April 30, 2003

Mr. John T. Conway
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 50-220/03-02, 50-410/03-02

Dear Mr. Conway:

On March 29, 2003, the NRC completed an inspection of your Nine Mile Point Nuclear Station, Units 1 and 2. The enclosed integrated inspection report documents the inspection findings which were discussed on April 11, 2003, with you and other members of your staff.

This 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 four NRC-identified findings and one self-revealing finding of very low safety significance (Green), four of which were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating violations as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any findings in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region I, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Nine Mile Point.

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders (dated February 25, 2002, January 7, 2003 and April 29, 2003) and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance access authorization. The NRC also issued Temporary Instruction 2515/148 on August 28, 2002, that provided guidance to inspectors to audit and inspect licensee implementation of the interim compensatory measures (ICMs) required by the February 25th Order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year (CY) '02, and the remaining inspections are scheduled for completion in CY '03. Additionally, table-top security drills were conducted at several licensees to evaluate the impact of expanded adversary characteristics and the ICMs on licensee protection and mitigative strategies. Information gained and discrepancies identified during the audits and drills were reviewed and dispositioned by the Office of Nuclear Security and Incident Response. For CY '03, the NRC will continue to monitor overall safeguards and security controls, conduct inspections, and resume force-on-force exercises at selected power plants. Should threat conditions change, the NRC may issue additional Orders, advisories, and temporary instructions to ensure adequate safety is being maintained at all commercial power reactors.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publically Available Records (PARS) component of NRC's document management 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/

James M. Trapp, Chief
Projects Branch 1
Division of Reactor Projects

Docket Nos. 50-220, 50-410
License Nos. DPR-63, NPF-69

Enclosure: Inspection Report 50-220/03-02, 50-410/03-02
w/Attachment: Supplemental Information

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REGION I

Docket Nos:	50-220, 50-410
License Nos:	DPR-63, NPF-69
Report No:	50-220/03-02 and 50-410/03-02
Licensee:	Nine Mile Point Nuclear Station, LLC (NMPNS)
Facility:	Nine Mile Point, Units 1 and 2
Location:	P. O. Box 63 Lycoming, NY 13093
Dates:	December 29, 2002 - March 29, 2003
Inspectors:	G. Hunegs, Senior Resident Inspector B. Fuller, Resident Inspector E. Knutson, Resident Inspector J. Noggle, Senior Health Physicist D. Silk, Senior Emergency Preparedness Inspector K. Young, Reactor Inspector
Approved by:	James M. Trapp, Chief Projects Branch 1 Division of Reactor Projects

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Summary of Findings

IR 05000220/2003-002, 05000410/2003-002; Nine Mile Point Nuclear Station, LLC; 12/29/2002 - 03/29/2003; Nine Mile Point, Units 1 and 2; Equipment Alignment; Personnel Performance; Post-Maintenance Testing and Surveillance Testing.

This report covered a 13 week period of inspection by resident inspectors and announced inspections by three region-based inspectors. Five Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using 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. The inspectors identified a non-cited violation of Technical Specification 6.8.1 at Unit 1 for procedural non-compliance, in that drive water differential pressure was not returned to normal after unsticking three stuck control rods and prior to continued rod withdrawal, as specified by operating procedure N1-OP-5, "Control Rod Drive System."

  The finding is greater than minor because it could reasonably be viewed as a precursor to a significant event. Specifically, inadequate control of the addition of positive reactivity to the reactor could lead to a plant transient and could challenge the integrity of the fuel cladding. The finding was determined to be of very low safety significance in accordance with Phase 1 of the Reactor Safety SDP because the actual control rod speeds were only slightly greater than nominal and the rod withdrawals did not produce a plant transient. This finding was an example of a cross-cutting issue in human performance. (Section 1R19)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of Technical Specification 5.4.1 at Unit 2 for procedural non-compliance, in that the control switch for one of three area heaters in the Division 1 Emergency Diesel Generator (EDG) room was not in the "auto" position as specified by operating procedure N2-OP-57, "Diesel Generator Building Ventilation System."

  This finding is greater than minor because it could reasonably be viewed as a precursor to a significant event, in that incorrect plant equipment configuration could impact system operability. The finding was determined to be of very low safety significance in accordance with Phase 1 of the Reactor Safety SDP because the other two area heater switches were in the correct position, the EDG room temperature was being maintained, and the EDG remained operable. This finding was an example of a cross-cutting issue in human performance. (Section 1R22)

- Green. The inspectors identified a non-cited violation of Technical Specification 6.8.1 at Unit 1 for procedural non-compliance, in that a sample was not taken from the liquid poison tank within eight hours of the test completion as specified by surveillance procedure N1-ST-Q8B, "Liquid Poison Pump 12 and Check Valve Operability Test."

  The finding is greater than minor because it could reasonably be viewed as a precursor to a significant event in that the liquid poison tank Boron concentration could have been inadvertently diluted. The finding was determined to be of very low safety significance in accordance with Phase 1 of the Reactor Safety SDP because water was not added to the liquid poison tank and therefore the liquid poison concentration was not affected. This finding was an example of a cross-cutting issue in human performance. (Section 1R22)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding at Unit 1 regarding scaffolding that was interfering with operation of the pressure suppression chamber to reactor building vacuum breaker valve (68-07). The scaffolding prevented the valve from fully opening.

  This finding is greater than minor because the scaffolding restricted the vacuum breaker from fully opening, therefore degrading the system. The finding was determined to be of very low safety significance in accordance with Phase 1 of the Reactor Safety SDP because the valve would still partially open. With the valve in a partially open condition, the flow capacity would be reduced but the reduction in performance would not be substantial and therefore the valve remained operable. The scaffolding did not impact the valve closing function which is a containment isolation function. This finding was an example of a cross-cutting issue in human performance. (Section 1R04)

Cornerstone: Emergency Preparedness

- Green. The inspectors identified a self-revealing non-cited violation of Technical Specification 6.8.1 at Unit 1 for procedural non-compliance, in that the Station Shift Supervisor (SSS) failed to implement a radiation emergency procedure for Reactor Building evacuation.

  This finding is greater than minor because the performance deficiency prevented the SSS from carrying out his duties which could affect the response to an emergency. The finding was determined to be of very low safety significance in accordance with the Emergency Preparedness SDP because planning standards were met and the actual radiological conditions did not reach the unusual event threshold. This finding was an example of a cross-cutting issue in human performance. (Section 1R14)

B. Licensee-Identified Violations: None

REPORT DETAILS

Summary of Plant Status

Nine Mile Point Unit 1 (Unit 1) began the inspection period at 100 percent power. On February 12, 2003, a planned shutdown was commenced to repair emergency condenser steam inlet isolation valve seal leakage. Following repairs, Unit 1 was started up on February 18 and returned to service on February 19 with full power reached on February 20, 2003. On March 14, 2003, Unit 1 was shutdown for refueling outage 17 and remained there through the end of the inspection period.

Nine Mile Point Unit 2 (Unit 2) began the inspection period at 100 percent power and remained there through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed two site specific weather related condition during this inspection period. The inspector performed walkdowns of the Unit 1 Reactor Building and the Unit 1 and Unit 2 Turbine Buildings on January 17 and 23, 2003, respectively, during a period of extremely cold weather. The walkdowns included inspecting for gaps in the perimeter walls and exterior doors of the buildings that could allow the infiltration of cold air and verification that area heaters where functioning properly. The criteria used for evaluation included operating logs and cold weather procedures. The inspector also reviewed and discussed with plant operators, procedures and features that ensure continued availability of the ultimate heat sink (i.e., Lake Ontario) during extreme cold weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns:

The inspectors performed four partial system walkdowns during this inspection period.

The inspector selected the Unit 1, emergency cooling loop 11 to conduct a partial system walkdown while loop 12 was isolated. The loop 12 steam inlet isolation valves were shut due to seal leakage. The walkdown included the control room switch verification and physical inspection and verification of the emergency cooling configuration. N1-OP-13, “Emergency Cooling System,” was used for this review.

The inspector selected the Unit 2, reactor core isolation cooling (RCIC) system to conduct a partial system walkdown. The walkdown included the control room switch verification and physical inspection and verification of the injection lineup. N2-OP-35, “Reactor Core Isolation Cooling,” was used for this review.

The inspector selected the Unit 1 core spray system 122 due to being designated protected equipment during Division I electrical work. The walkdown included the control room switch verification and physical inspection and verification of the system lineup. N1-OP-2, “Core Spray System,” was used for this review.

On March 13, the inspectors walked down portions of the Unit 1 vacuum relief system prior to the beginning of the outage to assess system configuration. N1-ST-Q24, “Drywell/Torus and Torus/Reactor Building Vacuum Relief Test,” was used for this review.

b. Findings

Introduction. The inspectors identified a Green finding at Unit 1 in that scaffolding was interfering with operation of the pressure suppression chamber to reactor building vacuum breaker 68-07.

Description. Three sets of vacuum relief valves (vacuum breakers) are provided between the primary containment and Reactor Building. The purpose of the vacuum relief valves is to equalize the pressure between the suppression chamber (torus) and the Reactor Building so that the structural integrity of the containment is maintained following a design basis accident. Reactor-building-to-torus vacuum breaker 68-07 is a 30-inch swing check valve.

On March 13, 2003, the inspectors noted that a scaffolding rod was positioned such that it would interfere with the operation of vacuum breaker 68-07. Specifically, the valve disc counterweight extensions would strike the scaffolding rod during the valve’s opening stroke, thereby preventing it from fully opening. In response to this finding, vacuum breaker 68-07 was declared inoperable and the scaffolding was repositioned to eliminate the interference.

Analysis. The performance deficiency was that scaffolding was installed such that it would have restricted the vacuum breaker from fully opening, therefore degrading the system. The finding was greater than minor because it could affect the barrier integrity cornerstone by impacting the operational capability of containment. The finding was determined to be of very low safety significance in accordance with Phase 1 of the Reactor Safety SDP because the valve would still partially open. With the valve in a partially open condition, the flow capacity would be reduced but the reduction in performance would not be substantial and therefore the valve remained operable. The scaffolding did not impact the valve closing function which is a containment isolation function. This was an example of a cross-cutting issue in human performance.

Enforcement. No violation of NRC requirements occurred.

1R05 Fire Protection

a. Inspection Scope

The inspectors walked down accessible portions of nine fire areas described below to assess the licensee's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers and any related compensatory measures. The condition of fire detection devices, the readiness of the sprinkler fire suppression systems and the fire doors were also inspected against industry standards. In addition, the fire protection features were inspected, including the ventilation system fire dampers, structural steel fire proofing, and electrical penetration seals. Reference material reviewed for installed features included the Updated Safety Analysis Report. The following plant areas were inspected:

- Division I & II Switchgear Rooms (Unit 2)
- Turbine Building Elevation 250 Feet (Unit 1)
- Normally Inaccessible Areas of Turbine Building Elevations 261, 277, 289, and 300 Feet (Unit 1)
- Division III Emergency Diesel Generator Room (Unit 2)
- Emergency Diesel Generator 103 Room (Unit 1)
- Emergency Cooling Isolation Valve Room (Unit 1)

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Routine Observations

a. Inspection Scope

The inspectors reviewed one licensed operator requalification training activity during this inspection period to assess the licensee's training program effectiveness. The inspectors observed Unit 2 licensed operator simulator training on March 6, 2003. The inspectors reviewed performance in the areas of procedure use, self and peer-checking, completion of critical tasks, and training performance objectives. Following the simulator training, the inspectors observed the crew debrief and critique, and reviewed simulator fidelity through a sampling process.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed two performance-based problems during this inspection period involving selected in-scope structures, systems, and components (SSCs) to assess the effectiveness of the maintenance program. Reviews focused on: (1) proper maintenance rule scoping, in accordance with 10 CFR 50.65; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10 CFR 50.65 (a)(1) and (a)(2) classifications; and, (5) the appropriateness of performance criteria for SSCs classified as (a)(2), and goals and corrective actions for SSCs classified as (a)(1). The inspectors reviewed the licensee's system scoping documents, system health reports and corrective action program documents. Additionally, the inspectors performed a walkdown of the systems, and discussed the system status and recent performance with engineering and operations personnel.

The Unit 1 primary containment system and shutdown cooling system were selected for review because they are high-safety-significant systems.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed six risk assessments and emergent work activities during this inspection period. For selected maintenance work orders (WOs) or action requests (ACRs), the inspectors evaluated: (1) the effectiveness of the risk assessments performed before the maintenance activities were conducted; (2) risk management control activities; (3) the necessary steps taken to plan and control resultant emergent work tasks; and (4) the overall adequacy of identification and resolution of emergent work and the associated maintenance risk assessments. The following documents were used for this review:

- GAP-MAI-01, Conduct of Maintenance, Revision 3
- GAP-PSH-01, Work Control, Revision 27
- NEG-CA-010, Online Configuration Risk Management Guidance

The following work items/WOs were reviewed:

- Repair of 111 containment spray pump after failure of the pump to start during surveillance activities (Unit 1)
- WO 03-5165, control rod drive hydraulic pump, P1B suction relief valve replacement (Unit 2)
- ACR 2003-0877, replace control power fuses for 11 core spray system isolation valves (Unit 1)
- ACR 2003-0411, investigate and repair emergency condenser isolation valve steam leak (Unit 1)
- Placement of a freeze seal to support reactor building closed loop cooling (RBCLC) system maintenance, due to its impact on shutdown cooling system availability; included review of the contingency plan that was developed for operating procedure N1-OP-4, “Shutdown Cooling System,” Off-Normal Procedure 4.0, “Freeze Seal Contingency Plan” (Unit 1)
- Repair of feedwater heater string 6B level control valve LVX-65B (Unit 2)

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions and Events

a. Inspection Scope

The inspectors reviewed the circumstances involved with the release of airborne radioactive contamination in the Unit 1 Reactor Building on March 17, 2003.

b. Findings

Introduction. A Green self-revealing NCV was identified for failure, by the Station Shift Supervisor (SSS), to implement Emergency Plan Implementing Procedures (EPIP) in accordance with TS 6.8.1.

Description. On March 17, reactor disassembly was in progress and reactor vessel flood-up to support refueling activities was underway. Water additions to raise vessel level were put on hold when vessel water temperature decreased due to the large amount of cool water added to the vessel. The movement of the steam dryer to the reactor internals storage pit was planned to occur with the dryer submerged. Due to the delay in vessel flood-up, and a desire to not lose time against the schedule, the Outage Management Team decided to move the dryer only partly submerged. The dryer was partially lifted above the water and moved into position over the internals pit, with intermittent water sprays directed at the exterior of the dryer. The dryer was lowered back into the water, which rewetted the dryer surfaces and forced air to flow upward through the steam dryer internal baffles. Loosely adherent surface contamination was apparently dislodged from the dryer surfaces by the airflow and became airborne.

Airborne contamination was detected by the refuel floor air monitoring equipment at levels above the air monitoring system (AMS) setpoint. The refuel floor and subsequently the Reactor Building were evacuated by radiation protection personnel. The SSS in the control room was not notified by refuel floor personnel that airborne contamination existed in the Reactor Building. Procedure EPIP–EPP-21, "Radiation Emergencies," is applicable to emergencies involving radiation levels above normal operating ranges but below emergency action level (EAL) classifications. EPIP-EPP-21 stated that the SSS was responsible to determine if radiation emergencies require a local area/building evacuation (step 3.1.1), direct making evacuation announcements (step 3.1.2) and/or implementation of the site emergency plan (step 3.1.3). The failure to notify the SSS prevented him from implementing the appropriate procedure instructions.

Analysis. The deficiency associated with this event is a failure to implement a procedure, which prevented the SSS from assessing the magnitude of the radiation released and conducting appropriate response activities. The finding is greater than minor because the performance deficiency prevented the SSS from conducting activities which could affect the response to an emergency. The finding was determined to be of very low safety significance (Green), in accordance with the Emergency Preparedness SDP, because planning standards were met and the actual radiological conditions did not reach the unusual event threshold. Additionally, the radiological consequences of this procedure error were low because the Reactor Building evacuation was directed by radiation protection personnel. This procedure non-compliance was an example of a cross-cutting issue in human performance.

Enforcement. TS 6.8.1 requires in part that written procedures be established, implemented and maintained that meet or exceed the requirements of Appendix A of Regulatory Guide 1.33. Regulatory Guide 1.33, Appendix A, Item 6y, requires procedures be implemented for abnormal releases of radioactivity. EPIP-EPP-21, Radiation Emergencies, required that the SSS assess the emergency and determine if the radiation emergency requires an evacuation. Contrary to the above, the control room and SSS were not notified of the release of airborne contamination and therefore did not implement EPIP-EPP-21. Because this failure of personnel to implement the radiation emergency procedure is of very low safety significance and has been entered into the corrective action program (DER NM-2003-1108), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-220/03-02-01, Failure to Implement Radiation Emergency Procedure.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed five operability evaluations during this inspection period, which affected risk significant mitigating systems, to assess: (1) the technical adequacy of the evaluation; (2) whether other existing degraded systems adversely impacted the affected system or compensatory measures; (3) where compensatory measures were used, whether the measures were appropriate and properly controlled; and, (4) that the degraded systems remained operable. The following documents were used for this review:

- NIP-ECA-01, Deviation/Event Reports
- GAP-OPS-02, Administration of Operations, Revision 19
- S-ODP-OPS-0116, Operability Determinations
- 10 CFR 21, Report number 0083 dated August 31, 2002

The following licensee documents were reviewed:

- DER 2-2003-0617, High Pressure Core Spray Emergency Diesel Generator Backup Lube Oil Pump Failed to Start (Unit 2)
- DER 1-2003-0485, Emergency Condenser Isolation Valves Leaking (Unit 1)
- DER 2-2003-1085, Division I Emergency Diesel Generator Jacket Water Leak (Unit 2)
- DER 2-2003-0906, Water Found in the Division I Emergency Diesel Fuel Oil Day Tank (Unit 2)
- DER 1-2003-0731, Reactor Water Cleanup Isolation Valve IV-33-04 Failed to Close (Unit 1)

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

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

- N1-ST-Q2, “Control Rod Drive (CRD) Pumps Flow Rate Test,” after corrective maintenance on the 12 CRD pump (Unit 1)
- N2-FSP-FPL-R001, “Low Pressure Carbon Dioxide System Functional Test,” following repair division I switchgear isolation damper for carbon dioxide in accordance with WO 02-5819
- N1-ST-R11, “Valve Remote Position Indicator,” following repair of emergency condenser isolation valve 39-09R (Unit 1)
- WO 02-04870, Post maintenance testing on refuel bridge (Unit 1)
- N1-PM-R1, “Control Rod Scram Insertion Time Test,” following maintenance on the associated hydraulic control units (Unit 1)

b. Findings

Introduction. A Green NCV was identified for procedural non-compliance, in that drive water differential pressure was not returned to normal after unsticking three stuck control rods and prior to continued rod withdrawal, as specified by operating procedure N1-OP-5, “Control Rod Drive System.”

Description. Following restart from the Unit 1 forced outage in February, operators performed single rod scram time testing as part of the acceptance testing for maintenance that had been done on several hydraulic control units. Following a scram insertion, occasionally control rods initially require greater than normal force to achieve outward motion. Operating procedure N1-OP-5 contains a procedure for unsticking control rods in this condition by increasing the drive water differential pressure. Off-Normal Procedure 3.0, “Stuck Control Rod,” includes a caution that states, “Failure to return drive water differential pressure to normal after unsticking a stuck control rod may result in excessive control rod speed . . .” However, the inspector noted that, after one such control rod had been unstuck, it was fully withdrawn prior to returning drive water differential pressure to normal.

The inspectors discussed this observation with the on-shift SSS. In response to this finding, additional single rod scram time testing was stopped pending review of the issue and determination of immediate corrective action. The procedure was changed to reenforce the need to return drive water differential pressure to normal as soon as the affected control rod was unstuck.

The licensee subsequently determined that, during single rod scram time testing on February 19, a total of three control rods had been fully withdrawn using elevated drive water differential pressure. The maximum rod withdrawal speed that was achieved was 3.27 inches per second (ips); nominal rod speed is 3.0 ips, and withdrawal rates up to 5.0

ips had previously been analyzed and shown not to significantly impact a rod withdrawal error transient.

Analysis. The deficiency associated with this event is procedure non-compliance, which led to the withdrawal of three control rods at a rate that was greater than nominal. The finding was greater than minor because it could reasonably be viewed as a precursor to a significant event. Specifically, inadequate control of the addition of positive reactivity to the reactor could lead to a plant transient and could challenge the integrity of the fuel cladding. The finding was determined to be of very low safety significance in accordance with Phase 1 of the Reactor Safety SDP because the actual control rod speeds were only slightly greater than nominal and the rod withdrawals did not produce a plant transient. The procedure non-compliance was an example of a cross-cutting issue in human performance.

Enforcement. TS 6.8.1 states, in part, that, “Written procedures . . . shall be established, implemented and maintained . . .” Contrary to the above, Operating Procedure N1-OP-5, “Control Rod Drive System,” Off-Normal Procedure 3.0, “Stuck Control Rod,” was not correctly implemented on three occasions on February 19, 2003, in that drive water differential pressure was not returned to normal after a stuck control rod was freed and prior to continued rod withdrawal, as required by step 3.2.5 and caution 3.0.2. As a result, these three control rods were withdrawn at greater than nominal speed. Because this procedural non-compliance is of very low safety significance and has been entered into the corrective action program (DER 2003-0645), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-220/03-02-02, Procedural Non-Compliance Resulted in Control Rods Being Withdrawn at Greater than Nominal Speed.

1R22 Surveillance Testing

a. Inspection Scope

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

- N1-ST-Q26, Feedwater and Main Steam Line Power Operated Isolation Valves Partial Exercise Test and Associated Functional Testing of Reactor (Unit 1)
- N1-MFT-081, High Pressure Coolant Injection Level Controller Replacement Test (Unit 1)
- N2-OSP-TIP-Q001, Traversing Incore Probe (TIP) Valve Operability Test (Unit 2)
- N2-OSP-EGS-M@001, Monthly Diesel Generator and Diesel Air Start Valve Operability Test - Division I and II (Unit 2)

- N1-ST-Q8B, Liquid Poison Pump 12 and Check Valve Operability Test (Unit 1)

b. Findings

.1 Division I EDG Area Heater Not in Automatic as Required by Procedure

Introduction. A green NCV was identified for procedural non-compliance, in that the control switch for one of three area heaters in the Unit 2, Division 1, EDG room was not in the “auto” position as specified by operating procedure N2-OP-57, “Diesel Generator Building Ventilation System.”

Description. A monthly surveillance test of the Unit 2 Division 1 EDG was performed on February 11. During the test, the inspector noted that the control switch for one of three area heaters in the EDG room was in the off position with the other two in the “auto” position. The area heater is part of the diesel generator building ventilation system whose purpose is to provide temperature control and ventilation for the EDG room. The electric area heater provides additional heating for the diesel generator room. Following identification by the inspector, the switch was repositioned correctly. The licensee also determined that the error was not the result of tampering.

Analysis. The deficiency associated with this event is procedure non-compliance, in that the control switch for one of the three area heaters in the EDG room was not placed in the required position. The finding was greater than minor because it could reasonably be viewed as a precursor to a significant event, in that plant equipment configuration could impact system operability. The finding was determined to be of very low safety significance (Green) using Phase 1 of the Reactor Safety SDP because there was no effect on system operability. The other two heater switches were in the correct position and the heaters were maintaining proper room temperature. In addition, operators periodically tour the space. The procedure non-compliance was an example of a cross-cutting issue in human performance.

Enforcement. TS 6.8.1 states, in part, that, “Written procedures . . . shall be established, implemented and maintained . . .” Contrary to the above, operating procedure N2-OP-57, “Diesel Generator Building Ventilation System,” was not correctly implemented on February 11, 2003, in that one of three area heater switches for the division 1 EDG room was not in the “auto” position specified by the procedure. Because this procedural non-compliance is of very low safety significance and has been entered into the corrective action program (DER 2003-0525), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-410/03-02-03, Procedural Non-Compliance Resulted in the Incorrect Position for Division 1 Unit Heater Switch.

.2 Liquid Poison Tank Not Sampled as Required by Procedure

Introduction. A green NCV was identified for procedural non-compliance, in that a sample was not taken from the liquid poison tank within eight hours of the test completion as specified by surveillance procedure N1-ST-Q8B, “Liquid Poison Pump 12 and Check Valve Operability Test.”

Description. A monthly surveillance test of the 12 liquid poison pump was performed on March 4. For this test, the suction and discharge of the liquid poison pump are aligned to a water-filled test tank, and the water is recirculated by the pump at the specified discharge pressure. The pump discharge relief valve, which discharges to the liquid poison tank, is within the pressurized portion of the system while in the test configuration. Since the pump discharge pressure during the test is close to the relief valve lift set point, it is possible that the relief valve could inadvertently be lifted while adjusting the discharge pressure to the required test value. This would result in non-borated water being added to the liquid poison tank.

TS 4.1.2.b(2) requires that the liquid poison tank be sampled within one day of adding water to the tank. To ensure that the TS requirement is satisfied, N1-ST-Q8B contains steps to determine whether or not the pump discharge relief valve opened during the test. Step 9.1 requires that the chemistry department be notified to sample the liquid poison tank if these steps indicate that water was added. In this case, no water had been added, and the step was marked as not applicable.

However, an earlier step in the procedure also addresses the concern for inadvertent dilution of the liquid poison tank. Step 6.12 directs that a sample must be taken from the liquid poison tank within eight hours of the test completion, irrespective of whether or not water was added. The inspectors identified that this step had not been performed, and that the basis for not doing so was that step 9.1 had indicated that a sample was not required.

Analysis. The deficiency associated with this event is procedure non-compliance, in that the liquid poison tank was not sampled after completion of the test as required by step 6.12 of the surveillance procedure. The finding was greater than minor because it could reasonably be viewed as a precursor to a significant event. The finding was determined to be of very low safety significance (Green) using Phase 1 of the Reactor Safety SDP because there was no effect on system operability. Water was not added to the liquid poison tank and therefore its concentration was not affected. The procedure non-compliance is an example of a cross-cutting issue in human performance.

Enforcement TS 6.8.1 states, in part, that, “Written procedures . . . shall be established, implemented and maintained . . .” Contrary to the above, surveillance procedure N1-ST-Q8B, “Liquid Poison Pump 12 and Check Valve Operability Test,” was not correctly implemented on March 4, 2003, in that the liquid poison tank was not sampled within eight hours of the test completion as required by step 6.12. Because this procedural non-compliance is of very low safety significance and has been entered into the corrective action program (DER 2003-1095), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-220/03-02-04, Procedural Non-Compliance Resulted in the Liquid Poison Tank Not Being Sampled as Required.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The inspector conducted an in-office review of licensee submitted changes to emergency plan-related documents to determine if the changes decreased the effectiveness of the plan. A thorough review was conducted of aspects of the plan related to the risk significant planning standards (RSPS), such as classifications, notifications and protective action recommendations. A cursory review was conducted for non-RSPS portions. These changes were reviewed against 10 CFR 50.54(q) to ensure that the changes do not decrease the effectiveness of the plan, and that the changes as made continue to meet the standards of 10 CFR50.47(b) and the requirements of Appendix E. These changes are subject to future inspections to ensure that the impact of the changes continues to meet NRC regulations. The submitted and reviewed documents (Plan and Implementing Procedures) are provided in Attachment 1 of this report.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspector reviewed the access control program by examining the controls established for exposure significant areas including: postings, barricades, radiological briefings to workers, radiation protection technician direct and remote job coverage, and locking controls of access to radiologically significant areas. The review was against criteria contained in Plant Technical Specifications and 10 CFR 20.1601. In-plant areas and activities reviewed included:

- Drywell main steam isolation valve modifications during March 23-27, 2003
- #15 recirculation pump motor and seal replacement during March 23-24, 2003
- Drywell in-service inspection of the shroud H-9 weld demobilization activities during March 24-27, 2003
- Under-vessel neutron instrument calibrations and local power range monitor replacements during March 27, 2003

- Drywell isolation condenser check valves 39-03 and 39-04 repairs during March 26-27, 2003
- Isolation condenser thermocouple removal during March 27, 2003
- Reactor water cleanup in-service inspection activities during March 24-26, 2003
- Local power range monitor replacement activities from the refueling floor on March 27, 2003

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed licensee ALARA performance in accordance with 10 CFR 20.1101(b). Areas reviewed included an evaluation of ALARA planning for the five highest exposure outage tasks. (The specific ALARA job performance observations performed are listed in Section 20S1, above.) In addition, the following ALARA inspection activities were conducted:

- Independent shielding effectiveness radiation surveys were conducted in the drywell;
- Observation of closed circuit television equipment, teledosimetry, headset and intercom communication use in the drywell was conducted with respect to drywell remote health physics work surveillance capability and technical specification requirements

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Occupational Radiation Safety

a. Inspection Scope

The inspectors reviewed eight DERs related to occupational radiation safety that were initiated during the current Unit 1 refueling outage. The review was conducted to evaluate the licensee's program for identifying and resolving problems in the area of radiation protection.

b. Findings

No findings of significance were identified.

.2 Fire Brigade Issues

a. Inspection Scope

The inspectors selected seven DERs for detailed review (DERs NM-2002-962, NM-2002-3364, NM-2002-3631, NM-2002-4157, NM-2002-3668, NM-2002-4613, and NM-2002-5057). The DERs were associated with deficiencies in site fire brigade qualifications, training and staffing. The DERs were reviewed to ensure that the full extent of these issues were identified, appropriate evaluations were performed, and appropriate corrective actions were specified and prioritized. The inspectors evaluated the DERs against the requirements of the licensee's corrective action program as delineated in procedure NIP-ECA-01, Deviation/Event Report, and 10 CFR 50, Appendix B.

Additionally, the inspectors reviewed a DER list which showed issues dated from July 2000 through March 2003, to determine the number of issues associated with fire brigade qualifications, training and staffing. The inspectors also reviewed selected corrective action documents and program documents to determine how these issues were resolved.

The inspectors reviewed fire brigade qualification documents, training documents and staffing requirements to determine if the fire brigade met the requirements for performing fire fighting activities.

The inspectors interviewed the Unit 2 Operations Manager, the Fire Protection Supervisor, the Fire Protection Manager and Fire Chiefs to determine their familiarity with the issues' inspected and to gain insights into the issues' resolution.

b. Findings

No findings of significance were identified.

The inspectors found the corrective actions associated with the reviewed DERs were appropriate and acceptable upon completion. The apparent cause evaluations were detailed and thorough. The licensee appropriately conducted extent of condition reviews for the identified issues. The inspectors determined that the reviewed fire brigade issues were minor when compared to the guidance provided in MC 0612.

The inspectors noted that there had been a number of issues associated with fire brigade qualifications, staffing and training since the transition from dedicated fire fighters to a fire brigade in July 2001. To address these issues, the licensee put a dedicated fire protection supervisor in place and initiated DER NM-2003-740 to capture and resolve all issues (including fire brigade issues) in their fire protection program. Additionally, Constellation was developing initiatives which include implementation of a Corrective Action Review Team (CART) and an Assess, Plan, Implement and Review (APIR) Program to identify, assess and correct fire protection issues that are repetitive in nature. Constellation was also developing fire protection performance indicators and a fire protection “top 10" list to help address and track fire protection issues including fire brigade issues.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. Conway, Vice President, Nine Mile Point, and other members of licensee management at the conclusion of the inspection on April 11, 2003. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

F. Cann, Fire Protection Manager
J. Conway, Vice President Nine Mile Point
R. Dean, General Supervisor, Design Engineering
G. Detter, Manager, Support Services
J. Gerber, ALARA Supervisor
L. Hopkins, Plant General Manager
J. Jones, Supervisor, Emergency Preparedness
B. Merryman, Manager, Unit 1 Operations
S. Minahan, Manager, Unit 2 Operations
B. Montgomery, Manager, Engineering Services
W. Paulhardt, Radiation Protection Manager
M. Peckham, Manager, Work Control/Outage Management
B. Randall, General Supervisor, System Engineering
C. Terry, Manager, Quality and Performance Assessment
D. Wolniak, General Supervisor, Licensing

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

50-220/2003-02-01	NCV	Failure to Implement Radiation Emergency Procedure.
50-220/2003-02-02	NCV	Procedural Non-Compliance Resulted in Control Rods Being Withdrawn at Greater than Nominal Speed.
50-410/2003-02-03	NCV	Procedural Non-Compliance Resulted in the Incorrect Position for Division 1 Unit Heater Switch.
50-220/2003-02-04	NCV	Procedural Non-Compliance Resulted in the Liquid Poison Tank Not Being Sampled as Required.

LIST OF DOCUMENTS REVIEWED

Licensing Documents

Nine Mile Point Unit 1 - Final Safety Evaluation Report (Updated)
Nine Mile Point Unit 2 - Updated Safety Analysis Report

Deviation Event Reports

NM-2002-469	NM-2002-809	NM-2002-962	NM-2002-3363
NM-2002-3364	NM-2002-3487	NM-2002-3599	NM-2002-3631
NM-2002-3668	NM-2002-4157	NM-2002-4613	NM-2002-4723
NM-2002-4950	NM-2002-4981	NM-2002-5466	NM-2003-740

Procedures

GAP-OPS-01, Administration Of Operations, Rev. 24
NIP-ECA-01, Deviation/Event Report, Rev. 27
NIP-EPP-01, Emergency Response Organization Expectations and Responsibilities, Rev. 13
NIP-FPP-01, Fire Protection Program, Rev. 13
NIP-TQS-402, Nuclear Fire Protection/Appendix R Fire Brigade Training Programs, Rev.14
N1-SOP-9.1, Control Room Evacuation, Rev. 10
Site Emergency Plan, Rev 48
EPIP-EPP-05A, Local Area/Building Evacuation, Rev 2
EPIP-EPP-08, Offsite Dose Assessment and Protective Action Recommendation, Rev 13, 14
EPIP-EPP-09, Determination of Core Damage Under Accident Conditions, Rev 3
EPIP-EPP-11, Hazardous Material Incident Response, Rev 7
EPIP-EPP-18, Activation and Direction of the Emergency Plans, Rev 10
EPIP-EPP-20, Emergency Notifications, Rev 14, 15, 16
EPIP-EPP-21, Radiation Emergencies, Rev 5
EPIP-EPP-27, Emergency Public Information Procedure, Rev 10
EPIP-EPP-28, Fire Fighting, Rev 7
EPMP-EPP-01, Maintenance of Emergency Preparedness, Rev 15
EPMP-EPP-04, Emergency Exercise/Drill Procedure, Rev 9, 10
EPMP-EPP-05, Emergency Preparedness Program Self Assessment, Rev 9
EPMP-EPP-06, Emergency Response Organization Notification Maintenance and Surveillance, Rev 11

Fire Protection System Health Reports

Unit 1 Fire System Health Report, Second Quarter 2002
Unit 1 Fire System Health Report, Third Quarter 2002
Unit 2 Fire Protection System Health Report, Second Quarter 2002
Unit 2 Fire Protection System Health Report, Third Quarter 2002

Fire Brigade Qualifications Documents

Fire Brigade - EMT/Medical
Fire Brigade - Initial Training

Fire Brigade Drill Assessments

Fire Brigade Drill Assessment, Completed September 23, 2002
Fire Brigade Drill Assessment, Completed January 3, 2003
Fire Brigade Drill Assessment, Completed January 10, 2003
Fire Brigade Drill Assessment, Completed January 17, 2003
Fire Brigade Drill Assessment, Completed January 20, 2003 (Announced)
Fire Brigade Drill Assessment, Completed January 29, 2003 (Announced)
Fire Brigade Drill Assessment, Completed January 29, 2003 (Unannounced)
Fire Brigade Drill Assessment, Completed March 5, 2003
Fire Drill Scenario 1-00.04, Fire In Relay Cabinet, Aux. Control Room 261', November 19, 2000

Miscellaneous Documents

Constellation Energy Group Letter Dated March 13, 2003, Nine Mile Point Units 1 and 2 - Safety Review and Audit Board Meeting Minutes 03-01
Fire Protection Self Assessment - 2002-38
NFPA 600, Standard on Industrial Fire Brigades, 200 Edition
Nine Mile Point Organization Chart, January 10, 2003
Safety Evaluation Report No. 96-002, Fire Brigade Member Requirements and Revision of NIP-FPP-01, January 23, 1996
Unit 2 Ops. Night Orders, On Duty Fire Brigade, March 10, 2003

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
CFR	Code of Federal Regulations
CRD	control rod drive
DERs	deviation event reports
EAL	emergency action level
EDG	emergency diesel generator
EPIP	emergency plan implementing procedures
ICMs	interim compensatory measures
NCV	non-cited violation
NRC	U.S. Nuclear Regulatory Commission
PMT	post-maintenance testing
RSPS	risk significant planning standard
SDP	significance determination process
SSCs	structures, systems, and components
SSS	station shift supervisor
WOs	work orders