27 July 2004

Mr. James A. Spina
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 05000220/2004003 and 05000410/2004003

Dear Mr. Spina:

On June 30, 2004, the US Nuclear Regulatory Commission (NRC) completed an inspection at Nine Mile Point Nuclear Station, Units 1 and 2. The enclosed integrated inspection report (IR) documents the inspection findings which were discussed on July 9, 2004, 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 one NRC-identified and two self-revealing findings of very low safety significance (Green), all of which were determined to involve violations of NRC requirements. In addition, one licensee-identified violation which was determined to be of very low safety significance is listed in Section 4OA7 of this report. Because of the very low safety significance and because the violations were entered into your corrective action program, the NRC is treating these 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.

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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 05000220/2004003 and 05000410/2004003
w/Attachment: Supplemental Information

cc w/encl:
M. J. Wallace, President, Constellation Generation
M. Heffley, Senior Vice President and Chief Nuclear Officer
J. M. Petro, Jr., Esquire, Counsel, Constellation Energy Group, Inc.
M. J. Wetterhahn, Esquire, Winston and Strawn
P. R. Smith, President, New York State Energy, Research, and Development Authority
J. Spath, Program Director, New York State Energy Research and Development Authority
C. Adrienne Rhodes, Chairman and Executive Director, State Consumer Protection Board **(NMP2 ONLY)**
J. R. Evans, LIPA **(NMP2 ONLY)**
P. D. Eddy, Electric Division, NYS Department of Public Service
C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
Supervisor, Town of Scriba
T. Judson, Central NY Citizens Awareness Network

Distribution w/encl:
H. Miller, RA/J. Wiggins, DRA (1)
J. Trapp, DRP
N. Perry, DRP
C. Miller, RI EDO Coordinator
R. Laufer, NRR
P. Tam, PM, NRR
R. Clark / R. Guzman, PM, NRR (Backup)
G. Hunegs, SRI - Nine Mile Point
B. Fuller, RI - Nine Mile Point
E. Knutson, RI - Nine Mile Point
Region I Docket Room (with concurrences)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.:	50-220, 50-410
License Nos.:	DPR-63, NPF-69
Report No.:	05000220/2004003 and 05000410/2004003
Licensee:	Nine Mile Point Nuclear Station, LLC (NMPNS)
Facility:	Nine Mile Point, Units 1 and 2
Location:	348 Lake Road Oswego, NY 13126
Dates:	April 1, 2004 - June 30, 2004
Inspectors:	G. Hunegs, Senior Resident Inspector B. Fuller, Resident Inspector E. Knutson, Resident Inspector S. Chaudhary, Reactor Inspector J. Furia, Senior Health Physicist
Approved by:	James M. Trapp, Chief Projects Branch 1 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000220/2004003, 05000410/2004003; 04/01/2004 - 06/30/2004; Nine Mile Point, Units 1 and 2; Refueling and Other Outage Activities.

This report covered a 13-week period of inspection by resident inspectors, and announced inspections by two region-based inspectors. Three Green findings 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 non-cited violation (NCV) of Unit 1 Technical Specification 6.4, "Procedures," was identified concerning inadequate use of procedures, in that an extra gasket was installed in an electromatic relief valve (ERV) pilot valve assembly, contrary to the maintenance procedure instructions. The procedure did not direct installing a second gasket; however, a second gasket was installed which caused the ERV to fail to close during post-maintenance testing at power. The performance deficiency associated with this finding is the failure to follow procedures. The finding is greater than minor because it is associated with the human performance attribute of the Initiating Event Cornerstone and adversely affects the cornerstone objective to limit the likelihood of those events that upset plant stability during power operations. The finding is of very low safety significance as determined by Phase 2 of the significance determination process. The failure to follow procedures is an example of a cross-cutting issue in the area of human performance. (Section 1R20)

Cornerstone: Barrier Integrity

- Green. A self-revealing non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified at Unit 1 for a repeat occurrence of a scaffold installation that interfered with operation of one of the reactor building to pressure suppression chamber vacuum breakers. The performance deficiency associated with this finding is that scaffolding was installed such that it would have restricted the vacuum breaker from fully opening, thereby rendering the vacuum breaker valve inoperable. A contributing cause is ineffective corrective action since a previous occurrence of a vacuum breaker being blocked by scaffolding was identified by the NRC in 2003. The finding is greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of barrier performance, and adversely affects the associated cornerstone objective of providing reasonable assurance that the primary containment protect the public from radionuclide releases caused by accidents or events. The finding is of very low safety significance in accordance with Table 6.2 of the Containment Integrity SDP because it relates to failure of a component critical to suppression pool integrity/scrubbing, and because the condition existed for less than three days. The inadequate corrective action taken to prevent operational interferences due to scaffolding installations is an example of a cross-cutting issue in problem identification and resolution. (Section 1R20)

- Green. The inspectors identified a non-cited violation (NCV) of Unit 2 Technical Specification 5.4.1 concerning the test configuration specified in surveillance procedure N2-OSP-GTS-R001, "Secondary Containment Integrity Test," in that the test did not establish conditions duplicating the allowable worst case configuration for the access doors in secondary containment. The performance deficiency associated with this finding is an inadequate test procedure, in that degraded seals on one of the two doors in a secondary containment access opening would not always be identified by the surveillance. The finding is greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of procedure quality and adversely affects the associated cornerstone objective of providing reasonable assurance that the primary containment protect the public from radionuclide releases caused by accidents or events. The finding is of very low safety significance because it did not represent a degradation of the radiological, toxic or smoke barrier function; did not represent an actual open pathway in physical integrity or actual reduction of the atmospheric pressure control function of the containment; and was not potentially risk significant due to seismic, flood, fire or weather related initiating events. (Section 1R20)

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective actions 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 percent power. On April 10, power was reduced to 95 percent to secure 13 condensate pump due to shaft packing failure. On April 27, Unit 1 was removed from service for a scheduled maintenance outage to replace electromatic relief valve (ERV)-123 due to internal leakage. The replacement ERV-123 failed open during at-power testing on May 2, necessitating a manual reactor scram and return to the cold shutdown condition. Following valve repair and successful testing, Unit 1 was returned to service on May 4, and reached 100 percent the following day. On May 15, power was reduced to 95 percent to secure 13 reactor recirculation pump (RRP) due to mechanical problems with its associated motor-generator. On May 29, power was reduced to 85 percent to restore 13 RRP to service. On June 7, power was reduced to 95 percent to secure 13 condensate pump due to shaft packing failure. On June 19, power was reduced to 95 percent to secure 15 RRP due to potential mechanical problems with the pump motor. Unit 1 operated at 100 percent power for the remainder of the inspection period.

Nine Mile Point Unit 2 (Unit 2) began the inspection period in refueling outage RF09, which had begun on March 15. The plant was returned to service on April 25, and reached 100 percent power on April 28. Power was reduced to 92 percent on June 18, as compensatory action for a control rod that drifted inward. Power was restored to 100 percent that same day, and remained at 100 percent for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope (71111.01 - 2 Samples)

The inspectors examined two Unit 1 risk significant systems, the control room ventilation system and the reactor building closed loop cooling system (cooling water to control room air cooling (CRAC), to verify that design features and operating procedures support operation during periods of hot weather. Documents reviewed included the Unit 1 Final Safety Analysis Report (FSAR), operating procedure N1-OP-49, “Control Room Ventilation System,” and System Design Basis Document (SDBD)-602, “Control Room Air Treatment (CRAT) System.” In addition, the inspectors performed a material condition inspection of portions of the systems.

The inspectors performed a review of Unit 1 preparations for expected severe weather. On June 9, the inspectors reviewed Unit 1 contingency plans in preparation for potential severe thunderstorms while one of two sources of off-site power was not available due to distribution system maintenance. Loss of the remaining off-site power line would have necessitated an emergency power reduction due to the loss of balance-of-plant equipment necessary to support full power operation.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdown. (71111.04 - 4 Samples)

The inspectors performed partial system walkdowns to verify system and component alignment and to note any discrepancies that would impact system operability.

- On May 11, the inspector selected the Unit 2 reactor core isolation cooling (RCIC) system to conduct a partial system walkdown due to increased risk significance during planned maintenance on the high pressure core spray (HPCS) system. The walkdown included the control room switch verification, physical inspection, and partial verification of the system lineup. N2-OP-35, "Reactor Core Isolation Cooling," and N2-VLU-01, "Walkdown Order Valve Lineup and Valve Operations," Attachment 35, "N2-OP-35 Walkdown Valve Lineup," were used for this review.

- On May 13, the inspector selected the Unit 2 standby gas treatment system (GTS), Division II, to conduct a partial system walkdown due to increased risk significance during a planned maintenance period on the Division I GTS. The walkdown included the control room switch verification, physical inspection, and partial verification of the system lineup. N2-OP-61B, "Standby Gas Treatment System," and N2-VLU-01, "Walkdown Order Valve Lineup and Valve Operations," Attachment 61B, "N2-OP-61B Walkdown Valve Lineup," were used for this review.

- On May 19, the inspector selected the Unit 1 core spray (CS) system to conduct a partial system walkdown based on safety significance. The walkdown included the control room switch verification, physical inspection, and partial verification of the system lineup. N1-OP-2, "Core Spray System," and drawing C-18007-C were used for this review.

- On June 14, the inspector selected the Unit 1 11 high pressure coolant injection (HPCI) system to conduct a partial system walkdown due to increased risk significance during an unplanned maintenance period on the 12 HPCI train. The walkdown included the control room switch verification, physical inspection, and partial verification of the system lineup. N1-OP-16, "Feedwater System Booster Pump to Reactor," was used for this review.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope (71111.05Q - 9 Samples)

The inspectors walked down accessible portions of fire areas described below to assess the licensee's control of transient combustible material and ignition sources, fire detection and suppression capabilities, and fire barriers and any related compensatory measures. The condition of fire detection devices, and readiness of sprinkler fire suppression systems and fire doors, were also inspected against industry standards. In addition, the fire protection features were inspected, including ventilation system fire dampers, structural steel fire proofing, and electrical penetration seals. Reference material reviewed for installed features included the Unit 1 final safety analysis report (FSAR) and the Unit 2 updated safety analysis report (USAR).

- Unit 1 Battery Rooms (TB 277 foot elevation)
- Unit 1 Condensate Storage Tank area and Reactor Building Emergency Ventilation area (Turbine Auxiliary Extension Building)
- Unit 1 Control Room Emergency Ventilation (TB 300 foot elevation)
- Unit 1 Auxiliary Control Room (TB 261 foot elevation)
- Unit 1 Containment Spray Corner Rooms
- Unit 2 Reactor Building 175 foot elevation
- Unit 2 Standby Gas Treatment Bays (RB 261 foot elevation)
- Unit 2 Residual Heat Removal (RHR) B Room (RB 175 foot elevation)
- Unit 2 RHR C Room (RB 175 foot elevation)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope (71111.06 - 1 Sample)

The inspectors performed a walkdown of the Unit 1 Turbine Building Basement (250 foot elevation) to examine its susceptibility to internal flooding. This area was selected due to the safety significance of the remote shutdown panel and Cable Spreading Room. Documents reviewed included the FSAR and the Individual Plant Evaluation (IPE).

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope (71111.11Q - 1 Sample and 71114.06 - 1 Sample)

Resident Inspector Quarterly Review. The inspectors reviewed one licensed operator requalification training activity which included procedure 71114.06, "Drill Evaluation," simulator-based training evolution, to assess the licensee's training program effectiveness. The inspectors observed Unit 2 licensed operator simulator training on June 15, 2004. 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 reviewed simulator fidelity through a sampling process. The inspectors evaluated emergency response organization performance regarding initial and subsequent actions by licensed operators.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

Maintenance Rule Implementation. (71111.12B - 4 Samples)

The inspector reviewed the periodic evaluations required by 10 CFR 50.65 (a)(3) to verify adequate consideration was provided for the balancing of reliability and unavailability for structures, systems and components (SSCs) contained within the scope of the maintenance rule. The inspector reviewed the licensee's most recent maintenance rule (MR) program periodic evaluation report, which covered the period from January 2002 through September 2003.

The inspector reviewed the safety significant systems that were in (a)(1) status to verify that: (1) goals and performance criteria were appropriate, (2) industry operating experience was considered, (3) corrective action plans were effective, and (4) performance was being effectively monitored. As of May 24, there were forty-two systems in (a)(1) status; seventeen of these systems were in a monitoring status, while corrective actions were under development and/or implementation for the remaining systems. The following systems and components were reviewed:

- Containment Ventilation and Purge Fan (FN-201-35) (Unit 1)
- Reactor Vessel Level Wide Range Level Indication System (Unit 1)
- Reactor Water Cleanup System Inboard Containment Isolation Valve (Unit 1)
- 115 kilovolt (kV) Off-site Power Line # 4 (Unit 1)
- Control Rod Drive Scram Discharge Volume High Level Detector (Unit 1)
- 2FWS-M1A Feedwater Pump Motor (Unit 2)
- Reactor Core Isolation Cooling System (Unit 2)
- Spent Fuel Cooling Flow Control Valve (Unit 2)

The inspector reviewed the following (a)(2) systems to confirm that their performance met the applicable MR performance criteria:

- Containment Spray (Unit 1)
- Post-Accident Sampling System (Unit 1)

Routine Maintenance Effectiveness Inspection. (71111.12Q - 2 Samples)

The inspectors reviewed one performance-based problem during this inspection period, involving the Unit 2 standby gas treatment system, to assess the effectiveness of the maintenance program. In addition, the inspectors reviewed the performance and condition history of one high safety significant system, the Unit 1 core spray system, to identify system performance. Reviews focused on: (1) proper MR scoping in accordance with 10 CFR 50.65(2), (2) characterization of failed SSCs; safety significance classifications, (3) 10 CFR 50.65 (a)(1) and (a)(2) classifications, and, (4) the appropriateness of performance criteria for SSCs classified as (a)(2). The inspectors reviewed the licensee's system scoping documents, system health reports and corrective action program documents.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope (71111.13 - 5 Samples)

The inspectors reviewed five risk assessments and emergent work activities during this inspection period. For selected maintenance, work items or work orders (WOs) the inspectors evaluated: (1) the effectiveness of the risk assessments performed before the maintenance activities were conducted, (2) risk management control activities, (3) the necessary steps taken to plan and control resultant emergent work tasks, and, (4) the overall adequacy of identification and resolution of emergent work and the associated maintenance risk assessments. GAP-OPS-117, "Integrated Risk Management," was used for this review:

- Failure of the 12 control rod drive (CRD) pump while the 12 high pressure coolant injection system and off-site power line 4 were inoperable (Unit 1)
- Repair reactor core isolation cooling inlet steam trap rupture disc, performed under WO 04-10965 (Unit 2)
- Repair of refuel bridge with fuel bundle raised above the top guide, performed per GAP-PSH-10, Attachment 1, "Troubleshooting and Testing Control Form," reference deviation event report (DER) NM-2004-1812 (Unit 2)
- A four day planned maintenance period for the high pressure core spray system (Unit 2)
- Failure of annunciators on Panel P851 in the main control room (Unit 2)

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-routine Evolutions and Events

a. Inspection Scope (71111.14 - 2 Samples)

At 2:17 a.m. on May 2, Unit 1 operators inserted a manual reactor scram after ERV-123 failed to close during post-maintenance testing at low power. The inspector responded to the site and assessed current plant conditions, reviewed plant and operator response to the event, and examined the licensee's plans for corrective action. The inspector also reviewed emergency action level entry conditions to ensure proper classification of the event. Aspects of this event that were not related to operator performance are further discussed in Section 4OA3 of this report.

On May 10, Unit 2 experienced a partial loss of control room annunciators during troubleshooting activities on an annunciator circuit. Technicians were attempting to remove a circuit card associated with annunciator power supply to a balance of plant (BOP) annunciator panel when the power supply circuit card for an unrelated BOP panel was also unintentionally unseated. Operators increased monitoring of available instruments in the control room and at affected equipment locations in the plant. The inspectors responded to the control room and observed the operator's response, verified completion of annunciator response procedures, and reviewed emergency action level entry conditions to ensure proper classification of the event. The plant was stable throughout the time that annunciation was degraded. The licensee developed a restoration plan and successfully reinserted the affected circuit card.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15 - 7 Samples)

The inspectors reviewed operability evaluations during this inspection period, which affected risk significant mitigating systems, assessing: (1) the technical adequacy of the evaluation, (2) whether other existing degraded systems adversely impacted the affected system or compensatory measures, and, (3) where compensatory measures were used, whether the measures were appropriate and properly controlled, and that the degraded systems remained operable. S-ODP-OPS-0116, "Operability Determinations," was used for this review. Operability evaluations associated with the following DERs were reviewed:

- DER NM-2004-2403, Emergency diesel generator 102 jacket water cooling line support broken (Unit 1)

- DER NM-2004-2519, Water dripping in control room from ceiling (Unit 1)
- DER NM-2004-2604, Emergency service water (ESW) pump 12 tripped unexpectedly during attempted start following electrical PMs (Unit 1)
- DER NM-2004-2064, Issue/Questions concerning secondary containment and door integrity (Unit 2)
- DER NM-2004-2129, 2ICS*V157 [RCIC inboard injection check valve] had dual position indication (Unit 2)
- DER NM-2004-2313, Standby gas treatment system rupture disc failure (Unit 2)
- DER NM-2004-2506, Reactor core isolation cooling system inlet steam trap rupture disc failed (Unit 2)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope (71111.16 - 1 Sample)

The inspector reviewed operator workarounds at Units 1 and 2 to determine if any had a potential adverse effect on the functionality of mitigating systems. Included in this review were the effect on the (1) reliability, availability and potential for mis-operation of a system; (2) the potential increase in initiating event frequency; and (3) the ability of operators to respond in a correct and timely manner to plant transients and accidents. Additionally, the inspector looked for any combined effects of the operator workarounds. NAI-REL-02, “Workaround Program,” was used for this review.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope (71111.19 - 9 Samples)

The inspectors reviewed post-maintenance testing (PMT) procedures and associated testing activities for nine selected risk significant mitigating systems assessing 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, and (7) test equipment was removed following testing and equipment was returned to the status required to perform its safety function. The following PMT activities were reviewed:

- N1-ST-Q3, “HPCI Pump and Check Valve Operability Test” (Unit 1)
- N1-PM-V2, “Pump Curve Validation Test,” and N1-ST-Q2, “Control Rod Drive Pumps Flow Rate Test,” performed as PMT following 12 CRD pump rotor replacement (Unit 1)
- N1-ST-Q1D, “CS 122 Pump and Valve Operability Test,” as PMT for circuit breaker protective relay calibrations for the 122 core spray and core spray topping pumps (Unit 1)
- Panel 851 annunciator card replacement (Unit 2)
- N2-OSP-CSH-Q@002, “High Pressure Core Spray Pump and Valve Operability Test” (Unit 2)
- N2-OSP-GTS-M001, “Standby Gas Treatment System Functional Test” (Unit 2)
- Control rod 06-31 failed transponder card replacement (Unit 2)
- N2-OSP-EGS-M@002, “Diesel Generator and Diesel Air Start Valve Operability Test - Division III,” performed as PMT for electrical maintenance performed during a four-day maintenance period for the high pressure core spray system (Unit 2)
- N2-OSP-EGS-M@001, “Diesel Generator and Diesel Air Start Valve Operability Test - Division I and II,” performed as PMT following maintenance on the Division I emergency diesel generator (EDG) engine driven jacket cooling water circulating pump (Unit 2)

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope (71111.20 - 2 Samples)

Unit 2 Refueling Outage. The inspectors reviewed the RFO9 Shutdown Safety Report (SSR) and contingency plans for the Unit 2 refueling outage, conducted March 15 - April 25, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that ensured maintenance of defense-in-depth. During the refueling outage, the inspectors viewed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the SSR for key safety functions and compliance with the applicable TS when taking equipment out of service.
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and an accounting for instrument error.
- Controls over the status and configuration of electrical systems to ensure that TS and outage safety plan requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes.
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls including flow paths, configurations, alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Maintenance of secondary containment as required by TS.
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage.
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers.
- Licensee identification and resolution of problems related to refueling outage activities.

Unit 1 Maintenance Outage. On April 27, Unit 1 shut down for a one week planned maintenance outage to replace an electromatic relief valve (ERV) that was showing indications of pilot valve leakage, and to perform several other shutdown corrective maintenance activities. The inspectors reviewed the following activities related to the Unit 1 maintenance outage for conformance to the applicable procedure and witnessed selected activities associated with each evolution. Surveillance tests were reviewed to verify TS were satisfied. Inspections were focused on reactor decay heat removal, inventory control, power availability, and secondary containment. The inspectors reviewed the outage plan, and outage risk mitigation strategies and evaluations. Portions of the shutdown and cool down processes were observed.

b. Findings

1. Unit 1 Reactor Building to Torus Vacuum Breaker

Introduction. A Green self-revealing non-cited violation (NCV) was identified at Unit 1 in that scaffolding was interfering with operation of Reactor Building to Pressure Suppression Chamber vacuum breaker valve 68-09.

Description. Three sets of vacuum relief valves (vacuum breakers) are provided between the Primary Containment and Reactor Building. The safety functions of these vacuum relief valves are (1) in the open direction, to equalize the pressure between the suppression chamber (torus) and the reactor building so that the structural integrity of the primary containment is maintained following a design basis accident, and (2) in the close direction, to maintain primary containment integrity as containment isolation valves.

On April 26, 2004, at about 3:00 p.m., scaffolding was erected in the vicinity of vacuum breaker 68-09 to support upcoming outage maintenance on one of the torus to drywell vacuum breakers, 68-03. The following day, licensee personnel noted that the scaffolding was positioned such that it would interfere with the operation of vacuum breaker 68-09. Specifically, the valve's air operator mechanism would strike a scaffolding post during the valve's opening stroke (after about one inch of the actuator's six inch stroke), thereby preventing it from fully opening. The scaffolding did not impact the valve's closing function. At the time of discovery, the vacuum breakers were not required to be operable because the plant had transitioned into cold shutdown at 11:28 a.m., in preparation for the planned maintenance outage; however, for the approximately 20 hours that the obstruction had existed prior to that time, they had been required to be operable. In response to this finding, the scaffolding was repositioned to eliminate the interference. The issue was entered in the licensee's corrective action program as DER NM-2004-2178.

A similar event had occurred in March 2003. In that event, the inspectors identified that scaffolding was obstructing operation of reactor building to torus vacuum breaker 68-07. The issue was entered in the licensee's corrective action program as DER NM-2003-978 and was documented in NRC inspection report 50-220/03-02, 50-410/03-02. Long term corrective action for that event had included revision of the scaffold procedure to increase personnel awareness of the potential for plant equipment impacts. The inspectors reviewed maintenance administrative procedure S-MAP-MAI-0109, "NMPNS Scaffold Program," revision 00, effective date August 29, 2003, to verify that this change had been implemented. This new procedure included a requirement for a pre-installation area inspection to identify any plant equipment susceptible to operational limitation, and a post-installation inspection to verify that operation of valves and valve actuators is not hindered. This procedure requirement was not adequately implemented during the erection of scaffolding near the vacuum breaker valve on April 26, 2004.

Analysis. The performance deficiency was that scaffolding was installed such that it would have restricted the vacuum breaker from fully opening, thereby rendering the valve inoperable. Additionally, corrective actions for the previous occurrence of a vacuum breaker being blocked by scaffolding were not effective. Although action had been taken to proceduralize scaffold inspection requirements to avoid operational interferences, the inspections were to be performed by personnel who were qualified and competent in scaffold installation, not in integrated plant operations. In this event, the licensee determined that the scaffold inspector did not understand how the vacuum breaker actuator operated, and furthermore was under the impression that the entire containment vacuum breaker system was already inoperable. The inspectors concluded that the previous corrective action had been ineffective in that it did not require that scaffold inspections, which were meant to verify that the installation presented no operational impact, were performed by appropriately qualified personnel.

The finding was greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of barrier performance, in that it relates to primary containment vacuum relief system availability, and adversely affects the associated cornerstone objective of providing reasonable assurance that the primary containment protect the public from radionuclide releases caused by accidents or events. From Phase 1 of the Reactor Safety SDP, the finding screened to the Containment Integrity SDP (Appendix H of MC 0609) because it represented an actual reduction of the atmospheric pressure control function of the reactor containment. From Appendix H, Table 4.1, the finding screened as a Type B finding, in that it related to a degraded condition that has potentially important implications for the integrity of the containment, without affecting the likelihood of core damage. The SDP required an assessment of the risk significance per Section 6, because the finding is related to a containment SSC that has an impact on large early release frequency (LERF). The finding was determined to be of very low safety significance (Green) in accordance with Table 6.2 of the Containment Integrity SDP, because it related to failure of a component critical to suppression pool integrity/scrubbing, and because the condition existed for less than three days. This assessment is conservative, in that the vacuum breaker would have partially opened if required, and because the reactor was not operating at full power for the duration of the event. The inadequate corrective action taken to prevent operational interferences due to scaffolding installations was an example of a cross-cutting issue in problem identification and resolution.

Enforcement. 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” Criteria XVI, “Corrective Action,” states, in part, “Measures shall be established to assure that conditions adverse to quality, such as . . . malfunctions . . . are promptly identified and corrected.” Contrary to the above, action taken to correct a deficiency identified in March, 2003, concerning the station scaffold program and obstruction of SSCs, was inadequate, in that, on April 26, 2004, Unit 1 reactor building to torus vacuum breaker valve 68-09 was obstructed by scaffolding. Because this finding is of very low safety significance and has been entered into the licensee’s corrective action program (DER NM-2004-2178), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy:

NCV 05000220/2004003-01, Ineffective Corrective Action to Prevent Recurrence of Plant Equipment Obstruction by Scaffolding.

2. Unit 1 Electromatic Relief Valve (ERV)-123

Introduction. A self-revealing Green NCV was identified at Unit 1for the failure to implement maintenance procedures during ERV pilot valve assembly in accordance with T.S. 6.4, which resulted in the ERV failing open during testing.

Description. On April 27, 2004, Unit 1 was shut down to repair ERV-123 which had exhibited elevated main valve tail pipe temperature. The repair work found a small cut on the small bore pipe union inside the ERV enclosure, and the main valve and pilot valve were replaced. To verify leak tightness of the small bore piping, the pilot valve internals were removed and a test flange was installed to facilitate pressurizing the piping between the union and the pilot valve. After the leak test, the pilot was reassembled in accordance with procedure N1-MPM-001-245, "Main Steam Electromatic Relief Valves and Associated Pilot Valves Preventive Maintenance (PM) Removal Overhaul and Replacement." Reactor startup commenced on May 1, and on May 2, with Unit 1 at 19 percent power, ERV-123 was tested using the control switch. ERV-123 opened but failed to close. Operators inserted a manual scram as directed by procedure. ERV-123 is one of six TS required solenoid-actuated pressure relief valves in the automatic depressurization system.

Valve disassembly identified that an additional gasket was installed in the pilot valve assembly. The gasket had been inappropriately installed, in that the ERV maintenance procedure did not direct the installation. The additional gasket prevented the adequate crush of the pilot valve bushing assembly gasket which allowed steam to bypass the pilot valve. When the valve was manually actuated for testing, the ERV opened as expected. When the valve was given a closed signal, the bypass leakage around the pilot valve prevented steam pressure equalization across the main seat, causing the main valve to remain open.

Analysis. The performance deficiency was inadequate use of procedures. The maintenance procedure did not direct installing a second gasket; however, a second gasket, which had been needed only during pre-startup testing, was installed which caused the ERV to fail to close. The finding was greater than minor because it is associated with the human performance attribute of the Initiating Event Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability during power operations.

The finding screened from Phase 1 of the Reactor Safety SDP to Phase 2, because it contributed to the likelihood of a primary system LOCA initiator. Using Table 1 of the Unit 1 Risk-Informed Inspection Notebook, the initiating event likelihood (IEL) of a stuck-open SRV was increased by an order of magnitude. Per Table 2, with ERVs as the affected system, all initiating event scenarios except large break loss of coolant accident (LOCA) were analyzed and the finding was determined to be of very low safety significance (Green). The inadequate procedure adherence was an example of a cross-cutting issue in human performance.

Enforcement. Unit 1 TS 6.4, "Procedures," Section 1a, requires, in part, that, "Written procedures . . . shall be established, implemented and maintained that . . . cover applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 3, 1972." Regulatory Guide 1.33, Appendix A, November 3, 1972, Section I.1, states, in part, that, "Maintenance which can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures." N1-MPM-001-245, "Main Steam Electromatic Relief Valves and Associated Pilot Valves Preventive Maintenance Removal Overhaul and Replacement," which provides ERV maintenance guidance, does not direct installing a second gasket in the pilot valve assembly. Contrary to the above, on May 1, 2004, a second gasket was installed in the ERV-123 pilot valve assembly which caused the valve to fail to close during operation. However, because of the very low safety significance and because the corrective actions taken through DER NM-2004-2239 appeared to be reasonable, the issue is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000220/2004003-02, Failure to Follow ERV Maintenance Procedure Leads to ERV Failure to Close and Subsequent Scram.

3. Unit 2 Secondary Containment Integrity Testing

Introduction. A Green NCV of Unit 2 Technical Specification (TS) 5.4.1 was identified concerning the test configuration specified in a surveillance procedure N2-OSP-GTS-R001, "Secondary Containment Integrity Test," that did not establish conditions duplicating the allowable worst case configuration for the access doors in secondary containment.

Description. The function of the secondary containment is to contain, dilute and holdup fission products that may leak from primary containment following a design basis accident. To prevent ground level leakage, support systems maintain the secondary containment pressure less than the external pressure. For the secondary containment to be considered operable, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained. Any fission products present within the secondary containment atmosphere are contained in the standby gas treatment system prior to the air being released to the environment.

Unit 2 TS 3.6.4.1, "Secondary Containment," Surveillance Requirement (SR) 3.6.4.1.3 verifies that one secondary containment access door in each access opening is closed. SR 3.6.4.1.4 verifies that secondary containment can be drawn down to $\geq$ 0.25 inch of vacuum water gauge in $\leq$ 66.7 seconds using one standby gas treatment (SGT) subsystem. SR 3.6.4.1.5 verifies that secondary containment can be maintained $\geq$ 0.25 inch of vacuum water gauge for 1 hour using one SGT subsystem at a flowrate $\leq$ 2670 cubic feet per minute. SR 3.6.4.1.3 is required to be performed on a 31-day frequency. SR 3.6.4.1.4 and 3.6.4.1.5 are required to be performed on a 24-month frequency.

Surveillance procedure N2-OSP-GTS-R001, “Secondary Containment Integrity Test,” is the implementing procedure for SR 3.6.4.1.4 and 3.6.4.1.5. Prerequisite Step 7.9 verifies that reactor building integrity is established with all hatches shut and sealed, and that at least one door in each access to secondary containment is closed and sealed, which meets the requirement of SR 3.6.4.1.3.

The normal operating condition for secondary containment is for both doors in each access opening to be closed. When access to secondary containment is needed, one door of the pair is opened at a time to permit passage. One door is required to be shut to maintain secondary containment integrity. Opening one door of the pair reduces the resistance to leakage airflow into the secondary containment and increases air in-leakage, depending upon the differential pressure between secondary containment and the outside. The allowable configuration of access doors which would permit one door in a pair to be open would be the worst case (maximum inleakage) for testing.

Analysis. The performance deficiency was an inadequate test procedure. Specifically, the surveillance procedure did not establish conditions duplicating the allowable worst case configuration for the access doors in secondary containment. As a result, degraded seals on one of the two doors in an access opening would not necessarily be identified by the surveillance. The performance deficiency was more than minor because it was associated with the Barrier Integrity Cornerstone attribute of procedure quality and adversely affected the associated cornerstone objective of maintaining the functionality of containment. Using Phase I of the Reactor Safety SDP, the finding was determined to be of very low safety significance (Green), because it did not represent a degradation of the radiological, toxic or smoke barrier function; did not represent an actual open pathway in physical integrity or actual reduction of the atmospheric pressure control function of the containment; and was not potentially risk significant due to seismic, flood, fire or weather related initiating events.

Enforcement. Unit 2 TS 5.4, “Procedures,” states, in part, that, “Written procedures shall be established, implemented and maintained covering . . . the applicable procedures, recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.” Regulatory Guide 1.33, Appendix A, February 1978, Section 8.b, states, in part, that, “. . . implementing procedures are required for each surveillance test . . . listed in the technical specifications.” Surveillance test procedure N2-OSP-GTS-R001, “Secondary Containment Integrity Test,” is the implementing procedure for TS surveillance requirements 3.6.4.1.4 and 3.6.4.1.5. Contrary to the above, on March 25, 2004, surveillance test procedure N2-OSP-GTS-R001, “Secondary Containment Integrity Test,” was not appropriate to the circumstances, in that the allowable worst case door configuration per TS SR 3.6.4.1.3 was not specified for conduct of the test. However, because of the very low safety significance and because the corrective actions taken through DER NM-2004-1305 appeared to be appropriate, the issue is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000410/2004003-03, Inadequate Secondary Containment Integrity Test Procedure.

1R22 Surveillance Testing

a. <u>Inspection Scope</u> (71111.22 - 6 Samples)

The inspectors witnessed performance of surveillance test procedures and reviewed test data of selected risk significant SSC's to assess whether the testing satisfied TS, FSAR/USAR, 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 surveillance tests were reviewed:

- N1-ST-Q6B, Containment Spray System Loop 121 Quarterly Operability Test (Unit 1)
- N1-ST-Q8B, Liquid Poison Pump 12 and Check Valve Operability Test (Unit 1)
- N2-OSP-RHS-Q@005, RHR System Loop B Pump and Valve Operability Test (Unit 2)
- N2-OSP-EGS-R002, Operating Cycle Diesel Generator 24 Hour Run and Load Rejection Division I and II (Unit 2)
- N2-OSP-HVK-Q001, Control Building Chilled Water Loop A and B Pump and Valve Operability Test (Unit 2)
- N2-OSP-EGS-M@002, Diesel Generator and Diesel Air Start Valve Operability Test - Division III (Unit 2)

b. <u>Findings</u>

No findings of significance were identified.

1R23 <u>Temporary Plant Modifications</u>

a. <u>Inspection Scope</u> (71111.23 - 2 Samples)

The inspectors reviewed a temporary plant modification to supply external cooling water to the Unit 1 13 condensate pump packing and thereby restore the 12 high pressure coolant injection train to operable status. Temporary change N1-04-071, "Temporary Cooling Water for PMP-50-03 [13 condensate pump] Packing," was reviewed to determine whether the temporary change adversely affected system or support system availability, or adversely affected a function important to plant safety. The inspectors reviewed the associated system design bases, including the FSAR and TS, and assessed the adequacy of the safety determination screening and evaluations.

The inspectors reviewed a temporary plant modification to swap electrical power sources to the Unit 1 13 condensate pump, to thereby restore the 12 high pressure coolant injection train to operable status. Temporary change N1-04-104, "13 Condensate Pump Swap Power Sources," was reviewed to determine whether the temporary change adversely affected system or support system availability, or adversely affected a function important to plant safety. The inspectors reviewed the associated system design bases, including the FSAR and TS, and assessed the adequacy of the safety determination screening and evaluations.

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 (71121.01 - 6 Samples)

The inspector reviewed and assessed the adequacy of the licensee’s internal dose assessment for any actual internal exposure greater than 50 mrem Committed Effective Dose Equivalent (CEDE).

The inspector examined the licensee’s physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within the spent fuel storage pool.

The inspector reviewed the licensee’s self assessments, audits, Licensee Event Reports, and Special Reports related to the access control program since the last inspection. The inspector determined that identified problems were entered into the corrective action program for resolution.

The inspector discussed with the Radiation Protection Manager (RPM) locked high radiation areas (LHRA), and very high radiation areas (VHRA) controls and procedures. The inspector verified that any changes to licensee procedures did not substantially reduce the effectiveness and level of worker protection.

The inspector discussed with first-line health physics supervisors the controls in place for special areas that have the potential to become VHRA during certain plant operations. The inspector determined that these plant operations require communication beforehand with the health physics (HP) group, so as to allow corresponding timely actions to properly post and control the radiation hazards.

During job performance observations, the inspector observed radiation protection technician performance with respect to all radiation protection work requirements. The inspector determined that they were aware of the radiological conditions in their workplace and the radiation work permit (RWP) controls/limits, and that their performance is consistent with their training and qualifications with respect to the radiological hazards and work activities.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. <u>Inspection Scope</u> (71121.02 - 5 Samples)

The inspector reviewed the as low as is reasonably achievable (ALARA) work activity evaluations, exposure estimates, and exposure mitigation requirements for the Unit 2 refueling outage. The inspector determined that the licensee has established procedures, engineering and work controls, based on sound radiation protection principles, to achieve occupational exposures that are ALARA.

The inspector compared the results achieved (dose rate reductions, person-rem used) with the intended dose established in the licensee's ALARA planning for these work activities.

The inspector reviewed the assumptions and basis for the current annual collective exposure estimate, and reviewed applicable procedures to determine the methodology for estimating work activity-specific exposures and the intended dose outcome.

The inspector reviewed the licensee's method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work are encountered.

Utilizing licensee records, the inspector determined the historical trends and current status of tracked plant source terms, and determined that the licensee was making allowances and developing contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry.

b. <u>Findings</u>

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

a. Inspection Scope (71121.03 - 1 Sample)

The inspector conducted a review of selected radiation protection instruments located in the radiologically controlled area (RCA). Items reviewed were: verification of proper function; certification of appropriate source checks; and, calibration for those instruments used to ensure that occupational exposures were maintained in accordance with 10 CFR 20.1201.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152 - 1 Sample)

1. Public Radiation Safety

a. Inspection Scope

The inspector selected issues associated with occupational radiation safety performance during 2004 which were identified in the licensee's corrective action program for detailed review. The inspector met with the plant radiation protection manager to discuss these reports. The review focused on assurance that the full extent of the issues was identified, that an appropriate evaluation was performed, and that appropriate corrective actions were specified and prioritized.

b. Findings

No findings of significance were identified.

2. Corrective Action Review by Resident Inspectors

a. Inspection Scope

Continuous Review

As required 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 items entered into the Nine Mile Point's corrective action program. This review was accomplished by reviewing paper copies of each DER, attending daily screening meetings and assessing Nine Mile Point's computerized database.

Semi-Annual Review

The inspectors performed an in-depth, semi-annual problem identification and resolution (PI&R) review of licensee documents written from January 2004 through June 2004 to verify that the licensee is identifying issues at the appropriate threshold, entering them into the corrective actions program, and ensuring that there are no significant, adverse trends outside of the corrective action program which would indicate the existence of a more significant safety issue.

The inspectors reviewed licensee performance indicators, self-assessment reports, quality assurance audit/surveillance reports, corrective action reports, and system health reports, and compared the results of the review with the results reported in the NRC baseline inspection program. Additionally, the inspectors evaluated the reports against the requirements of the licensee's corrective action program as delineated in NIP-ECA-01, "Deviation/Event Report."

b. Findings

No findings of significance were identified.

3. Cross-References to PI&R Findings Documented Elsewhere

Section 1R20 describes a finding for ineffective correction action associated with a vacuum breaker being blocked by scaffolding, since a previous occurrence of a vacuum breaker being blocked by scaffolding was identified by the NRC in 2003.

4OA3 Event Follow-up

1. (Closed) LER 50-410/2003-002, Reactor Scram Due to Electric Grid Disturbance.

On August 14, 2003, Unit 2 automatically scrammed from 100 percent power due to a turbine control valve fast closure signal that was generated as the electrohydraulic system (EHC) attempted to control turbine speed and reactor pressure in response to a severe disturbance in the electric grid. All three division EDGs started and powered the associated emergency buses. Plant conditions were stabilized using RCIC and turbine bypass valves. An unusual event (UE) emergency classification was declared based on offsite grid instability. The UE was exited approximately 15 hours after grid stability had been established and off-site power had been restored to all three emergency buses.

The licensee event report (LER) was reviewed by the inspectors and no findings of significance were identified. This event did not constitute a violation of NRC requirements. This LER is closed.

2. (Closed) LER 50-220/2003-003, Automatic Initiation of Emergency Diesel Generator 103 due to Momentary Loss of Offsite Power.

On November 13, 2003, Unit 1 EDG 103 automatically initiated due to a momentary loss of offsite 115 kV power source line 4 in combination with an abnormal plant configuration due to maintenance. The apparent cause of the event was high wind conditions which resulted in a momentary loss of line 4, together with the abnormal configuration for a switchyard disconnect switch to allow for maintenance. All equipment operated as designed. The LER was reviewed by the inspectors and no findings of significance were identified. This event did not constitute a violation of NRC requirements. This LER is closed.

3. (Closed) LER 50-220/2004-001, Manual Reactor Scram and Cooldown Rate Exceeding TS Limits Due to Electromatic Relief Valve Failure to Close.

On May 2, 2004, while conducting post-maintenance testing following plant startup, ERV-123 would not close following remote manual actuation. Accordingly, operators manually scrammed the plant from 19 percent reactor power. Plant systems functioned as expected with the exception of the stuck open ERV. High pressure coolant injection initiated due to the low reactor water level from the level transient associated with the scram. Main steam isolation valves were closed to minimize the rate of reactor depressurization and the associated plant cooldown. Operators subsequently closed the manual block valve for ERV-123 which terminated the steam flow from the ERV.

The reactor cooldown rate exceeded the TS limit during the first hour following the scram. An engineering evaluation concluded that the event did not create conditions outside the design basis of the plant. The excessive cooldown rate constituted a violation of TS 3.2.1 which limits the reactor vessel cooldown rate to 100 degrees F in any one hour. The inspectors evaluated the violation in accordance with the guidance of IMC 0612, Appendix B, "Issue Screening." The finding was determined to be minor because it could not reasonably be viewed as a precursor to a significant event, if left uncorrected the finding would not become a more significant safety concern, the finding does not relate to performance indicators, and it does not affect the associated cornerstone (barrier integrity) objective of providing reasonable assurance that the physical design barrier will protect the public from radionuclide releases caused by accidents or events.

This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. Performance deficiencies associated with the ERV failure are documented in Section 1R20 of this report. The issue was entered into the corrective action program as DER NM-2004-2239. This LER is closed.

4. (Closed) LER 50-220/2003-004, Unplanned Inoperability of Emergency Cooling System Caused by Inadequate Review of Clearance for Replacement of Instrument Relay.

On August 13, 2003, maintenance was commenced to replace a Unit 1 emergency cooling (EC) system initiation logic time delay dropout relay. Electrical fuses that were removed to provide isolation for this work also supply power to one of two trip systems in each emergency cooling (EC) loop for the high steam flow isolation function. TS Table 3.6.2c indicates that the minimum number of operable EC isolation trip systems is two. The actions required by the TS table for less than two operable trip systems would render the EC system inoperable. Since the actions could not be completed, TS

3.6.2a(3) states that the system shall be considered inoperable. TS 3.1.3.e states that, if the EC system is inoperable, then a normal orderly shutdown shall be initiated within one hour.

This impact of the planned relay replacement on EC system operability had not been recognized at the time that the maintenance commenced. However, shortly thereafter, the on-shift operations crew questioned the validity of the previously developed assessment of the maintenance impact on plant operations. The Station Shift Supervisor directed that the activity be stopped and the system restored (fuses reinstalled) pending further investigation. It was subsequently determined that the high steam flow isolation trip system had been inoperable (deenergized) for approximately one hour and eight minutes, and that therefore the requirement of TS 3.1.3.e to initiate a shutdown within one hour had not been satisfied.

The inspectors evaluated the violation of TS 3.1.3.e in accordance with the guidance of IMC 0612, Appendix B, "Issue Screening." The finding was determined to be minor because it could not reasonably be viewed as a precursor to a significant event, if left uncorrected the finding would not become a more significant safety concern, the finding does not relate to performance indicators, and it does not affect the associated cornerstone (barrier integrity) objective of providing reasonable assurance that the physical design barrier will protect the public from radionuclide releases caused by accidents or events. Although inoperable per TS, the EC high steam flow isolation system remained functional throughout the event, in that the other trip system had remained energized, and only one trip signal is required to initiate an isolation.

This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The issue was entered into the licensee's corrective action program as DER NM-2003-3482. This LER is closed.

5. (Closed) LER 50-410/2004-001, Inoperability of a Control Room Envelope Filtration System Train Due to a Locked Closed Damper.

On January 28, 2004, a backdraft damper in the Unit 2 Control Room Outside Air Special Filter Train (Division II) was determined to be in the locked closed position causing the Division II Control Room Envelope Filtration (CREF) subsystem to be inoperable. Further investigation determined that the damper had been locked closed for approximately four and one-half months. TS 3.7.2 provides an allowed outage time of seven days for one CREF subsystem to be inoperable. The redundant CREF subsystem had also been inoperable on several occasions during that time, for periods of time that exceeded the maximum specified by TS 3.0.3.

The apparent cause of the event was a deficient blade lock for the damper, which allowed the blade lock to become inadvertently engaged. The monthly CREF surveillance procedure did not identify the change in air flow when the blade lock became engaged. The blade lock was subsequently modified to prevent inadvertent engagement of the blade lock. The CREF monthly surveillance procedure was revised to add guide values for acceptable flow and differential pressure conditions.

The inspectors evaluated the violation of TS 3.7.2 in accordance with the guidance of IMC 0612, Appendix B, "Issue Screening." The finding was determined to be minor because it could not reasonably be viewed as a precursor to a significant event, if left uncorrected the finding would not become a more significant safety concern, the finding does not relate to performance indicators, and it does not affect the associated cornerstone (mitigating systems) objective of protection against external factors.

This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The issue was entered into the licensee's corrective action program as DER NM-2004-406. No new findings were identified in the inspector's review. This LER is closed.

4OA4 Cross-Cutting Aspects of Findings

Section 1R20 describes a cross-cutting issue in human performance. Specifically, the inadequate use of procedures led to the improper assembly of an ERV.

4OA5 Other Activities

1. TI 2515/156, Offsite Power System Operational Readiness

Cornerstone: Initiating Events, Mitigating Systems

a. Inspection Scope

The inspector performed Temporary Instruction (TI) 2515/156, "Offsite Power System Operational Readiness." The inspector collected and reviewed information pertaining to the offsite power system specifically relating to the areas of the MR (10 CR 50.65), the station blackout rule (10 CFR 50.63), offsite power operability, and corrective actions. The inspector reviewed this data against the requirements of 10 CFR 50, Appendix A, General Design Criterion 17, "Electric Power Systems," and Unit 1 and 2 Technical Specifications. This information was forwarded to NRR for further review.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

On July 9, 2004, the inspectors presented the inspection results to Mr. J. Spina, Vice President Nine Mile Point, and other members of licensee management. The licensee acknowledged the findings and confirmed that proprietary information was not provided during the inspection.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV. Since this event does involve a locked high radiation area, in accordance with NEI 99-02, Revision 2, “Regulatory Assessment Performance Indicator Guideline,” the licensee has entered the information as a performance indicator in the Occupational Radiation Safety Cornerstone.

Unit 1 TS 6.7.2.a requires that areas having dose rates in excess of 1000 millirem per hour measured 30 centimeters from the source of radiation be posted, barricaded, locked and access controlled as a high radiation area. Access to, and the activities in, each such area shall be controlled by means of an RWP that includes specification of the radiation dose rates in the immediate work area. Contrary to the above, on April 29, 2004, during a scheduled maintenance outage at Unit 1, licensee personnel identified that the padlock used to secure one of three entry points to the drywell, a posted locked high radiation area with at least one area where the radiation levels measured at 30 centimeters from the source of radiation exceeded 1000 millirem per hour, was unlocked and unattended. No instances of workers gaining unauthorized entry to the drywell, nor any instances of workers receiving unplanned radiation exposures were identified during the period that the entry was unlocked. This event is documented in the licensee’s corrective action program as DER-NM-2004-2214. This finding is only of very low safety significance because it did not involve a very high radiation area or personnel overexposure.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

L. Hopkins, Plant General Manager
R. Godley, Manager, Operations
B. Holston, Manager, Engineering Services
G. Detter, Manager, Support Services
W. Paulhardt, Manager, Radiation Protection
J. Jones, Director, Emergency Preparedness
P. Doran, General Supervisor, System Engineering
G. Perkins, General Supervisor, Engineering Programs
C. Fisher, Maintenance Rule Coordinator
J. Gerber, ALARA Supervisor
T. Hogan, Radiation Protection Supervisor
T. Syrell, Nuclear Regulatory Matters
T. Kulczycky, Reliability Engineering
J. Raby, Engineering Programs
D. Williams, Engineering Programs
T. DeSanto, Radiation Specialist

NRC Personnel

W. Schmidt, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000220/2004003-01	NCV	Ineffective Corrective Action to Prevent Recurrence of Plant Equipment Obstruction by Scaffolding.
05000220/2004003-02	NCV	Failure to Follow ERV Maintenance Procedure Leads to ERV Failure to Close and Subsequent Scram.
05000410/2004003-03	NCV	Inadequate Secondary Containment Integrity Test Procedure.

Closed

05000410/2003002	LER	Reactor Scram Due to Electric Grid Disturbance
05000220/2003003	LER	Automatic Initiation of Emergency Diesel Generator 103 due to Momentary Loss of Offsite Power

05000220/2004001	LER	Manual Reactor Scram and Cooldown Rate Exceeding TS Limits Due to Electromatic Relief Valve Failure to Close
05000220/2003004	LER	Unplanned Inoperability of Emergency Cooling System Caused by Inadequate Review of Clearance for Replacement of Instrument Relay
05000410/2004001	LER	Inoperability of a Control Room Envelope Filtration System Train Due to a Locked Closed Damper

<u>Discussed</u>

NONE

LIST OF DOCUMENTS REVIEWED

<u>Section 1R12: Maintenance Effectiveness</u>

Periodic Assessment of the MR Program for Nine Mile Point, Units 1& 2 for the period January 2002 through September 2003
NMPNS MR Reliability Monitoring Report as of April 30, 2004
NMPNS MR Unavailability Reporting as of April 30, 2004
Nuclear Quality Assurance Report 04-055, MR Implementation
System Health Reports for selected Systems
Summary Report of DERs Initiated by MR Coordinators
Summary and eCap Report for Following DERs:
1-2003-336; 1-2003-825; 1-2003-1014; 2-2003-2023; 3-2003-2809; 1-2003-4745; 1-2004-873; 1-2004-1661; 1-2004-1692; 1-2004-1693; 1-2004-1696; 102004-1699; 1-2004-1745

LIST OF ACRONYMS

ADAMS	agencywide documents access and management system
ALARA	as low as is reasonably achievable
BOP	balance of plant
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
CRAT	control room air treatment
CRD	control rod drive
CREF	control room envelope filtration
CS	core spray
DER	deviation event report
EC	emergency cooling
EDG	emergency diesel generator
EHC	electrohydraulic system
ERV	electromatic relief valves

ESW	emergency service water
FSAR	final safety analysis report
GTS	gas treatment system
HP	health physics
HPCI	high pressure coolant injection
HPCS	high pressure core spray
IEL	initiating event likelihood
IMC	inspection manual chapter
IPE	individual plant evaluation
IR	inspection report
kV	kilovolt
LER	licensee event report
LERF	large early release frequency
LHRA	locked high radiation areas
LOCA	loss of coolant accident
MC	manual chapter
MR	maintenance rule
NCV	non-cited violation
NMPNS	Nine Mile Point Nuclear Station
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PARS	publically available records
PI&R	problem identification and resolution
PM	preventive maintenance
PMT	post-maintenance testing
RB	reactor building
RCA	radiologically controlled area
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RFO	refueling outage
RHR	residual heat removal
RPM	radiation protection manager
RRP	reactor recirculation pump
RV	reactor vessel
RWP	radiation work permit
SDBD	system design basis document
SDP	significance determination process
SGT	standby gas treatment
SR	surveillance requirement
SRV	safety/relief valve
SSCs	structures, systems, and components
SSR	shutdown safety report
TB	turbine building
TI	temporary instruction
TS	technical specification
UE	unusual event
USAR	updated safety analysis report
VHRA	very high radiation area
WO	work order