U.S. NUCLEAR REGULATORY COMMISSION

REGION I

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| Docket/Report Nos.: | 50-220/99-05<br>50-410/99-05 |
| License Nos.: | DPR-63<br>NPF-69 |
| Licensee: | Niagara Mohawk Power Corporation<br>P. O. Box 63<br>Lycoming, NY 13093 |
| Facility: | Nine Mile Point, Units 1 and 2 |
| Location: | Scriba, New York |
| Dates: | May 9, 1999 to June 19, 1999 |
| Inspectors: | G. K. Hunegs, Senior Resident Inspector<br>R. A. Fernandes, Resident Inspector<br>R. A. Skokowski, Resident Inspector<br>J. E. Carrasco, Reactor Engineer |
| Approved by: | Michele G. Evans, Chief<br>Projects Branch 1<br>Division of Reactor Projects |

## EXECUTIVE SUMMARY

**Nine Mile Point Units 1 and 2**
**50-220/99-05 & 50-410/99-05**
**May 9, 1999 - June 19, 1999**

This integrated inspection report includes aspects of licensee operations, engineering, maintenance, and plant support. The report covered a six-week period of resident inspection and the results of an inservice inspection review.

Operations

Unit 1 core reload was well performed with good communications, independent verification, and procedure use noted. (O1.2)

The reactor restart from the Unit 1 refueling outage was conducted in a conservative, well controlled manner. Effective supervision and oversight was provided by senior management. (O1.3)

Maintenance

The reactor core isolation cooling (RCIC) system trip encountered during surveillance testing was the result of a poorly developed system flushing methodology. The subsequent on-line RCIC system maintenance outage was not effectively and efficiently executed to ensure the system unavailability time was minimized. NMPC's root cause determination for the RCIC turbine trip was reasonable and the corrective actions appropriately implemented and documented in the associated deficiency event reports. (M1.2)

Acceptable control of the technical details and appropriate oversight of the contractor performing the non-destructive examinations (NDE) of the core shroud at Unit 1 was noted. The contractor used state-of-the-art ultrasonic technology to detect and size weld indications and cracks. The contractor used acceptable means for the interpretation of the NDE data and the NDE personnel were determined to have been properly certified. (M2.1)

During the refueling outage for Unit 1, appropriate reviews of the indications detected in the recirculation piping safe-end to elbow and nozzle to safe-end welds were performed. (M2.2)

During the Unit 1 reactor vessel hydrostatic test, a leak developed in the reactor vessel bottom head drain line. The cause was determined to be thermal stress induced fatigue which was caused by a system valve packing leak onto the adjacent downstream piping. The inspectors noted that the valve packing leakage was a long-standing material condition problem, the consequence of which was not fully recognized until the crack was identified, analyzed, and repaired. NMPC's corrective actions were acceptable. (M2.3)

Executive Summary (cont'd)

Engineering

Inspection of core shroud vertical and horizontal weld inspections at Unit 1 showed that required structural margins were satisfied. However, inspection results for the V10 weld showed some crack depth change. NMPC decided to pre-emptively repair the V9 and V10 welds using a contingency repair which was previously approved by the NRC. The installation of the repair clamp was well controlled. (E1.1)

A core shroud tie rod upper spring assembly repair at Unit 1 was well conducted. A team approach to develop a repair plan, good utilization of mock-up training, and good radiological controls practices were noted by the inspectors. (E1.2)

On May 18, while performing work on the Unit 1 refuel floor, the reactor building hoist trolley connection failed. The apparent cause of the failure was fatigue of the threaded rod connection. Previously conducted crane inspections were not sufficient to identify the equipment degradation and long-term corrective actions from a February 1988 failure had not been effective. (E1.3)

Plant Support

Radiological controls during the Unit 1 outage were good. Protective clothing, dosimetry and radiological posting requirements and radiation protection technician oversight were effective in minimizing personnel exposure. (R1.1)

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<u>Report Details</u>

<u>Summary of Plant Status</u>

Nine Mile Point Unit 1 (Unit 1) began the inspection period in cold shutdown in a scheduled refueling outage. Unit 1 restarted on June 14. The plant was at 80% power by the end of the inspection period. Major outage activities, in addition to refueling, included the repairs of the core shroud vertical welds and core shroud tie rod, inspection of the reactor vessel beltline, replacement of two feedwater heaters, and modification of the emergency core cooling system suction strainers.

Nine Mile Point Unit 2 (Unit 2) began the period at 65 percent power following a forced outage and subsequent single recirculation loop operation. On May 9, Unit 2 was returned to two loop operation and reached 100 percent power on May 11. The unit remained at 100 percent power through the remainder of the inspection period.

# I. Operations

## O1 Conduct of Operations [1]

### O1.1 General Comments (71707)

Using NRC Inspection Procedure 71707, the resident inspectors conducted frequent reviews of ongoing plant operations. The reviews included tours of accessible 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 (TSs), and verification that logs and records accurately reflected equipment status. In general, the conduct of operations was professional and safety-conscious.

### O1.2 Core Reload Activities (Unit 1)

a. <u>Inspection Scope (71707)</u>

The inspectors observed portions of the core reload to verify that fuel movements were done in accordance with station procedures and Technical Specifications.

b. <u>Observations and Findings</u>

The core reload was performed in accordance with fuel handling procedures N1-FHP-27B, Whole Core Reload, and N1-FHP-25, General Description of Fuel Moves. The inspector observed fuel handling operations from the refuel floor, as well as, the control room. The operators utilized good three-way communications and independent verification during the process of reloading the core. Verification of fuel moves was

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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.

independently performed on the refuel bridge, as well as, step verification from the control room.

During the core reload, operators noted that one of the two refueling mast cables was in a degraded condition and ceased fuel moves. The cabling and cable handling equipment was inspected and subsequently repaired. The inspector reviewed the work order and post work testing and found them to be acceptable. The inspector noted that the discovery of the degraded cable was good and the repair was completed satisfactorily.

c. Conclusions

Unit 1 core reload was well performed with good communications, independent verification, and procedure use noted.

O1.3 Post Outage Startup (Unit 1)

a. Inspection Scope (71707)

The inspectors observed reactor startup activities following the refueling outage. This review included the conduct of operations, resolution of plant problems, and management oversight.

b. Observations and Findings

The reactor startup was conducted in a conservative, well controlled manner. Pre-evolution briefs were thorough and a safety focus was emphasized. Operators were aware of the status of testing and properly addressed identified deficiencies. During the approach to criticality, the reactor went critical on a control rod double notch. Operators responded appropriately by inserting the control rod and changing the pull sheet to continue the startup. Throughout the reactor restart evolution, senior Niagara Mohawk Power Corporation (NMPC) managers provided oversight of activities.

c. Conclusions

The reactor restart from the Unit 1 refueling outage was conducted in a conservative, well controlled manner. Effective supervision and oversight was provided by senior management.

**08 Miscellaneous Operations Issues (92700)**

O8.1 (Closed) Licensee Event Report (LER) 50-410/99-05: Reactor Trip Due to a Main Generator Protection Volts/Hertz Relay Failure. The technical issues associated with this LER were described in NRC inspection report 50-410/99-04, Sections O1.2, M2.2, and E1.1. The inspectors completed an on site review of the LER and verified that the report was completed in accordance with the requirements of 10CFR50.73. Specifically, the description and analysis of the event as documented 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.

The inspector noted that the reactor core isolation cooling (RCIC) system post-maintenance and surveillance testing following the 1998 outage did not identify that the mechanical linkage for the turbine trip and throttle valve was misadjusted. Plant staff troubleshooting revealed that the trip throttle valve overspeed trip mechanism was improperly set-up to ensure proper long-term engagement of the trip hook and latch lever (reference Non-Cited Violation 50-410/99-04-02). Because of the misadjustment, the trip latch was only nominally engaged, but satisfactorily functioned during testing. However, during the event the excessive engagement tolerance coupled with normal system operating vibration caused the trip throttle valve to unlatch and close. NMPC has revised the periodic RCIC test procedure to include the proper trip mechanism tolerances and verification of proper trip latch engagement. This LER is closed.

## II. Maintenance

### M1 Conduct of Maintenance

M1.1 General Comments (61726, 62707)

Using NRC Inspection Procedures 61726 and 62707, the resident inspectors periodically observed various maintenance activities and 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. In general, maintenance and surveillance testing 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:

- WO 99-08931, Gas Treatment System
- Surveillance Test (ST) Q27, Reactor Building Closed Loop Cooling Check Valve Operability Test
- ST M3, Suppression Pool Drywell Relief Valve Exercise
- N1-PM-V7, Turbine Trip Test
- N2-OSP-ICS-Q002, Reactor Core Isolation Cooling Test

M1.2 Reactor Core Isolation Cooling (RCIC) System Maintenance (Unit 2)

a. Inspection Scope (61726)

During routine testing to support returning the RCIC system to service following an on-line maintenance outage, the turbine driven pump tripped on low suction pressure following the changing of the pump's water supply. NMPC remained in the fourteen-day limiting condition for operation (LCO) outage to evaluate, troubleshoot, and make any necessary repairs to address this issue. The inspectors reviewed NMPC's activities to

evaluate the effectiveness of corrective actions and to ensure that the system was being tested and operated consistent with station procedures.

b. Observations and Findings

The portion of the test that was being performed at the time of the RCIC system trip involved the swapping of the water supply (pump suction) from the suppression pool back to the condensate storage tank (CST). The system suction is normally aligned to the CST and low water level in the CST causes the system to automatically swap-over to the suppression pool. This function was tested successfully. However, to flush the system with clean water, the test procedure directed the operators to re-align the suction back to the CST. Shortly after opening the suction valve the RCIC pump tripped on low suction pressure.

NMPC assembled a few teams to investigate the issues surrounding this RCIC system trip. Troubleshooting included: instrument venting and calibration; installation of system performance monitoring equipment; test procedure changes and additional testing requirements; and inspection of several system check valves. NMPC's investigation determined that because of system configuration, voids formed in the suction piping from the CST while the RCIC pump was aligned and drawing water from the suppression pool. Upon suction swap-over back to the CST, the voids collapsed and caused a rapid pressure transient. This pressure transient dropped low enough to cause the pump to trip on low suction pressure. NMPC determined that the pressure transient was further amplified by the unsatisfactory performance of a check valve in the suction piping of the keep-fill pump. The inspectors concluded that NMPC's root cause determination was reasonable and that the associated Deficiency Event Reports (DERs) properly documented the results and corrective actions. However, it appeared that the licensee introduced this RCIC system problem via a poorly researched and reviewed surveillance test procedure change for flushing the system piping using the CST water.

The inspectors noted that the control room operators made a 10 CFR 50.72 notification (Event No. 35706) on May 12, 1999, identifying a preliminary determination that the RCIC system was inoperable because of the system trip on suction swap-over during testing. The licensee subsequently determined that a successful suction swap-over from the suppression pool back to the CST was not a system design requirement. Changes were made to the surveillance procedure to perform an alternate method of flushing system piping after pumping water from the suppression pool. Consequently, the licensee concluded that the RCIC system was not inoperable as a result of the trip on suction swap-over from the suppression pool to the CST. On June 9, 1999, NMPC retracted their May 12, 1999, event notification. The inspectors reviewed the basis for the retraction and found it acceptable.

The inspectors observed that the operators experienced difficulty in performing post-work testing after system restoration from the internals inspections of the RCIC system check valves. During the post-work test, the RCIC pump lost flow and was manually tripped from the control room. Subsequent review and investigation by NMPC determined that the system piping was not adequately filled and vented. Licensee

investigation identified that the system operating procedure did not provide adequate direction for filling the system following an extensive system breach. The inspectors concluded that, in addition to the procedural inadequacies, the work control process could have been more thorough with regards to system restoration following this type of intrusive maintenance.

The inspectors noted that the licensee used 12 days of the available 14-day LCO action statement to complete the necessary maintenance and restore the RCIC system to an operable status. The licensee's internal guidance recommends on-line maintenance be limited to 50 percent of the available LCO time, to account for any unforeseen contingencies. Although the RCIC system suction swap-over trip and subsequent check valve internals inspections contributed to the lengthening of the outage, these events occurred early in the LCO outage window and the 50 percent target was not achieved. The inadequate system refill and venting contributed to this delay. Accordingly, the licensee's processes for efficiently and effectively resolving these RCIC system problems appeared to have been challenged, and thus adversely impacted the availability of a system important to safety. The plant management acknowledged this observation and shared the inspectors' concern for safety system availability.

Subsequent to this inspection period, additional issues were identified with the RCIC system following a June 24, 1999 automatic reactor shutdown. NRC review of these issues will be documented in NRC IR 50-220 & 50-410/99-06.

c. Conclusions

The reactor core isolation cooling (RCIC) system trip encountered during surveillance testing was the result of a poorly developed system flushing methodology. The subsequent on-line RCIC system maintenance outage was not effectively and efficiently executed to ensure the system unavailability time was minimized. NMPC's root cause determination for the RCIC turbine trip was reasonable and the corrective actions appropriately implemented and documented in the associated deficiency event reports.

## M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Inspection of Core Shroud Vertical Welds (Unit 1)

a. Inspection Scope and Background (73753)

The inspector reviewed and assessed the adequacy of the In-Service Inspection (ISI) examinations of the vertical welds of the core shroud during refueling outage 15 (RFO15).

The core shroud is a stainless steel cylinder that surrounds the active core and provides a barrier to separate the upward flow of coolant through the core from the downcomer feedwater inlet and recirculation flow. A loss of structural integrity of the core shroud could potentially result in the loss of core geometry and inability to maintain proper alignment of the fuel. The event that could trigger this consequence is a main steam line

break accident and the complete failure of shroud horizontal welds H4 and H5 and vertical welds V9 and V10. This event coupled with a seismic event could potentially cause a deflection of the fuel rods, which may prevent rod insertion.

b. Observations and Findings

At Unit 1, the core shroud horizontal and vertical welds have been inspected and determined to have intergranular stress corrosion cracking (IGSCC) in and near the heat-affected zone (HAZ) of the welds. To address the horizontal weld cracking, the NRC staff reviewed and approved the licensee's alternative repair method involving the installation of core shroud stabilizer assemblies (tie-rods). With the tie-rods installed, the licensee is no longer obligated to examine the horizontal welds per their ISI program. The ISI examination interval for the vertical welds was established based on NMPC engineering analysis of the existing cracks and consideration for potential crack growth.

The inspector verified that the licensee had completed the scanning of all the vertical welds and the pre-selected intersections between the vertical and the horizontal welds. The inspector observed some of the data interpretation performed by the contractor. The inspector also reviewed the results of the ultrasonic (UT) examination and the comparison of these results to the UT examination results of the previous refueling outage (RFO14). During RFO14, the contractor (GE) used the Smart2000 computerized data acquisition and imaging system, and a multiple probe in a single housing that utilized a 45 degree shear, a 60 degree longitudinal, and a creeping wave. During RFO15, the contractor (Framatome Technologies) used the Accusonex computerized data acquisition system and probes consisting of 45 degree, 60 degree, and 80 degree.

During RFO14, the shroud ring vertical welds (outside surfaces) were inspected using enhanced visual techniques (EVT1). Because the vertical welds were machine flashed, some of the welds were not located and consequently not inspected (i.e., welds V15 and V16). However, NMPC did commit to the NRC staff to develop a technique to locate these welds and inspect them during RFO15. Of the welds inspected in RFO14 using EVT1 methodology only, no cracks were identified. During RFO15, NMPC satisfied their commitment. Using ultrasonic testing (UT) methodology, examination data of the accessible segments of shroud ring vertical welds V1, V2, V5, V6, V13, V14, V15, and V16 showed no cracks.

The inspector observed that during RFO14, vertical welds V7 and V8 were UT inspected with a coverage of about 50% of the weld length and the results showed that there were no indications. Re-examination during RFO15 identified no cracking. Vertical welds V3 and V4 were UT inspected in RFO14 with an inside diameter crack identified in weld V4. This V4 crack was analyzed and dispositioned as acceptable per Boiling Water Reactor Vessel Inspection Program (BWRVIP) criteria. Vertical weld V3 examination results showed a few small indications that were dispositioned as acceptable. During RFO15, re-examination of welds V3 and V4 with better UT coverage identified acceptable results.

Vertical welds V9 and V10 were UT and EVT1 examined (both inside and outside diameter) in RFO14. The cracks identified in these welds were analyzed and determined

to be acceptable for one cycle of operation. During RFO15, these welds were re-examined via UT and V9 had only minor changes, compared to RFO14. However, weld V10 demonstrated a significant change in depth. The average crack growth of V10 was determined to be 0.25 inch. This translated into a crack growth rate of 1.72E-5 inch/hr, which was less than the specified 2.2E-5 inch/hr NMPC acceptance criterion. While a crack growth rate of 1.72E-5 inch/hr would have been acceptable for one more cycle of operation, NMPC conservatively decided to repair vertical welds V9 and V10.

The inspector noted that the licensee used UT to examine intersections between horizontal and vertical welds for welds V9, V10, V3, and V4. In addition, 6 to10 inches of the base metal was examined to ensure the quality of the base metal on each side of the vertical welds inspected.

c. Conclusions

Acceptable control of the technical details and appropriate oversight of the contractor performing the non-destructive examinations (NDE) of the core shroud at Unit 1 was noted. The contractor used state-of-the-art ultrasonic technology to detect and size weld indications and cracks. The contractor used acceptable means for the interpretation of the NDE data and the NDE personnel were determined to have been properly certified.

M2.2 Recirculation Piping Weld Examinations (Unit 1)

a. Inspection Scope and Background (73753)

In 1983, the recirculation piping was replaced at Unit 1 due to extensive intergranular stress corrosion cracking (IGSCC) in pipe welds and safe-ends. The cause of cracking was determined to have been an aggressive water chemistry environment along with weld and furnace sensitized stainless steel components and weld residual stresses. During this inspection, the inspector assessed the RFO15 ultrasonic inspections performed on reactor recirculation system (RRS) pipe welds. The inspector reviewed the pertinent drawings and records and conducted interviews with ISI and engineering personnel engaged in the NDE of the reactor recirculation piping welds.

b. Observations and Findings

The inspector noted that two safe-end to elbow welds (32-WD046, loop 12 and 32-WD086, loop 13) were identified with circumferential indications near the weld root that exceeded the acceptance criteria in the American Society of Mechanical Engineers (ASME) Code, Section XI, paragraph IWB-3514.3. As required by the ASME Code, the licensee performed expanded scope inspections of RRS pipe welds and identified rejectable indications in two additional welds (32-WD126, loop14 and 32-WD168, loop15). The inspector verified that these rejectable weld indications were properly reported in Deviation/Event Report (DER) No. 1-1999-1255, dated May 13, 1999. The inspector determined that the disposition of this DER also addressed welds 126 and 168, which were reported under DERs 1-1999-1411 and 1-1999-1559, respectively.

Following the identification of these rejectable indications, NMPC performed a review of the weld inspection history. As documented in DER 1-1999-1255, the 1983 replacement fabrication records were examined to determine the extent and location of repairs in these welds. Based on this records examination and comparison with the new UT data, NMPC concluded that these indications were lack of fusion from prior repairs and none were indication of IGSCC. Alternatively, these indications were characterized as construction induced, not service induced. Accordingly, these rejectable indications were evaluated and determined "accept-as-is," in accordance with the criteria contained in the ASME Code, Section XI, Subsection IWB 3600. The inspector confirmed that NMPC plans to submit to the NRC the results of the analysis associated with the acceptability of the safe-end to elbow indications, in accordance with the reporting requirements of ASME Code, Section XI, Subsection IWB-3600.

During RFO15, NMPC performed UT examinations of the safe-end to nozzle welds and identified indications on one RRC pipe suction nozzle. NMPC dispositioned these indications as "acceptable" per ASME Code, Section XI, Paragraph IWB 3500.

c. Conclusions

During the refueling outage for Unit 1, appropriate reviews of the indications detected in the recirculation piping safe-end to elbow and nozzle to safe-end welds were performed.

M2.3 Reactor Vessel Bottom Head Drain Line Leak (Unit 1)

a. Inspection Scope

During the performance of the reactor vessel hydrostatic test, a leak was identified in the reactor vessel drain line. The inspector performed a partial system walkdown, discussed the leakage with NMPC personnel and reviewed the corrective actions.

b. Observations and Findings

On June 6, during the vessel hydrostatic test, a leak was identified in the reactor vessel bottom head drain line downstream of the manual isolation valve. The leak was from a crack located on the top of the pipe approximately one inch from the pipe to valve socket weld. The vessel hydrostatic test was secured and the plant was depressurized. NMPC installed freeze seals to facilitate removal and replacement of the affected section of pipe.

A vendor laboratory analysis showed that the crack was typical of fatigue cracking. The cracking was concentrated on the outside diameter surface on the top of the piping. In addition, it was determined that poor weld fit up contributed to high stress at the weld. The cracking was caused by the direct surface exposure of the pipe to leakage from the adjacent manual isolation valve packing. Review of operational history identified that the manual isolation valve had exhibited packing leakage during several operational cycles. The long-time leakage onto the pipe was evidenced by the discoloration and deposits

built-up on the pipe surface. In hindsight, this valve packing leak had not been appropriately addressed.

NMPC documented their corrective actions in DER 99-1907. A walkdown was performed of the remaining sections of the drain line piping and no discrepancies were identified. A temporary modification was installed to shield the new piping from possible future packing leakage from the adjacent valve. From a risk perspective, the NRC staff concluded that a catastrophic break in the drain line (at power) would be significant. In particular, any efforts to isolate the postulated pipe break would be difficult, if at all possible, due to the only isolation valve upstream of the postulated break being manually operated. Absent a means to isolate this postulated pipe break, long-term reactor water inventory control may have to be achieved via containment flood-up.

c. Conclusions

During the Unit 1 reactor vessel hydrostatic test, a leak developed in the reactor vessel bottom head drain line. The cause was determined to be thermal stress induced fatigue which was caused by a system valve packing leak onto the adjacent downstream piping. The inspectors noted that the valve packing leakage was a long-standing material condition problem, the consequence of which was not fully recognized until the crack was identified, analyzed, and repaired. NMPC's corrective actions were acceptable.

## III. Engineering

### E1 Conduct of Engineering

E1.1 Core Shroud Vertical Weld Repair (Unit 1)

a. Inspection Scope (37551)

The inspector reviewed the safety evaluation related to the alternative repair of the core shroud vertical welds. Portions of the electric discharge machining (EDM) process and installation of the clamp were observed and the post repair inspection plan and results were reviewed.

b. Observations and Findings

During the 1997 refueling outage, NMPC identified that some vertical welds joining sections of the cylindrical stainless steel reactor core shroud were cracked. Core shroud weld inspections which were conducted this outage showed that vertical weld V9 remained essentially unchanged from the previous outage and some crack growth was evident for weld V10. NMPC concluded that the crack growth rate was consistent with their previous analyses and that the reactor core shroud continued to be structurally sound. (See Section M2.1)

Based on the results of the examination of the reactor core shroud and analysis, NMPC determined that shroud vertical weld repairs were warranted. Contingency shroud vertical weld repair plans were submitted to and approved by the NRC in a letter dated April 30, 1999: The repair is a clamp assembly consisting of a plate with attached pins that are inserted into holes, machined in the shroud by an EDM process on both sides of the vertical weld. The clamps bridge across the flawed vertical weld. Two clamps each were used for the V9 and V10 welds. Procedures, quality assurance oversight and controls were sufficient to support proper installation of each repair clamp.

c. Conclusions

Inspection of core shroud vertical and horizontal weld inspections at Unit 1 showed that required structural margins were satisfied. However, inspection results for the V10 weld showed some crack depth change. NMPC decided to pre-emptively repair the V9 and V10 welds using a contingency repair which was previously approved by the NRC. The installation of the repair clamp was well controlled.

E1.2 Core Shroud Tie Rod Upper Spring Assembly Repair (Unit 1)

a. Inspection Scope (37551)

During routine inspection of the core shroud tie rod assemblies NMPC discovered that a fastener had become dislodged from one of the four assemblies. The inspector reviewed NMPC's corrective actions and root cause evaluation for the failure of the fastener.

b. Observations and Findings

The fastener was a socket head cap screw located in the upper spring assembly. NMPC's preliminary investigation determined that the most likely failure mechanism was stress corrosion failure under high stress (thermal induced) resulting in part from the different materials used. The inspector observed the staging of a mock-up fixture on the refuel floor and subsequent repair work. The inspector noted good radiological and quality assurance support. The repair personnel were utilizing good as-low-as-reasonably-achievable (ALARA) and contamination controls in carrying out the task. Procedures were properly used and mechanics utilized machined fixtures to increase the accuracy of the repairs.

c. Conclusions

A core shroud tie rod upper spring assembly repair at Unit 1 was well conducted. A team approach to develop a repair plan, good utilization of mock-up training, and good radiological controls practices were noted by the inspectors.

E1.3 Reactor Building Crane Auxiliary Hoist (Unit 1)

a. Inspection Scope (37551)

On May 18, while performing work on the refuel floor, NMPC personnel observed that one of the four hangers supporting the reactor building crane auxiliary hoist had failed. The inspector reviewed NMPC's corrective actions and equipment maintenance history.

b. Observations and Findings

The reactor building crane auxiliary hoist is mounted to the underside of the reactor building crane by four threaded rod supports. The load is transmitted from the auxiliary hoist to the reactor building crane by two spherical machined nuts threaded onto the rod, and load bearing on an upper and lower piece of channel iron. In this particular case, the second support from the north end of the crane failed. NMPC's immediate corrective actions included stopping work on the refuel floor and processing a temporary modification to support the auxiliary hoist. The inspector reviewed the temporary modification and concluded that the actions taken to temporarily support the load were acceptable.

The inspector determined that, although, NMPC has a procedure for inspecting the auxiliary hoist, it lacked clarity and did not provide for inspection of the threaded rod supports. The design of the refuel floor is such that portions of the auxiliary hoist cannot be readily inspected without extensive scaffolding. The support hanger that failed had not been inspected.

Inspector follow-up determined that one of the supports had failed in February 1988. The failure occurred following the mis-operation of the reactor building crane when the bridge operator mistakenly went east instead of west with the main trolley. At the time, the crane was already near the end of the track and it's movement caused the bridge to strike the rail end stops, with the subsequent failure of the auxiliary hoist trolley support and some structural welds. NMPC determined the root cause of the February 1988 failure to be fatigue as a result of cyclic loading. The apparent cause of the recent failure was also determined to be fatigue. In addition to weld repairs and replacement of the trolley supports, the recommended corrective actions included structural engineering review of the attachment design and recommendations for a long term modification. The inspector concluded, that, based on the recent failure that the long term corrective actions were ineffective.

c. Conclusions

On May 18, while performing work on the Unit 1 refuel floor, the reactor building hoist trolley connection failed. The apparent cause of the failure was fatigue of the threaded rod connection. Previously conducted crane inspections were not sufficient to identify the equipment degradation and long-term corrective actions from a February 1988 failure had not been effective.

## E8 Miscellaneous Engineering Issues (92712)

E8.1 (Closed) LER 50-410/99-01 Supplement 1: Unit 2 Outside Design Basis Due to Safe Shutdown Service Water Pump Bay Unit Coolers Being Out-of-Service.

The technical issues associated with this LER were described in NRC inspection report 50-410/99-03, Section E1.3. Supplement 1 provided additional information regarding NMPC's corrective actions. The inspectors completed an in-office review of the additional information provided in the LER and found it to be acceptable. This LER is closed.

E8.2 Review of Year 2000 Program and Implementation

During this inspection period, a review was conducted of Nine Mile Point's year 2000 (Y2K) activities and documentation using Temporary Instruction (TI) 2515/141, "Review of Year 2000 Readiness of Computer Systems at Nuclear Power Plants." The review addressed aspects of Y2K management planning, documentation, implementation planning, initial assessment, detailed assessment, remediation activities, Y2K testing and validation, notification activities, and contingency planning. The reviewers used NEI/NUSMG 97-07, "Nuclear Utility Year 2000 Readiness," and NEI/NUSMG 98-07, "Nuclear Utility Year 2000 Readiness Contingency Planning," as the primary references for this review.

The results of this review will be combined with similar reviews of Y2K programs at other U.S. commercial nuclear power plants and summarized in a report to be issued by the NRC staff by July 31, 1999.

# IV. Plant Support

## R1 Radiological Protection & Chemistry Controls

R1.1 Refuel Outage Radiological Controls (Unit 1)

a. Inspection Scope (71750)

The inspectors observed radiological work practices and controls during the Unit 1 refueling outage including protective clothing and personal dosimeter use, and radiological postings.

b. Observations and Findings

During the outage, the inspectors noted that good radiation protection controls were in effect. The inspectors noted that protective clothing was properly used and dosimetry was properly worn. Radiological boundaries were clearly defined and posted. Radiation protection technicians were actively providing oversight to help minimize personnel exposure. The inspectors noted that an ALARA goal of 280 person rem was set for RFO

15. (See NRC IR 99-04). Actual outage exposure was 330 person rem with the increase due to emergent work.

c. Conclusions

Radiological controls during the Unit 1 outage were good. Protective clothing, dosimetry and radiological posting requirements and radiation protection technician oversight were effective in minimizing personnel exposure.

## V. Management Meetings

### X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on July 15, 1999. The licensee acknowledged the findings presented.

ATTACHMENT 1

## PARTIAL LIST OF PERSONS CONTACTED

Niagara Mohawk Power Corporation

| | |
|---|---|
| D. Bosnic | Manager, Operations, Unit Two |
| S. Doty | Manager, Maintenance, Unit One |
| N. Paleologos | Plant Manager, Unit Two |
| F. Fox | Manager, Maintenance, Unit Two |
| R. Smith | Plant Manager, Unit One |
| N. Rademacher | Manager, Quality Assurance |
| D. Topley | Manager, Operations, Unit One |

## INSPECTION PROCEDURES USED

| | |
|---|---|
| IP 37551 | On-Site Engineering |
| IP 61726 | Surveillance Observations |
| IP 62707 | Maintenance Observations |
| IP 71707 | Plant Operations |
| IP 71750 | Plant Support |
| IP 73753 | Inservice Inspection |
| IP 92700 | Onsite Follow-up of Written Reports of Non-Routine Events at Power Reactor Facilities |
| IP 92712 | In-office Review of Written Reports of Non-Routine Events at Power Reactor Facilities |

## ITEMS OPENED, CLOSED, AND UPDATED

**CLOSED**

| | | |
|---|---|---|
| 50-410/99-05 | LER | Reactor Trip Due to a Main Generator Protection Volts/Hertz Relay Failure |
| 50-410/99-01, Sup 1 | LER | Unit 2 Outside Design Basis Due to Safe Shutdown Service Water Pump Bay Unit Coolers Being Out-of-Service |

## LIST OF ACRONYMS USED

| | |
|---|---|
| ALARA | As Low As Reasonably Achievable |
| ASME | American Society of Mechanical Engineers |
| CST | Condensate Storage Tank |
| DER | Deviation/Event Report |
| EDM | Electric Discharge Machining |
| ESF | Engineered Safeguards Feature |
| EVT1 | Enhanced Visual Techniques |
| GE | General Electric |
| HAZ | Heat-Affected Zone |
| IGSCC | Intergranular Stress Corrosion Cracking |
| IR | Inspection Report |
| ISI | In-Service Inspection |
| LCO | Limiting Condition for Operation |
| LER | Licensee Event Report |
| NCV | Non Cited Violation |
| NDE | Nondestructive Examination |
| NMPC | Niagara Mohawk Power Corporation |
| NRC | Nuclear Regulatory Commission |
| RCIC | Reactor Core Isolation Cooling System |
| RFO14 | Refueling Outage Number Fourteen |
| RFO15 | Refueling Outage Number Fifteen |
| RRS | Reactor Recirculation System |
| TI | Temporary Instruction |
| TS | Technical Specification |
| Unit 1 | Nine Mile Point Unit 1 |
| Unit 2 | Nine Mile Point Unit 2 |
| UT | Ultrasonic |
| WO | Work Order |
| Y2K | Year 2000 |