December 22, 2000

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, 2nd Floor
P.O. Box 63
Lycoming, NY 13093

SUBJECT: NRC's NINE MILE POINT INSPECTION REPORT 05000220/2000-008, 05000410/2000-008

Dear Mr. Mueller:

On November 11, the NRC completed an inspection of your Nine Mile Point Nuclear Station, Units 1 and 2. The enclosed report presents the results of that inspection. Preliminary results were discussed with Mr. J. Conway and other members of your staff on November 30, 2000.

NRC inspectors examined numerous activities as they related to reactor safety and compliance with the Commission's rules and regulations and with the conditions of your operating license. The inspection consisted of a selected examination of procedures and records, observations of activities, and interviews with personnel.

Based upon the results of this inspection, the inspectors identified five issues of very low safety significance (GREEN). Four of these issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations (NCVs), consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000, (65FR25368). If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001: with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at the Nine Mile Point Nuclear Power Plant.

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Sincerely,

/RA by
William A. Cook
Acting For/

Michele G. Evans, Chief
Projects Branch 1
Division of Reactor Projects

Docket Nos.: 05000220, 05000410
License Nos.: DPR-63, NPF-69

Enclosure: NRC's Nine Mile Point Inspection Report 05000220/2000-008, 05000410/2000-008

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REGION I

Docket Nos:	050000220, 050000410
License Nos:	DPR-63, NPF-69
Report No:	050000220/2000-008, 050000410/2000-008
Licensee:	Niagara Mohawk Power Corporation (NMPC)
Facility:	Nine Mile Point, Units 1 and 2
Location:	P. O. Box 63 Lycoming, NY 13093
Dates:	October 1, 2000 - November 11, 2000
Inspectors:	G. Hunegs, Senior Resident Inspector R. Fernandes, Resident Inspector B. Fuller, Resident Inspector L. Cheung, Senior Reactor Inspector D. Dempsey, Reactor Engineer J. Trapp, Senior Reactor Analyst
Approved by:	Michele G. Evans, Chief Projects Branch 1 Division of Reactor Projects

Summary of Findings

IR 05000220-00-08, 05000410-00-08; on 10/01 - 11/11/2000; Niagara Mohawk Power Corporation; Nine Mile Point, Units 1 & 2; Equipment alignment, flood protection measures, event follow-up.

This inspection was conducted by the resident inspectors and two region based inspectors. The inspection identified five Green findings, four of which involved a non-cited violation. The significance of most/all findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609 “Significance Determination Process” (See Attachment 1). Findings for which the Significance Determination Process (SDP) does not apply are indicated by “no color” or by the severity level of the applicable violation.

Cornerstone: Mitigating Systems

Green. The inspectors identified that the licensee did not recognize that the valve seat leakage of the Unit 2 feedwater control valves (FWS-L10A, B, and C) adversely impacted the ATWS mitigation function of the redundant reactivity control system, as discussed in the UFSAR.

This finding was of very low safety significance because, although the valves have a documented history of excessive leak-by, this leakage was not significant enough to compromise their ATWS mitigation function. (Section 1R04)

Green. The inspectors identified that degraded flood protection equipment at Unit 2, involving the water tight doors of the reactor building auxiliary bays, potentially compromised the emergency core cooling systems located in these rooms in the event of internal flooding.

This finding was of very low safety significance because the watertight doors were determined to be degraded, but functional. The inspectors identified this as a Non-Cited Violation for failure to maintain the design configuration of the water tight doors per 10CFR50, Appendix B, Criterion III. (Section 1R06)

Green. The inspector identified that the Unit 1 control room operators were not maintaining a suppression pool temperature log in accordance with Technical Specification (TS) 4.3.2.c. for approximately 20 minutes following the electromatic relief valve opening on October 2, 2000. The root cause of this TS violation was the temporary lock-up of the plant process computer.

This finding was of very low safety significance because the suppression pool temperature rise was very slow and remained within TS limits. This violation of TS was treated as a Non-Cited Violation. (Section 1R53)

Green. During Unit 1 start-up on October 2, 2000, an electromatic relief valve (ERV) inadvertently opened. The stuck open ERV resulted in a continuous discharge of steam and subsequent failure of the tailpipe vacuum breaker. Failure of the vacuum breaker resulted in discharge of steam directly to the drywell. The inspectors identified a Non-Cited Violation for ineffective corrective actions involving previously identified conditions which were not corrected and contributed to the ERV inadvertent opening.

The ERV opening was of very low safety significance because of low levels of reactor power (3.0%), temperature (254°F), pressure (38 psig), and decay heat (<0.5%), and all emergency core cooling systems were available. Licensee corrective actions for this event were appropriate. (Section 1R53)

Green. On September 27, 2000, while Unit 1 was shutdown, a low reactor vessel level condition occurred when placing the reactor water cleanup (RWCU) system in service. The low level was a result of reactor vessel water inventory being displaced to the RWCU system due to leaking isolation valves and inadequate fill and vent of the RWCU system prior to being placed in service. The inspectors identified a Non-Cited Violation for ineffective corrective action for the longstanding RWCU isolation valve degradation which contributed to this event.

This issue was of very low safety significance because the unit was shutdown at the time of the reactor vessel level transient and all shutdown emergency core cooling systems were available for reactor inventory control, if needed. The licensee's corrective action program was not effective in precluding this specific event. (Section 4OA3)

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Attachment 1 - NRC's REVISED REACTOR OVERSIGHT PROCESS

Report Details

SUMMARY OF PLANT STATUS

Nine Mile Point Unit 1 (Unit 1) began this inspection report period shut down in a planned maintenance outage. During reactor startup on October 2, an electromatic relief valve (ERV) on the main steam system stuck open with the reactor at very low power. A manual scram was initiated. An increase in unidentified leakage greater than 10 gallons per minute from the reactor due to a failed vacuum breaker in the ERV tailpipe resulted in entry into the emergency plan for an Unusual Event. The Unusual Event was exited when the unit was placed in cold shutdown. The reactor was restarted on October 9, after repairs to the ERVs were completed. Unit 1 was synchronized to the grid on October 10 and reached 100 percent power on October 12. With the exception of several planned power reductions for maintenance, Unit 1 remained at 100 percent power through the end of the inspection period.

Nine Mile Point Unit 2 (Unit 2) began this inspection report period at 100 percent power. On October 25, Unit 2 entered single loop operation when the "A" reactor recirculation flow control valve failed closed. Unit 2 remained in single loop operation until the unit was shutdown on October 29, after an unsuccessful attempt to recover the idled loop. The reactor was restarted on November 3 and was synchronized to the grid on November 4. Unit 2 reached 100 percent power on November 7 and remained there through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors conducted a review of Unit 1 actions to ensure protection of mitigating systems from adverse weather effects. The NRC inspection procedure was implemented before the onset of low temperatures expected during the winter months. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and individual plant examination (IPE) for insights into cold weather related risks to mitigating systems. Areas reviewed included room heater operation and weather sealing.

The inspectors performed a walkdown of the following areas as part of the inspection for adverse weather protection:

- Unit 1 containment spray and core spray pump rooms
- Unit 1 screen house

b. Issues and Findings

No findings of significance were identified. Unit 1 does not have a program for cold weather preparations. A draft document existed, but had not been issued at the time of the inspection. The licensee currently uses a checklist of systems and components, developed by the work control manager, to ensure that cold weather sensitive equipment was ready for the winter season.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted equipment alignment partial walkdowns to evaluate the operability of selected trains or backup systems, while the redundant train or system was inoperable or out of service. Walkdowns were also conducted on equipment recently realigned due to outage activities and surveillance testing. The walkdowns included, as appropriate, consideration of plant procedures and reviews of documents to determine correct system lineups and verification of critical components to identify any discrepancies which could adversely affect operability of the redundant train or backup system. With respect to the Unit 2 walkdowns, a partial breaker line-up and control room panel walk down, visual inspection of the electrical protection assembly (EPA) panels, un-interruptible power supples, motor generator sets, and scram pilot solenoids were also inspected. In addition, the inspector reviewed the system health reports and corrective actions for previously identified problems. The inspectors used the Unit 2 Individual Plant Examination (IPE) to review the RPS and RRCS systems for insights into important functions of various structures, systems and components (SSCs).

The inspectors performed the following partial system walkdowns:

- Reactor building closed loop cooling system (Unit 1)
- Reactor protection system (RPS) (Unit 2)
- Redundant reactivity control system (RRCS) (Unit 2)

b. Issues and Findings

As part of the walkdown of the Unit 2 RRCS system, the inspectors' review of the maintenance history associated with the feed pump high pressure, high flow, level control valves, FWS*LV10A, B and C, identified that there have been repetitive failures (valve seat leakage) of these valves to completely isolate feedwater flow to the reactor. In some instances, these failures have caused consequential high reactor water level trip signals following a reactor trip and normal feedwater system isolation response (reference Deviation/Event Report Nos. 2-1997-1330, 2-1998-2078, 2-1998-3624, 2-2000-1517, and 2-2000-3181).

Inspector review determined that the IPE and Updated Final Safety Analysis Report (UFSAR) document that one of the functions of the RRCS is to send a signal to the feedwater level control valves to close in response to an anticipated transient without scram (ATWS). The inspector questioned the operability of this RRCS function, with respect to the material condition of the feedwater level control valves, and the observed inability of the feedwater control system to stop flow to the vessel in an ATWS mitigation scenario. NMPC evaluation of this issue determined that the valves were operable, but degraded, and prepared an engineering support analysis (ESA) to formalize their determination. ESA-2M00-011 concluded that the RRCS would be capable of performing its intended design function because the ATWS analysis report does not take significant credit for the subcooling effect of the feedwater system runback.

This degraded condition of the feedwater level control valves and impact on the RRCS ATWS mitigation function was evaluated using the Significance Determination Process (SDP). This issue, if left uncorrected, could increase the likelihood of core damage in the event of an ATWS. The licensee's risk evaluation (SAS-00-090) for this inspector identified issue concluded that the delta-core damage frequency (CDF) estimate was ~1.8E-7/year and the delta-large early release frequency (LERF) estimate was ~2E-8/year. An independent risk assessment performed by the Region I Senior Reactor Analyst (SRA), using less conservative assumptions for the feedwater control valve failures, concluded an approximate delta-CDF value of ~1.9E-8/year. Accordingly, these risk evaluations concluded that the risk associated with this issue was of very low significance (GREEN).

Inspector review of this issue also identified that there may have been prior opportunities for the licensee to have taken appropriate corrective action for these feedwater control valve problems. Reference section 1R12 of this report for further discussion of this issue.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted walk-downs of fire areas to determine if there was adequate control of transient combustibles and ignition sources. The condition of fire detection devices, the readiness of the sprinkler fire suppression system and the fire doors were also inspected. In addition, the passive fire protection features were inspected, including the ventilation system fire dampers, structural steel fire proofing and electrical penetration seals. The following plant areas were inspected:

- Emergency core cooling system (ECCS) corner rooms (Unit 1)
- 175 foot elevation of the reactor building, fire zones 212SW and 213SW (Unit 2)
- 175 foot elevation of the South auxiliary bay, fire zones 206SW, 207SW and 208SW (Unit 2)
- 175 foot elevation of the North auxiliary bay, fire zones 201SW, 202SW, and 203SW (Unit 2)

b. Issues and Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the licensee's flood mitigation equipment, such as watertight barriers, drainage and pumping systems, to ensure that the equipment was capable of meeting design requirements. The inspectors reviewed the UFSAR and IPE to identify those areas with the potential to be affected by internal flooding. Selected areas were walked down to inspect penetration seals, adequacy of watertight doors and the operability of floor drains including common drain system valve positions and check valves which were credited for isolation of flood areas within the reactor building. In addition, the inspector reviewed operating procedures, alarm response procedures, and emergency operating procedures. For those areas where operator actions are credited, the inspector verified that the procedures for coping with flooding, could reasonably be used to achieve the desired actions. The inspector verified that NMPC entered past problems in the corrective action program and that problems were properly addressed for resolution.

The following areas were selected:

- Unit 1 containment spray and core spray pump rooms

- Unit 2 safe shutdown equipment located in reactor building auxiliary bays

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

During the Unit 1 inspection, the inspectors identified additional foam weather strip attached to the neoprene seal on door D-287 in the Southeast corner room. This additional weather stripping was not shown on the associated design control drawings. DER 1-2000-3780 was initiated to document this discrepancy. The additional sealing material was determined by the licensee to have had no impact on the door sealing ability.

During the Unit 2 inspection, the inspectors determined that the material condition of the water tight doors in the reactor building auxiliary bays was degraded. Specifically, the seals on watertight door Nos. SA-175-2, "A" residual heat removal (RHR) and low pressure core spray (CSL) pump room, and SA-175-3, "C" RHR pump room, were damaged or partially missing. In addition, the inspectors noted a broken flexible conduit on 2 DFR*LS144, the RHR "C" pump room flooding (water level) switch. NMPC documented these discrepancies in the corrective action program as DER Nos. 2-2000-3597 and 2-2000-3592, and performed operability determinations for both issues. NMPC engineering determined that the water tight door seals and broken flexible conduit were degraded, but would still perform their intended function.

The discrepant water tight door seal conditions were evaluated using the SDP. This issue, if left uncorrected, could have a credible impact on the availability of mitigating systems in the event of a flood in the affected emergency core cooling system rooms. The NRC Region I SRA evaluated plant transients, loss of off-site power (LOOP), and a spectrum of loss of coolant accidents to determine the resultant change in core damage probabilities (CCDPs) with a postulated loss of the RHS and CSL pumps. Using the SPAR, revision 3, Nine Mile Point PRA model, and GEM code, the SRA identified that all of these events demonstrated an increase in CCDP, but still below the 1E-6/year. Accordingly, this issue was determined to be of very low significance (GREEN).

The failure to maintain the Unit 2 reactor building auxiliary bay flood protection watertight doors in good material condition, constitutes a violation of 10CFR50, Appendix B, Criterion III, "Design Control." However, because of the low safety significance of this issue and because the licensee has included this issue in their corrective action program, this design control violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy, issued on May 1, 2000, (65FR25368). These deficiencies are in the Unit 2 correction action system as DER Nos. 2-2000-3597 and 2-2000-3592. **(NCV 05000410/2000-008-01)**

1R07 <u>Heat Sink Performance</u>

1. <u>Annual Heat Exchanger Review</u>

a <u>Inspection Scope</u>

The inspector reviewed the past performance testing and the recent inspection of the Unit 1 number 11 reactor building closed loop cooling (RBCLC) heat exchanger.

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

No findings of significance were identified.

2. <u>Biennial Heat Exchanger Review</u>

a. <u>Inspection Scope</u>

The inspector reviewed the licensee's maintenance, testing, inspection, and trending of the following heat exchangers, all cooled by raw water, to ensure that proper heat transfer was maintained for these heat exchangers:

- Unit 1 emergency diesel generator (EDG) 102 cooler No. 79-03,
- Unit 2 residual heat removal (RHR) heat exchanger,
- Unit 2 emergency diesel generator No. 2EGS*EG3 cooler.

For each selected Unit 2 heat exchanger, the inspector reviewed heat exchanger performance test methodology, frequency of testing, test conditions, acceptance criteria, and trending of test results. For the selected Unit 1 cooler, which did not have dedicated performance testing and trending of results, the inspector reviewed the monthly diesel generator testing and the quarterly in-service testing for the cooling water pump and associated valves to verify that adequate cooling was provided to EDG 102. The periodic inspection, cleaning, and eddy-current testing of heat-exchanger tubing were reviewed to verify that the heat exchanger degradation could reasonably be detected. The inspector also determined that corrective actions were specified in the procedures for conditions not meeting the acceptance criteria. The inspector reviewed the heat load calculation for Unit 1 EDG cooler. This calculation identified the maximum number of tubes that could be plugged (10 tubes per cooler) without jeopardizing the required cooling capability. The inspector verified that this result was specified in the heat exchanger inspection procedure. The inspector also walked down all three heat exchangers and verified that the installed configurations were consistent with the design drawings. Also, the inspector reviewed a sample of DERs associated with these three heat exchangers and verified that the identified problems were entered into the licensee's corrective action program and adequately resolved.

b. Issues and Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

c. Inspection Scope

The inspectors reviewed the licensed operator requalification training activities to assess the licensee's training effectiveness. The inspectors observed Unit 1 licensed operator simulator training on October 25, 2000, and Unit 2 on October 27, 2000. The inspectors assessed performance in the areas of procedure use, self- and peer-checking, completion of critical tasks, and training performance objectives. Following the simulator exercises, the inspectors observed the training instructor's debrief and critique and reviewed simulator fidelity through a sampling process.

d. Issues and Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed performance based problems involving selected in-scope structures, systems, and components (SSCs) to assess the effectiveness of the maintenance program. Reviews focused on: (1) proper maintenance rule scoping, in accordance with 10 CFR 50.65; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10 CFR 50.65 (a)(1) and (a)(2) classifications; and, (5) the appropriateness of performance criteria for SSCs classified as (a)(2), and goals and corrective actions for SSCs classified as (a)(1). The inspectors reviewed the licensee's system scoping documents and system health reports. The following Deviation/Event Reports (DERs) were reviewed:

- DER No. 1-2000-3780 — Unit 1 watertight doors in ECCS corner rooms
- DER No. 1-2000-3619 — Unit 1 mechanical pressure regulator
- DER No. 2-2000-4040 — Unit 2 failure to track the operability of the FWS-LV10 valves as they relate to the runback function of the RRCS

b. Issues and Findings

As previously discussed in Section 1R04, the inspector identified that NMPC had not recognized that the FWS-LV10 valve leakage problems adversely impacted the valves' associated redundant reactivity control system ATWS mitigation function. The inspectors also identified that this degraded ATWS mitigation function may be construed as a functional failure per the licensee's Maintenance Rule Program. The licensee acknowledged the inspector's observation and initiated DER No. 2-2000-4040 to track this observation through the corrective action program and Unit 2 Maintenance Rule Expert Panel's assessment process. This issue will remain unresolved, pending the licensee's determination and inspector follow-up of whether the FWS -L10 valves have been appropriately maintained per the licensee's Maintenance Rule Program. **(Unresolved Item 05000410/2000-008-02)**

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

For selected maintenance work orders (WOs), the inspectors evaluated: (1) the effectiveness of the risk assessments performed before the maintenance activities were conducted; (2) risk management control activities; (3) the necessary steps taken to plan and control resultant emergent work tasks; and, (4) the overall adequacy of identification and resolution of emergent work and the associated maintenance risk assessments. The following WOs were reviewed:

- No. 00-05694, Replace transmitter and associated flow switch to support improved technical specifications implementation for the high pressure core spray minimum flow control valve (Unit 2).

- No. 00-11120-00, Mechanical pressure regulator bushing not rotating (Unit 1).

- No. 00-11181-00, Pressure transmitter PT36-07A spurious high reactor pressure trip signals (Unit 1).

b. Issues and Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability evaluations affecting risk significant mitigating systems, to assess: (1) the technical adequacy of the evaluation; (2) whether continued system operability evaluations were warranted; (3) whether other existing degraded systems adversely impacted the affected system or compensatory measures; (4) where compensatory measures were used, whether the measures were appropriate and properly controlled; and (5) the degraded system's impact on technical specifications (TS) limiting condition for operations. The following DERs were reviewed:

- No.1-2000-3417 — Manual reactor scram and Unusual Event due to stuck open ERV-111 (Unit 1).
- No.1-2000-3438 — CKV-66-26 disk separated from hinge pin (Unit 1).
- No. 2-2000-1517 — Feedwater excursion to the reactor vessel (Unit 2).
- No. 2-2000-3462 — High pressure core spray system trip unit alarm after pump start (Unit 2).

b. Issues and Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

Unit 1 operating history has shown that on a plant scram, the high pressure coolant injection (HPCI) system automatically initiates; however, depending on initial conditions, one or both feedwater pumps may trip on low suction pressure. NMPC installed modification N1-00-022, Feedwater(FW)/HPCI Low Suction Pressure Trips, Phase 2, to help prevent this unnecessary system response. The inspector reviewed the design document change and associated applicability review to verify that the design bases, licensing bases, and performance capability of the Unit 1 high pressure coolant injection mode of the feedwater system was not degraded through the implementation of the modification.

b. Issues and Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed post-maintenance testing (PMT) procedures and associated testing activities for selected risk significant mitigating systems to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness, consistent with the design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy for the application; (5) tests were performed, as written, with applicable prerequisites satisfied; (6) jumpers installed or leads lifted were properly controlled; (7) test equipment was removed following testing; and (8) equipment was returned to the status required to perform its safety function.

- N1-ST-C2, Rev. 15, Manual Opening of the Solenoid-Actuated Pressure Relief Valves and Flow Verification (Unit 1).

- N1-PM-W7, Rev 03, Main Turbine, Thrust Bearing, Mechanical Pressure Regulator, and Oil tests (Unit 1).

- N2-ISP-CSH-R103, Operating Cycle Channel Calibration of the High Pressure Core Spray (HPCS) Pump Discharge Flow Instrument Channel (Unit 2).

- N2-OSP-CSH-Q@002, HPCS Pump and Valve Operability and System Integrity Test (Unit 2).

b. Issues and Findings

No findings of significance were identified.

1R20 Outage Activities

a. Inspection Scope

On October 2, Unit 1 was shutdown to repair ERVs and associated vacuum breakers. The reactor was restarted on October 9, after repairs to the ERVs were completed.

On October 25, Unit 2 entered single loop operation when the “A” reactor recirculation flow control valve failed closed. Unit 2 remained in single loop operation until October 28. An attempt to restart the idle loop was not successful and the unit was shutdown on October 29. The reactor was restarted on November 3. The inspectors reviewed the Unit 1 and Unit 2 outage activities listed below for conformance to the applicable procedures and witnessed selected activities associated with each evolution. Surveillance tests were reviewed to verify TS were satisfied. The inspectors observed start-up activities in the control room to verify that TS, license conditions, commitments, and other procedural prerequisites and requirements for mode changes were met prior to changing modes.

- shutdown cooling system operation
- shutdown risk evaluations

- containment closeout
- reactor startup, including the control of approach to criticality
- various outage related preventive and corrective maintenance activities

b. Issues and Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed performance of surveillance test procedures and reviewed test data of selected risk significant SSCs to assess whether the SSCs satisfied Technical Specifications, Updated Final Safety Analysis Report (UFSAR), and licensee procedure requirements; and to determine if the testing appropriately demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. The following tests were witnessed:

- N1-PM-W7, Rev 03, Main Turbine, Thrust Bearing, Mechanical Pressure Regulator, and Oil tests (Unit 1)

- N1-OP-31, Rev 18, Tandem Compound Reheat Turbine (Unit 1)

b. Issues and Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspector reviewed Unit 2 temporary modification No. 2000-036, Installation of a Blank Flange at 2 CPS-FE103, which was installed to comply with technical specifications following an unsatisfactory local leak rate test of a containment isolation valve. The temporary modification utilized document design change (DDC) No. 2M11820, to remove the drywell purge line flow element, No. 2 CPS-FE103, and install a blank flange. The inspector verified that the installation was in accordance with the DDC and the associated work order, and that the post-work testing was completed. The inspector verified that the safety evaluation was consistent with the design documentation and that plant drawings were updated.

b. Issues and Findings

No findings of significance were identified.

1R53 Event Followup - Stuck Open Electromatic Relief Valve (Unit 1)

a. Inspection Scope

Using the guidance of Inspection Procedures 71153 and 93812, the resident inspectors and a region based specialist inspector reviewed the events and circumstances involving an electromatic relief valve failure during Unit 1 start-up.

Description and Chronology of Event

On October 2, 2000, during reactor startup, electromatic relief valve (ERV) -111 opened. Licensed operators directed a manual scram of the reactor, after attempting to shut the ERV and commenced a reactor plant cooldown. At the time of the event, the reactor was critical with intermediate range monitors (IRMs) on range 9 and reactor pressure approximately 38 psig. All plant systems responded as expected.

The stuck open ERV resulted in continuous discharge of steam into the ERV tailpipe. The tailpipe has two vacuum breakers installed to prevent drawing a column of water into the tailpipe as the steam in the pipe condenses. Repeated cycling due to the continuous low pressure steam discharge resulted in fatigue failure of one of the two 10-inch vacuum breaker valves (CKV-66-26). Failure of the vacuum breaker resulted in discharge of steam directly to the drywell, with a resulting increase in drywell pressure and temperature. Licensee personnel properly declared an Unusual Event (UE) due to the increase in unidentified leakage into the drywell. The UE was terminated when the reactor temperature was reduced and steam was no longer being discharged to the drywell.

Risk Significance of Event

The inspector and a Region I Senior Reactor Analyst reviewed the licensee’s safety and availability assessment of the event. The licensee estimated the event to have a conditional core damage probability of 3.6E-9, approximately the same probability as a scram during power operation. The event occurred at low levels of reactor power (3.0%), temperature (254°F), pressure (38 psig), and decay heat (<0.5%). All emergency core cooling systems were available for reactor vessel inventory makeup. Therefore, the event had very low risk significance.

Operator Performance Issue

Unit 1 Technical Specification 4.3.2.c requires that whenever heat from relief valve operation is being added to the suppression pool, that the pool temperature shall be continually monitored and logged every five minutes until the heat addition is terminated. While the ERV was open during this event, operators were crediting a plant process computer (PPC) special log routine for logging torus temperature every five minutes.

At 06:45 am, twenty minutes after the reactor was shutdown and the ERV was still open, the inspector observed that the PPC large value display for reactor vessel level on the main control board was not updating. The inspector informed the Assistant Station Shift Supervisor (ASSS) that the PPC appeared to be locked-up. The ASSS investigated and verified that the control room operators were not controlling reactor level using the large value display. The Shift Technical Advisor (STA) determined that the PPC outputs were locked-up and identified that the PPC failure had occurred at the time the reactor was shutdown. Consequently, the torus heating special log had not been updating for a period of twenty minutes.

NMPC initiated DER 1-2000-3422 to document that the Technical Specification logging requirement was not met for approximately twenty minutes. During the time that the PPC was not updating plant displays and trend recorders, torus temperature was rising slowly. The temperature data from before the computer locked-up (available from the PPC's memory) enabled the licensee to determine that peak torus temperatures did not exceed bulk or local temperature Technical Specification limits. Based on this information, the NRC concluded that the failure to monitor and log torus temperature during a heat addition event was a violation of very low safety significance (Green). This inspector identified violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy, issued on May 1, 2000, (65FR25368). This issue was entered in to the licensee's corrective action program under DER No. 1-2000-3416. **(NCV 05000220/2000-008-03)**

Equipment Failures

The event was initiated by the spurious opening of electromatic relief valve ERV-111. The valve is a six-inch, Model 1525-VX, solenoid-actuated, pilot-operated relief valve manufactured by Dresser Industries, Inc. Operation of the solenoid depresses a lever which unseats the reverse acting pilot valve. This action vents the main valve balance chamber allowing system pressure to open the main disk against a spring. When the overpressure condition is relieved, the main disk closes and the pilot valve is re-seated by a return spring. Computer logs showed that the ERV-111 pilot valve was open during the reactor startup. At a reactor steam pressure of 23 psig, the main disk opened. However, the pilot valve did not shut when steam pressure decreased allowing the main disk to remain open until the main disk spring finally overcame system pressure approximately an hour and thirty minutes later.

Computer logs indicated that the pilot valve of valve ERV-112 was also open during the startup. It is likely that valve ERV-112 also would have opened had reactor steam pressure been able to exceed its main spring preload. (Main spring preload may vary, but is typically about 50 psig.)

Instead of discharging to the torus as designed, steam discharged from valve ERV-111 entered the drywell through failed open vacuum breaker CKV-66-26. The vacuum breaker is a ten-inch, 300 psig swing check valve manufactured by Anchor-Darling. After the event, the licensee found the valve disk separated from the hinge arm. The hinge arm stop block, which is designed to transfer impact loads to the valve body, was missing from the hinge arm, and the disk retaining bolt was broken. Subsequently, the licensee also identified that the hinge arm was bent approximately 3 degrees from the disk arm hole centerline. Because the vacuum breaker internals were not visually inspected periodically in a preventive maintenance program, the licensee was unable to determine with certainty when the backstop might have separated from the hinge arm. From the physical evidence, it was possible that the backstop was missing from the hinge arm prior to the event.

Human Factor/Procedure Deficiencies

The inspector identified three potentially generic issues concerning the electromatic relief valves.

The licensee identified problems in its ERV maintenance and surveillance procedures that were attributable to deficiencies in the vendor manual. After disassembling the ERV-111 and ERV-112 pilot valves, the licensee found that the stems were slightly bent. The vendor specifies a maximum stem runout, or straightness, of 0.003 inches during fabrication. However, this factory check is performed prior to final machining of the steam. The vendor manual does not discuss the criterion or recommend that it be verified during pilot assembly or maintenance. Thus, licensee processes did not identify stem straightness as a critical quality attribute, and stem straightness was not verified during receipt inspection or pilot assembly maintenance. However, post-maintenance testing verifies smooth operation of the pilot valve, and the licensee had successfully performed these tests following the ERV maintenance activities that were performed prior to the October 2, 2000 event.

The vendor's manual also provides incorrect post-installation test guidance by permitting the pilot valves to be operated under dry conditions. The licensee determined that "dry cycling" the pilot valve could partially separate the valve disk from the stem, making the valve incapable of closing.

Past stuck ERV events at two other nuclear facilities were caused by internal leakage past the pilot seat bushing. Significant bypass leakage can actuate an ERV even if the pilot valve is closed. The leakage was due to insufficient compression of the pilot base gasket using the bracket stud nut torque specified in the vendor's manual. The torque specified in the licensee's maintenance procedure was consistent with the vendors' manual. The licensee identified that some bypass leakage appeared to have occurred in valve ERV-111, but the leakage in this case was not great enough to contribute to the event.

Quality Assurance Deficiencies

No significant licensee quality assurance findings were identified.

Radiological Consequences

The event had no adverse on-site or off-site radiological consequences.

Probable Contributing Causes

ERV-111

The licensee root cause investigation team performed a Kepnor-Tragoe analysis of the event to identify missing barriers that would have precluded the event.

The spurious operation and sticking of valve ERV-111 most probably was caused by a bent stem and partial disk-stem separation. It is most likely that the pilot valve stem was bent prior to being installed in the valve. Thus, the stem either was received in a bent condition or was damaged during the maintenance staging process. The licensee changed procedures to require stem runout checks on receipt and during installation.

The vendor informed the licensee that disk-stem separation had been observed during dry cycling of other pilot valve models. The licensee demonstrated during a simulated bench test that the phenomenon could occur in the Model 1525-VX pilot valve as well. The licensee has eliminated dry cycling from its procedures.

In response to operating experience reports, the licensee increased the solenoid bracket stud nut torque requirement to ensure adequate gasket crush and preclude pilot valve stem bushing leakage.

The licensee missed a previous opportunity to have identified and corrected the conditions that precipitated the event. During refueling outage 14 in March 1997, the pilot valve of ERV-113 was rebuilt. The valve stroked acceptably when operated manually, but became sluggish after being actuated by the solenoid. Following disassembly, the valve stem was found to be bent. A licensee mechanic also recalled finding excessive upward stem play that could have indicated a partially disengaged disk. These conditions were not documented in or evaluated by the licensee's corrective action program.

Vacuum Breaker CKV-66-26

The inspector identified two potentially generic issues involving the ERV vacuum breakers.

The vacuum breaker failed due to design and manufacturing deficiencies associated with the hinge arm stop block, and greater than anticipated impact and cyclic loads on the hinge arm. The licensee found that the seating surface of the vacuum breaker stop blocks on valve CKV-66-26 and other similar vacuum breakers were not flush with the hinge arm surface and that the attachment welds were not large enough. The licensee corrected the deficiencies by modifying the stop blocks and hinge arms.

The loads experienced by the vacuum breaker were more severe than assumed in the original valve design which was based on ERV actuation during rated reactor power operation. ERV discharge under low reactor power and pressure conditions resulted in condensation instability at the steam/water interface of the ERV discharge line quencher or in steam bubbles just outside of the quencher in the torus. Condensation instability can produce significant vent system pressure in the ERV tailpipe and result in high velocity vacuum breaker impacts of the hinge arm against the valve body. The licensee concluded that the vacuum breaker cycled approximately 500 to 1100 times in the hour and one half between initial ERV actuation and the manual reactor scram. The licensee performed a structural analysis using ASME Code allowables for fatigue loading to qualify the vacuum breakers for this condition during the current operating cycle.

b. Issues and Findings

The licensee completed a risk assessment that concluded that operation during the past year with the degraded ERV had low risk significance (delta-core damage frequency (delta-CDF) 3.1E-7/year. The licensee's analysis reviewed historical plant data to derive a frequency that the ERVs would open. This frequency was multiplied by an assumed probability of the ERV sticking open to derive a steam, small break loss of coolant accident (SBLOCA) frequency. This result was then multiplied by the conditional core damage probability for steam SBLOCAs. A similar process was performed for multiple ERVs sticking open. The contribution to risk of the degraded vacuum breaker was negligible because the frequency of ERVs sticking open is low, and the plant is designed to mitigate a SBLOCA. The NRC assessed the event using an independent approach that resulted in a similar risk estimate (delta-CDF 2.6E-7/year). Thus the event was of very low safety significance (GREEN).

Criterion XVI, "Corrective Action," of 10 CFR 50, Appendix B, requires that conditions adverse to quality be promptly identified and corrected. The cause of significant conditions adverse to quality must be identified and corrective actions taken to prevent recurrence. Contrary to the above, in March 1997, the licensee failed to identify and take actions to prevent recurrence for a significant condition adverse to quality involving degraded electromatic relief valve pilot valve components. Similar conditions in another ERV pilot valve subsequently contributed to the stuck open ERV event that occurred on October 2, 2000. Since this issue was of very low safety significance and was appropriately corrected, the violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy, issued on May 1, 2000, (65FR25368). (**NCV 05000220/2000-008-04**)

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verifications
Cornerstones: Initiating Events, Mitigating Systems, Emergency Preparedness

a. Inspection Scope

The inspectors verified the following second and third quarter 2000 PIs for Units 1 and 2:

- Unplanned scrams per 7000 critical hours.
- Scrams with loss of normal heat removal.
- Transients with loss of normal heat removal.
- Safety system unavailability.
- Safety system functional failures.
- Emergency response organization (ERO) drill/exercise performance.
- Alert and Notification system reliability (ANS).

b. Issues and Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

Reference Sections 1R53 and 4OA3 of this report for examples of Problem Identification and Resolution shortcomings this inspection period.

4OA3 Event Followup

a. Inspection Scope

Using the guidance of Inspection Procedure 71153, the inspector reviewed Licensee Event Report No. 50-220/2000-003, dated October 27, 2000, to determine if NMPC's evaluation and corrective actions were reasonable and if any violations of regulatory requirement were involved.

b. Issues and Findings

LER 50-220/00-003, Reactor Trip on Low Reactor Water Level While Placing the Reactor Water Cleanup System in Service, reported that on September 27, 2000, while the plant was in a cold shutdown configuration, a low reactor vessel level condition (reactor scram signal) occurred when placing the reactor water cleanup (RWCU) system in service. The low level transient was a result of reactor vessel water inventory being displaced to the RWCU system to make-up for voids in the RWCU system piping. The LER stated that the cause for the low reactor vessel level condition was inadequate filling and venting of the RWCU system piping, prior to the system being placed in service.

The issue screened as GREEN (very low safety significance) using the Shutdown Operations Significance Determination Process (NRC Inspection Manual Chapter 0609, Appendix G), because all shutdown emergency core cooling systems (ECCS) were available for inventory control and reactor vessel level was recovered in a short period of time using the control rod drive system (all ECCS remained in a standby configuration). Reactor vessel level instruments were operable during this event and operators responded to the level transient in accordance with unit operating procedures.

The LER states that the reactor vessel water was displaced to an isolated and drained standby filter in the RWCU system via leaking isolation valves. As documented in the LER, historically, operators have observed reactor water level decreasing (approximately four inches) when the RWCU system was placed in service. However, the typical four-inch level drop was with the condensate system in service, which acts to mitigate the vessel water level transient. For this event, operators did not anticipate the reactor vessel level and RWCU system response with the condensate system not in operation.

Inspector review of the LER and follow-up with the plant staff identified that a few aspects of the event and their associated corrective actions were not specifically described in the LER. However, the inspectors did find these items addressed in other licensee corrective action or tracking processes. One item of specific concern to the inspectors was the apparent longstanding poor material condition of the RWCU system isolation valves which contributed to the inadequate fill and vent of the system and deficient isolation of the standby filter train. The licensee provided the inspectors with the DER (No. 1-2000-3305) which addressed this issue.

The September 27, 2000, reactor vessel level transient and consequential reactor scram signal was the result of operators placing the RWCU system in service with a longstanding degraded isolation valve condition. This condition contributed to the inadequate vent and fill the RWCU system. This failure to take appropriate corrective action for a known deficient condition constitutes a violation of 10CFR50, Appendix B, Criterion XVI, “Corrective Action.” This violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). This issue was added to the licensee’s corrective action program under DER No. 1-2000-3305 and has also been addressed in the Unit 1 Top Ten Issues List. **(NCV 05000220/2000-008-05)**

The inspectors noted that LER 00-03, although lacking some event details and specific corrective actions, adequately addressed the requirements of 10 CFR 50.73, in order for the inspectors to make a risk assessment of the event and an overall determination of the adequacy of the licensee’s corrective actions. This LER is closed.

4OA5 Other

Performance Indicator Data Collecting and Reporting Process Review

a. Inspection Scope

Using Temporary Instruction 2515/144, the inspectors reviewed NMPC’s PI process to determine if they were appropriately implementing NRC/industry guidance specified in NEI 99-02, Revision 0, “Regulatory Assessment Performance Indicator Guideline,” issued by the Nuclear Energy Institute. This inspection verified the data collection and reporting process for the following PIs:

- Unplanned power changes per 7000 critical hours.
- Safety system unavailability and safety system functional failures.
- Emergency response organization drill participation.
- Protected area security equipment performance index.

b. Issues and Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. Conway, Vice President, Nuclear Generation and other members of licensee management at the conclusion of the inspection on November 30, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Abbott, VP Nuclear Engineering
J. Conway, VP Nuclear Generation
L. Hopkins, Unit 1 Plant Manager
J. Mueller, Senior VP and Chief Nuclear Officer
M. Peckham, Unit 2 Plant Manager
C. Terry, VP Quality Assurance Nuclear
D. Wolniak, Manager, Licensing

ITEMS OPENED, CLOSED, AND DISCUSSED

Items Opened

05000410/2000-008-02	URI	Unit 2, unresolved item involving the maintenance of the FWS-L10 valves per the licensee's Maintenance Rule Program

Items Closed

05000220/2000-003	LER	Reactor Trip on Low Reactor Water Level While Placing the Reactor Water Cleanup System in Service.

Items Opened and Closed

05000410/2000-008-01	NCV	Unit 2, violation of 10CFR50, Appendix B, Criterion III.
05000220/2000-008-03	NCV	Unit 1, violation of Technical Specification 4.3.2.c.
05000220/2000-008-04	NCV	Unit 1, failure to take corrective actions as a result of previous ERV bent stem
05000220/2000-008-05	NCV	Unit 1, failure to take corrective actions for longstanding reactor water cleanup system isolation valves

LIST OF ACRONYMS USED

ANS	Alert and Notification System
ASSS	Assistant Station Shift Supervisor
ATWS	Anticipated Transient Without Scram
CCDP	Change in Core Damage Probability
CDF	Core Damage Frequency
CLS	Low Pressure Core Spray
DER	Deviation/ Event Report
DDC	Document Design Change
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPA	Electrical Protection Assembly
ERO	Emergency Response Organization
ERV	Electromatic Relief Valve
ESA	Engineering Support Analysis
FSAR	Final Safety Analysis Report
FW	Feedwater
HPCI	High Pressure Coolant Injection
IPE	Individual Plant Examination
LER	Licensee Event Report
LERF	Large Early Release Frequency
LOOP	Loss of Offsite Power
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NMPC	Niagara Mohawk Power Corporation
PARS	Publically Available Records
PI	Performance Indicator
PMT	Post-Maintenance Testing
PPC	Plant Process Computer
PRA	Probabilistic Risk Analysis
RBCLC	Reactor Building Closed Loop Cooling
RHR	Residual Heat Removal
RPS	Reactor Protection System
RRCS	Redundant Reactivity Control System
IRM	Intermediate Range Monitor
RWCU	Reactor Water Cleanup
SBLOCA	Small Break Loss of Coolant Accident
SDP	Significance Determination Process
SSC	Structures, Systems and Components
STA	Shift Technical Advisor
TS	Technical Specification
UE	Unusual Event
Unit 1	Nine Mile Point Unit 1
Unit 2	Nine Mile Point Unit 2
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order

PARTIAL LIST OF DOCUMENTS REVIEWED

Unit 1 Documents

N1-MPM-079-412	Diesel Generator Cooling Water Heat Exchanger and Temperature Control Valve Maintenance, Revision 2, dated September 7, 2000.
N1-ST-Q25	Emergency Diesel Generator Cooling Water Quarterly Test, Revision 8, dated May 18, 1998, and Test results dated October 3, 2000. (Note: this is the in-service testing of cooling water pumps and valves)
N1-ST-M4	EDGs/PB 102 and 103 Operability Test, Revision 24.
Drawing D227627	Design Drawing for XF-1303-TR-2P Heat Exchanger, Revision M, dated April 17, 1972.
N1-99-013	Replace NMP-1 EDG Raw Water Tube Bundles, dated July 28, 2000. (A design change document)
N1-MPM-GEN-242	Check Valve Preventive Maintenance, Revision 4, dated August 31, 2000.
N1-ST-C2	Manual Opening of the Solenoid-Actuated Pressure Relief Valves and Flow Verification, Revision 15, dated March 20, 1997.
N1-ST-R20	Manual Exercising of ERV Line Vacuum Breakers, Revision 4, dated October 25, 1995.
N1-MPM-001-245	Main Steam Electromatic Relief Valves and Assorted Pilot Valves Preventive Maintenance, Removal, Overhaul, and Replacement, Revision 2, dated February 14, 1995.
NIP-DES-03	Processing of Vendor Technical Information, Revision 0, dated September 15, 1999.
Cal S15-79HX01	Maximum Allowable Raw Water Temperature for EDG Cooling Water Heat Exchanger, Revision 2 dated July 13, 1999.
DER 1-1999-1478	Increase in Chromate tank level for EDG 103, dated May 6, 1999.
DER1-1997-0710	DG 102 Heat Exchanger Tube Leak, Dated March 13, 1997.
DER 1-1998-2223	Inappropriate Application of Engineering Judgement, Dated July 15, 1998
DER 1-1998-1962	Diesel Generator 102 Jacket Water HXs Degraded Tubes, Dated June 23, 1998.
DER 1-2000-3417	Manual Reactor Scram and Unusual Event Due to Stuck Open ERV #111.

DER 1-2000-3424 Broken Parts From ERV-111 Tailpipe Vacuum Breaker Missing.

DER 1-2000-3438 CKV-66-26 Disk Separated From Hinge Pin.

DER 1-2000-3443 ERV 112 Pilot Valve Slow to Reposition.

1M00829 Design Document Change, Heat Exchanger Tube Plugging, EDG 102/103, Dated August 4, 1999.

Niagara Mohawk's response to NRC's Generic Letter 89-13, dated February 16, 1990 (NMP1L 0478)

Miscellaneous Documents

Analysis Group Technical Report, "Safety and Availability Assessment, NMP1 ERV Sticking Event and Past Condition (DER 1-2000-3417)," dated October 7, 2000.

Engineering Supporting Analysis (ESA) for 10" ADS Vacuum Breakers, dated October 8, 2000.

Design Document Change 1M01005, Modification of Vacuum Breakers and Supporting Document Revisions, dated October 4, 2000.

Licensee Event Report (LER) 50-220/90-16, "Unusual Event Classification and Reactor Shutdown Due to Excess Drywell Leakage Resulting From Unadjusted ERV Pilot Valves," dated August 29, 1990.

Vendor Manual #N1D24500VALVE001, "Instruction For Installation and Maintenance of Consolidated Electromatic Relief Valve (ERV) Nuclear," January 1986 Edition.

Unit 2 Documents

N2-TTP-RHS-4Y001 Residual Heat Removal System Heat Exchangers Performance Monitoring, Revision 0, dated May 1, 1997, and test results dated May 2 and May 4, 1998, and dated March 3 and March 4, 2000.

N2-OSP-LOG-W001 Surveillance Test Procedure, Weekly Check, Revision 6, Dated May 30, 2000, Section 1.3 and Attachment 6 (related to flushing of RHR heat exchanger with demineralized water.)

N2-OSP-EGS-M@001 Diesel Generator and Diesel Air Start Valve Operability Test, Revision 3, dated April 20, 2000, and test results dated July 27, 2000.

Drawing PID-104D-5 P& ID for Jacket Water, Standby Diesel Generator System, dated February 7, 1996.

DER 2-1999-1950 Unexpected Higher Pressure Observed on RHR Heat Exchangers Following RCIC Run, dated May 13, 1999.

DER 2-2000-0882 2RHS*E1A, Deviation Noted During PM, dated March 11, 2000.

DER 2-1997-3357	Deviation From GEK and GE Design Specification Recommendations for RHR Heat Exchangers, dated December 11, 1997.
DER 2-1997-0798	Inadequate Procedural Guidance - SW Flow to RHR HX Post LOCA, dated March 18, 1997.

General Electric Document No. 22A3730, RHR Heat Exchanger, Revision 0, dated December 8, 1977.

General Electric Document No. 23A5554RHR Heat Exchanger Calculated Performance, Revision 0, dated April 23, 1986.

Data Sheets for Diesel Generator Cooling Tubular Heat Exchanger, by Stone and Webster, dated November 16, 1976.

Niagara Mohawk's response to NRC's Generic Letter 89-13, dated February 16, 1990 (NMP2L 1225)

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.