August 7, 2006

Mr. Timothy J. O'Connor
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P.O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000220/2006003 and 05000410/2006003

Dear Mr. O'Connor:

On June 30, 2006, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Nine Mile Point Nuclear Station Unit 1 and Unit 2. The enclosed inspection report documents the inspection results that were discussed on July 18, 2006, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two findings of very low safety significance (Green). One of these findings was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it was entered into your corrective action program, the NRC is treating this finding as a non-cited violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy. In addition, violations of very low safety significance identified by Nine Mile Point Nuclear Station, LLC (NMPNS) are listed in Section 4OA7 of the report. If you contest the non-cited violation noted in this report, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, 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; U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-001; and the NRC Resident Inspector at Nine Mile Point.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any) will be available electronically for public inspection in the

NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web Site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Brian J. McDermott, Chief
Projects Branch 1
Division of Reactor Projects

Docket No.: 50-220, 50-410
License No.: DPR-63, NPF-69

Enclosure: Inspection Report 05000220/2006003 and 05000410/2006003
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-220, 50-410
License No.:	DPR-63, NPF-69
Report No.:	05000220/2006003 and 05000410/2006003
Licensee:	Nine Mile Point Nuclear Station, LLC (NMPNS)
Facility:	Nine Mile Point, Units 1 and 2
Location:	Lake Road Oswego, NY
Dates:	April 1 - June 30, 2006
Inspectors:	L. Cline, Senior Resident Inspector B. Fuller, Resident Inspector E. Knutson, Resident Inspector R. Fuhrmeister, Special Project Engineer J. Furia, Senior Health Physicist S. Lewis, Reactor Inspector J. Noggle, Senior Reactor Inspector J. Schoppy, Senior Reactor Inspector
Approved by:	Brian J. McDermott, Chief Projects Branch 1 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000220/2006003, 05000410/2006003; 04/01/06 -06/30/06; Nine Mile Point, Units 1 and 2; Operability Evaluations and Event Followup.

The report covered a thirteen-week period of inspection by resident inspectors, and announced inspections by a senior health physicist and several regional specialist inspectors. One Green non-cited violation (NCV) and one Green finding were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing finding of very low safety significance occurred on March 9, 2006, when Nine Mile Point Unit 2 automatically scrammed due to a main turbine trip caused by low condenser vacuum. The loss of condenser vacuum occurred when the normal turbine gland seal supply isolated due to high water level and the emergency gland seal steam supply (non-safety related) failed. The emergency gland seal steam supply failed because a maintenance technician improperly assembled a pressure indicating controller for the system following maintenance in April 2004. Maintenance repaired the pressure indicating controller and Operations restored the plant to full power on March 13, 2006. Nine Mile Point Nuclear Station (NMPNS) entered the issue into its corrective action program (CAP) as CR 2006-0993.

  The finding is greater than minor because it was associated with the human performance attribute of the Initiating Event cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during power operations. The inspectors determined the finding to be of very low safety significance using the Phase 1 SDP screening worksheet for at power situations. The finding screened to Green because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, and was not potentially risk significant due to external events. (Section 4OA3).

Cornerstone: Mitigating Systems

- Green. An NRC-identified NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," was identified on February 8, 2006, when the reactor core isolation cooling (RCIC) system was operated in an unanalyzed configuration that degraded plant safety. Specifically, steam exhaust line vacuum breaker isolation valve 2ICS*MOV148 was shut while RCIC remained aligned for automatic operation. This configuration would have prevented the vacuum breakers from mitigating the water hammer event that occurs following system shutdown, which

can produce stresses in the RCIC steam exhaust line that exceed code-allowable values during certain accident scenarios. Operations revised the operating procedure to direct operators to inhibit RCIC automatic initiation if the steam exhaust line vacuum breakers were isolated. NMPNS entered the issue into its CAP as CR 2006-0545.

The finding is greater than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The Region I SRA conducted a Phase 3 risk assessment and determined the finding to be of very low safety significance. The only accident conditions that could cause the suppression pool to pressurize and RCIC to automatically start were medium and large break loss of coolant accidents (LOCAs). The SRA conservatively assumed, based on NMPNS data, that RCIC was in the degraded condition for 3 hours. Using the annual initiating event frequencies from the NMP2 SPAR model for medium and large break LOCAs, the SRA determined that the delta-CDF could not be greater than the low E-8 range, because even if RCIC caused the failure of all injection sources, the increase in the probability of core damage could not be greater than the initiating event frequency adjusted for the exposure time. (Section 1R15)

B. Licensee-Identified Violations

Violations of very low safety significance that were identified by NMPNS have been reviewed by the inspectors. Corrective actions taken or planned by NMPNS have been entered into NMPNS' CAP. These violations are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Nine Mile Point Unit 1 (Unit 1) began the inspection period at 100% power. On June 2, 2006, operators reduced power to 65% to conduct testing to identify and suppress a leaking fuel assembly. On June 5, operators restored power to 100%. On June 10, operators reduced power to 15% and deinerted the drywell in preparation for a drywell entry to identify the cause of elevated reactor coolant system (RCS) unidentified leakage. Operators identified a leaking recirculation loop drain valve as the cause, and after an unsuccessful attempt to reduce leakage, operators shutdown the reactor to complete repairs. On June 15, operators restored power to 100% and the plant remained at 100% power for the remainder of the report period.

Nine Mile Point Unit 2 (Unit 2) began the inspection period in refueling outage (RFO) 10 that began on March 20, 2006. On April 12, operators commenced plant startup and restored 100% power on April 15. The plant remained at 100% power through the end of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events/Mitigating Systems/Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample)

a. Inspection Scope

The inspectors completed one adverse weather protection sample. The inspectors reviewed and verified completion of the operations department warm weather preparation checklists contained in procedures N1-OP-64, “Meteorological Monitoring, Attachment 2: Hot Weather Preparation Checklist” and N2-OP-102, “Meteorological Monitoring, Attachment 3: Hot Weather Preparation Checklist,” for Units 1 and 2 respectively. The inspectors reviewed the operating status and lineups for the Unit 1 reactor and turbine building (TB) closed loop cooling systems and the reactor and TB ventilation systems at Unit 1 and 2, reviewed the procedural limits and actions associated with elevated lake temperature, and walked down accessible areas of the buildings to assess the effectiveness of the ventilation systems. The walkdowns included discussions with operations personnel to ensure that they were aware of temperature restrictions and required actions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04 - 4 samples, 71111.04S - 1 sample)

.1 Partial System Walkdown

a. Inspection Scope

The inspectors performed four partial system walkdowns to verify a train was properly restored to service following maintenance or to evaluate the operability of one train while the opposite train was inoperable or out of service for maintenance and testing.

The inspectors compared system lineups to system operating procedures (OPs), system drawings, and the applicable chapters in the Updated Final Safety Analysis Report (UFSAR). The inspectors also verified the operability of critical system components by observing component material condition during the system walkdown and reviewing the maintenance history for each component. Documents reviewed during this inspection are listed in the Attachment. The inspectors performed partial walkdowns of the following systems:

- Unit 1 containment spray 112 inspected on April 11, 2006, during surveillance testing of containment spray 122;
- Unit 2 low pressure core spray (LPCS) system on April 26, 2006;
- Unit 1 115 kV Line 4 and emergency diesel generator (EDG) 102 and EDG 103 on May 17, 2006, during planned maintenance outage on 115 kV Line 1; and
- Unit 2 Division II standby liquid control (SLC) system on June 19, 2006, due to maintenance scheduled for later that week on Division I SLC system.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown (1 sample)

a. Inspection Scope

The inspectors performed a complete walkdown of the Unit 1 containment spray system to identify any discrepancies between the existing equipment lineup and the specified lineup. During the walkdown system drawings and OPs were used to verify proper equipment alignment and operational status. The inspectors reviewed the open maintenance work orders (WOs) on the system for any deficiencies that could affect the ability of the system to perform its function. Documentation associated with unresolved design issues such as temporary modifications (TMs), operator workarounds, and items tracked by plant engineering were also reviewed to assess their collective impact on system operation. In addition, the inspectors reviewed the condition report (CR) database to verify that equipment alignment problems were being identified and appropriately resolved.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Protection - Tours (71111.05Q - 9 samples, 71111.05A - 1 sample)

a. Inspection Scope

Quarterly. The inspectors completed nine quarterly fire protection inspection samples. The inspectors toured nine areas important to reactor safety on the Nine Mile Point site to evaluate NMPNS' control of transient combustibles and ignition sources and the

material condition, operational status, and operational lineup of fire protection systems including detection, suppression and fire barriers. The inspectors used procedure GAP-INV-02, “Control of Material Storage Areas,” the UFSARs for Unit 1 and Unit 2, the fire hazards analysis and pre-fire plans to perform the inspection. The areas inspected included:

- Unit 1 TB 300 foot elevation;
- Unit 1 reactor feed pump area TB 261 foot elevation;
- Unit 1 Reactor Building (RB) 340 foot elevation;
- Unit 1 control rod drive hydraulic control units area RB 237 foot elevation;
- Unit 2 Division 1 and 2 service water pump rooms;
- Unit 2 LPCS room, RB 175 foot elevation;
- Unit 2 A residual heat removal pump room, RB 175 foot elevation;
- Unit 2 Division 1 cable chase room, control building 306 foot elevation; and
- Unit 2 relay room, control building 288 foot elevation.

b. Findings

No findings of significance were identified.

.2 Fire Protection - Drill Observation

a. Inspection Scope (1 sample)

The inspectors completed one annual fire drill observation inspection sample. The inspectors observed a fire brigade drill on June 5, 2006, in the Unit 2 Division 2 cable routing area. The inspectors observed brigade performance during the drill to evaluate the following: donning and use of protective equipment, fire brigade leader command and control, fire brigade response time, radio communications, and use of pre-fire plans. The inspectors attended the post-drill critique and reviewed the disposition of issues and deficiencies identified during the drill. The inspectors also verified that all fire fighting equipment used during the drill was returned to a condition of readiness required to respond to an actual fire when the scenario was complete. Documents reviewed for this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 sample)

.1 Internal Flooding

a. Inspection Scope

The inspectors completed one internal flooding inspection sample. The inspectors reviewed the Individual Plant Examination (IPE), the Probabilistic Risk Assessment (PRA) and UFSAR for Unit 2 concerning internal flooding events and completed a walkdown of one area in which flooding could have a significant impact on risk, the pipe

tunnel from the TB to the RB. The flooding scenario of concern was flooding from the service water supply to the TB closed loop cooling water heat exchangers (HXs), with the possibility that this could lead to flooding of the RB via the pipe tunnel. The inspectors verified the validity of assumptions made in the IPE regarding this flooding scenario.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A - 1 sample)

a. Inspection Scope

The inspectors completed one annual heat sink performance inspection sample. The inspectors observed NMPNS HX inspections and the state of cleanliness of the tubes for the 11 reactor building closed loop cooling (RBCLC) HX. The inspectors verified that NMPNS procedure N1-MPM-070-409, “RBCLC Water HXs 70-13R, 70-14R, 70-15R,” which was performed on an annual basis to clean and inspect the RBCLC HXs, used the methods outlined in EPRI Report NP-7552, “HX Performance Monitoring Guidelines.” The inspectors reviewed recent performance data and verified tube plugging limits with the actual number of tubes plugged in the HX to ensure that HX operation was consistent with design. Other documents reviewed for this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08 - 5 samples)

a. Inspection Scope

The purpose of this inspection was to assess the effectiveness of NMPNS’ program for monitoring degradation of the RCS boundary, risk significant piping system boundaries, and the containment boundary. The inspectors assessed the inservice inspection (ISI) activities using the criteria specified in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI.

The inspectors observed a sample of nondestructive examination activities. These included video and measurement recordings and reports of volumetric, surface, and visual examinations. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation could result in an increase in core damage probability. The observation and documentation review was conducted to verify the activities were performed in accordance with the ASME Boiler and Pressure Vessel Code requirements. The inspectors reviewed the in-service inspection (ISI) results from Unit 2 RFO 10 and noted that there were no indications outside the acceptance criteria of the ASME, Section XI Code that required repair or engineering evaluation for continued service. Also, during Unit 2 RFO 10, no

welding was performed on any pressure boundary component for any Class 1 or 2 system. The inspectors also evaluated effectiveness in the resolution and corrective action of problems identified during ISI activities for selected samples.

The inspectors reviewed the following ISI examination measurement, video recordings, and documentation of ISI examination reports conducted during Unit 2 RFO 10. Other documents reviewed for this inspection are listed in the Attachment.

Ultrasonic Testing

High Pressure Core Spray (HPCS), N–16, 2RPV-KC32
Recirculation Suction, N1A, 2RPV-KB01
Recirculation Discharge, N2K, 2RPV-KB12
Recirculation Discharge, N2G, 2RPV-KB09
Feedwater Discharge, N4C, 2RPV-KB19
Low Pressure Core Spray, N5B, 2RPV-KB23

Liquid Penetrant Testing

Control Rod Drive Housing (CRD), 2RPV-CRDH036A
Residual Heat Removal, Class 2 weld, 2RHSV376
HPCS, Class 2 weld, 2CSH-25-18-FW300-301

Magnetic Particle Testing

Low Pressure Core Spray, Class 2 weld, 2CSLV121

Visual Testing

HPCS, Class 2 restraint, 2CSH-PSR177A2

In-Vessel Visual Inspection

Video recordings of the following reactor components were sampled and reviewed during this inspection: steam dryer drain channel no. 4; jet pump wedge nos. 6 and 10; jet pump ramshead nos. 3 and 4; and jet pump beam nos. 16 and 20.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q - 3 samples)

a. Inspection Scope

The inspectors completed three licensed operator requalification training program (LORT) inspection samples. Documents reviewed for this inspection are listed in the Attachment. For each scenario observed, the inspectors assessed the clarity and effectiveness of communications, the implementation of appropriate actions in response

to alarms, the performance of timely control board operation and manipulation, and the oversight and direction provided by the shift manager. During the scenario the inspector also compared simulator performance with actual plant performance in the control room. The following simulator scenarios were observed:

- On May 30, 2006, the inspectors observed Unit 2 LORT to assess operator performance during a scenario involving a loss of off-site power while operators were performing a Division I EDG surveillance. The inspectors evaluated the performance of significant operator actions directed by plant OPs, including N2-SOP-03, “Loss of AC Power,” and N2-ARP-01, “Control Room Alarm Response Procedures.”

- On June 1, 2006, the inspectors observed licensed operator performance in the Unit 2 simulator during an emergency preparedness exercise. The exercise scenario involved a loss of offsite power and an off-gas system trip followed by indications of a small reactor coolant leak in the drywell and a reactor water cleanup system valve packing failure that resulted in a reactor coolant leak outside the primary containment. The inspectors evaluated the performance of risk significant operator actions including the use of emergency operating procedures (EOPs), N2-EOP-RPV, “RPV Control,” N2-EOP-PC, “Primary Containment Control,” N2-EOP-SC, “Secondary Containment Control,” and N2-EOP-RR, “Radioactivity Release Control.”

- On June 6, 2006, the inspectors observed Unit 1 LORT to assess operator performance during a scenario involving a main steam isolation valve closure that resulted in scram dump volume drain valve packing failure, followed by a loss of the plant process computer. The inspectors evaluated the performance of risk significant operator actions, including the EOPs, N1-EOP-02, “RPV Control,” N1-EOP-05, “Secondary Containment Control,” and N1-EOP-08, “RPV Blowdown.”

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 3 samples, 71111.12B - 5 samples)

a. Inspection Scope

Routine Inspection. The inspectors completed three maintenance effectiveness routine inspection samples. The inspectors reviewed performance-based problems or performance and condition history reviews involving selected in-scope structures, systems or components (SSCs) to assess the effectiveness of the maintenance program. Reviews focused on: proper Maintenance Rule (MR) scoping in accordance with 10 CFR 50.65; characterization of reliability issues; tracking system and component unavailability; 10 CFR 50.65 (a)(1) and (a)(2) classifications; identifying and addressing common cause failures, trending key parameters, and the appropriateness of performance criteria for SSCs classified (a)(2), as well as, the adequacy of goals and corrective actions for SSCs classified (a)(1). The inspectors reviewed system health

reports, maintenance backlogs, and MR basis documents. Other documents reviewed for the inspection are listed in the Attachment. The following three MR inspection samples were reviewed:

- Unit 1 EDG system performance;
- Unit 1 fire protection water system performance; and
- Unit 2 residual heat removal system performance.

Periodic Evaluation. The inspector reviewed the two most recent 10 CFR 50.65 (a)(3) periodic evaluations to verify that NMPNS adequately balanced the reliability and unavailability for structures, systems and components (SSCs) contained within the scope of the MR. The inspector reviewed the safety significant systems that were in (a)(1) status to verify that NMPNS: (1) established appropriate goals and performance criteria; (2) considered applicable industry operating experience; (3) developed and implemented effective corrective action plans; and (4) adequately monitored performance. The inspector reviewed the following four SSCs that were in (a)(1) status in June 2006: (1) Unit 1 high pressure coolant injection condensate pump No. 13; (2) Unit 1, 115 kV switchyard disconnect switches; (3) Unit 1 emergency service water pump train No. 12; and (4) Unit 2 diesel generator ventilation motor-operated damper actuators. The inspector also reviewed Unit 2 service water Clow butterfly valves that NMPNS returned to (a)(2) status in December 2004.

The inspector reviewed the following safety significant (a)(2) systems to confirm that their performance met the applicable MR performance criteria: (1) Unit 1 emergency cooling, and (2) Unit 2 RBCLC. The inspector also walked down accessible portions of the above SSCs with system engineers to evaluate the effectiveness of NMPNS' maintenance efforts. Documents reviewed for this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 10 samples)

a. Inspection Scope

The inspectors reviewed risk assessments for ten work weeks during the inspection period. The inspectors verified that risk assessments were performed in accordance with GAP-OPS-117, "Integrated Risk Management," that risk of scheduled work was managed through the use of compensatory actions and schedule adherence; and that applicable contingency plans were properly identified in the integrated work schedule. Documents reviewed for the inspection are listed in the Attachment.

The following work weeks were reviewed:

Unit 1

- Week of April 17, 2006, that included maintenance on the turbine electro-hydraulic control system and the environmental risk associated with an extensive concrete pour for the Unit 1 condensate system iron pre-filter modification on the Unit 1 TB 300 foot elevation.

- Week of April 24, 2006, that included a maintenance outage for 115 kV offsite power Line 3, EDG 102 quarterly operability testing, containment spray 112 quarterly operability testing and manual feedwater level control operations during maintenance on control room level recorders.

- Week of May 15, 2006, that included a maintenance outage for 115 kV offsite power Line 1, 11 reactor recirculation pump motor generator set maintenance, and high pressure coolant injection system pump and valve testing.

- Week of June 5, 2006, that included SLC system monthly operability testing, emergency condenser level control system valve exercising, 345 kV switchyard Line 8 maintenance and a downpower to 15 percent to facilitate drywell inspections due to increased unidentified RCS leakage measurements.

- Week of June 12, 2006, that included Unit 1 reactor startup following forced outage 1F601, EDG 103 monthly operability testing and core spray 122 quarterly operability testing.

Unit 2

- Week of April 17, 2006, that included Division 1 EDG monthly operability testing, A residual heat removal (RHR) operations and instrumentation and controls testing, and outage demobilization activities.

- Week of April 24, 2006, that included HPCS pump and valve maintenance, Division 3 EDG monthly operability testing and local power range and average power range meter calibrations.

- Week of May 8, 2006, that included 115 kV offsite power B reserve station transformer breaker and relay calibrations and reactor core isolation cooling (RCIC) maintenance and quarterly operability testing.

- Week of May 22, 2006, that included Division 1 safety-related switchgear under and degraded voltage testing, RHR A instrumentation and controls, maintenance and operations monthly operability testing and A safety-related uninterruptible power supply, 2VBA*UPS2A, maintenance.

- Week of June 19, 2006, that included planned maintenance on Division 1 SLC system and LPCS maintenance and testing.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Evolutions and Events (71111.14 - 4 samples)

a. Inspection Scope

The inspectors assessed operator performance during the four non-routine evolutions described below. The inspectors reviewed operator logs, interviewed operators and plant management. When possible, the inspectors conducted control room observations to determine what occurred, how the operators responded, and if the response was in accordance with plant procedures and management expectations. Other documents reviewed for the inspection are listed in the Attachment.

- On June 2, 2006, Unit 1 operators performed a downpower to approximately 65 percent power to conduct local power suppression testing to identify and suppress a leaking fuel assembly. Unit 1 was restored to full power on June 5, 2006.

- On June 11, 2006, Unit 1 operators performed a TS required shutdown and cooldown due to a greater than 2 gallons per minute increase in unidentified RCS leakage over a 24-hour period. Operators restarted Unit 1 on June 12, after the necessary repairs were completed.

- On June 14, 2006, NMPNS repaired a leak on a drain line for a Unit 1 balance-of-plant steam seal regulator. Due to the potential impact of the repair method on condenser vacuum, operations controlled the performance of the maintenance as a special evolution in accordance with procedure GAP-OPS-117, "Integrated Risk Management." The inspectors attended the station operations review committee meeting that reviewed the work plan and observed control room activities during the evolution including the special evolution pre-job brief.

- On April 28, 2006, the Unit 2 B instrument air compressor (IAC) tripped due to high discharge temperature. The A and C IACs both started, but header pressure continued to degrade because the B IAC discharge check valve failed to shut. Operators entered special operating procedure (SOP)-19, "Loss of Instrument Air." Instrument air header pressure recovered when the operator was dispatched by the control room to investigate shut the manual discharge valve for the B IAC.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 7 samples)

a. Inspection Scope

The inspectors completed seven operability evaluation inspection samples. The inspectors reviewed operability determinations to assess the acceptability of the evaluations; when needed, the use and control of compensatory measures; and the compliance with TSs. The inspectors' review included a verification that the operability determinations were made as specified by Procedure S-ODP-OPS-0116, "Operability Determinations." The technical adequacy of the determinations was reviewed and compared to the TSs, UFSAR, and associated design basis documents (DBDs). The following eight evaluations were reviewed:

- CR 2006-0545 concerning operation of RCIC with the turbine exhaust line vacuum breakers isolated;
- CR 2006-1306 concerning leakage from a pin hole on the suction line to Unit 1 reactor feedwater pump 12;
- WOs 02-05317-00 and 05-17611-00 concerning the operability of Unit 1 hydraulic control accumulator units (HCUs) during reactor manual control system maintenance that resulted in all hydraulic control unit accumulators being in alarm for a period of approximately eight hours;
- CR 2006-2278 concerning a failed rupture disc in the reactor core isolation cooling system turbine steam supply drain line;
- CR 2006-2879 concerning the effect of a failed closed damper, BV-210-34, "Motor Operated Blocking Damper - Supply Air Blocking Valve to Aux Control Room," on the control room emergency ventilation system;
- CR 2006-1416 concerning snubber 2CCP-PSSP267A4 that failed its acceleration function test; and
- CR 2006-2069 concerning service life of adhesive labels attached to cable, conduit and cable trays in the Unit 2 drywell.

b. Findings

Introduction. An NRC-identified Green NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," was identified on February 8, 2006, when the reactor core isolation cooling (RCIC) system was operated in an unanalyzed configuration that degraded plant safety. Specifically, steam exhaust line vacuum breaker isolation valve 2ICS*MOV148 was shut while RCIC remained aligned for automatic operation. This configuration would have prevented the vacuum breakers from mitigating the water hammer event that occurs following system shutdown, which can produce stresses in the RCIC steam exhaust line that exceed code-allowable values during certain accident scenarios.

Description. On February 8, electrical maintenance performed preventive maintenance on the electrical supply breaker to one of the two RCIC system steam exhaust line vacuum breaker isolation valves, 2ICS*MOV148. This valve is normally open, but because it is a containment isolation valve, it was placed in the shut position in preparation for the breaker maintenance. Operations maintained the system aligned for automatic operation and considered it inoperable but available.

During a routine control room panel walkdown on February 8, the inspectors observed RCIC in this configuration and questioned the potential for a water hammer in the steam exhaust line following system shut down. Pending a formal engineering evaluation, NMPNS decided to shut the RCIC trip throttle valve to prevent automatic initiation while 2ICS*MOV148 was shut. As an immediate corrective action, Operations developed a change to operating procedure N2-OP-35, "Reactor Core Isolation Cooling," that directed operators to inhibit automatic initiation of RCIC if the steam exhaust line vacuum breakers were isolated.

NMPNS completed a formal water hammer stress analysis for the steam exhaust line. The analysis demonstrated that, with the vacuum breakers isolated, the water hammer caused by the collapse of residual steam in the exhaust line after the RCIC turbine shuts down can produce stresses in the steam exhaust line that exceed ASME code allowable levels. Specifically, NMPNS determined that if the suppression pool was pressurized as a result of an accident, the RCIC turbine exhaust piping could be over-stressed if RCIC started during the accident and then tripped due to high RCS water level or low steam line pressure. The failure of the RCIC turbine exhaust piping could lead to draining the suppression pool to the RB through the break. The lower suppression pool level could reduce the net positive suction head available to the emergency core cooling system (ECCS) pumps negatively impacting their ability to inject to the RCS or cool the suppression pool.

<u>Analysis</u>. The performance deficiency associated with this event was the alignment of RCIC for automatic operation in a configuration that was not adequately evaluated and could have impacted the safety function of ECCS equipment and primary containment during certain accident scenarios. Traditional enforcement does not apply because the issue did not have an actual safety consequence or a potential for impacting the NRC's regulatory function, and it was not the result of any willful violation of NRC requirements. The finding was greater than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors determined that a Phase 2 risk assessment was required because the finding degraded both the Barrier Integrity and Mitigating Systems cornerstones. The Region I SRA determined that a Phase 2 risk assessment was not appropriate because under certain accident conditions the potential failure of the RCIC turbine exhaust line could lead to a common mode failure of all emergency core cooling and suppression pool cooling safety functions. The Region I SRA conducted a Phase 3 risk assessment and determined the finding to be of very low safety significance. The only accident conditions that could cause the suppression pool to pressurize and RCIC to automatically start were medium and large break LOCAs. The SRA conservatively assumed, based on NMPNS data, that RCIC was in the degraded condition for 3 hours. Using the annual initiating event frequencies from the NMP2 SPAR model for medium and large break LOCAs, the SRA determined that the delta-CDF could not be greater than the low E-8 range, because even if RCIC caused the failure of all injection sources, the increase in the probability of core damage could not be greater than the initiating event frequency adjusted for the exposure time.

Enforcement. 10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that, "measures shall be established to assure that applicable regulatory requirements and design basis . . . for those structures, systems, and components to which this appendix applies are correctly translated into . . . procedures and instructions." Contrary to the above, on February 8, 2006, the RCIC system was placed in an alignment for automatic initiation that was contrary to its design. Specifically, insufficient information regarding the RCIC turbine exhaust line vacuum breakers was included in procedures and instructions to ensure that alignment of the system would remain consistent with the plant's design basis during a maintenance activity. Because this finding is of very low safety significance and has been entered into the NMPNS's CAP as CR 2006-0545, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000410/2006003-01, RCIC Alignment During Maintenance Not Consistent With Design Bases.

1R19 Post Maintenance Testing (71111.19 - 8 samples)

a. Inspection Scope

The inspectors completed eight post maintenance testing inspection samples. The inspectors reviewed post maintenance test procedures and associated testing activities for selected risk significant mitigating systems to assess whether the effect of maintenance on plant systems was adequately addressed by control room and engineering personnel. The inspectors verified that test acceptance criteria were clear; demonstrated operational readiness and were consistent with design basis documents (DBD); that test instrumentation had current calibrations and the range and accuracy for the application; and that tests were performed, as written, with applicable prerequisites satisfied. Upon completion, the inspectors verified that equipment was returned to the proper alignment necessary to perform its safety function. The adequacy of the identified post maintenance testing requirements were verified through comparisons with the recommendations of GAP-SAT-02, "Pre/Post-Maintenance Test Requirements," and the design basis information contained in the TSs, UFSAR and associated DBDs. Other documents reviewed for this inspection are listed in the Attachment. The following eight post maintenance test activities were reviewed:

- Unit 1 WO 05-16591-00, for breaker preventive maintenance (PM) on reactor recirculation sample isolation, IV 110-127, on April 29. The retest was performed by stroking the valve in accordance with N1-ST-Q4, "Reactor Coolant System Isolation Valves Operability Test."

- Unit 1 WO 06-00837-00, for inspection and cleaning of fire water pre-action zone 2083. The retest was performed by flow testing the zone in accordance with a one time change to N1-OP-21A, "Fire Protection System - Water."

- Unit 1 WO 06-16407-00, for replacement of the electronic pressure regulator MOOG valve on April 21. The retest was performed by stroke timing the valve in accordance with N1-IPM-302-001, Attachment 18 "Mid cycle MOOG valve replacement," and observing proper pressure control by the EPR.

- Unit 1, WO 05-01609-00 and 05-26501-00, that the performed Line 1, 115 kV supply breaker five year overhaul preventive maintenance. The retest was performed in accordance with N1-EPM-GEN-298, "Power Circuit Breaker P.M.," Section 7.2.

- Unit 2 N2-OSP-RPV-@003, "Reactor Pressure Vessel and All Class I Systems Leakage Test with the RPV Solid," performed as PMT for vessel reassembly and various other component maintenance performed during the refueling outage.

- Unit 2 N2-OSP-SLS-0001, "SLC Pump, Check Valve, Relief Valve Operability Test and ASME XI Pressure Test," performed as PMT for replacement of the Division I pump discharge relief valve, 2SLS*RV2A.

- Unit 2, WO 06-07714-00, the replaced that A Instrument Air compressor discharge check valve 2IAS-V1791A. The retest was performed in accordance with N2-MPM-IAS-V606, "Instrument Air Compressor P.M. 2IAS-C3A, 2IAS-C3B, and 2IAS-C3C," Section 7.8, and the WO step text.

- Unit 2, WO 05-17016-00 that performed the annual diesel driven fire pump engine inspection. The retest was performed in accordance with N2-MPM-FPW-A854, "Diesel Driven Fire Pump Engine Inspection," Section 4.5, and N2-OSP-FOF-W001, "Engine Driven Fire Pump Operability and Storage Tank Level Test."

b. <u>Findings</u>

No findings of significance were identified.

1R20 <u>Refueling and Other Outage Activities</u> (71111.20 - 2 samples)

a. <u>Inspection Scope</u>

<u>Unit 2 RFO 10</u>. The inspectors observed and/or reviewed the following refueling outage activities to verify that operability requirements were met and that risk, industry experience, and previous site specific problems were considered. Documents reviewed for this inspection are listed in the Attachment.

- The inspectors reviewed outage schedules and procedures, verified that TS-required safety system availability was maintained and shutdown risk was minimized. The inspectors verified that when specified by NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," and NMPNS procedure NIP-OUT-01, "Shutdown Safety," contingency plans existed for restoring key safety functions.

- Through plant tours, the inspectors verified that NMPNS maintained and adequately protected electrical power supplies to safety-related equipment and that TS requirements were met.

- The inspectors verified proper alignment and operation of shutdown cooling and other decay heat removal systems. The verification also included reactor cavity and fuel pool makeup paths and water sources, and administrative control of drain down paths.

- The inspectors reviewed N2-FHP-003, “Refueling Manual,” N2-FHP-13.3, “Core Shuffle,” N2-ODP-NFM-0101, “Refueling Operations,” and TS, and verified all requirements for refueling operations were met through refuel bridge observations, control room panel walkdowns and surveillance procedure reviews.

- Before the drywell was closed to general access for start-up, the inspectors performed an “as-left” walkdown to identify evidence of RCS leakage and verify the condition of drywell coatings, structures, valves, piping, supports and other equipment in areas where maintenance was completed. The inspectors also verified that no debris was left in the drywell that could affect the performance of the ECCS suction strainers.

Unit 1 Forced Outage 1F601. The inspectors observed and reviewed the following activities during the Nine Mile Point Unit 1 forced outage 1F601 from June 11 to June 13, 2006. Documents reviewed for this inspection are listed in the Attachment.

- The inspectors observed portions of the plant shutdown and cooldown on June 11, and verified that the TS cooldown rate limits were satisfied.

- The inspectors reviewed outage schedules and procedures and verified that TS required safety system availability was maintained, shutdown risk was considered, and that contingency plans existed to restore key safety functions such as electrical power and containment integrity.

- The inspectors performed a walkdown of the drywell to identify evidence of RCS leakage, and verify the condition of drywell coatings, structures, valves, piping, supports and other equipment. The inspectors also verified that no debris was left in the drywell that could affect the performance of the ECCS suction strainers.

- The inspectors observed portions of the reactor startup following the outage, and verified through plant walkdowns, control room observations, and ST reviews that safety-related equipment required for mode change was operable.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 8 samples)

a. Inspection Scope

The inspectors witnessed performance of and/or reviewed test data for eight risk-significant STs to assess whether the SSCs tested satisfied TS, UFSAR, Technical

Requirements Manual, and NMPNS procedure requirements. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness and were consistent with the DBDs; that test instrumentation had current calibrations and the range and accuracy for the application; and that tests were performed, as written, with applicable prerequisites satisfied. Upon ST completion, the inspectors verified that equipment was returned to the status specified to perform its safety function. Documents reviewed for this inspection are listed in the Attachment. The following eight STs were reviewed:

- N1-ISP-044-005, “High Water Level Scram Discharge Volume Instrument Channel Functional Calibration;”
- N1-ST-Q1A, “Core Spray 111 Pump Valve and SDC Water Seal Check Valve Operability Test;”
- N1-ST-Q26, “Feedwater and Main Steam Line Power Operated Isolation Valves Partial Exercise Test and Associated Functional Testing of Reactor Protection System Trip Logic;”
- N2-OSP-EGS-R004, “Operating Cycle Diesel Generator Simulated Loss of Offsite Power with ECCS Division I & II;”
- N2-OSP-ICS-Q@002, “RCIC Pump and Valve Operability Test and System Integrity Test and ASME XI Functional Test;”
- N1-ST-Q28, “Containment Spray Raw Water Intertie Check Valve Quarterly Operability Test;”
- N2-ISP-ISC-Q017, Quarterly Functional Test of Feedwater/Main Turbine Trip on Reactor Vessel Water Level High Level 8 Instrument Channels; and
- N1-ST-1B, “Liquid Poison Pump 12 Operability Test.“

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 3 samples)

a. Inspection Scope

The inspectors completed three TM inspection samples. For the temporary change packages listed below the inspectors verified that the installation and removal of TMs did not affect the safety functions for the associated systems. The inspectors assessed the adequacy of the 10 CFR 50.59 evaluations; verified that the changes did not adversely affect the system’s ability to perform its design functions as described in the UFSAR and TS; that the installation and removal were consistent with the modification documentation; that the drawings and procedures were updated as applicable; and that the post-installation and restoration testing was adequate. Documents reviewed for this inspection are listed in the Attachment.

- Unit 2 TCP No. N2-05-027, Jumper cell 21 on Battery 2BYS*BAT2B.
- Unit 1 TCP No. N1-06-011, 12 feed water pump suction relief valve leak repair.
- Unit 1 TCP No. N1-06-013, Disconnect pin 6 input signal to buffer card for control rod 38-35 indication.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors completed one drill evaluation inspection sample. The inspectors observed simulator, technical support center and emergency operations facility activities associated with Unit 2 emergency planning drill on June 1, 2006. The inspectors verified that emergency classification declarations and notifications were completed in accordance with 10 CFR 50.72, 10 CFR 50, Appendix E, and the Nine Mile Point emergency plan implementing procedures. Documents reviewed for this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 6 samples)

a. Inspection Scope

The inspectors reviewed NMPNS' self assessments, audits, Licensee Event Reports (LERs), and Special Reports related to the access control program since the last inspection, and determined that identified problems were entered into the CAP for resolution.

The inspectors reviewed corrective action reports related to access controls. Included in this review were high radiation area (HRA) radiological incidents in HRAs <1R/hr that have occurred since the last inspection in this area.

For repetitive deficiencies or significant individual deficiencies in problem identification and resolution the inspectors determined that the NMPNS' self-assessment activities were also identifying and addressing these deficiencies.

The inspectors reviewed NMPNS' documentation packages for all PI events occurring since the last inspection; determined if any of these PI events involved dose rates >25 R/hr at 30 centimeters or >500 R/hr at 1 meter; and, determined what barriers had failed and if there were any barriers left to prevent personnel access. For unintended exposures >100 mrem Total Effective Dose Equivalent or >5 rem Skin Dose Equivalent

or >1.5 rem Lens Dose Equivalent, the inspectors determined if there were any overexposures or substantial potential for overexposure.

The inspectors reviewed radiological problem reports since the last inspection which found that the cause of the event was due to radiation worker errors; determined if there was an observable pattern traceable to a similar cause; and, determined if this perspective matched the corrective action approach taken by NMPNS to resolve the reported problems. The inspectors discussed with the Radiation Protection (RP) Manager any problems with the corrective actions planned or taken. The inspectors verified adequate posting and locking of entrances to all reasonably accessible high dose rate areas including high radiation, and very HRAs.

The inspectors reviewed radiological problem reports since the last inspection found that the cause of the event was RP technician error; determined if there was an observable pattern traceable to a similar cause; and, determined if this perspective matched the corrective action approach taken by NMPNS to resolve the reported problems.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 2 samples)

a. Inspection Scope

The inspectors reviewed NMPNS' self assessments, audits, and special reports related to the as low as is reasonably achievable (ALARA) program since the last inspection and determined if NMPNS' overall audit program's scope and frequency for all applicable areas under the Occupational Radiation Safety cornerstone met the requirements of 10 CFR 20.1101.

The inspectors determined if identified problems were entered into the CAP for resolution; reviewed dose significant post-job reviews and post-outage ALARA report critiques of exposure performance; and, determined if identified problems were properly characterized, prioritized, and resolved in an expeditious manner.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03 - 1 sample)

a. Inspection Scope

The inspectors reviewed the plant UFSAR to identify applicable radiation monitors associated with transient high and very HRAs including those used in remote emergency assessment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151 -10 samples)

a. Inspection Scope

The inspectors reviewed performance indicator (PI) data for the below listed cornerstones and used NEI 99-02, "Regulatory Assessment PI Guidance," to verify individual PI accuracy and completeness.

Cornerstone: Initiating Events

- Unplanned Scrams per 7000 Critical Hours
- Scrams with a Loss of Normal Heat Removal
- Unplanned Transients per 7000 Critical Hours

The inspectors reviewed a selection of LERs, operator log entries, monthly operating reports, and PI data sheets to determine whether NMPNS adequately identified the number of scrams and unplanned power changes greater than 20 percent that occurred at Unit 1 and Unit 2 from September 2004 to March 2006. This number was compared to the number reported for the PI following the first quarter of 2006. The inspectors also verified the accuracy of the number of critical hours reported and NMPNS' basis for crediting normal heat removal capability for each of the reported reactor scrams.

Cornerstone: Barrier Integrity

- RCS activity
- RCS leakage

The inspectors reviewed operator logs, plant computer data, and daily sampling and surveillance procedure results for Unit 1 and Unit 2 to verify the accuracy of NMPNS' reported maximum RCS identified leakage and reactor coolant activity from September 2004 to March 2006. These numbers were compared to the number reported for these PIs following the first quarter of 2006.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152 - 2 samples)

.1 Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As specified by Inspection Procedure 71152, “Identification and Resolution of Problems,” and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of all items entered into NMPNS’s CAP. The review was accomplished by accessing the computerized database for CRs and attending CR screening meetings. In accordance with the baseline inspection modules, the inspectors also selected 218 CAP items across the Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, and Occupational Radiation Safety cornerstones for additional follow-up and review. The inspectors assessed NMPNS’s threshold for problem identification, the adequacy of the cause analyses, extent of condition review, and operability determinations, and the timeliness of the specified corrective actions. The CRs reviewed are noted in the Attachment.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope

As specified by Inspection Procedure 71152, “Identification and Resolution of Problems,” the inspectors performed a review of the NMPNS’ CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors’ review was focused on repetitive equipment and corrective maintenance issues. To perform the review, the inspectors examined CRs prepared from January 1 - June 26, 2006. The inspectors compared and contrasted the results of their review with the results contained in the NMPNS first quarter integrated quarterly assessment report. Corrective actions associated with a sample of the issues identified in the quarterly assessment report were reviewed for adequacy.

b. Assessment and Observations

The inspectors did not identify any adverse performance trends that were not already documented in the NMPNS CAP.

.3 Annual Sample

a. Inspection Scope

The inspectors selected CR 2005-3583 for detailed review. The CR was associated with activities conducted in 2005 by the Quality and Performance Assessment group reviewing fire protection program and equipment improvement actions. The CR

documented that improvement activities for some long-standing issues had not been fully successful. The CR was reviewed to determine whether the full extent of the problems were identified, that an appropriate evaluation was performed, and appropriate corrective actions were specified. The inspectors evaluated the reports against the requirements of procedure NIP-ECA-01, "Corrective Action Program," and 10 CFR 50, Appendix B.

b. Findings and Observations

There were no findings identified with the sample reviewed. NMPNS' extent of condition review determined that several past improvement plans had not been effective in resolving equipment and program issues. The root cause evaluation determined that the root and contributing causes were not unique to Fire Protection, but were common to several other Category 1 CRs. While specific actions were developed to address fire protection issues, the organizational issues related to resource allocation, responsibility and authority to effect changes and coordination between work groups were addressed more broadly on a site-wide basis. The inspector determined that there were no immediate safety concerns since compensatory measures were in place for degraded SSCs.

4OA3 Event Followup (71153 - 4 samples)

.1 (Closed) LER 05000410/2005-001-00, Both Standby Gas Treatment (SGT) Subsystems Inoperable Due to an Original Design Deficiency.

NMPNS identified that the design of each Unit 2 SGT subsystem made each subsystem potentially incapable of performing its design basis function of establishing and maintaining a negative pressure in secondary containment following a design basis accident, when it was lined up in accordance with the procedure for primary containment inerting, de-inerting, or purging. Specifically, plant OPs for primary containment inerting, de-inerting, and purging permitted operation with the filter train recirculation pressure control valve in the manual control mode. In this line-up the capability of a SGT subsystem to establish and maintain a negative secondary containment pressure of at least 0.25 inches of water was potentially compromised. Based on this condition NMPNS determined, through a review of past operating history, that both SGT subsystems were concurrently inoperable on three occasions contrary to the requirements of TS limiting condition for operation (LCO) 3. 6.4.3. The LCO action statement for this condition required that action be immediately taken in accordance with TS LCO 3.0.3 that required that a shutdown be initiated within one hour, and the plant be placed in cold shutdown. As corrective action for this issue, NMPNS implemented procedure changes that prevent recurrence by requiring that the associated train of SGT be declared inoperable when used for primary containment inerting, de-inerting, or purging with the recirculation line pressure control valve in manual.

The inspectors evaluated this issue in accordance with the guidance of IMC 0612, Appendix B, "Issue Screening." The finding was determined to be greater than minor because it was associated with the design control attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that radiological barriers such as SGT protect the public from radio nuclide

releases caused by accidents. The finding was determined to be of very low safety significance in accordance with Phase 1 of the Reactor Safety SDP because it only represented a degradation of the radiological barrier function provided by the SGT system. This licensee-identified finding involved a violation of TS 3.0.3 that requires that action be initiated within one hour to shut down, and ultimately requires that the plant be placed in cold shutdown. The enforcement aspects of this violation are discussed in Section 4OA7. This LER is closed.

.2 (Closed) LER 05000410/2005-001-01, Both SGT Subsystems Inoperable Due to an Original Design Deficiency.

As a result of further review and evaluation of SGT system operation during primary containment purging operations, NMPNS identified that a previously unrecognized procedural deficiency that allowed the SGT subsystems to remain cross-connected during primary containment purging operations could have resulted in damage to both trains of SGT in the event of a large-break LOCA and thus could prevent fulfillment of the safety function of the SGT system. As corrective action for this issue, NMPNS implemented procedure changes that prevent recurrence by verifying that during primary containment purging operations at least one of the SGT train cross-connect line isolation valves is closed.

The inspectors evaluated this issue in accordance with the guidance of IMC 0612, Appendix B, "Issue Screening." The finding was determined to be greater than minor because it was associated with the procedure adequacy attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that radiological barriers such as SGT protect the public from radio nuclide releases caused by accidents. The finding was determined to be of very low safety significance in accordance with Phase 1 of the Reactor Safety SDP because it only represented a degradation of the radiological barrier function provided by the SGT system. This licensee-identified finding involved a violation of TS 5.4, "Procedures." The enforcement aspects of this violation are discussed in Section 4OA7. This LER is closed.

.3 (Closed) LER 05000220/2005-004-00, Operation Prohibited by TS due to Unrevealed Inoperability of One Off-site Power Source.

On December 19, 2005, NMPNS identified that one 115 kV off-site power line was inoperable due to an open phase. The unknown failure of one phase had existed since November 29, 2005. During that time Nine Mile Point Unit 1 had exceeded the TS allowed outage time for an inoperable off-site power line and the allowed outage time for an inoperable diesel generator coincident with an inoperable off-site power line. NMPNS determined the cause to be a functional design deficiency regarding the adequacy of control room indications and alarms. The off-site power system at Unit 1 has a ring-bus design, and the loss of one phase did not cause a loss of power to the unit or any control room alarms. Corrective action included implementation of a plant process computer alarm for low amperage on all three phases of the off-site power lines.

The inspectors evaluated this issue in accordance with the guidance of IMC 0612, Appendix B, "Issue Screening." The finding was greater than minor because it was associated with the configuration control attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors determined the finding to be of very low safety significance because as a transient initiator it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. This licensee-identified finding involved violations of TS 3.6.3.b and c, Emergency Power Sources. The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

.4 (Closed) LER 05000410/2006-001-00, Automatic Reactor Scram due to a Loss of Main Turbine Gland Sealing Steam.

a. Inspection Scope

On March 9, 2006, Unit 2 automatically scrammed from approximately 85% reactor power. Unit 2 was coasting down in power for an upcoming refueling outage. The scram was caused by a turbine trip due to low condenser vacuum. The turbine generator gland seal and exhaust system failed resulting in the loss of condenser vacuum.

b. Findings

Introduction. A self-revealing finding of very low safety significance occurred on March 9, 2006, when Unit 2 automatically scrammed due to a main turbine trip caused by low condenser vacuum. The loss of condenser vacuum occurred when the normal turbine gland seal supply isolated due to high water level and the emergency gland seal steam supply (non-safety related) failed. The emergency gland seal steam supply failed because a maintenance technician improperly assembled a pressure indicating controller for the system following maintenance in April 2004.

Description. The turbine steam seal system provides steam to the turbine glands to prevent the entrance of air and non-condensible gases, which degrade condenser vacuum, into the main condenser. At Unit 2, the normal source of this sealing steam is one of two clean steam reboilers. In the event that the clean steam reboilers are unavailable, the main steam system provides a back-up supply of emergency sealing steam. The main steam supply is normally aligned to automatically supply low pressure steam to the seals in the event of a loss of the normal steam seal supply. Main steam to the turbine seals is supplied through two pressure regulators, 2TME-PV122 that reduces main steam pressure to about 150 psig, and 2TME-PV111 that reduces pressure to the required steam seal supply pressure of 4 psig.

On March 9, 2006, when the in-service clean steam reboiler isolated on high level, the emergency sealing steam supply valves opened; however, the pressure control valve 2TME-122 failed closed isolating the emergency steam seal supply to the turbine seals.

This resulted in a turbine trip due to low condenser vacuum followed by a reactor scram due to the turbine trip.

NMPNS performed a root cause analysis (RCA) and determined that the emergency gland seal steam supply pressure regulator, 2TME-PV122, failed because the pressure indicating controller for the valve, 2TME-PIC122, was improperly reassembled following maintenance in April 2004. A linkage in 2TME-PIC122 disconnected and caused 2TME-PV122 to shut preventing the flow of emergency sealing steam through the valve. The RCA team determined that the cause of the disconnected linkage was a mispositioned spreader device on the fastener clip on one of the two ends of the linkage. This spreader device, when placed in its expanded position, is used to spread the ends of the clip on the linkage to allow for installation and removal during maintenance. After installation, the spreader device should be returned to its engaged position. Following the scram on March 9, maintenance technicians found the linkage disconnected and the spreader device on one end of the linkage in the expanded position. After discussion with the RCA team, the mispositioned spreader device most likely caused the linkage to become disconnected during controller operations on March 9, which caused 2TME-PV122 to fail closed and resulted in the reactor scram.

The RCA team reviewed maintenance records for the gland seal system and determined that the spreader device was left in the expanded position after 2TME-PIC122 was rebuilt and calibrated in April 2004. The work control documents, procedures, and the vendor technical manual did not include guidance or discuss the spreader device. The RCA team, based on discussions with the NMPNS's procedure development group, concluded that connecting the linkage and use of the spreader device was within the skill set of the individual technician performing the maintenance and did not require detailed instructions for proper installation. The inspectors agreed with this assessment and determined, based on discussion with the RCA team, that the individual maintenance technician failed to adequately apply the applicable human performance techniques such as self-checking to ensure that, based on his training and experience, he properly reassembled 2TME-PIC122 following maintenance in April 2004. This resulted in the March 9, 2006, Unit 2 reactor scram.

Analysis. The performance deficiency associated with this event was the Unit 2 reactor scram due to a loss of main condenser vacuum when the normal turbine gland seal supply isolated due to high water level and the emergency gland seal steam supply failed due to the improperly reassembled 2TME-PIC122. Traditional enforcement does not apply because the issue did not have an actual safety consequence or a potential for impacting the NRC's regulatory function, and it was not the result of any willful violation of NRC requirements. The finding is greater than minor because it affected the human performance attribute of the Initiating Event cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during power operations. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors determined the finding to be of very low risk significance because it does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, and is not potentially risk significant due to external events. This finding was entered into the NMPNS CAP as CR 2006-0993. FIN 05000410/2006003-02, Inadequate Use of Human

Performance Tools During Maintenance Results in an Equipment Failure that Causes a Reactor Scram.

Enforcement. No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance issue because it occurred on non safety-related balance of plant equipment.

4OA5 Other Activities

.1 Implementation of Temporary Instruction (TI) 2515/165 - Operational Readiness of Offsite Power and Impact on Plant Risk

a. Inspection Scope

The objective of TI 2515/165, “Operational Readiness of Offsite Power and Impact on Plant Risk,” was to gather information to support the assessment of nuclear power plant operational readiness of offsite power systems and impact on plant risk. The inspectors evaluated NMPNS procedures against the specific offsite power, risk assessment, and system grid reliability requirements of TI 2515/165.

The information gathered while completing this TI was forwarded to the Office of Nuclear Reactor Regulation for further review and evaluation on April 3, 2006.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. Timothy O’Connor and other members of NMPNS management on July 18, 2006. NMPNS acknowledged that some of the material reviewed by the inspectors during this period was proprietary, but the content of this report contains no proprietary information.

4OA7 Licensee-identified Violations

The following violations of very low safety significance were identified by NMPNS and are violations of NRC requirements that meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- TS 3.6.4.3, requires, in part, that if both SGT trains are inoperable TS 3.0.3 shall be entered immediately. TS 3.0.3 requires that action shall be taken within one hour to place the unit in Mode 2 within 7 hours, Mode 3 within 13 hours and Mode 4 within 37 hours. Contrary to this requirement on three occasions, March 15 - 16, 2002 (27.9 hours), November 24 - 25, 2002 (16.7 hours), and March 15, 2004 (11 hours), NMPNS failed to take action within the required timeframe. In accordance with IMC 0609, Appendix A, “Significance

Determination of Reactor Inspection Findings for At-Power Situations," the inspectors determined the finding to be of very low risk significance because it only represented a degradation of the radiological barrier function provided by the SGT system. Because the violation is of very low risk significance and NMPNS entered the deficiency into its CAP as CR 2005-0026, this finding is being treated as an NCV consistent with Section VI.A.1 of the Enforcement Policy.

- TS 5.4 requires, in part, that procedures be established, implemented, and maintained covering the activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Item 4 of Regulatory Guide 1.33 recommends, in part, that instructions for energizing, filling, venting, draining, startup, shutdown, and changing modes of operation be prepared for SGT. Contrary to the above, operating procedure N2-OP-61B, "Standby Gas Treatment System," did not include instructions to preclude operating the SGT subsystems cross-connected during primary containment purging operations, which could have resulted in damage to both trains of SGT in the event of a large-break LOCA and thus could have prevented fulfillment of the SGT system safety function. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors determined this finding to be of very low risk significance because it only represented a degradation of the radiological barrier function provided by the SGT system. Because the violation is of very low risk significance and NMPNS entered the deficiency into its CAP as CR 2005-3559, this finding is being treated as an NCV consistent with Section VI.A.1 of the Enforcement Policy.

- TS 3.6.3.b requires that if a 115 kV external line is de-energized, that line shall be returned to service within seven days. Contrary to this requirement, from November 29 to December 19, 2005, Line 4, a 115 kV external line, was de-energized and not restored within seven days. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors determined the finding to be of very low risk significance because as a transient initiator it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available and is not potentially risk significant due to external events. Because the violation is of very low risk significance and NMPNS entered the deficiency into its CAP as CR 2005-5180, this finding is being treated as an NCV consistent with Section VI.A.1 of the Enforcement Policy.

- TS 3.6.3.c requires that if a diesel generator power system becomes inoperable coincident with a 115 kV line de-energized, that diesel-generator system shall be restored to an operable condition within 24 hours. Contrary to this, from November 29 to December 3, 2005, EDG 102 was inoperable coincident with Line 4 being inoperable and not restored within 24 hours. On December 12 - 13, 2005, EDG 103 was inoperable coincident with Line 4 being inoperable and was not restored within 24 hours. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power

Situations,” the inspectors determined the finding to be of very low risk significance because as a transient initiator it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available and is not potentially risk significant due to external events. Because the violation is of very low risk significance and NMPNS entered the deficiency into its CAP as CR 2005-05180, this finding is being treated as an NCV consistent with Section VI.A of the Enforcement Policy.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

N. Conicella, Manager, Operations
M. Faivus, General Supervisor, Chemistry
J. Gerber, Manager, Radiation Protection
W. Paulhardt, Manager, Work Control, Outage Management
J. Hutton, Plant General Manager
T. Maund, Manager, Maintenance
M. Miller, Director, Licensing
T. O'Connor, Site Vice President
M. Schimmel, Manager, Engineering Services
T. Shortell, Manager, Training, Nuclear
R. Dean, Director, Quality and Performance Assessment

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000410/2006003-01	NCV	RCIC Alignment During Maintenance Not Consistent With Design Bases. (Section 1R15)
05000410/2006003-02	FIN	Inadequate use of human performance tools during maintenance results in an equipment failure that causes a reactor scram. (Section 4OA3)

Closed

05000410/2005-001-00	LER	Both Standby Gas Treatment Subsystems Inoperable Due to an Original Design Deficiency (Section 4OA3)
05000410/2005-001-01	LER	Both Standby Gas Treatment Subsystems Inoperable Due to an Original Design Deficiency (Section 4OA3)
05000220/2005-004-00	LER	Operation Prohibited by TS due to Unrevealed Inoperability of One Off-site Power Source (Section 4OA3)

05000410/2006-001-00	LER	Automatic Reactor Scram due to a Loss of Main Turbine Gland Sealing Steam (Section 4OA3)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1RO4: Equipment Alignment

N1-VLU-01, “Valve Lineup and Valve Operations”
N2-VLU-01, “Walkdown Order Valve Lineup and Valve Operations”
N1-OP-33A, “115 kV System”
N1-OP-45, “Emergency Diesel Generators”
WO 05-01609-00, Maintenance of power circuit breaker R10, 5 year overhaul per N1-EPM-GEN-298
WO 05-26501-00, Connections from HDS-11/511 to HDS-13 needs to be inspected for proper bolting
N1-OP-14, “Containment Spray”
N2-OP-32, “Low Pressure Core Spray”
N2-OP-36A, “SLC System”

Section 1R05: Fire Protection

Fire Brigade Scenario No. OS-FT-FIR-SCN-2-04, Division 2 Cable Routing Cable Fire
NDD-FPP, “Fire Protection Program”
S-ODP-FPP-0101, “Report on Fire Department Activity/Drills”
NIP-FPP-01, “Fire Protection Program”

Section 1RO7: Heat Sink Performance

WO 05-10378-00, Perform Annual PM on RBCLC Water HX, HTX-70-13R
N1-MPM-070-409, “RBCLC Water HXs 70-13R, 70-14R, 70-15R”
S-TDP-REL-0102, “Service Water HX and Component Inspection Guide”
S-TDP-REL-0103, “Service Water System Problems Affecting Safety-related Equipment Program Plan”
N1-OP-11, “RBCLC System”
Calc No. S13.4-70HX06, “RBCLC TCV 70-137 Minimum Position and Wintertime Supply Temperature Evaluation”
S13.4-7Q-HX07, “Power Ascension Test procedure N1-PAT-11-1 Results Evaluation”

Section 1R08: Inservice Inspection Activities

NDT Examination Reports

W2-3.00-06-0011, Liquid Penetrant Examination, 2RPV-CRDH036
W2-3.00-06-0003, Liquid Penetrant Examination, 2RHSV376
W1-3.00-06-0002, Liquid Penetrant Examination, 2CSH-25-18-FW300-301
W2-4.00-06-0007, Magnetic Particle Examination, 2CSLV121
W1-2.01-06-0016, Visual Examination, 2CSH-PSR177A2

ISI Program Documents

Safety Evaluation Report, NMP Unit 2, Relief Requests for Second Ten-Year ISI Program Plan, Revision 1, March 3, 2000
Safety Evaluation Report, NMP Unit 2, Approval to use Risk-Informed ISI Program for Second Ten-Year ISI Program, May 31, 2001
Safety Evaluation Report, NMP Unit 2, Approval to use PDI for Weld Overlay Qualification, May 6, 2003

In-Vessel Visual Inspection

VT-1, Visual Examination of Steam Dryer
VT-1, Visual Examination of Jet Pump Beam nos. 16 and 20
VT-1, Visual Examination of Jet Pump Wedge nos. 16 and 10
VT-1, Visual Examination of Jet Pump Ramshead nos. 3 and 4

Section 1R11: Licensed Operator Requalification Program

NMPNS Operations Manual
NMP Simulator Scenario, O1-OPS-009-TRA-1-76, MSIV Closure/Scram Dump Volume Rupture/Loss of PPC
NEI 99-02, PI Guidelines, Revision 2
CNG-HU-1.01, “Human Performance Program”
CNG-HU-1.01-1000, “Human Performance”
CNG-HU-1.01-1001, “Human Performance Tools and Verification Practices”
S-ODP-OPS-0001, “Conduct of Operations”
N1-EOP-02, “RPV Control”
N1-EOP-05, “Secondary Containment Control”
N1-EOP-08, “RPV Blowdown”
N2-ARP-01, “Control Room Alarm Response Procedures.”
N2-SOP-03, “Loss of AC Power”
N2-EOP-RPV, “RPV Control”
N2-EOP-PC, “Primary Containment Control”
N2-EOP-SC, “Secondary Containment Control”
N2-EOP-RR, “Radioactivity Release Control.”

Section 1R12: Maintenance Effectiveness

NIP-REL-01, “Maintenance Rule”
S-MRM-REL-0101, “Maintenance Rule”
S-MRM-REL-0104, “Maintenance Rule Scope”
GAP-PSH-03, “Control of On-line Work Activities”
Unit 1 Integrated Performance Criteria Matrix
S-MRM-REL-0105, “Maintenance Rule Performance Criteria”
Unit 1 Integrated Scoping Matrix
Unit 1 High Safety Significant Functions and Related Key Safety Functions Matrix
Unit 2 Integrated Scoping Matrix
Unit 2 High Safety Significant Functions and Related Key Safety Functions Matrix
Dwg C-18026, sheet 1, "Emergency Diesel Generator #102 Starting Air, Cooling Water, Lube Oil, and Fuel P&I Diagram"
Dwg C-18026, sheet 2, "Emergency Diesel Generator #103 Starting Air, Cooling Water, Lube Oil, and Fuel P&I Diagram"
N1-TDP-REL-0101, "Emergency Diesel Generator Reliability Program"
Nine Mile Point Unit 1 MR Function Report
Nine Mile Report MR Reliability Report - Emergency Diesel Generator Report, dated June 1, 2004 to May 31, 2005
System Health Report - 1st Quarter 2006
System Health Report - 1st Quarter 2005
System Health Report - 2nd Quarter 2005
System Health Report - 3rd Quarter 2004

Audits and Self-Assessments

Periodic Assessment of MR Program, Nine Mile Point Nuclear Station, January 2002 through September 2003
Periodic Assessment of MR Program, Nine Mile Point Nuclear Station, October 2003 through September 2005
Report of Audit MAI-05-01-N, Maintenance, dated 6/29/05

Completed Surveillance Test Procedures

N1-ST-M2, Emergency Cooling System Makeup Tank Level Control Valves Exercising Test, dated 5/4/06
N1-ST-V19, Emergency Cooling System - Heat Removal Capacity Test at High Power, dated1/2/04 and1/6/04
N2-OSP-EGS-M001, Diesel Generator and Diesel Air Start Valve Operability Test - Division I and II, dated 5/3/06

Maintenance Rule
NRC Regulatory Guide 1.160, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 2
NUMARC 93-01, Industry Guideline For Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 2

1-2006-002, MR Change Control, dated 4/21/06
Nine Mile Point Nuclear Station MR Unavailability Report 6/6/04 - 5/5/06
Nine Mile Point Nuclear Station MR Reliability Report 6/1/01 - 5/31/06
MR Expert Panel Meeting Record, dated 3/8/06 (2006-007), 5/3/06 (2006-008), 5/10/06 (2006-009), 5/23/06 (2006-010), 5/23/06 (2006-010), 5/31/06 (2006-011), 6/7/06 (2006-012)
NIP-REL-01, MR, Revision 9
NDD-REL, MR and Probabilistic Risk Assessment, Revision 8
S-MRM-REL-0102, Structural Assessment and Monitoring Program, Revision 3
S-MRM-REL-0104, MR Scope, Revision 0
S-MRM-REL-0105, MR Performance Criteria, Revision 0
S-MRM-REL-0101, MR, Revision 17

Miscellaneous

Risk-Informed Inspection Notebook For Nine Mile Point Nuclear Station, Unit 1 & Unit 2, Revision 2
NRC PIs for Nine Mile Point 1 & 2 for 1st Quarter 2006
LERs, Unit 1 and Unit 2, dated June 2004 - June 2006
Calculation No. S22.4WW198STAT04, Torus Wall Thinning Trending Analysis, Revision 19
NDEP-UT-6.05, Ultrasonic Thickness Measurement, dated 2/13/06
PM Deferral Nos. 06-321257-87, 06-125212-94, 06-321256-98, and 06-321257-93

Operating Experience

NRC Information Notice 97-11: Cement Erosion from Containment Subfoundations at Nuclear Power Plants, dated 3/21/97
NRC Information Notice 98-26: Settlement Monitoring and Inspection of Plant Structures Affected by Degradation of Porous Concrete Subfoundations, dated 7/24/98
NMPNS Response to NRC Information Notice 97-18: Problems Identified During MR Baseline Inspections, dated 7/1/97

Procedures

NIP-ECA-01, Corrective Action Program, Revision 42
S-EPM-GEN-697, 115KV Motorized Disconnect Switch PM, Revision 0
N2-CTP-GEN-@200, Floor and Equipment Drain Sump Inspections, Revision 1
N2-MPM-GEN-V317, Spline Adapter Inspection on Clow Valves With HBC Operators, Revision 4

System Health Reports & Trending

System Health Report (Q1 - 2006), Unit 1 Emergency Diesel Generator
System Health Report (Q1 - 2006), Unit 1 Core Spray
System Health Report (Q1 - 2006), Unit 1 Emergency Cooling
System Health Report (Q1 - 2006), Unit 1 DC Electric Power & UPS
System Health Report (Q1 - 2006), Unit 2 Emergency Diesel Generator
System Health Report (Q1 - 2006), Unit 2 HPCS
System Health Report (Q1 - 2006), Unit 2 Service Water

System Health Report (Q1 - 2006), Unit 2 DC Electric Power & UPS
System Health Report (Q1 - 2006), Unit 2 RX BLDG Closed Loop Cooling

Work Orders

02-03675-00	03-08519-00	04-20868-00	06-04877-00
03-08127-00	04-03926-00	05-08805-00	

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

GAP-OPS-117, "Integrated Risk Management"
GAP-PSH-03, "Control of On-line Work Activities"
NAI-PSH-03, "On-line Work Management Process"
N2-OSP-EGS-M@001,"
N2-ISP-RHS-M010, "Monthly Functional Test of RHR/LPCS Discharge Leakage Pressure Monitor Instrument Channels"
N2-OSP-RHS-Q@001, "Residual Heat Removal System Loop A Valve Operability Test and Partial ASME XI Pressure Test"
N1-ST-M4A, "Emergency Diesel Generator 102 and PB 102 Operability Test"
N1-ST-Q6C, "Containment Spray System Loop 112 Quarterly Operability Test"
OP-16, "Feedwater System Booster Pump to Reactor"
WO 05-18656-00, Perform PMs on Reactor vessel water level recorder
WO 05-00349-00, Unit 1 condensate iron pre-filter modification
WO 05-19700-00, Unit 1 condensate iron pre-filter modification
WO 05-16407-00, Annual EPR-MOOG valve replacement
N1-IPM-302-001, "Electronic Pressure Regulator"

Section 1R14: Operator Performance During Non-Routine Evolutions and Events

CNG-HU-1.01, "Human Performance Program"
CNG-HU-1.01-1000, "Human Performance"
CNG-HU-1.01-1001, "Human Performance Tools and Verification Practices"
S-ODP-OPS-0001, "Conduct of Operations"
N1-REP-34, "Power Suppression Testing"
N2-SOP-101C, "Plant Shutdown"
N2-ARP-01, "Control Room Alarm Response Procedures"
N2-SOP-19, "Loss of Instrument Air"
CR 2006-2109
WO 06-09575-00, PSV-02-57, replace piping from relief valve to union

Section 1R15: Operability Evaluations

Ltr dtd October 6, 1977, from Per Mar Systems Inc., Electromark Division to Niagra Mohawk Power Corporation, regarding Qualification of Pressure Sensitive Markers
RCM Technologies Qualified Life Manual Calculation Report, System 1000, Revision16 for Electromark 1000 series labels using PI database drywell temperatures.

ASME/ANSI OMc-1990 Addenda to ASME/ANSI OM-1987 Operation and Maintenance of Nuclear Power Plants
ASME/ANSI OM-1987 Operation and Maintenance of Nuclear Power Plants, Part 4, Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers)
ASME OMc-1994 Addenda to ASME OM-1990 Code for Operation and Maintenance of Nuclear Power Plants
NAI-MAI-04, “NMP Snubber Program”
SAS-2006-03, SDP Evaluation Unit 2 RCIC Vacuum Breaker Isolation
CR 2006-0545, Unit 2 RCIC Vacuum Breaker Isolation

Section 1R19: Post Maintenance Testing

GAP-SAT-02, “Pre/Post-Maintenance Test Requirements”
CNG-HU-1.01, “Human Performance Program”
CNG-HU-1.01-1000, “Human Performance”
CNG-HU-1.01-1001, “Human Performance Tools and Verification Practices”
WO 05-24636-00, Preventative maintenance diesel driven fire pump overspeed switch
ACR 05-04646, Current fire system out leakage rates at Unit 2 are about 3 gallons per minute
ACR 06-03516 , Negative terminal on “B” battery is broken off
ACR 05-05039, Oil leaks from oil cooler
N2-OSP-FOF-W001, “Engine Driven Fire Pump Operability and Storage Tank Level Test” completed April 23, 2006
N2-OSP-FOF-W001, “Engine Driven Fire Pump Operability and Storage Tank Level Test” completed April 17, 2006
N2-OSP-FOF-W001, “Engine Driven Fire Pump Operability and Storage Tank Level Test” completed March 12, 2006
WO 04-208636-00, Diesel driven fire pump annual engine inspection
N2-OP-43, “Fire Protection Water”
GAP-MAI-01, “Conduct of Maintenance”
CR 2006-2109, Trip of Instrument Air Compressor caused NMP2 to enter numerous SOPs
Bonnington, Alex, 11 May 2006, NMP2 Instrument Air Compressor Check Valve Failure, LSS Report No. 06-0269, memorandum, Constellation Generation Group, LLC, Generation Services Department, Lab Services Section
N21187, Atlas Copco, Instruction Manual for Compressor Model No. ZR4-60
N10768, GEI-88079, GE Apparatus Power Circuit Breaker Department Instruction Oil Blast Circuit Breaker Type Fk-115, -138, and -161, 115, 138 and 161 kV

Section 1R20: Refueling and Other Outage Activities

N2-OP-115, “Alternate Decay Heat Removal System
NIP-OUT-01, “Shutdown Safety”
N2-OP-39, “Fuel Handling and Reactor Service Equipment”
N2-FHP-003, “Refueling Manual”
N2-FHP-13.3, “Core Shuffle”
N2-OSP-RCS-@001, “RCS Pressure/Temperature Verification”
N2-OSP-RPV-@003, “Reactor Pressure Vessel and All Class I Systems Leakage Test With The RPV Solid”

WO 05-15034-00, Perform comprehensive flush of “B” Recirc pump seal injection line and seal internals
N2-OP-101A, “Plant Start-Up”
N2-OP-101C, “Plant Shutdown”
N1-OP-4, “Shutdown Cooling System”
N2-OP-43A, “Plant Startup”
N2-OP-43C, “Plant Shutdown”
NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management”

Section 1R22: Surveillance Testing

NMP1 Calc. No. S14-93-V002, IST Torque determination for Containment Spray Check Valves 93-58, 93-62, and 93-64
Unit 1 10 CFR 50.59 Safety Evaluation 91-023
Unit 1 10 CFR 50.59 Safety Evaluation 89-013
NMPNS Unit 1 IST Basis Document
CR 2000-3154, Potential radiological release pathway not analyzed or appropriately tested

Section 1R23: Temporary Plant Modifications

NIP-CON-01, “Design and Configuration Control Process”
GAP-CON-01, “Control of Temporary Alterations”
NIP-CON-02, “Review of Temporary Changes”
WO 05-03362-01, Replace 2BYS*BAT2B Cell 21
WO 05-13043-00, Division II Battery Profile Load Test 2BYS*BAT2B
N2-ESP-BYS-R685, “Division I/II/III Battery Modified Profile Test”
IEEE Standard 450-1995, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications”
IEEE Standard 450-1980, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications”

Section 1EP6: Drill Evaluation

NEI 99-02, PI Guidelines, Revision 2
CNG-HU-1.01, “Human Performance Program”
CNG-HU-1.01-1000, “Human Performance”
CNG-HU-1.01-1001, “Human Performance Tools and Verification Practices”
S-ODP-OPS-0001, “Conduct of Operations”
EPIP-EPP-01, “Classification of Emergency Conditions at Unit 1"
EPIP-EPP-17, “Emergency Communications Procedure”
EPIP-EPP-20, “Emergency Notifications”

Section 2OS2: ALARA Planning and Controls

Nine Mile Point Unit 2 RP RFO10 Post Outage Report (DRAFT)
RP Self-Assessment Reports: FSA-2005-63; FSA-2005-64; FSA-2005-54; FSA-2005-49

Q&PA Report No. RPP-05-02-N
Q&PA Quarterly Reports: 3d Quarter 2005; 4th Quarter 2005; 1st Quarter 2006

Q & PA Observation Numbers

2005-1112	2005-1331	2005-1365	2005-1391
2006-0169	2005-1343	2005-1368	2005-1395
2006-0266	2005-1348	2005-1370	2005-1396
2005-1037	2005-1351	2005-1373	2005-1406
2005-1114	2005-1353	2005-1374	2005-1409
2005-1115	2005-1358	2005-1377	2005-1416
2005-1269	2005-1360	2005-1383	2005-1430
2005-1284	2005-1361	2005-1385	2005-1455
2005-1309	2005-1362	2005-1390	2005-1460
2005-1315	2005-1363		

Section 2OS3: Radiation Monitoring Instrumentation

Unit 1 Updated Final Safety Analysis Report Section XII, B, 2.0, Area Radioactivity Monitoring Systems
Unit 2 Updated Final Safety Analysis Report, Section 12.2.4, Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Procedures

AP-07.00, “RP Program”
AP-07.01, “Radiation Work Permit Program”

Section 4OA2: Identification and Resolution of Problems

Condition Reports

2004-0284	2005-0111	2005-4115	1997-1418
2004-0388	2005-0116	2005-4117	1997-1419
2004-0493	2005-0383	2005-4118	1997-1608
2004-1746	2005-2060	2005-4120	1998-2415
2004-2437	2005-2065	2005-4905	2001-6018
2004-2439	2005-2115	2005-4962	2002-0482
2004-2497	2005-2220	2006-0306	2002-3087
2004-2642	2005-2305	2006-0513	2004-1870
2004-3421	2005-2305	2006-2325	2004-1907
2004-4125	2005-2590	2006-1583	2004-3352
2004-4811	2005-3862	2006-1636	2004-5718
2004-5415	2004-4054	2006-1425	2005-2051
2004-5419	2005-4065	2006-1627	2005-2483
2004-5763	2005-4070	2006-1428	2005-3175
2004-5764	2005-4084	1997-1279	2005-5142

2003-1254
2004-0010
2004-0338
2004-0390
2004-1100
2004-1405
2004-1661
2004-1840
2006-0014
2006-2013
2006-2014
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2006-1920
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2006-2134
2006-2172
2006-2220
2006-2225
2006-2250
2006-2270
2006-2298
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2006-2408
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2006-2456
2006-2492
2006-2589
2006-2634
2006-2789
2006-2795
2006-2843
2006-2845

<u>Section 4OA3: Event Followup</u>

N2-REP-6, “Post-Scram Review”
N2-REP-6, “Post-Scram Review,” completed March 10, 2006
CR 2006-0993, Unit 2 automatic reactor scram from unexpected loss of condenser vacuum
GAP-MAI-01, “Conduct of Maintenance,” Revision 5

GAP-MAI-01, “Conduct of Maintenance,” Revision 10
CNG-HU-1.01, “Human Performance Program”
CNG-HU-1.01-1000, “Human Performance”
CNG-HU-1.01-1001, “Human Performance Tools and Verification Practices”
S-ODP-OPS-0001, “Conduct of Operations”
N2-ARP-01, “Control Room Alarm Response Procedures.”
N2-EOP-RPV, “RPV Control”
N2-SOP-101C, “Reactor Scram”

Section 4OA5: Other Activities

N1-OP-33A, “115KV System,” Revision 21
N1-SOP-33A.3, “Major 115KV Grid Disturbances,” Revision 01
N2-SOP-70, “Major Grid Disturbances,” Revision 00
N2-OP-70, “Station Electrical Feed and 115KV Switchyard,” Revision 08
GAP-PSH-03, “Control of On-Line Work Activities,” Revision 12
GAP-OPS-117, “Integrated Risk Management,” Revision 07
S-ODP-OPS-0112, “Off-site Power Operations and Interface,” Revision 09
NIP-IRG-02, “Interface With Outside Agencies Other Than The NRC,” Revision 12

LIST OF ACRONYMS

ADAMS	agency wide documents and management system
ALARA	as low as reasonably achievable
CAP	corrective action program
CFR	Code of Federal Regulations
CR	condition report
CRD	control rod drive
DBD	design basis document
ECCS	emergency core cooling system
EDG	emergency diesel generator
EOP	emergency operating procedure
HPCS	high pressure core spray
HRA	high radiation area
HX	heat exchanger
IAC	instrument air compressor
IMC	inspection manual chapter
IPE	individual plant evaluation
ISI	inservice inspection
LCO	limiting condition for operation
LER	licensee event report
LOCA	loss of coolant accident
LORT	licensed operator requalification training program
LPCS	low pressure core spray
MR	maintenance rule
NCVs	non-cited violations

NMPNS	Nine Mile Point Nuclear Station
NRC	Nuclear Regulatory Commission
OP	operating procedure
PARS	publically available records
PI	performance indicator
PM	preventive maintenance
PRA	probabilistic risk assessment
Q&PA	quality & performance assessment
RB	reactor building
RBCLC	reactor building closed loop cooling
RCA	root cause analysis
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RFO	refueling outage
RHR	residual heat removal
RP	radiation protection
SDP	significance determination process
SGT	standby gas treatment
SLC	standby liquid control
SOP	special operating procedure
SRA	Senior Reactor Analyst
SSC	structure, system, and component
TB	turbine building
TI	temporary instruction
TM	temporary modifications
TS	technical specification
UFSAR	Updated Final Safety Evaluation Report
WO	work order