U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket/Report Nos.:	50-220/98-14 50-410/98-14
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:	August 16 - September 26, 1998
Inspectors:	B. S. Norris, Senior Resident Inspector R. A. Fernandes, Resident Inspector, FitzPatrick G. K. Hunegs, Senior Resident Inspector, FitzPatrick R. A. Skokowski, Resident Inspector

EXECUTIVE SUMMARY

Nine Mile Point Units 1 and 2
50-220/98-14 & 50-410/98-14
August 16 - September 26, 1998

This NRC inspection report includes reviews of licensee activities in the functional areas of operations, engineering, maintenance, and plant support. The report covers a six-week period of inspections and reviews by the Nine Mile Point and FitzPatrick resident staffs.

OPERATIONS

In general, the conduct of operations was professional and safety-conscious. During the period, the inspectors noted improved attention-to-detail on the part of Unit 1 operators, especially the licensed control room operators, in the areas of shift briefings, routine communications, and the use of procedures.

MAINTENANCE/SURVEILLANCE

NMPC effectively modified the Unit 1 average power range monitors to account for thermal hydraulic instabilities, as required by NRC Generic Letter 94-02. The inspectors noted that the modification was completed, as designed, the work package and safety evaluation were thorough, and the training provided to the control room operators was acceptable.

During a Unit 2 surveillance test of the Division I standby liquid control system (SLS), operators discovered that the Division II pump suction manual isolation valve was locked closed vice locked open. This resulted in both divisions of SLS being declared inoperable. This issue remains open pending inspector review of NMPC's completed root cause analysis and determination of corrective actions to prevent recurrence. **(EEI 50-410/98-15-01)**

During planned maintenance on the Unit 1 containment spray system, NMPC discovered that primary containment integrity had been breached due to an inadequate boundary valve markup. Subsequent review by NMPC identified that in 1994 a similar condition existed, but was also not recognized as a breach of primary containment integrity. In both cases, NMPC's failure to maintain primary containment integrity resulted in leakage in excess of that allowed by the Unit 1 Technical Specifications, Section 3.3.3.a. These licensee identified and corrected Technical Specification non-compliances are being treated as a non-cited violation. **(NCV 50-220/98-14-02)**

ENGINEERING

NMPC determined that several Unit 2 pipe welds had not been examined within the appropriate time interval between the first and third refueling outages, as required by Generic Letter 88-01 and the Technical Specifications. This licensee identified and corrected violation was of minor significance and not subject to formal enforcement action.

Since initial plant startup, the Unit 2 offgas pre-treatment radiation monitors had been set non-conservatively because the associated procedure improperly reduced the conversion factor to eliminate the effect of short-lived isotopes. Although the offgas system would not have isolated at the Technical Specification value, several related alarms would have provided the operators sufficient warning and allowed for timely operator action to effectively mitigate the consequence of high activity in the offgas system. Upon identification, NMPC took prompt and appropriate corrective actions. This licensee identified and corrected offgas radiation monitor Technical Specification non-compliance is being treated as a non-cited violation. **(NCV 50-220/98-14-03)**

PLANT SUPPORT

In general, the performance in the area of plant support was professional and safety conscious.

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REPORT DETAILS

Nine Mile Point Units 1 and 2
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SUMMARY OF ACTIVITIES

Niagara Mohawk Power Corporation (NMPC) Activities

Both units maintained essentially full power throughout the inspection period.

Nuclear Regulatory Commission (NRC) Staff Activities

Inspection Activities

The NRC inspectors conducted inspection activities during normal, backshift, and deep backshift hours.

Updated Final Safety Analysis Report Reviews

While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the Updated Final Safety Analysis Report (UFSAR). The inspectors verified that the UFSAR descriptions were consistent with the observed plant practices, procedures, and/or parameters.

I. OPERATIONS

O1 Conduct of Operations

O1.1 General Comments (71707)[1]

Using NRC Inspection Procedure 71707, the resident inspectors conducted frequent reviews of ongoing plant operations. The reviews included tours of accessible 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 (TS), and verification that logs and records accurately identified equipment status or deficiencies. In general, the conduct of operations was professional and safety-conscious. During the period, the inspectors noted improved attention-to-detail on the part of Unit 1 operators, especially the licensed control room operators, in the areas of shift briefings, routine communications, and the use of procedures.

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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.

O8 Miscellaneous Operations Issues

O8.1 (Closed) URI 50-220 & 50-410/97-04-06: SORC Review of TS Violations (92901)

In June 1997, the NRC identified that the Unit 1 and Unit 2 Station Operating Review Committees (SORCs) were not reviewing all TS violations, as required by the TS 6.5.1.6. At that time, the inspectors questioned whether this included all procedural non-compliances which were cited as violations of TS 6.8.1. This was left as an unresolved item pending further review by the NRC.

This issue was discussed between the regional staff and the NRC Headquarters Technical Specification Branch, Quality Assurance (QA) Branch, and the Office of Enforcement staffs. The NRC staff agreed that a literal reading of the TS would imply that all procedure violations needed SORC review. However, it was concluded that this was not consistent with the general intent of TS 6.5.1.6. The requirements of the on-site review committee are listed in American National Standards Institute (ANSI) N18.7, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," which is endorsed by the NRC in Regulatory Guide (RG) 1.33, "Quality Assurance Program Requirements (Operation)." ANSI N18.7 states that on-site review committees should review violations that have safety significance.

Based upon the inspectors' broad review of the deviation/event report (DER) database and detailed examination of selected DERs, no significant TS violations were identified by the inspectors that were not reviewed by the SORCs. Consequently, there was no violation of NRC requirements. This unresolved item is closed.

O8.2 (Closed) VIO 50-410/97-02-01: Failure to Implement Unit 2 Control Room Deficiency Program (92901)

In May 1997, the inspectors identified several noncompliances with the Nine Mile Point Unit 2 (Unit 2) control room deficiency program, as described in Procedure N2-ODP-OPS-0001, "Conduct of Operations." The discrepancies were classified as violations of TS 6.8.1, regarding procedure adherence. NMPC's response, dated May 16, 1997, provided the root cause and corrective actions for this violation. The inspectors reviewed this letter, conducted an on-site review of the current control room deficiency program, and concluded that the root cause determination and corrective actions were appropriate. The inspectors verified that the work control computer software was revised to prevent personnel outside of the operations department from changing the tracking code that identifies the deficient condition as a control room deficiency. In addition, the inspectors reviewed the current control room deficiencies and found the list to be accurate and complete. This violation is closed.

II. MAINTENANCE [2]

M1 Conduct of Maintenance

M1.1 General Comments (61726, 62707)

Using NRC Inspection Procedures 61726 and 62707, the resident inspectors periodically observed 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 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 95-4157-03	Thermal Hydraulic Stability Modification - #11 APRM
WO 98-6269	Repair of Electrical Penetration RW-67
N1-CSP-A326	OGESMS Stack Detector #112-07 Calibration Verification
N1-IPM-081-005	Core Spray System Flow Instruments
N1-ISP-092-321	APRM #11 Instrument Channel Calibration/Test
N1-PM-Q11	Primary Containment Manual Valve Check
N1-REP-8	Core Thermal Power
N1-RSP-6Q	Control Room Ventilation Radiation Monitor Instrument Channel Test
N1-RSP-13Q	Stack Radiation Monitor Quarterly Calibration Check & Channel Test
N1-ST-Q6A	Containment Spray Loop #111 Operability Test
N1-ST-Q15	Condensate Transfer System Operability Test
N2-OSP-EGS-M@002	Diesel Generator & Diesel Air Start Valve Operability Test, Division III
N2-OSP-SLS-Q001	Standby Liquid Control Pump, Check Valve, Relief Valve Operability & 40 Month Functional Test
N2-OSP-SLS-Q002	Standby Liquid Control Valve Operability Test
N2-RSP-RMS-R102	Channel Calibration Test of the Main Control Room Area Radiation Monitor
- N2-RSP-RMS-R111	Channel Calibration Test of the Drywell Atmosphere Offline Gas & Particulate Process Radiation Monitors

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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 - Installation of the Thermal Hydraulic Stability Modification per GL 94-02

a. Inspection Scope (61726, 62707)

The inspectors reviewed the modification to the Nine Mile Point Unit 1 (Unit 1) average power range monitors (APRMs) to account for thermal-hydraulic instabilities. The review included observations of some in-field work and the pre-evolution brief, and review of the safety evaluation, work package, post-maintenance acceptance work, and the operations training material.

b. Observations and Findings

During the inspection period, NMPC implemented a modification to the Unit 1 APRM circuitry to incorporate the requirements of NRC Generic Letter (GL) 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors." NMPC treated the modification as a special evolution, in accordance with NMPC Procedure GAP-SAT-03, "Control of Special Evolutions," due to the potential for a significant plant transient (e.g., reactor scram). Special evolutions require direct upper management involvement. Accordingly, the Unit 1 Technical Support Manager was designated the Senior Manager for this modification.

The inspectors observed one of the pre-evolution briefs by the Senior Manager, portions of the actual modification, and portions of the post-modification test (PMT). The pre-evolution briefing was thorough and detailed. The modification work and the PMT were appropriately detailed in the work package and completed, as designed. The oversight by Instrumentation and Controls (I&C) supervision and the system engineer was good. The inspectors reviewed the safety evaluation and the operations training material and found them to be consistent and complete.

c. Conclusion

NMPC effectively modified the Unit 1 average power range monitors to account for thermal hydraulic instabilities, as required by NRC Generic Letter 94-02. The inspectors verified that the modification was properly completed, as designed, the work package and safety evaluation were thorough, and the training provided to the control room operators was acceptable.

M1.3 Unit 2 Standby Liquid Control System Inoperable Due to a Valve Inadvertently Locked Closed (61726, 71707)

On September 11, 1998, during a Unit 2 surveillance of the Division I standby liquid control system (SLS), operators discovered that the Division II pump suction manual isolation valve (2SLS*V46) was locked closed vice locked open. With Division I already inoperable because of the surveillance, and with 2SLS*V46 locked closed, both divisions of SLS were inoperable. The Station Shift Supervisor (SSS) directed that 2SLS*V46 be locked open and independently verified locked open.

Subsequently, NMPC wrote DER 2-98-2730 to document the event and to initiate a formal root cause analysis.

Additional corrective action included a valve line-up verification of select safety systems at both units; no other mispositionings were identified nor was there any indication of tampering. Initial review by NMPC indicated that the valve was last manipulated in August 1998 during a routine surveillance, and that the valve was locked closed since that surveillance. This issue remains open pending completion of the DER disposition, issuance of the associated Licensee Event Report (LER), and review by the inspectors of the root cause and corrective actions.
(EEI 50-410/98-14-01)

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) EEI 50-220/98-09-01: Inadequate Markup Resulted in a Breach of the Unit 1 Primary Containment Integrity (90712, 92902)

On August 4, 1998, during planned maintenance on the Unit 1 containment spray system, NMPC discovered that primary containment integrity had been breached due to an inadequate boundary valve markup. This event was initially discussed in NRC IR 98-09, Section M1.2, and a tracking item assigned pending issuance of the associated LER and inspector review.

As documented in LER 98-15, NMPC's investigation determined that the root cause was complacency on the part of the licensed operators who developed, reviewed, and approved the markup for isolation of the containment spray system. The operators relied on experience and other operators, rather than reviewing the system drawings and the plant impact statements in the work packages. Corrective actions included a briefing for each shift as to the requirements and expectations for markups and enhanced training to include lessons learned from this event. NMPC evaluated the increased containment leakage and noted that the resultant dose to the control room from a design basis accident (DBA) would remain within the limits of 10CFR100. During their review, NMPC identified that in 1994, a similar condition existed, but was likewise not recognized as a breach of primary containment integrity. The NMPC analyses for each event concluded that neither posed a significant safety threat to the public or plant personnel. EEI 50-220/98-09-01 is closed.

The failures to maintain primary containment integrity resulted in leakage in excess of that allowed by the Unit 1 Technical Specifications, Section 3.3.3.a. These licensee identified and corrected violations are being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1 of the NRC Enforcement Policy.
(NCV 50-220/98-14-02)

M8.2 (Closed) LER 50-220/98-15: Breach of Primary Containment Due to Personnel Error (90712)

The inspectors previously reviewed the technical issues associated with this event as documented in NRC IR 98-09. Additional inspector observations are discussed in Section M8.1 of this inspection report. The inspectors completed an in-office review of the LER and considered the root cause and corrective actions to be reasonable. The description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. The LER is closed.

M8.3 (Closed) LER 50-220/98-17: Breach of Primary Containment Due to Personnel Error in 1994 (90712)

The technical issues associated with this LER were described in Section M8.1 of this inspection report. The inspectors completed an in-office review of the LER and considered the root cause and corrective actions to be reasonable. The description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. The LER is closed.

III. ENGINEERING

E1 Conduct of Engineering

E1.1 General Comments (37551)

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

E8 Miscellaneous Engineering Issues

E8.1 (Closed) LER 50-410/98-21: Missed Inservice Inspections Required by TS Caused by Inadequate Change Management

a. Inspection Scope (90712, 92700)

In June 1998, NMPC determined that some Unit 2 inservice inspection (ISI) pipe weld examinations had not been performed, as required. The inspectors discussed the missed ISI examinations with NMPC management, including the engineering manager responsible for ISI, and the Quality Assurance Manager. In addition, the inspectors performed an on-site review of the LER, associated DER for both units, portions of the Unit 2 ISI Program, and the appropriate sections of the ASME [American Society of Mechanical Engineers] Code.

b. Observations and Findings

In June 1998, NMPC determined that ISI examinations had not been completed as required by Unit 2 TS Surveillance Requirement (TSSR) 4.0.5.f, which states that the ISI Program for piping shall be performed in accordance with the schedule requirement of Generic Letter (GL) 88-01. Specifically, GL 88-01 requires all designated Category D pipe welds shall be inspected every two refueling outages. NMPC identified that the examinations of some reactor vessel Category D welds were not conducted during the 18-month operating cycles between Refueling Outage One (RFO1) and RFO3. This was initially identified by NMPC in 1994 during the preparation of the post-RFO3 ISI Summary Report. At that time, NMPC resolved this discrepancy using the rationale that the new 24-month operating cycle time interval could be applied, and thus they were complying with the current Unit 2 TSs. During the recent refueling outage, RFO6, completed June 1998, NMPC identified that this 1994 assumption was incorrect.

All of the Category D weld examinations were completed during RFO4 (1995). Since then, 50 percent of the welds were examined during RFO5 and the other 50 percent were inspected during RFO6. NMPC stated in the LER that all of the weld examination results were acceptable. The failure to complete the required examinations between RFO1 and RFO3 is contrary to TSSR 4.0.5.f. However, this failure constitutes a violation of minor significance and is not subject to formal enforcement action.

The inspectors verified that the LER was completed in accordance with the requirements of 10CFR50.73. Specifically, the description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event, based on the inspection observations discussed above. The root cause and corrective and preventive actions described in the LER were reasonable and appropriate. This LER is closed.

During review of the above event, the inspectors noted that there were several DERs addressing ASME Code required examinations that may not have been completed. Specifically, visual inspections of several piping system structural supports were not performed in the manner prescribed by the ASME Code. The inspectors determined that NMPC's initial interpretation of the ASME Code was that insulation on structural supports could remain installed during visual examinations. NMPC sent a letter to the ASME Code Committee requesting a formal interpretation. The ASME Code Committee ruled that NMPC's interpretation was incorrect, and that the insulation needed to be removed. Subsequently, NMPC issued DERs 1-98-2593 and 2-98-2594 (Units 1 and 2 respectively) to initiate a review of the missed inspections.

For Unit 1, NMPC determined that the required examinations were completed for the 1st and 2nd periods of the current Ten-Year Interval, and that the remaining examinations were planned for the 3rd period (during RFO15, scheduled for Spring 1999). For Unit 2, NMPC determined that some of the inspections had not been performed prior to the end of the First Ten-Year Interval (April 1998). However, the

ASME Code, Section IWA-2430, provides for extending the interval up to 1 year. The inspector observed that NMPC planned to exercise this option and complete the required examinations within the next 12 months. The inspectors found this approach consistent with the ASME Code.

c. Conclusion

NMPC determined that several Unit 2 pipe welds had not been examined within the appropriate time interval between the first and third refueling outages, as required by Generic Letter 88-01 and the Technical Specifications. This licensee identified and corrected violation was of minor significance and not subject to formal enforcement action.

E8.2 (Closed) VIO 50-410/97-03-09: Failure to Install Eight-Hour Battery-Pack Emergency Lighting in the Vicinity of Appendix R Remote Shutdown Equipment (92903)

In May 1997, the inspectors identified that eight-hour battery pack emergency lighting was not provided in the vicinity of the RHR minimum flow valves. Emergency lighting is required because local operation of these valves is necessary for safe shutdown of the plant, in the event of a control room fire requiring evacuation. The inspectors completed an on-site review of the root cause and corrective actions documented in NMPC's August 11, 1997, violation reply letter, and determined the actions to be adequate. NMPC installed the required emergency lighting and completed a broad review of their Appendix R program. The inspectors verified proper installation of the new emergency lighting. In addition, the inspectors reviewed the findings of the licensee's Appendix R program review and confirmed that the findings were being appropriately addressed through NMPC's corrective action program. This violation is closed.

E8.3 (Closed) LER 50-410/98-22: Radioactive Gaseous Effluent Monitoring Instrumentation Set Non-Conservative

a. Inspection Scope (90712, 92700)

On June 23, 1998, while Unit 2 was shutdown for RFO6, NMPC personnel determined that the Unit 2 offgas pre-treatment radiation monitor setpoint had been set non-conservatively since initial plant startup. The inspectors completed an on-site review of the issues associated with this LER. Particularly, the inspectors assessed the licensee's root cause analysis and corrective actions as described in the LER, including a review of the TS, UFSAR, applicable licensee procedures, and Offsite Dose Calculation Manual (ODCM). The inspectors also discussed the issue with the responsible Unit 2 Radiation Protection Supervisor. In addition, the inspectors verified the completion of the LER in accordance with 10CFR50.73.

b. Observations and Findings

During a review of the Unit 2 emergency operating procedures, NMPC personnel determined that the offgas pre-treatment radiation monitors had been set non-conservatively since initial plant startup. As required by TS 3.3.7.10, the alarm/trip setpoints for the offgas pre-treatment radiation monitors are determined and adjusted in accordance with the methodology and parameters provided in the ODCM. In 1986, NMPC Chemistry and Radiation Protection personnel developed procedures to adjust the offgas pre-treatment radiation monitor setpoints. Review of this procedure in June 1998 determined that the setpoint conversion factor used by the radiation monitors was improperly reduced to eliminate the effect of short-lived isotopes and background radiation contributions, which resulted in a non-conservative setpoint. Upon identification, NMPC determined an appropriate conversion factor consistent with the methodology provided in the ODCM and reset the offgas pre-treatment radiation monitor alarm/trip setpoints.

The offgas pre-treatment radiation monitors isolate offgas flow to the main stack in the event of a high radiation condition. In addition, the monitors provide a control room alarm to alert the operators of higher than normal offgas activity. NMPC analyzed the consequence of the event and determined that although the offgas system would not have isolated at the value required by TS, the alarm would have provided the operators sufficient warning and allowed for timely operator action to reduce a monitored radioactive release, in accordance with the alarm response procedure (ARP). Furthermore, changes in offgas activity would have been detected by the main stack gaseous effluent monitor, and other plant radiation monitors, including the main steam line radiation monitors and particular area radiation monitors. The alarms associated with these monitors would have prompted operators to investigate and take action to reduce a monitored radioactive release. Based upon the alarm indications available from the offgas monitors and other radiation monitors, NMPC concluded that timely operator action would have been taken to effectively mitigate the consequence of high activity in the offgas system.

The inspectors reviewed plant procedures and discussed the issue with the responsible Unit 2 Radiation Protection Supervisor and several Unit 2 senior reactor operators (SROs), and determined that NMPC's conclusion was technically sound. Nonetheless, the failure to determine and adjust the offgas pre-treatment radiation monitors in accordance with the methodology and parameters provided in the ODCM is a violation of TS 3.3.7.10. This non-repetitive, licensee identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. (**NCV 50-410/98-14-03**)

The inspectors verified that the LER was completed in accordance with the requirements of 10CFR50.73. Specifically, the description and analysis of the event, as contained in the LER, were consistent with the inspectors' understanding of the event. The root cause and corrective and preventive actions as described in the LER were reasonable. This LER is closed.

c. Conclusion

Since initial plant startup, the Unit 2 offgas pre-treatment radiation monitors had been set non-conservatively because the associated procedure improperly reduced the conversion factor to eliminate the effect of short-lived isotopes. Although the offgas system would not have isolated at the Technical Specification value, several related alarms would have provided the operators sufficient warning and allowed for timely operator action to effectively mitigate the consequence of high activity in the offgas system. Upon identification, NMPC took prompt and appropriate corrective actions. This licensee identified and corrected offgas radiation monitor Technical Specification non-compliance is being treated as a non-cited violation. **(NCV 50-220/98-14-03)**

IV. PLANT SUPPORT

Using NRC Inspection Procedure 71750, the resident inspectors routinely monitored the performance of activities related to the areas of radiological controls, chemistry, emergency preparedness, security, and fire protection. Minor deficiencies were discussed with the appropriate management, significant observations are detailed below.

F2 Status of Fire Protection Facilities and Equipment

F2.1 Off-site Fire Department Causes Automatic Start of Fire Pumps (71750)

On August 22, 1998, with the fire suppression water headers cross-connected, the Scriba Fire Department used a fire hydrant located outside of the protected area to fill the department's tanker truck. This action resulted in the automatic start of the diesel and electric driven fire pumps to maintain normal fire suppression water system pressure. A verbal agreement between the local fire department and NMPC allowed use of the hydrant, provided prior permission was received. In this instance, no permission was granted and the fire pump starts were unanticipated.

The inspectors discussed this event with the Nine Mile Fire Department and were informed that the planned corrective action was limited to a call to the Scriba Fire Department to remind them of the need to request permission prior to using the hydrant. The inspectors were concerned that NMPC's corrective actions were weak, in that, no positive means were instituted to prevent an individual outside of the protected area from inadvertently impacting site fire equipment located inside the protected area. Further discussions with the Supervisor of the Nine Mile Fire Department resulted in more definitive action (i.e., the hydrant was locked closed and a letter was sent to the Scriba Fire Department informing of the actions and the need for prior permission to operate the hydrant). The inspectors considered these subsequent actions to be appropriate.

F8 Miscellaneous Fire Protection Issues

F8.1 (Closed) URI 50-410/97-11-08: Breach Permit Greater than 3 Years Old (92904)

In October 1997, the NRC identified that a fire-door in the radioactive-waste building was removed from its hinges. The associated breach permit was dated September 9, 1994, and stated that the door was removed to allow hoses to pass through the doorway for a temporary modification. At that time, the inspectors questioned: (1) whether the door being removed for over three years was considered in the fire hazards analysis, and (2) whether the excessive time was in essence a permanent modification without the requisite safety evaluation.

During NMPC's disposition of the associated DER, they determined the cause to be due to a misapplication of design inputs during the installation of a plant modification. Specifically, the modification did not include an evaluation of the breached door. A contributing cause was the failure of management to monitor fire protection activities, in that the breach was allowed to remain open for an extended period of time. Corrective actions included a design change to re-route the hoses, the door was re-hung, and the breach permit was closed. Also, fire protection personnel reviewed all existing breach permits at both units, and identified others that were greater than 90 days old. The conditions requiring the permit were either corrected, or the breach was re-evaluated for extension, as allowed by the procedure. Finally, the controlling NMPC Procedure (GAP-FPP-03, "Breach Permit") was revised to include a requirement for a DER to be initiated for any breach permit greater than six months old that was not also part of a temporary modification.

The inspectors performed an on-site review of the associated DER and toured the facility for similar conditions with no additional concerns identified. The inspectors concluded that although the temporary modification and associated breach permit were poorly controlled, no violation of NRC requirements occurred. This unresolved item is closed.

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 October 9, 1998. During this meeting, the resident inspector findings were presented. NMPC did not dispute any of the inspectors findings or conclusions. Based on the NRC Region I review of this report, and discussions with NMPC representatives, it was determined that this report does not contain safeguards or proprietary information.

X3 Management Meeting Summary

X3.1 NRC/NMPC Meeting Related to Extension Request for Inspection of the Unit 1 Core Shroud, Followed by Public Question/Answer Period

On September 24, 1998, the NRC met with NMPC to discuss their request to extend the inspection interval of the Unit 1 core shroud from 10,400 hot operating hours to 14,500 hot operating hours. This equates to an extension from late November 1998, until the next refueling outage scheduled for May 1999. This meeting was conducted at the Oswego Campus of the State University of New York (SUNY) -Oswego and was open for public observation. Following the meeting between the NRC and NMPC, a second meeting was held to receive public comments regarding NMPC's request. The handouts from the NRC - NMPC meeting is included as Attachment 2 to this report.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Niagara Mohawk Power Corporation

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

INSPECTION PROCEDURES USED

IP 37551	On-Site Engineering
IP 61726	Surveillance Observations
IP 62707	Maintenance Observations
IP 71707	Plant Operations
IP 71750	Plant Support
IP 90712	In-Office Review of Written Reports of Non-Routine Events at Power Reactor Facilities
IP 92700	Onsite Follow-up of Written Reports of Non-Routine Events at Power Reactor Facilities
IP 92901	Follow-up - Operations
IP 92902	Follow-up - Maintenance
IP 92903	Follow-up - Engineering
IP 92904	Follow-up - Plant Support

ITEMS OPENED, CLOSED, AND UPDATED

OPENED

50-410/98-14-01	EEI	Standby liquid control inoperable due to a valve inadvertently locked closed
50-220/98-14-02	NCV	Failure to maintain primary containment integrity resulted in leakage in excess of TS allowable
50-410/98-14-03	NCV	Failure to adjust offgas rad monitors per the ODCM

CLOSED

50-220/98-09-01	EEI	Inadequate markup resulted in a breach of primary containment integrity
50-220 & 50-410/97-04-06	URI	SORC reviews of TS violations
50-410/97-02-01	VIO	Failure to implement the Unit 2 CR deficiency program
50-220/98-15	LER	Breach of primary containment due to personnel error
50-220/98-17	LER	Breach of primary containment due to personnel error in 1994
50-410/97-03-09	VIO	Failure to install eight-hour battery-pack emergency lighting in the vicinity of Appendix R remote shutdown equipment
50-410/98-22	LER	Radioactive gaseous effluent monitoring instrumentation set non-conservative
50-220/98-14-02	NCV	Failure to maintain primary containment integrity resulted in leakage in excess of TS allowable
50-410/98-14-03	NCV	Failure to adjust offgas rad monitors per the ODCM
50-410/97-11-08	URI	Breach Permits Greater than 3 Years Old
50-410/98-21	LER	Missed inservice inspections required by TS caused by inadequate change management

UPDATED

none

LIST OF ACRONYMS USED

ASME	American Society of Mechanical Engineers
APRMs	Average Power Range Monitors
ARP	Alarm Response Procedure
CFR	Code of Federal Regulations
DBA	Design Basis Accident
DER	Deviation/Event Report
EEI	Escalated Enforcement Item
ESF	Engineered Safeguards Feature
GL	Generic Letter
IR	Inspection Report

Attachment 1

I&C	Instrumentation and Controls
ISI	Inservice Inspection
LER	Licensee Event Report
NCV	Non-Cited Violation
NMPC	Nine Mile Point Corporation
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
QA	Quality Assurance
RFO	Refueling Outage
RG	Regulatory Guide
RHR	Residual Heat Removal
SLS	Standby Liquid Control System
SORC	Station Operating Review Committee
SRO	Senior Reactor Operator
SSS	Station Shift Supervisor
SUNY	State University of New York
TS	Technical Specification
TSSR	Technical Specification Surveillance Requirement
UFSAR	Updated Final Safety Analysis Report
Unit 1	Nine Mile Point Unit 1
Unit 2	Nine Mile Point Unit 2
VIO	Violation
WO	Work Order

ATTACHMENT 2

HANDOUTS FROM

SEPTEMBER 24, 1998

MEETING WITH NMPC

CONCERNING THE

UNIT 1 CORE SHROUD

AGENDA

September 24, 1998
Meeting Regarding Inspection of Core Shroud Vertical Welds at
Nine Mile Point Nuclear Station Unit 1

I. NRC SESSION WITH NIAGARA MOHAWK POWER CORPORATION (NMPC)

5:00	NRC Opening Remarks	Darl Hood
	Purpose Introduction of Participants	
5:05	Background	Robert Hermann
5:10	NMPC's Review of Request to Extend Core Shroud Inspection Interval	Richard Abbott et al.
	Introduction Core Shroud Boat Sample Tests and Evaluations Application of BWRVIP-14 to Unit 1 Core Shroud Weld Cracks Conclusions	
6:30	NRC Questions/Comments	
7:00	Break	

II. NRC SESSION WITH PUBLIC

7:30	NRC Opening Statements	Darl Hood
7:35	Questions/Comments from Audience	
9:30	NRC Closing Remarks	Singh Bajwa

Enclosure 1

NRC ATTENDEES

Office of Nuclear Reactor Regulation, Rockville, MD:

Singh S. Bajwa	Director Project Directorate I-1
Darl S. Hood	Senior Project Manager Project Directorate I-1
Robert A. Hermann	Senior Level Advisor-Materials Science Materials and Chemical Engineering Branch Division of Engineering
William H. Koo	Senior Materials Engineer Materials and Chemical Engineering Branch Division of Engineering
Ralph Caruso	Section Chief Reactor Systems Branch Division of Systems Safety and Analysis
Kerri A. Kavanagh	Reactor Systems Engineer Reactor Systems Branch Division of Systems Safety and Analysis
Dr. Lambros Lois	Senior Reactor Systems Engineer Reactor Systems Branch Division of Engineering

Region I, King of Prussia, PA:

Lawrence T. Doerflein	Chief, Project Branch 1 Division of Reactor Projects
Barry S. Norris	Senior Resident Inspector Nine Mile Point Nuclear Station
Neil A. Sheehan	Senior Public Affairs Officer Public Affairs Staff

NRC Contractor:

Dr. William J. Shack	Associate Division Director of the Energy Technology Division Argonne National Laboratory

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT NUCLEAR STATION UNIT 1

NRC/NMPC
Nine Mile Point Unit 1
Core Shroud Meeting

September 24, 1998

Agenda

Opening Remarks	J. H. Mueller
Introductions	R. B. Abbott
Purpose	R. B. Abbott
Background	C. D. Terry
Results of Evaluation	G. Inch R. Horn M. Manahan
Results of Structural Margin Assessment	G. Inch
Conclusion	R. B. Abbott

Meeting Purpose

- Present supplemental information applied as basis for extending shroud reinspection
  - NMP1 shroud metallurgical, fluence, and crack growth assessment submitted February, 1998
  - NMP1 supplemental shroud structural margin analysis submitted April, 1998
  - Neutron transport analysis - September, 1998
- Applicability of BWRVIP-14

Background

- The BWRVIP developed industry standardized shroud inspection, evaluation and repair criteria which were approved by the NRC
- Unit 1 shroud horizontal welds preemptively repaired in 1995
- All vertical welds inspected in 1997 consistent with BWRVIP criteria for repaired shrouds
- Cracks were observed and boat samples removed for metallurgical evaluation

Background

- April 1997, NMPC provided justification, consistent with BWRVIP-01 guidelines, for 10,600 hours of hot operation
- May 8, 1997, NRC issued an SER allowing operation for 10,600 hours prior to reinspection of the vertical welds
- February 27, 1998, the NMPC submittal requested to extend operation from 10,600 hours to 14,500 hours, based upon metallurgical evaluation and reassessment of crack growth rates for welds V9 and V10
- April 30, 1998, NMPC submitted results of supplemental structural margin assessment of welds V4, V9 and V10, consistent with BWRVIP-01 guidance, to further support operation for 14,500 hours
- On June 8, 1998, the NRC issued an SER on BWRVIP-14 which is directly applicable to the NMP1 cracking

Basis of the Vertical Weld 10,600 Hour Inspection Interval

- 100% inspection of all accessible vertical and horizontal welds consistent with BWRVIP-01 and BWRVIP-07
- Finite element Linear Elastic Fracture Mechanics (LEFM) analysis of V9 and V10 part through wall cracks based on fracture toughness of (150 ksi $\sqrt{in}$) consistent with BWRVIP-01 evaluation guidelines
- Limit Load Analysis for V4, V15, and V16
- Operating interval was defined based on CGR of 5.0×10^{-5} in/hr
- No credit for horizontal weld integrity
- Part through wall cracking assumed at locations where UT identified uncracked ligament
- Operate within EPRI water chemistry guidelines
- Complete boat sample evaluations

Actions Since April 1997 Inspection and Evaluation

- NRC approved the NMPC finite element fracture mechanics and limit load analysis of the vertical welds and the safety assessment of the vertical weld cracking
- NMPC has operated well below the EPRI water chemistry guideline commitment (conductivity <.3 µS/cm, sulfate <5 ppb, chloride <5ppb)
  - avg. conductivity 0.076 µS/cm
  - avg. sulfate 2.01 ppb, avg. chloride <0.5 ppb
- NMPC completed detailed metallurgical evaluations of the vertical weld boat samples
- Additional structural margin analysis completed
- The NRC issued BWRVIP-14 SER which supports lower CGR

Basis of the Vertical Weld 14,500 Hour Inspection Interval

- Metallurgical and fluence evaluations justify 14,500 hours based upon lower CGR
  - PLEDGE analysis CGR confirms 2.2×10^{-5} in/hr with significant margin
  - Cracking confirmed as IGSCC, consistent with basis of BWRVIP-14
  - Analysis satisfies the BWRVIP-14 SER conditions
    - » Fluence will remain below 5×10^{20} n/cm^2
- Supplemental structural analysis which satisfies BWRVIP-01 analysis guidelines justifies greater than 14,500 hours at the assumed 5×10^{-5} in/hr CGR

Vertical Weld Boat Sample Evaluations

- Two boat samples removed
- Boat samples exhibit expected IGSCC characteristics
  - Crack located in heat affected zone (HAZ)
  - Surface cold work
  - No extensive crack tip branching, grain encirclement or grain dropout characteristic of irradiation effects
- Results confirm UT sizing (within .1 inch)
- Results confirm excellent material ductility
- Tensile properties are consistent with irradiation of material in the 3×10^{20} n/cm^2 range
- Boat sample based fluence measurements, confirm that analysis predictions of vertical weld peak fluence are conservative
- Metallography and other measurements confirm assessment of material sensitization
- Conclusion: Vertical weld is IGSCC which is typical of BWR core shroud cracking with no observed evidence of irradiation effects

NRC SER Crack Growth Assessment

- NRC SER issued June 8, 1998 on the BWRVIP-14 crack growth rate concludes that the three approaches are acceptable subject to staff review and the following conditions:
  - Fabrication weld repairs, etc., are considered in evaluating the residual stress
    - » Fabrication records show no repairs to vertical weld
    - » NMP1 analysis reviewed fabrication practices (MPM-497439)
  - Components are operated in accordance with EPRI BWR water chemistry guidelines
    - » NRC approved NMP1 Technical Specification which incorporates EPRI guidelines
  - Crack tip stress intensity is explicitly less than 25 ksi √in where applicable in structural analysis
    - » NMP1 analysis shows stress intensity will remain less than 25 ksi √in(GE-NE-B13-01869-113, GE-NE-523-B13-01869-043)
  - Fluence less than 5×10^{20} n/cm^2
    - » MPM-998676

NMPC Crack Growth Assessment Summary

- Evaluations based on both GE PLEDGE model and the BWRVIP-14 correlation
- Evaluations consider all the factors which affect potential crack growth rate
  - Vertical weld residual and fabrication stresses (BWRVIP-14, NMP1 analysis)
  - NMP1 operating chemistry (Plant Data)
  - Corrosion potential (NMP1 data and BWR data)
  - Material fluence (Analysis and Boat sample)
  - Material sensitization (Boat sample data, GE data, BWRVIP-14)
- Conclusions
  - PLEDGE predicts CGR at or below 0.42×10^{-5} in/hr
  - Use of 2.2×10^{-5} in/hr bounds predicted CGR (factor of 5)
  - Application of 2.2×10^{-5} in/hr supports a cycle greater than 24 months
  - Substantial margin exists

Role of Irradiation Effects on NMP1 Shroud

- GE presentation (Dr. R. Horn)

Effects of Irradiation on Shroud Cracking

- High fluence can contribute to the susceptibility of the material
  - Can produce chromium depletion at grain boundaries
  - Sensitization can be found outside of the weld HAZ
- Cracking will exhibit additional features:
  - Significant grain fallout
  - Significant crack branching in higher fluence regions
- Irradiation will also produce significant hardening of the base material

Comparison of Boat Sample Data

Key Factors	Comparison Plant	NMP1
Fluence	8×10^{20}	$<3 \times 10^{20}$
Cracking in Non-sensitized Material	Yes	No
Significant Grain Fallout	Yes	No
Crack Branching	Yes	No
Significant Hardening	Yes	No

Summary

- The NMP1 shroud boat samples allowed a comparison with the earlier evaluation, performed on a boat sample from another shroud, irradiated to higher fluence
  - Locations of sensitization
  - Cracking morphology
  - Base material characteristics
- Unit 1 crack evaluation indicates no irradiation effects
  - Limited levels of base material hardening
  - No significant grain fallout
  - No significant crack branching
  - Cracking correlated with regions of weld induced sensitization
  - Fluence was below levels where irradiation effects are important

NMP1 Shroud Neutron Transport Analysis

- MPM Technologies, Inc. Presentation
  (Dr. M.P. Manahan, Sr.)

Analysis of Boat Sample Dosimetry Data

- Two boat samples were cut from the shroud at the end of cycle 12
  - ID surface of V9 26.4 inches above midplane (peak ID measured fluence = 3.49×10^{20}n/cm2)
  - OD surface of V10 8.3 inches below midplane (peak OD measured fluence = 1.42×10^{20}n/cm2)
- Dosimetry data taken at three depths within each boat sample

Analysis of Boat Sample Dosimetry Data (continued)

- Analysis by Framatome in January, 1998 using cycle 7 transport data showed a discrepancy between the Fe and Ni dosimeters
- Analysis of the 210 degree surveillance capsule dosimetry in May,1998 by MPM using a mid-cycle 12 transport analysis showed a similar discrepancy
- In May, 1998 MPM suggested that a large flux drop through cycle 12 would explain the discrepancy

Boat Sample Analysis Results

- Through cycle analysis has resulted in close agreement between Fe and Ni dosimeters
- Average ratio of the fluxes from Ni to those from Fe are 0.991 with a standard deviation of 3.3%
- Calculations at the boat sample locations have been shown to be conservative by comparison with the measured fluxes

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Neutron Flux Calculations

Analyses Include:

- R-θ, R-Z, and R calculations for 5 cycle 12 representative power profiles (15 transport calculations)
- Uncertainty Analysis

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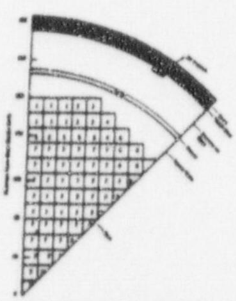
NMP-1 R-θ Geometry



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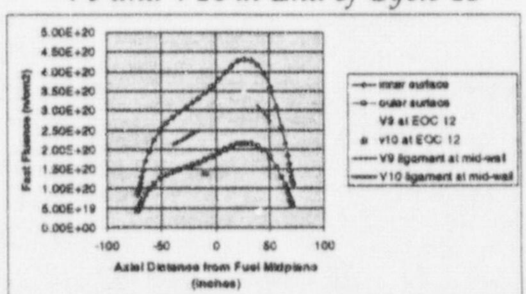
Calculated Fast Fluence to Welds V9 and V10 at End of Cycle 13



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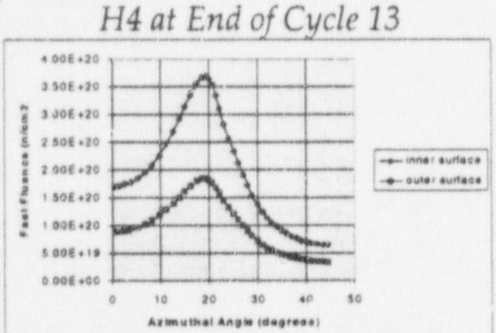
Calculated Fast Fluence to Weld H4 at End of Cycle 13



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Neutron Transport Results for Shroud Welds

Weld Identification	Location			Cycle 12 Fast (E > 1) Fluence (n/cm²)	Cycle 13* Fast (E > 1) Fluence (n/cm²)
	ID Surface Radius (in)	Height Above Midplane (in)	angle (degrees)		
H1	93	102.18	19.38	1.2e+19	1.4e+19
H2	93	70.93	19.38	7.2e+19	7.9e+19
H3	88	68.93	19.38	1.4e+20	1.5e+20
H4	88	50.43	19.38	3.4e+20	3.7e+20
H5	88	-39.69	19.38	2.4e+20	2.9e+20
H6 A	88	-103.19	19.38	5.5e+18	6.1e+18
H6 B	88	-107.69	19.38	3.3e+18	3.7e+18
H7	80	-129.82	19.38	2.7e+17	2.9e+17
V9/V10	88	27.00	20	3.9e+20	4.3e+20

* Calculations for cycle 13 are for 14,500 EFPH past the end of cycle 12

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Neutron Transport Results for V9/V10 at End of Cycle 13

Limiting Ligament in Terms of Fast Fluence	Height Above Fuel Midplane (inches)	Comments	Fast (E>1 MeV) Fluence at Shroud ID Surface Along V9/V10 (n/cm²)	Fast (E>1 MeV) Fluence - 1t at Shroud ID Surface Along V9/V10 (n/cm²)
V9	50.47	No intersection with V9/V10	3.93e+20	4.28e+20
V9	43.20	mesh point at V9/V10 uncracked ligament	3.95e+20	4.66e+20
V9	40.70	9.75 inches below the V9 uncracked ligament elevation at beginning of cycle 13	4.11e+20	4.85e+20
V10	17.10	11.55 inches below the V10 uncracked ligament elevation at beginning of cycle 13	4.18e+20	4.93e+20
V10	13.00	mesh point at V10 uncracked ligament	4.14e+20	4.89e+20
V10	9.00	mesh point at V10 uncracked ligament	3.98e+20	4.69e+20
V10	3.00	mesh point at V10 uncracked ligament	3.80e+20	4.48e+20
V10	<3.00	axial height from 3 inches above fuel midplane to B5	< 3.80e+20	< 4.48e+20

Summary and Conclusions

- Through cycle transport calculations for cycle 1[illegible] have brought the Fe and Ni dosimeter measured fluxes into agreement
- The calculated fluences at the boat sample locations exceed the measured values by 16% indicating that the calculations at the shroud are conservative
- The peak fluence to the V9 and V10 remaining ligaments will not exceed $5.0 \times 10^{20} n/cm^2$ at 14,500 EFPH past the end of Cycle 12

Supplemental Structural Margin Analysis

- The follow-up supplemental fracture mechanics analysis, demonstrates that the required ASME code required margins are maintained, for more than 14,500 hours, even assuming a CGR of 5×10^{-5} in/hr
  - Analysis consistent with BWRVIP-01 guidelines
  - Credit taken for uncracked locations confirmed by both volumetric inspections (UT) and visual inspections (EVT-1) for V9 and V10
  - Credit taken for far side detection capability of UT as qualified by BWRVIP-03 for V4 weld
  - V4, V9 and V10 limit load evaluations show signif[illegible]ant margin

Concluding Remarks

- There is substantial basis for reduced crack growth rate
- Fluence effects are not significant
- Structural analysis demonstrates inspection interval of 14,500 hrs is justified without reducing CGR

EVALUATED CRACK GEOMETRIES