January 26, 2004

Mr. Peter E. Katz
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 05000220/2003006 and 05000410/2003006

Dear Mr. Katz:

On December 31, 2003, the US Nuclear Regulatory Commission (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 January 16, 2004, with Mr. L. Hopkins 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 two NRC-identified findings of very low safety significance (Green), both of which were determined to involve violations of NRC requirements. In addition, three licensee-identified violations which were determined to be of very low safety significance are listed in 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.

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by the order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year 2002, and the remaining inspection activities for Nine Mile Point were completed in May 2003. The NRC will continue to monitor overall safeguards and security controls at Nine Mile Point.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the 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 05000220/2003006 and 05000410/2003006
w/Attachment: Supplemental Information

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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.:	05000220/2003006 and 05000410/2003006
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:	September 28, 2003 - December 31, 2003
Inspectors:	G. Hunegs, Senior Resident Inspector B. Fuller, Resident Inspector E. Knutson, Resident Inspector S. Chaudhary, Reactor Inspector J. Noggle, Senior Health Physicist H. Williams, Senior Operations Engineer
Approved by:	James M. Trapp, Chief Projects Branch 1 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000220/2003-006, 05000410/2003-006; 09/28/2003 - 12/31/2003; Nine Mile Point, Units 1 and 2; Fire Protection, Operability Evaluations.

This report covered a 13-week period of inspection by resident inspectors and announced inspections by three region-based inspectors. Two 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: Mitigating Systems

- Green. The inspectors identified a Green non-cited violation (NCV) of Facility Operating License DPR-63, 2.D(7), Fire Protection, concerning a degraded fire seal for a 3-hour fire barrier that separates the diesel fire pump from the remainder of the screenhouse at Unit 1. The performance deficiency associated with this finding is failure to promptly identify a degraded fire seal for a pipe penetration. The finding is greater than minor because it is associated with the protection against the external factors attribute, and affects the mitigating systems cornerstone objective of ensuring the availability of systems that respond to initiating events. The finding is of very low safety significance in accordance with Phase 2 of the Fire Protection Significance Determination Process (SDP) because there is no realistic scenario by which a fire on one side of the barrier could propagate through the degraded seal to the other side of the barrier. The failure to identify the degraded fire seal is an example of a cross-cutting issue in problem identification and resolution. (Section 1R05)

- Green. The inspectors identified a Green non-cited violation (NCV) of 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action," for the failure to implement timely corrective actions to replace degraded control rod system components which resulted in several control rods failing to meet the Technical Specification (TS) five percent insertion time requirement. The performance deficiency associated with this finding is that appropriate corrective actions were not performed to replace degraded scram solenoid pilot valve diaphragms in a timely manner. This led to four control rods exceeding their TS five percent insertion time limit in October 2003. The finding is greater than minor, because it is associated with the equipment performance attribute of the mitigation system cornerstone and adversely affected the cornerstone objective of reliability. The finding is of very low safety significance because it is not a design or qualification deficiency, it did not represent a loss of safety function and was not potentially risk significant due to seismic, fire, flooding or weather related initiating events. The failure to implement timely corrective actions is an example of a cross-cutting issue in the area of problem identification and resolution. (Section 1R15)

Summary of Findings (cont'd)

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

- 10 CFR 71.5 requires NRC licensee's to comply with Department of Transportation (DOT) regulations 49 CFR 170-189. 49 CFR 173.441(b), limits radiation levels to the external surface of transportation packages to 200 mrem/hr unless shipped in a closed transport vehicle. Contrary to this, on April 29, 2003, a shipment containing two packages that were not transported in a closed transport vehicle was received at a radioactive waste processing vendor in Oak Ridge, Tennessee, with dose rates on the exterior bottom surface of one package of 280-300 mrem/hr. This event is documented in the licensee's corrective action program as DER 2003-4228. This finding is of very low safety significance because the only surface of the package greater than the limit was inaccessible and was less than 2 times the radiation limit, package integrity was not lost, and no contamination limit was exceeded.

- Technical Specification 3.1.5 states in part that, "during power operating condition whenever the reactor coolant pressure is greater than 110 psig . . . all six solenoid-actuated pressure relief valves shall be operable." Contrary to the above, on April 21, 2003, one of the solenoid-actuated pressure relief valves, ERV-111, would not operate following maintenance on its associated solenoid cut-out switch contacts. This was identified in the licensee's corrective action program as DER 2003-2017. This finding is of very low safety significance because the other five solenoid-operated pressure relief valves were operable.

- Technical Specification 3.3.1.1 requires the oscillation power range monitor instrumentation function to be operable during power operation. Contrary to the above, the instrumentation function was not operable for approximately 18 months, due to non-conservative settings. This was identified in the licensee's corrective action program as DER NM-2003-4149. Management review concluded that the violation was of low risk due to alternate methods available to detect and suppress thermal-hydraulic instability oscillations.

REPORT DETAILS

Summary of Plant Status

Nine Mile Point Unit 1 (Unit 1) began the inspection period at 100 percent power. On October 1, power was reduced to 90 percent for planned maintenance on the 12 reactor recirculation pump (RRP) motor generator. On October 8, a Technical Specification (TS) required plant shutdown was commenced due to leakage, that was in excess of the allowable, past a core spray system boundary valve. The leakage rate was subsequently determined to be acceptable and the shutdown was terminated at 95 percent power. On October 10, 17, and December 22 power was reduced to 95 percent for planned turbine valve testing. On November 13, an unplanned power reduction to 75 percent was performed due to loss of the 13 RRP which resulted from a brief power loss from off-site 115 KV line 4. On November 15, power was reduced to 65 percent for single control rod scram time testing and control rod drive hydraulic system maintenance. Based on the results of the scram time testing, the scope of maintenance was expanded and, as a result, the plant remained at 65 percent power until November 22. On November 23, power was reduced to 80 percent for a control rod pattern adjustment. On December 2, power was reduced to 95 percent to secure the 12 condensate pump for planned maintenance. 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 at 100 percent power. Reactor power was reduced to 55 percent on October 8, November 22 and December 5 for feedwater pump swaps and control rod manipulations. On December 21, power was reduced to 75 percent for planned maintenance on the condenser waterbox. Unit 2 operated at 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 2 Samples)

a. Inspection Scope

The inspectors examined two Unit 1 and two Unit 2 risk significant systems to verify that design features and operating procedures support operation of these systems during periods of cold weather. Unit 1 documents reviewed included the Unit 1 Final Safety Analysis Report (FSAR), the Service Water System Design Basis Document, the Unit 1 Individual Plant Examination for External Events, Unit 1 operating procedures N1-OP-18, “Service Water System,” N1-OP-21A, “Fire Protection System - Water,” and N1-OP-64, “Meteorological Monitoring.” This inspection activity represented two samples of the following systems.

- The service water system for possible susceptibility to extreme cold lake conditions, such as frazil ice intrusion.

- The fire water system for freeze protection/prevention measures to ensure the operability of outside fire suppression during periods of extreme cold weather.

Unit 2 documents reviewed included the Unit 2 Updated Safety Analysis Report (USAR), the Unit 2 Individual Plant Examination for External Events, Unit 2 operating procedures N2-OP-11, “Service Water System,” N2-OP-4, “Condensate Storage and Transfer,” and N2-OP-102, “Meteorological Monitoring.”

- The service water system for possible susceptibility to extreme cold lake conditions, such as frazil ice intrusion.
- The condensate storage and transfer system for freeze protection/prevention measures to ensure the operability of the source for emergency core cooling system (ECCS) and reactor core isolation cooling (RCIC) injection water during periods of extreme cold weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial System Walkdowns. (71111.04Q - 3 Samples) The inspectors performed partial system walkdowns to verify system and component alignment, and to note any discrepancies that would impact system operability. Partial system walkdowns of the following systems were completed:

- On October 14, the inspectors walked down the Unit 1 emergency diesel generator (EDG) 102 due to its increased risk significance during emergent maintenance on the EDG 103 governor control circuit. The walkdown included a physical inspection and switch verification. Operating procedure N1-OP-45, “Emergency Diesel Generators,” was used for this review.
- On November 5, the inspectors walked down the Unit 1 control rod drive system subsequent to extensive maintenance on the system. The walkdown included a physical inspection and switch verification. Operating procedure N1-OP-5, “Control Rod Drive System,” was used for this review.
- On November 17, the inspectors walked down the Unit 2 Division 2 EDG due to its increased risk significance while the Division 1 EDG was inoperable for planned maintenance. The walkdown included a physical inspection and switch verification. Operating procedure N2-OP-100A, “Standby Diesel Generators,” was used for this review.

Complete System Walkdown. (71111.04S - 1 Sample) The inspectors performed a complete system walkdown of the Unit 1 containment spray raw water system to verify that the system was properly aligned. The walkdown included reviews of valve positions, major system components, electrical power availability, and equipment deficiencies. The inspectors reviewed the system operating procedure, N1-OP-14, “Containment Spray System,” the system piping and instrumentation diagram, drawing number C-18012-C, the FSAR, and the System Design Basis Document. The inspectors also interviewed the system engineer to verify the bases of the current system configuration and historic system modifications.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 5 Samples)

a. Inspection Scope

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, 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 Unit 1 FSAR and the Unit 2 Updated Final Safety Analysis Report (UFSAR). This inspection activity reviewed the following areas:

- Diesel Fire Pump Room (Unit 1)
- Reactor Building 340 foot elevation (Unit 1)
- Reactor Building 237 foot elevation (Unit 1)
- Electric Fire Pump Room (Unit 2)
- Diesel Fire Pump Room (Unit 2)

b. Findings

Introduction. A Green finding was identified for failure to identify a degraded penetration seal in a fire barrier that separates Unit 1 equipment required for safe shutdown.

Description. On September 26, the inspectors identified that the fire seal for pipe penetration DP-20 was degraded due to circumferential cracking on both sides of the seal. Penetration DP-20 is in a 3-hour fire barrier that separates the diesel fire pump from the remainder of the screenhouse; both areas contain equipment that is required for safe shutdown of the reactor under certain plant fire scenarios. The seal is a 3-hour rated flamastic fire seal. The fire seal was declared inoperable and compensatory measures were established in accordance with FSAR, Appendix 10A, Section 2.4.1.10.b. This issue was entered in the licensee’s corrective action program as Deviation Event Report (DER) 2003-4113, and the fire seal was subsequently replaced.

Fire seal DP-20 had been inspected by the licensee on March 24, 2003, in accordance with N1-FST-FPP-C001, “Fire Barrier/Penetration Sealing Inspection.” This procedure specifies that a detailed inspection of 10 percent of the fire seals in the plant is to be performed during each operating cycle. This approach is consistent with the surveillance requirements of FSAR, Appendix 10A, Section 2.4.1.10.1. However, procedure N1-PM-S1, “Operator’s Rounds Guide,” Step 4.3.1, provides for a general inspection of fire barriers during the conduct of daily operator rounds (which include the diesel fire pump). Other periodic observations, such as daily fire door inspections by fire protection department personnel and plant tours by supervisory/management personnel provided additional opportunities for the licensee to have identified the degraded fire seal. The inspectors concluded that the licensee should have previously identified the degraded fire seal independent of the cycle frequency surveillance.

Analysis. The performance deficiency associated with this finding is failure to promptly identify a degraded penetration fire seal. The finding was greater than minor because it is associated with the protection against external factors attribute and affects the mitigating systems cornerstone objective of ensuring the availability of systems that respond to initiating events. A Significance Determination Process (SDP) Phase 1 screening directed that a Phase 2 analysis be performed. The finding was determined to be of very low safety significance (Green) in accordance with Phase 2 of the Fire Protection SDP because there is no realistic scenario by which a fire on one side of the barrier could propagate through the degraded seal to the other side of the barrier. The failure to identify a degraded penetration fire seal was an example of a cross-cutting issue in the area of problem identification and resolution.

Enforcement. Facility Operating License DPR-63, 2.D(7), Fire Protection, states that ... Nine Mile Point Unit 1 shall implement and maintain in effect all provisions of the approved Fire protection Program as described in the FSAR. The FSAR, Appendix 10A, Section 2.4.1.10, Fire Barriers/Penetrations, states that fire barrier penetrations are sealed to maintain the integrity of the barrier. Contrary to the above, on September 26, 2003, fire barrier penetration seal DP-20 was not maintained in that the seal was degraded. NCV 05000220/2003006-01, Degraded Penetration Fire Seal not Identified in a Timely Manner.

1R06 Flood Protection Measures (71111.06 - 1 Sample)

a. Inspection Scope

The inspectors examined the Unit 1 Screen House for its susceptibility to internal flooding. This area was selected based on its risk significance and because it contains multiple high capacity raw (lake) water systems. The inspection included a walkdown of the area to examine structure/system configurations and equipment material conditions. Documents reviewed during this inspection included the Unit 1 FSAR and the Unit 1 Individual Plant Examination.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

1) Resident Inspector Quarterly Review. (71111.11Q - 2 Samples)

The inspectors reviewed a 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 November 14, and Unit 1 on December 9, 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 reviewed simulator fidelity through a sampling process. The inspectors evaluated emergency response organization performance regarding initial and subsequent actions by licensed operators (See Section 1EP6).

2) Biennial Review by Regional Specialist. (71111.11B - 1 Sample)

The following inspection activities were performed using NUREG-1021, Rev. 8, “Operator Licensing Examination Standards for Power Reactors,” Inspection Procedure Attachment 71111.11, “Licensed Operator Requalification Program,” and NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process (SDP),” and 10 CFR55.46 ”Simulator Rule” as acceptance criteria.

The inspector reviewed documentation of operating history since the last requalification program inspection. Documents reviewed included NRC Plant Issue Matrix and a listing of licensee event reports (LER). The deviation event reports (DER) reviewed for possible training deficiencies and corrective actions are located under the List of Documents reviewed. The inspector ensured that operational events that were indicative of possible training deficiencies were captured in the training program.

A sample of the comprehensive written exams and operating tests given in 2002 and 2003 were reviewed. The sample of the written exams consisted of two reactor operator (RO) and two senior reactor operator (SRO) exams for weeks one and four for Units 1 and 2. The inspector observed the administration of the annual operating test for two operating crews during the week of December 1, 2003, for Unit 1. The quality of the written exams, the annual operating tests, and the administration and evaluation of the operating tests were reviewed to ensure they met the criteria of the Examination Standards and 10 CFR 55.59.

The inspector reviewed a sample of records related to requalification training attendance (Unit 1), remediation of failures and exam performance (Unit 2) and confirmed the operators were in compliance with license conditions and NRC regulations.

The inspector conducted an in-office review of licensee annual operating tests results for Unit 1 for 2003. The test results for Unit 2 were reviewed on site during the week of December 1, 2003. The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process (SDP).” The inspectors verified that:

- Crew failure rate was less than 20%. (Crew failure rate was 0% for Unit 1 and 11% for Unit 2.)
- Individual failure rate on the dynamic simulator test was less than or equal to 20%. (Individual failure rate was 0% for Unit 1 and 3.3% for Unit 2.)
- Individual failure rate on the walk-through test was less than or equal to 20%. (Individual failure rate was 0% for both units.)
- Individual failure rate on the comprehensive biennial written exam was less than or equal to 20%. (Individual failure rate was 0% for Unit 1 in 2002 and 13% for Unit 2 in 2003.)
- Overall pass rate among individuals for all portions of the exam was greater than or equal to 75%. (Overall pass rate was 100% for Unit 1 and 83% for Unit 2.)

This inspection activity represented one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 2 Samples)

a. Inspection Scope

The inspectors reviewed two performance-based problems during this inspection period involving selected 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. This inspection activity represented two samples of the following systems:

- Control room envelope after licensee reviews which were conducted in response to NRC Generic Letter 2003-01 "Control Room Habitability" identified potential unfiltered inleakage paths into the control room envelope (Unit 2)

- Emergency cooling system because of degraded system performance (Unit 1)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 3 Samples)

a. Inspection Scope

The inspectors reviewed three 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. The documents used for this review listed in the attachment to this report.

- WO 03-07382, Replace 11 control rod drive pump (Unit 1)

- Replacement of 115 KV bus sectionalizing disconnect switch MOD-8106 (Unit 1)

- Scram solenoid pilot valve (SSPV) diaphragm replacements (Unit 1)

b. Findings

No findings of significance were identified.

1R14 <u>Personnel Performance During Non-routine Plant Evolutions and Events</u> (71111.14 - 3 Samples)

a. <u>Inspection Scope</u>

The inspectors reviewed personnel performance for transient/non-routine operations. The inspectors compared operator response to that required by procedures and training and reviewed the plans for the evolutions. This inspection activity represented three samples.

- On November 20 and 21, the inspectors observed electric plant operations to support replacement of bus sectionalizing disconnect switch MOD-8106. This involved assuming busses 103/102 (one at a time) on EDGs 103/102 and divorcing them from off-site power, and de-energizing 115 KV line 4/1; restoration required a dead bus transfer of the busses back to off-site power (Unit 1).

- On November 20 and 21, the inspectors observed single rod scram time testing following SSPV diaphragm replacements, performed in accordance with N1-ST-R1, "Control Rod Scram Insertion Time Test" (Unit 1).

- On December 5, the inspectors observed a power reduction in preparation for swapping feed water pumps due to a control problem with the A-feed water pump level control valve, LCV-10A. This valve was blocked open, with the C-feed water pump level control valve, LCV-10C, controlling level. As power was reduced, the A-feed water pump block valve, MOV-47A, was used to manually throttle flow to maintain LCV-10C in the control band. This required removal of the seal-in closure feature on MOV-47A to allow it to be used as a throttle valve, and was performed in accordance with N2-OP-3, "Condensate and Feedwater System" (Unit 2).

b. <u>Findings</u>

No findings of significance were identified.

1R15 <u>Operability Evaluations</u> (71111.15 - 3 Samples)

a. <u>Inspection Scope</u>

The inspectors reviewed three 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 documents used for this review are located in the attachment to this report. The following operability evaluations were reviewed:

- On October 8, the inspectors evaluated the operability determinations for reactor coolant system leakage into the 12 core spray loop keepfill system. The first determination concluded the check valves were operable based on engineering judgement. Later ultrasonic measurement found the leakage rate to be 1.2 gallons per minute, which exceeded the TS limit for containment boundary valve leakage. A plant shutdown was commenced in parallel with efforts to directly measure the leakage rate. Direct measurement determined that the leakage was less than the TS limit, and the shutdown was stopped at 95 percent power. This issue was entered into the licensee's corrective action program as DERs 2003-4215 and 2003-4229 (Unit 1).

- On October 27, the inspectors reviewed an engineering evaluation supporting the operability determination for the degraded Unit 1 control rod drive scram solenoid pilot valves. The issue was entered into the licensee's corrective action program as DER 2003-4463 (Unit 1).

- On December 8, the inspectors reviewed the need for compensatory measures for 12 emergency cooling loop keepfill outboard check valve, CV 28.2-11, which failed its leakage rate test during the quarterly containment isolation valve surveillance N1-ST-Q5, "Primary Containment Isolation Valves Operability Test." This issue was entered into the licensee's corrective action program as DER 2003-4951(Unit 1).

b. Findings

Introduction. A Green non-cited violation (NCV) of 10 CFR 50 Appendix B Criterion XVI, "Corrective Action" was identified for the failure to implement timely corrective actions to replace degraded control rod system components which resulted in several control rods failing to meet their TS five percent insertion time requirement.

Description. On October 25, 2003, during a planned control rod scram time test for Unit 1, four control rods exceeded their TS five percent insertion time limit. The failures were attributed to premature aging of the scram solenoid pilot valve (SSPV) exhaust port diaphragm which is composed of Buna-N material. The Buna-N hardens when exposed to heat and air. As the diaphragm hardens, it becomes less flexible and reduces the rate at which the air can vent off the scram valve actuators which causes a delay in the start of control rod motion during a scram.

The Buna-N material was installed during the 1996-1997 time frame. On April 13, 2003, during the refueling outage, all control rods were scram time tested. Control rod 38-19 showed a response time of 0.395 seconds, with a TS allowed value of 0.398 seconds. Six months later, the same control rod and three others that also had response times close to the limit, failed the test.

The diaphragm vendor, General Electric (GE), identified the same problem in the past at several facilities. The Buna-N used in fabricating solenoid valves manufactured by Automatic Switch Company (ASCO), was originally designed to last for ten years; but after vendor analysis this was subsequently changed to seven years. Based on industry information provided to you prior to the April 2003 testing, there was sufficient opportunity to replace the diaphragms, which were close to the TS scram time, prior to failure. Based on industry information concerning the diaphragm degradation, the NRC also published Information Notices (INs) 1994-71 and 2003-17 to alert the industry of potential problems regarding this issue. The NRC concluded that there was sufficient information to reach the conclusion that the SSPV diaphragms were susceptible to the same degradation mechanism and that adequate corrective action was not taken to prevent their subsequent failure.

<u>Analysis</u>. The performance deficiency associated with this event is that appropriate corrective actions were not performed to replace degraded SSPV diaphragms in a timely manner. This led to four control rods exceeding their TS five percent insertion time limit in October, 2003. The finding is greater than minor, because it is associated with the equipment performance attribute of the mitigation system cornerstone and adversely affected the cornerstone objective of reliability. Using Phase I of the Reactor Safety SDP, the finding is determined to be of very low safety significance, (Green), because it is not a design or qualification deficiency, it did not represent a loss of safety function and was not potentially risk significant due to seismic, fire, flooding, or weather related initiating event. The failure to implement timely corrective actions is an example of a cross-cutting issue in problem identification and resolution.

<u>Enforcement</u>. 10 CFR 50 Appendix B, Criterion XVI, “Corrective Actions” requires that the licensee establish measures to assure that conditions adverse to quality, such as failures, malfunctions and deficiencies, are promptly identified and corrected.

Contrary to the above, in April 2003, the licensee did not correct a condition adverse to quality in that they did not correct a degraded condition which caused four control rods to exceed the five percent scram insertion time during a subsequent test. However, because of the very low safety significance and because the licensee entered this issue into their corrective action program as DER 2004-082, this issue is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. NCV 05000220/2003006-02, Untimely Corrective Action Resulted in the Failure of Control Rods to Meet the Five Percent Scram Insertion Time.

1R16 Operator Workarounds (71111.16 - 1 Sample)

a. Inspection Scope

The inspectors 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 (1) the 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. NAI-REL-02, Workaround Programs, was referenced for this review. Additionally, the inspectors looked for any combined effects of the operator workarounds.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 6 Samples)

a. Inspection Scope

The inspectors reviewed post-maintenance testing (PMT) procedures and associated testing activities for six 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. This inspection activity represented six samples of the following systems:

- On October 22, the inspectors observed N1-ST-Q1C, “CS 112 Pump and Valve Operability Test,” performed as PMT for multiple work items that were performed during the three day LCO maintenance period for 112 core spray (Unit 1)

- On October 29, the inspectors reviewed N1-ST-R1, “Control Rod Scram Insertion Time,” for work completed under WO-03-04663, scram solenoid pilot valve replacement (Unit 1)

- On November 5, the inspectors observed N1-OP-5, Control Rod Drive System, performed as PMT after the 11 CRD pump was rebuilt (Unit 1)

- On November 6, the inspectors observed PMT for WO-03-14540, “Division II EDG Jacket Water Heater Repair” (Unit 2)

- On November 14, inspectors observed N2-OSP-ICS-Q001, “Reactor Core Isolation Cooling Valve Operability Test,” performed as PMT after adjustment of packing on steam supply valve, MOV-121 (Unit 2)

- On November 19, the inspectors observed EDG 103 operation per N1-ST-M4B, “Emergency Diesel Generator 103 and PB 103 Operability Test,” performed as PMT to verify output breaker operation following restoration of a clearance on that breaker (Unit 1).

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 4 Samples)

a. Inspection Scope

The inspectors witnessed performance of four surveillance test procedures and reviewed test data of selected risk significant SSC’s to assess whether the SSC’s satisfied TS, 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. This inspection activity represented four samples of the following systems:

- On November 24, N1-ISP-002-002, “Turbine Anticipatory Trip - Low Oil Pressure Instrument Channel Calibration/Test” (Unit 1)

- On October 21, N2-OSP-EGS-M@001, “Division I Emergency Diesel Generator” (Unit 2)

- On November 20, N2-OSP-RHS-Q@004, “Residual Heat Removal “A” Loop” (Unit 2)

- On December 11, N2-OSP-ICS-Q@002, “Reactor Core Isolation Cooling” (Unit 2)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 2 Samples)

a. Inspection Scope

The inspectors reviewed two temporary plant modifications to determine whether the temporary changes 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 also assessed configuration control of the temporary changes by reviewing selected drawings and procedures (N1-OP-33A, 115kv System; NMPC dwg C-19408-C, One Line Diagram Main and Secondary Connections; N2-ARP-01, Control Room Alarm Response Procedures) to verify whether appropriate updates had been made. The inspectors compared the actual installations to the temporary modification documents to determine whether the implemented changes were consistent with the approved documented modification. The inspectors reviewed the post-installation test results to verify whether the actual impact of the temporary changes had been adequately demonstrated by the test. This inspection activity represented two samples of the following temporary modifications.

- The main turbine high vibration protective trip was disabled due to a power supply failure (Unit 2).
- Work associated with the replacement of bus sectionalizing disconnect switch MOD-8106; specifically, removal of a portion of the bus work to 115 KV line 4 and subsequent reinstallation using a new crimping technique (Unit 1).

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06 - 2 Samples)

a. Inspection Scope

The inspectors reviewed the operators emergency classification and notification completed during requalification training on November 14 (Unit 2), and December 9 (Unit 1) (See Section 1R11). The inspectors evaluated the results against EPIP-EPP-01 Classification of Emergency Conditions at Unit 1, and EPIP-EPP-02 Classification of Emergency Conditions at Unit 2.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety and Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 1 Sample)

a. Inspection Scope

During the period November 19 - 20, 2003, the inspector conducted the following activities to verify that the licensee was properly implementing physical, engineering, and administrative controls for access to locked high radiation areas, and other radiologically controlled areas, and that workers were adhering to these controls when working in these areas. Implementation of the access control program was reviewed against the criteria contained in 10 CFR 20, site TSs, and the licensee's procedures.

The inspector attended two pre-job High Risk Activity briefings, reviewed the exposure controls specified in the radiation work permit (RWP) and the associated as low as is reasonably achievable (ALARA) Review, and observed radiation worker and radiation protection technician performance during the following locked high radiation area work activity. The inspector interviewed workers associated with this work activity regarding their knowledge of the RWP, electronic dosimetry set points, the work area radiological conditions, and the individual's assigned task. This inspection activity represented one sample relative to this inspection area, completing the annual inspection requirement.

- Unit 1 Clean-Up Filter Sludge Pump maintenance: initial entry and source term removal (RWP 103120 and associated High Risk Activity Plans).

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

During the period November 17 - 21, 2003, the inspector conducted the following activities to verify that the licensee was properly implementing operational, engineering, and administrative controls to maintain personnel exposure as low as is reasonably achievable (ALARA) for tasks conducted during the Spring 2003 Unit 1 refueling outage. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and the licensee's procedures.

The inspector reviewed the 1R17 Radiation Work Permit Dose Summary Reports, detailing the worker estimated and actual exposures for work activities performed during the refueling outage. The inspector evaluated the exposure mitigation requirements, specified in ALARA Reviews (AR), and compared actual worker cumulative exposure to estimated dose for tasks associated with these work activities. This inspection activity represented completion of two samples relative to this inspection area, completing the biennial inspection requirement.

- Drywell main steam isolation valve modifications, RWP 103543, AR No. 2003-39.
- Emergency condenser return check valve No. 39-04 overhaul, RWP 103533, AR No. 2003-30.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation (71122.02 - 21 Samples)

a. Inspection Scope

The Unit 1 and Unit 2 liquid and solid radwaste processing plant equipment spaces were walked down and reviewed with respect to radwaste processing design and abandoned radwaste processing equipment descriptions in the Updated Final Safety Analysis Report (UFSAR) Sections 11.2 and 11.4 and the Process Control Program (PCP). Any radwaste processing changes since the previous inspection in this area were reviewed with respect to 10 CFR 50.59 evaluations. During the solid radwaste processing system walkdown, the processes for transferring radwaste into shipping containers were reviewed to ensure appropriate sampling and waste characterization of radwaste shipments.

The most recent radio-chemical radioactive waste stream analyses were reviewed for appropriate use in classifying waste shipments for transport in accordance with 10 CFR 61.55, which included: dry active waste, bead resin, filter sludge and powdered resin wastes specific for Units 1 and 2. Program processes to ensure continued validity of the 10 CFR 61.55 samples during plant operation changes since the previous inspection in this area were also reviewed with respect to Branch Technical Position guidelines.

On October 7, 2003, the inspectors observed a condensate demineralizer bead resin shipment (No. 03-2050), that was prepared for shipment, surveyed, and shipped off site. On October 8, 2003, the inspectors observed the packaging of a bead resin liner into a shipping cask in preparation for future shipment. These activities were reviewed with respect to licensee procedures, 10 CFR Parts 61, 71, and 49 CFR Parts 170-189 requirements.

The inspector reviewed the following nine radioactive shipment records for compliance with licensee radwaste shipping procedures and federal regulations in 10 CFR Parts 20, 61, and 71 and 49 CFR Parts 170-189.

- Shipment No. 02-1025, Unit 1 powdered resin, shipped July 16, 2002
- Shipment No. 03-1014, Unit 1 bead resin, shipped February 14, 2003
- Shipment No. 03-1030, Unit 1 control rod drives, shipped April 1, 2003
- Shipment No. 03-1086, Unit 1 recirculation pump motor, shipped May 6, 2003
- Shipment No. 03-1065, Unit 1 clean-up resin, shipped April 2, 2003
- Shipment No. 2WS-2153, Unit 2 bead resins, shipped June 4, 2002
- Shipment No. 2WS-2172, Unit 2 dry active waste, shipped August 2, 2002
- Shipment No. 2WS-2173, Unit 2 resin and sludge, shipped October 23, 2002
- Shipment No. 03-2050, Unit 2 bead resins, shipped October 7, 2003

The licensee's oversight of the radwaste transportation program was reviewed during the previous two years which consisted of a Quality Assurance audit of the radioactive material shipping program conducted in November 2001. The criteria used for this review were the audit requirements specified in 10 CFR 71.137 and 10 CFR 20.1101(c). This inspection activity represented 21 samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Annual Inspection. (71151 - 19 samples) The inspectors sampled licensee submittals for the performance indicators (PI's) listed below for the period from September 2002 through September 2003. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Rev. 2, were used to verify the basis in reporting for each data element. Data for Unit 1 and Unit 2 was reviewed.

Reactor Safety Cornerstone

- Unplanned Scrams per 7,000 Critical Hours PI
- Scrams with a Loss of Normal Heat Removal PI
- Unplanned Power Changes per 7000 Critical Hours PI

The inspector reviewed a selection of LERs, portions of Unit 1 and Unit 2 operator log entries, daily morning status reports (including the daily DER descriptions), the monthly operating reports, monthly maintenance rule reports and PI data sheets to determine whether the licensee adequately identified the number of scrams and unplanned power changes greater than 20 percent that occurred during the previous four quarters. This number was compared to the number reported for the PI during the current quarter. The inspectors also verified the accuracy of the number of critical hours reported and the licensee's basis for crediting normal heat removal capability for each of the reported reactor scrams. In addition, the inspectors also interviewed licensee personnel associated with the PI data collection, evaluation, and distribution.

- Safety System Unavailability - Emergency AC Power System PI
- Safety System Unavailability - High Pressure Injection System PI
- Safety System Unavailability - Heat Removal System PI (Unit 2 only)
- Safety System Unavailability - Residual Heat Removal System PI
- Safety System Functional Failures PI

The inspector reviewed a selection of LER's, portions of Unit 1 and Unit 2 operator log entries, daily morning status reports (including the daily DER descriptions), the monthly operating reports, monthly maintenance rule reports and PI data sheets to determine whether the licensee adequately identified safety system unavailability and functional failures. In addition, the inspectors also interviewed licensee personnel associated with the PI data collection, evaluation, and distribution.

- Reactor Coolant System Activity PI
- Reactor Coolant System Leakage PI

The inspector reviewed portions of Unit 1 and Unit 2 operator log entries, daily morning status reports (including the daily DER descriptions), and PI data sheets to determine whether the licensee accurately reported reactor coolant system activity and identified leak rate. In addition, the inspectors also interviewed licensee personnel associated with the PI data collection, evaluation, and distribution. This inspection activity represented 19 samples relative to this inspection area, completing the annual inspection requirements.

<u>Occupational Exposure Control Effectiveness</u>. (71151 - 1 Sample) The inspector reviewed implementation of the licensee's Occupational Exposure Control Effectiveness Performance Indicator (PI) Program. Specifically, the inspector reviewed DERs, radiologically controlled area (RCA) dosimeter exit logs, and internal and external dose evaluation records for the past four (4) calendar quarters. These records were reviewed for occurrences involving locked high radiation areas, very high radiation areas, and unplanned exposures against the criteria specified in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2, to verify that all occurrences that met the NEI criteria were identified and reported as performance indicators. This inspection activity represented one sample relative to this inspection area, completing the annual inspection requirement.

<u>RETS/ODCM Radiological Effluent Occurrences</u>. (71151 - 1 Sample) The inspector reviewed a listing of relevant effluent release reports for the past four calendar quarters, for issues related to the public radiation safety performance indicator, which measures radiological effluent release occurrences per site that exceed 1.5 mrem/qtr whole body

or 5.0 mrem/qtr organ dose for liquid effluents; 5 mrads/qtr gamma air dose, 10 mrad/qtr beta air dose, and 7.5 mrads/qtr for organ dose for gaseous effluents. This inspection activity represents the completion of one sample relative to this inspection area, completing the annual inspection requirement.

The inspector reviewed the following documents to ensure the licensee met all requirements of the performance indicator from the fourth quarter 2002 through the third quarter 2003:

- Monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases
- Quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases
- Dose assessment procedures

b. Findings

No findings of significance were identified.

40A2 Identification and Resolution of Problems (71152)

a. Inspection Scope

1) Problem Identification and Resolution 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 licensee’s corrective action program. This review was accomplished by reviewing hard copies of each condition report.

2) Radioactive Material Processing and Transportation

The inspector reviewed eleven Deviation Event Reports (DERs) relating to the processing and shipping of radioactive material between November 2001 and September 2003 to evaluate the licensee’s threshold for identifying and resolving problems in implementing the radioactive material transportation program.

The condition reports were evaluated against the criteria contained in the PCP, 10 CFR Parts 20, 61, and 71 and 49 CFR Parts 170-189.

3) ALARA Planning and Controls

The inspector reviewed two DERs, relating to maintaining personnel exposure ALARA, to evaluate the threshold for identifying, evaluating, and resolving problems in implementing the ALARA program. The selected DERs (NM-2003-1860 and NM-2003-1634) were preventable and resulted in only minor additional dose during the Spring 2003 refueling outage. This review was conducted against the criteria contained in 10 CFR 20, TSs, and the licensee's procedures.

4) Control Rod Drive System Performance Problems

The control rod drive (CRD) system had experienced several equipment performance problems resulting in system unavailability. The CRD system is a source of high pressure coolant injection which is credited for during a small loss of coolant accident. The inspector selected three DERs (DER-NM-2003-3396, DER-NM-2003-1760, and DER-NM-2002-4570) related to malfunctions of Number 11 and 12 CRD pumps. The inspector selected the above DERs for a detailed review for assessing the definition of the problem, technical adequacy of the resolution and the effectiveness of corrective action. The inspector observed that a detailed analysis was performed to determine the root cause of the pump failures and the CRD system problems. The proposed immediate, midterm, and the long term corrective actions were adequate, and appeared technically valid. The maintenance rule evaluation of the operability and availability was sound.

5) Cross References to Findings Documented Elsewhere

- Section 1 R05 describes a performance deficiency that was a contributing cause to a finding associated with the failure to identify a degraded fire penetration seal.
- Section 1R15 describes a performance deficiency that was a contributing cause to a finding associated with untimely corrective action. Specifically, SSPVs were not repaired in a timely manner which resulted in several control rods failing TS criteria for five percent scram insertion time. The licensee had sufficient information concerning SSPV degradation which should have lead to their decision to replace the diaphragm material prior to the control rod failures.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

1. (Closed) LER 50-220/2003-001, TS Cooldown Rate Exceeded During Required Cooldown for a Failed Solenoid Actuated Pressure Relief Valve

Solenoid Actuated Pressure Relief Valve ERV-111 Failure to Open. On April 21, Unit 1 was starting up from a refueling outage. With reactor power at approximately 23 percent, an operating cycle surveillance to manually open each of the six solenoid-actuated pressure relief valves (ERVs) was performed as required by TS 4.1.5.a. One of the valves, ERV-111, failed to open during this test. TS 3.1.5.a requires that all six solenoid-actuated pressure relief valves be operable whenever reactor coolant pressure is greater than 110 psig. The reactor was subsequently shut down and returned to the cold shutdown condition.

Unit 1 determined that ERV-111 failed to open due to high resistance in the cut-out switch contacts for the associated solenoid operated valve, (SOV)-01-102A. This condition limited coil current and prevented the SOV from operating. Unit 1 determined that the cause of the event was an inadequate preventive maintenance procedure that had been performed on ERV-111 during the refueling outage. The procedure required cleaning of the affected contacts, but did not specify subsequent measurement of contact resistance. Corrective action included replacement of SOV-01-102A, measurement of cut-out switch contact resistance for the remaining five solenoid-actuated pressure relief valves, and revision of the preventive maintenance procedure to include contact resistance measurement.

This finding was considered more than minor because it was associated with the Procedure Quality attribute of the Mitigation Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was evaluated in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” using Phase 1, Phase 2, and Phase 3 significance determination process (SDP) analysis.

This issue was of very low safety significance, based on the Phase 3 analysis results assuming that ERV-111 would not have opened from the control room for one year. The Phase 1 analysis required a Phase 2 evaluation because the finding represented an actual loss of a safety function of a single ERV train (one of six valves) for a year. The Phase 2 analysis required two of six valves to function for success for normal manual depressurization and two of six valves for an anticipated transient without scram (ATWS) manual depressurization. The Phase 2 analysis produced overly conservative results, because there were still five remaining ERVs that could have functioned as needed.

To address the conservatism, the Region I Senior Risk Analyst (SRA) conducted a Phase 3 evaluation, using the 3.01 SPAR model for Nine Mile Point Unit 1.

It was assumed, based on information from the inspectors and LER 2003-001, that testing was sufficient to confirm that the other five valves were not affected by the same condition that caused the failure of ERV-111. This analysis produced an internal delta-core damage frequency (CDF) increase in the range of very low safety significance. The dominant core damage sequence was a loss of instrument air (LOIA) event directly resulting in loss of the ability to vent the containment and to operate suppression pool cooling. The LOIA was further compounded by the potential that operators do not properly control feedwater resulting in the loss of isolation condenser and the power conversation system (PCS) due to high reactor vessel level. This analysis did not credit the automatic feedwater reactor vessel level setdown which limits that potential for a high reactor vessel level and the chance that operators could recover the isolation condensers or PCS prior to core damage. In accordance with NRC Inspection Manual Chapter (IMC) 609A, the Phase 3 analysis included additional assessments of delta-CDF for external events (fire and seismic) and delta-large early release frequence (delta-LERF) for both internal and external events. The SRA reviewed the licensee's analysis which included internal and external delta-CDF and delta-LERF evaluations and considered that the external delta CDF contribution would not cause the total delta CDF to increase above very low safety significance. Further, the delta-LERF increase, based on Phase 3 analysis, resulted in a very low safety significance, for both internal and external events.

This licensee-identified finding involved a violation of TS 3.1.5, Solenoid Actuated Pressure Relief Valves. The enforcement aspects of the violation are discussed in Section 4OA7.

<u>Maximum Allowed Cooldown Rate Exceeded During Plant Cooldown</u>. During plant cooldown following the April 21, 2003 shutdown, the TS maximum allowable cooldown rate of 100 degrees Fahrenheit (°F) per hour was exceeded by approximately one degree F for a period of approximately three minutes. The cooldown rate was subsequently reduced to less than 100 degrees F per hour by securing auxiliary steam loads. The licensee determined that the cause of the event was procedural inadequacy, in that the procedure did not provide sufficient guidance for controlling the plant cooldown rate under conditions of low decay heat. Corrective action included revision of the procedure to include additional guidance for controlling cooldown rate and operator training on the event.

The excessive cooldown rate constituted a violation of NRC requirements, and the inspectors evaluated it 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 radio nuclide 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.

The issue was entered into the corrective action program as DER 2003-2021. This LER is closed.

2.0 (Closed) Violation 50-220/2003-03-01, Failure to Determine the Cause of a Significant Condition Adverse to Quality and Implement Corrective Action to Prevent Repetition, Associated with Severe Corrosion of the Reactor Building Closed Loop Cooling (RBCLC) System.

NRC Special Inspection Report 50-220/03-003 reviewed the degraded condition of the RBCLC system and identified a finding which involved inadequate implementation of corrective actions for degraded piping in the RBCLC system. On May 23, 2003, the NRC issued a letter documenting the final significance determination for a White finding and notice of violation (EA-03-053). The inspector reviewed the licensee's reply to the Notice of Violation dated June 23, 2003. The licensee's response was determined to be acceptable. This violation is closed.

3.0 (Closed) LER 50-410/2003-003 and Supplement 1, Oscillation Power Range Monitor Inoperable Due to Non-conservative Settings for Adjustable Parameters

On October 2, 2003, the licensee received a Part 21 notification from General Electric (GE) that the oscillation power range monitor (OPRM) may not prevent exceeding the safety limit minimum critical power ratio for all anticipated instability events. The OPRM at Unit 2 had non-conservative settings for the adjustable period confirmation variables (period tolerance and cutoff frequency). The OPRM would have been inoperable since activation of the trip function in April 2000. This would exceed the action statement of TS 3.3.1.1, Reactor Protection System Instrumentation.

Based on their analysis of the July 24, 2003, Unit 2 scram event, GE concluded that the adjustable period confirmation variables did not adequately filter out high frequency noise, creating a signal that caused frequent confirmation count resets.

The licensee determined that the apparent cause of this event was failure to ensure, through proper qualification and testing, that the OPRM system is capable of performing its expected trip function during an instability event. Corrective action was to change the conditioning filter cutoff frequency parameter and the period tolerance parameter to values recommended by GE.

The inspectors reviewed this LER and no findings of significance were identified since it was not the result of a licensee performance deficiency and therefore not evaluated as a potential finding. However, the event constituted a violation of TSs and is being dispositioned as a licensee-identified violation (see section 4OA7). This LER is closed.

4OA6 Meetings, Including Exit

On January 16, 2004, the inspectors presented the inspection results to Mr. L. Hopkins, Plant General Manager, 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 violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as Non-Cited Violations (NCVs).

a. 10 CFR 71.5 requires NRC licensee's to comply with US Department of Transportation (DOT) regulations 49 CFR 170-189. 49CFR173.441(b), limits radiation levels on the external surface of transportation packages to 200 mrem/hr unless shipped in a closed transport vehicle. Contrary to this, on April 29, 2003, a shipment containing two packages that were not transported in a closed transport vehicle were received at a radioactive waste processing vendor in Oak Ridge, Tennessee, with dose rates on the exterior bottom surface of one package of 280-300 mrem/hr. This event is documented in the licensee's corrective action program as DER NM-2003-4228. This finding is of very low safety significance because the only surface of the package greater than the limit was inaccessible and was less than 2 times the radiation limit, package integrity was not lost, and no contamination limit was exceeded.

b. TS 3.1.5.a states, in part, "during power operating condition whenever the reactor coolant pressure is greater than 110 psig . . . all six solenoid-actuated pressure relief valves shall be operable." Contrary to the above, on April 21, 2003, one of the solenoid-actuated pressure relief valves, ERV-111, would not operate following maintenance on its associated solenoid cut-out switch contacts. This was identified in the licensee's corrective action program as DER 2003-2017. This finding is of very low safety significance because the other five solenoid-operated pressure relief valves were operable.

c. TS 3.3.1.1 requires the OPRM instrumentation function to be operable during power operation. Contrary to the above, the instrumentation function was not operable for approximately 18 months, due to non-conservative settings.

This was identified in the licensee's corrective action program as DER NM-2003-4149. Management review concluded that the violation was of low risk due to alternate methods available to detect and suppress thermal-hydraulic instability oscillations.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

G. Detter, Manager, Support Services
B. Holston, Manager, Engineering Services
L. Hopkins, Plant General Manager
J. Jones, Supervisor, Emergency Preparedness
M. Navin, Manager, Site Operations
W. Paulhardt, Radiation Protection Manager
A. Shiever, Manager, Nuclear Training

NRC Personnel

W. Schmidt, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000220/2003006-01	NCV	Degraded Penetration Fire Seal not Identified in a Timely Manner.
05000220/2003006-02	NCV	Untimely Corrective Action Resulted in the Failure of Control Rods to Meet the Five Percent Scram Insertion Time.

Closed

05000220/2003003-01	VIO	Failure to Determine the Cause of a Significant Condition Adverse to Quality and Implement Corrective Action to Prevent Repetition, Associated with Severe Corrosion of the Reactor Building Closed Loop Cooling (RBCLC) System.
05000220/2003-001	LER	TS Cooldown Rate Exceeded During Required Cooldown for a Failed Solenoid Actuated Pressure Relief Valve
05000410/2003-003 and Supp. 1	LER	Oscillation Power Range Monitor Inoperable Due to Non-conservative Settings for Adjustable Parameters

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R11: Licensed Operator Requalification Program

NM-2002-1311	Disposition for DER 1-2001-4018 is Inadequate
NM-2002-2379	Differences between simulator and plant response for shutdown training
NM-2002-3957	Unexpected simulator results during 2002 WANO Peer Review
NM-2002-4529	Post Exam Analysis of the Unit 1 NRC Written Exam
NM-2003-614	Simulator failure during just-in-time training for Unit 1 startup
NM-2003-2353	Negative training on simulator due to inaccurate modeling of condensate/feedwater system
NM-2003-3491	Crew failure of simulator evaluation during cycle 3

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

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

Section 1R15: Operability Evaluations

NM-2003-4464, control rod scram time failure for Unit 1
Engineering Safety Analysis (ESA) for the scram time failure NM-2003-4464
NIP-ECA-01, Deviation/Event Reports
GAP-OPS-02, Administration of Operations, Revision 19
S-ODP-OPS-0116, Operability Determinations

Section 4OA2: Deviation/Event Reports

NM-2003-3010	NM-2002-784	NM-2002-1973	NM-2002-3168
NM-2002-3568	NM-2002-4547	NM-2002-4862	NM-2003-356
NM-2003-753	NM-2003-2550	NM-2003-2623	

Section 4OA2: Control Rod Drive System Review

DER-NM-2003-3396, DER NM-2003-1760, DER-NM-2002-4570
Repair specification for Worthington Model 2WT810
CRD Pump Trend Data for Pump 12 PMP-28-17 and 11 PMP-28-15
Flowserve Pump Specification for centrifugal and axial pumps
Unit 1 CRD Pump Recommendations
Niagara Mohawk Drawing No. F-45128-C

LIST OF ACRONYMS

ALARA	As low as is reasonable achievable
AR	ALARA reviews
CFR	Code of Federal Regulations
CRD	control rod drive
DERs	deviation event reports
DOT	US Department of Transportation
EDG	emergency diesel generator
ERV	solenoid-actuated pressure relief valves
FSAR	final safety analysis report
GE	General Electric
KV	kilovolt
LER	licensee event report
NCV	non-cited violation
NEI	Nuclear Energy Institute
NMP1	Nine Mile Point Unit 1
NRC	U.S. Nuclear Regulatory Commission
OPRM	oscillation power range monitor
PCP	process control program
PI	performance indicator
PMT	post-maintenance testing
RBCLC	reactor building closed loop cooling
RCA	radiologically controlled area
RCIC	reactor core isolation cooling
RO	reactor operator
SDP	significance determination process
SOV	solenoid operated valve
SRA	senior risk assessment
SSCs	structures, systems, and components
SSPV	scram solenoid pilot valve
TS	technical specifications
UFSAR	updated final safety analysis report