be overstressed in the event of a LOCA coincident with a LOOP; performance of the engineering evaluation appeared warranted and prudent. The root cause evaluation and corrective actions to prevent similar occurrence were appropriate.

E8.2 <u>(Closed) Special Report: Inoperability of Unit 1 #11 Containment Hydrogen Monitoring System</u>

On November 14, 1996, with Unit 1 operating at 100% reactor power, #11 Containment Hydrogen Monitoring System (HMS) was removed from service for calibration. During the calibration, a toggle switch failed, delaying completion of the surveillance until the switch was replaced. The #12 HMS was operable and in calibration. The #11 HMS was returned to an operable status on November 16 following toggle switch replacement and system calibration.

NMPC initiated DER 1-96-3100 to evaluate the toggle switch failure and to determined corrective actions. NMPC concluded that frequent use of the toggle switch resulted in mechanical failure. The toggle switch was a momentary contact, spring return switch, actuated extensively during monthly calibrations, and did not impact the safety-related function of the system. The inspectors' discussion with the system engineer indicated that the associated toggle switches were recently installed as part of a modification during RFO13 in 1995.

Within the past year, numerous toggle switches of this type have failed and were subsequently replaced. NMPC was evaluating the availability of replacement switches of different design. NMPC will continue to monitor toggle switch performance and replace failed switches, as required. The licensee found a different design switch which was being evaluated as a replacement; they intend to replace the balance of the switches in the near future. In addition, NMPC is reviewing the repetitive failures of the toggle switches for possible 10 CFR 21 reporting consideration.

NMPC submitted a special report to the NRC within 14 days, as required by Unit 1 TS 3.6.11-1, Action Statement Table 3.6.11-2 (4a). The inspectors reviewed the special report and confirmed that all required information was provided.

E8.3 (Closed) URI 50-220/95-25-01: Emergency Cooling System Material Deficiencies

a. Inspection Scope

In January 1996, NRC resident inspectors performed a walkdown of all accessible areas of the Unit 1 emergency cooling (EC) system and identified two material condition concerns. Specifically, several EC system drain valve packing gland nuts appeared to have insufficient thread engagement; and secondly, supports for the fire water header connection to the #11 EC makeup tank appeared to exceed the maximum allowable span between supports permitted by NMPC internal standards.

The inspectors reviewed the licensee DERs to assess corrective actions and discussed the results with members of the engineering staff.

b. Observations and Findings

Packing Gland Nut Thread Engagement

NMPC documented the packing gland nut thread engagement issue on DER 1-96-0130. NMPC Standard Design Specification Procedure, SDS-006, "Bolt-Torque Requirements for Unit 1 and Unit 2," Revision 1, provided general guidance for thread engagement. However, in practice, NMPC did not apply this requirement to packing gland nuts, even though there was no exception stated within the procedure. Maintenance staff were trained to adjust packing so that a valve could be operated without binding and no packing leakage existed. NMPC stated that vendor manuals could be used as guidance, but most vendor manuals did not specifically address packing gland nut thread engagement.

Procedure SDS-006, Section 6.1.C, stated that " . . . the minimum thread engagement for a fastener will be one thread beyond the top of the nut, . . . [and that for] any fasteners that do not obtain thread engagement greater than one thread beyond the top of the nut, approval by design engineering is required." The failure to follow Procedure SDS-006, is a violation of TS 6.8.1 regarding procedural adherence. 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.

NMPC discussed the packing gland thread engagement concern with vendors and other licensees, and concluded that good maintenance practice was to ensure one thread visible beyond the nut. SDS-006 was revised to reinforce compliance with Section 6.1.C. with respect to packing gland nut thread engagement. Packing gland nuts currently with insufficient thread engagement were to be evaluated for operability on a case-by-case basis.

Fire Water Header Piping Supports to #11 EC Makeup Tank

NMPC issued DER 1-96-0102 to document the potential operability concern regarding the lack of fire water header piping supports to #11 EC makeup tank. Design Engineering determined that the supports for #11 EC makeup tank fire water supply were adequate for the applied loading and for system operability; however, a support was added mid-span as a system enhancement. The inspectors verified the support installation and that the engineering drawing represented the current plant configuration.

c. Conclusions

NMPC evaluation and corrective actions to NRC-identified EC system discrepancies appeared appropriate. However, the failure to implement technical guidance to ensure adequate valve packing gland nut thread engagement was a violation of procedures.

E8.4 (Closed) URI 50-410/95-12-01: Temporary Scaffolding Erected Around Unit 2 Liquid Poison Tank for Extended Period with no Engineering Evaluation

a. Inspection Scope

In April 1995, during an inspection of the Unit 2 reactor building, the inspectors identified that the temporary scaffolding around the standby liquid control (SLC) storage tank did not appear to have been inspected recently. In addition, the inspectors questioned whether an engineering analysis had been performed considering the potential safety risk associated with temporary scaffolding near safety-related equipment. The Unit 2 Independent Safety Engineering Group (ISEG) initiated an investigation after the inspectors raised the concern.

b. Observations and Findings

The inspectors reviewed the associated ISEG report, dated May 15, 1995. The report identified that the attached scaffold tag (#93-231) indicated the scaffold had been erected or last inspected sometime in 1993. Also, ISEG noted that no analysis had ever been performed. Scaffold Procedure N2-MAP-MAI-0301, required an evaluation of scaffolds in safety-related areas that were installed for greater than

60 days. The ISEG report stated that the scaffold was scheduled to be replaced with a permanent structure by August 31, 1995, in accordance with a simple design change (SDC 2-0398-91). The ISEG concluded that the engineering evaluation should have been performed when it was recognized that the scaffold was to be installed for long term.

The licensee initiated a DER (2-95-2093) after the NRC identified the issue. NMPC determined the root cause to be a combination of factors: inadequate procedure adherence; engineering judgement used in lieu of calculation; management did not budget resources after approving the design change; and engineering did not properly disposition a May 1992 DER (2-92-2132). DER 2-95-2093 also noted that a modification was requested in September 1986 to install a permanent ladder and platform over the SLC tank, and that the temporary scaffolding was initially installed in September 1989.

Immediate corrective actions included a seismic evaluation of the scaffolding until the permanent structure could be installed. NMPC reviewed all installed scaffolds and identified two others that exceeded the 60-day requirement; one at Unit 1 and another at Unit 2. Both were evaluated for seismic considerations and found acceptable. Actions to preclude recurrence included a review of the associated maintenance and engineering procedures, and emphasis on procedural adherence.

The inspectors verified that the scaffolding around the SLC tank had been replaced with a permanent ladder and work platform. However, the failure to perform evaluations of scaffolding erected for greater than 60 days is a violation of Unit 2 Procedure N2-MAP-MAI-0301, Section 5.5.1b. 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.

c. <u>Conclusion</u>

The failure to follow plant procedures resulted in the installation of temporary scaffolding around the Unit 2 SLC tank for an extended period without proper engineering analysis.

IV. PLANT SUPPORT

Using Inspection Procedure 71750, the 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.

R4 Staff Knowledge and Performance in Radiological Protection (71750)

R4.1 Repeat Failure to Properly Secure a Unit 1 High Radiation Area Gate

a. Inspection Scope

At the end of the previous inspection period, the inspectors toured the Unit 1 turbine building and identified that the gate to the turbine deck, a posted high radiation area, was unlatched. The inspectors continued inspection of this issue during this inspection period. The issue was discussed with the SSS, RP supervision, the Operations Manager, and the Plant Manager.

b. Observations and Findings

On November 29, 1996, the inspectors identified that on the 300 foot elevation of the turbine building, the east gate to the turbine deck was not properly latched and locked, allowing access to the turbine deck and adjacent reheater rooms. During power operations, the gates to the turbine deck are normally locked and posted as "High Radiation Areas." RP and the SSS were notified and the gate was subsequently shut and latched. NMPC initiated DER 1-96-3217 to address the issue. Reactor power at the time was 100%. Subsequent radiation surveys indicated the highest localized radiation levels (measured at 30 centimeters) were approximately 300 millirem/hour (mrem/hr) on the turbine deck and 800 mrem/hr in the reheater rooms. The highest on contact readings were 380 mrem/hr and 1000 mrem/hr on the turbine deck and in the reheater rooms, respectively.

Previously, on September 17, 1996, the inspectors identified the same gate not properly latched and locked. Subsequently, and only following further discussion with the inspectors, NMPC initiated DER 1-96-2301 on September 27 to address the issue. NMPC noted the apparent cause as inadequate work practices, in that personnel failed to verify gate closure. The corrective actions were to (1) counsel shift personnel with regard to ensuring lockable barriers remained latched, and (2) repair the gate, which had considerable "play" and was known to not always latch upon closure.

The Plant Manager and RP Manager informed the inspectors that the gate being unlatched did not meet their expectations. The inspectors noted that corrective actions to the September 17 event were ineffective, in that personnel again failed to verify proper gate latching upon exiting the area. Subsequent to the November 29 event, counselling of shift personnel was again conducted. An already open work order to repair the "play" in the gate was immediately initiated upon identifying the repeat event.

NMPC Procedure GAP-RPP-08, "Control of High, Locked High, and Very High Radiation Areas," Revision 03, Section 3.1.3 states that "... when practicable, High Radiation Areas should be locked." Additionally, Section 3.6.1 requires personnel to maintain positive access control to High, Locked High, and Very High Radiation Areas. The procedure specified that barriers are to remain closed and locked after

each entry, and that the barriers be checked closed by shaking. The failure to ensure that the east gate to the Unit 1 turbine deck, a posted High Radiation Area, remained locked was not in accordance with Procedure GAP-RPP-08 and is a violation of Unit 1 TS 6.11. TS 6.11 requires that written procedures be approved, maintained and adhered to for all operations involving personnel radiation exposure. (VIO 50-220/96-14-03)

c. Conclusions

On two occasions, the inspectors identified the same high radiation area access gate to be unlocked. The inspectors considered this a recurring failure of procedural adherence and attention to detail. Furthermore, the corrective actions to the first occurrence were ineffective.

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 27, 1997. 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.

X3 Management Meeting Summary

On January 6, 1997, a meeting between the NRC and NMPC management was held at the NRC headquarters. This meeting was requested by NMPC to present their bases for disagreeing with the Level IV violation regarding the failure to report the condition of the Unit 1 blow out panels being outside the design basis when it was identified in October 1993 (NRC Inspection Report 50-220/96-05). Results of this meeting will be provided to NMPC in a separate correspondence. This meeting was open to the public.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

<u>Niagara Mohawk Power Corporation</u>

R. Abbott, Vice President & General Manager, Nuclear
J. Aldrich, Maintenance Manager, Unit 1
M. Balduzzi, Operations Manager, Unit 1
D. Barcomb, Radiation Protection Manager, Unit 2
C. Beckham, Manager, Quality Assurance
J. Burton, Director, ISEG
G. Correll, Chemistry Manager, Unit 1
J. Conway, Plant Manager, Unit 2
K. Dahlberg, General Manager, Projects
R. Dean, Engineering Manager, Unit 2
A. DeGracia, Work Control & Outage Manager, Unit 1
G. Helker, Work Control & Outage Manager, Unit 2
M. McCormick, Vice President, Nuclear Engineering
L. Pisano, Maintenance Manager, Unit 2
N. Rademacher, Plant Manager, Unit 1
R. Smith, Operations Manager, Unit 2
P. Smalley, Radiation Protection Manager, Unit 1
K. Sweet, Technical Support Manager, Unit 1
R. Sylvia, Executive Vice President & Chief Nuclear Officer
C. Terry, Vice President, Nuclear Safety Assessment & Support
K. Ward, Technical Support Manager, Unit 2
C. Ware, Chemistry Manager, Unit 2
D. Wolniak, Manager, Licensing
W. Yaeger, Engineering Manager, Unit 1

INSPECTION PROCEDURES USED

IP 37551:	On-Site Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
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 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND UPDATED

OPENED

50-410/96-14-01	URI	Hot Shorts Vulnerability of Shutdown Cooling Valves
50-220 & 50-410/96-14-02	URI	Potential Overpressurization Concerns Relative to NRC Generic Letter 96-06
50-220/96-14-03	VIO	Repeat Failure to Properly Secure High Radiation Area Gate
50-220/96-11	LER	Reactor Scram Caused by the Main Generator Lockout Relay Trip

CLOSED

50-410/95-12-01	URI	Temporary Scaffolding Erected Around Unit 2 Liquid Poison Tank for Extended Period with no Engineering Evaluation
50-220/95-25-01	URI	Emergency Cooling System Material Deficiencies
50-220/96-09	LER	Potential Overstressed Pipe Supports Caused by Design Deficiency

UPDATED

None

LIST OF ACRONYMS USED

ASSS	Assistant Station Shift Supervisor
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CRD	Control Rod Drive
CS	Core Spray
DER	Deviation/Event Report
DWED	Drywell Equipment Drains
DWFD	Drywell Floor Drains
EC	Emergency Cooling
FME	Foreign Material Exclusion
FWP	Feedwater Pump
GE	General Electric
GL	Generic Letter
HMS	Hydrogen Monitoring System
ISEG	Independent Safety Engineering Group
IST	Inservice Testing
kV	kilo-Volt
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MG	Motor Generator
MOV	Motor Operated Valve
mrem/hr	millirem/hour
NMPC	Niagara Mohawk Power Corporation
NRC	Nuclear Regulatory Commission
PAS	Post-Accident Sampling
psi	pounds per square inch

RBCLC	Reactor Building Close Loop Cooling
RCS	Reactor Recirculation System
RFO	Refueling Outage
RHR	Residual Heat Removal
RP	Radiation Protection
RRP	Reactor Recirculation Pump
SDC	Shutdown Cooling
SFP	Spent Fuel Pool
SLC	Standby Liquid Control
SSS	Station Shift Supervisor
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation
WO	Work Order