January 26, 2005

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 05000220/2004005 and 05000410/2004005

Dear Mr. Spina:

On December 31, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Nine Mile Point Nuclear Station (NMPNS), Units 1 and 2. The enclosed integrated inspection report (IR) documents the inspection findings which were discussed on January 14, 2005, with Mr. Tim O'Connor 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 and one self-revealing findings of very low safety significance (Green). Two of the findings were determined to involve violations of NRC requirements. 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 IR, with the basis for your denial, to the U.S. 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, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Nine Mile Point.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure 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

Mr. James A. Spina

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

Enclosure: Inspection Report 05000220/2004005 and 05000410/2004005
w/Attachment: Supplemental Information

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REGION I

Docket Nos.:	50-220, 50-410
License Nos.:	DPR-63, NPF-69
Report No.	05000220/2004005 and 05000410/2004005
Licensee:	Nine Mile Point Nuclear Station, LLC (NMPNS)
Facility:	Nine Mile Point, Units 1 and 2
Location:	348 Lake Road Oswego, NY 13126
Dates:	October 1, 2004 - December 31, 2004
Inspectors:	G. Hunegs, Senior Resident Inspector B. Fuller, Resident Inspector E. Knutson, Resident Inspector S. Barr, Senior Operations Engineer M. Brown, Operations Engineer J. Furia, Senior Health Physicist K. Kolaczyk, Senior Resident Inspector S. McCarver, Reactor Inspector N. Perry, Senior Project Engineer D. Silk, Senior Emergency Preparedness Inspector L. Vick, NRR (Observer)
Approved by:	James M. Trapp, Chief Projects Branch 1 Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000220/2004-005, 05000410/2004-005; 10/01/2004 - 12/31/2004; Nine Mile Point, Units 1 and 2; Licensed Operator Requalification Program and Operability Evaluations.

This report covered a 13-week period of inspection by resident inspectors, and announced inspections and two in-office reviews, by six region and one headquarters 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: Mitigating Systems

- Green. A finding of very low safety significance was identified at Unit 2. The finding was associated with crew performance on the simulator during facility-administered requalification examinations. Of the nine crews evaluated, three failed to pass their simulator examinations.

  The finding is more than minor because it reflected the potential inability of the crews to take appropriate safety-related actions in response to actual abnormal or emergency conditions. The finding is of very low safety significance because the failures occurred during annual testing of the operators on the simulator, because there were no actual consequences to the failures, and because the crews were removed from watch standing duties, retrained and re-evaluated before they were authorized to return to control room watches. (Section 1R11)

- Green. An NRC identified finding for failure of the NMP Unit 1 and Unit 2 simulators to comply with 10 CFR 55.46(c)(1), ”Plant-referenced simulators.” The NCV involved two examples of the failure of Nine Mile Point simulators to correctly demonstrate the expected plant response to two separate events, one at each NMP unit.

  This finding is more than minor because it affects the human performance (human error) attribute of the Mitigating Systems Cornerstone. The finding is of very low safety significance (Green) because the simulators’ uncorrected model discrepancies did not have an adverse impact on operator actions such that safety-related equipment was made inoperable during normal operations or in response to a plant transient. (Section 1R11)

- Green. An NRC identified non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” was identified for failure to take prompt action to correct a condition adverse to quality. A graph of predicted jet pump loop flow versus flow control valve position, used to perform a daily Technical Specification (TS) surveillance to verify jet pump operability, had not been updated as required after the

2004 refueling outage (April 2004) and the deficiency was not corrected until October 26, 2004. The performance deficiency associated with this finding is the failure to take prompt action to correct a condition that affected the ability of operators to verify the operability of safety-significant reactor vessel internal components (the jet pumps).

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the cornerstone attribute of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance because it was not a design or qualification deficiency that had been confirmed to result in a loss of function per Generic Letter 91-18, did not represent a loss of safety function, did not represent actual loss of safety function of a single train for greater than its TS allowed outage time, did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk-significant per 10 CFR 50.65 for greater than 24 hours, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The failure to promptly inform operators when a problem was identified that affected performance of the daily jet pump surveillance is an example of a cross-cutting issue in problem identification and resolution (PI&R). (Section 1R15)

B. <u>Licensee-Identified Violations</u>

None

REPORT DETAILS

Summary of Plant Status

Nine Mile Point Unit 1 (Unit 1) began the inspection period operating on four recirculation loops at 100 percent power. Operation on five recirculation loops was restored on October 8, after completion of planned maintenance. Routine small power maneuvers were conducted during the inspection period for scheduled maintenance activities. The inspection period ended with Unit 1 operating on five recirculation loops at 100 percent power.

Nine Mile Point Unit 2 (Unit 2) began the inspection period at 100 percent power. Power was reduced on October 1, to approximately 64 percent and maintained there for several days to support main feedwater pump maintenance and to locate a main condenser tube leak. The "E" main condenser water box was damaged during the tube leak repair and subsequently remained out-of-service. The plant was returned to 100 percent power on October 5. On October 24, power was reduced to 82 percent to support switching steam jet air ejectors. On December 4, power was reduced to approximately 60 percent and maintained there for several days to support flux suppression testing due to indication of a possible fuel cladding leak. The suspect fuel assembly location was identified and control rod 18-07 was inserted to suppress power and limit the potential for defect growth. During the power reduction, the "E" main condenser water box was also repaired and returned to service. The plant was returned to 100 percent power on December 7, and continued to operate at full power 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 and two Unit 2 risk significant systems to verify that design features and operating procedures support operation of the associated systems during periods of cold weather. Unit 1 documents reviewed included the Unit 1 Final Safety Analysis Report (FSAR), the Unit 1 Individual Plant Examination for External Events (IPEEE), N1-OP-64, "Meteorological Monitoring," and N1-PM-A5, "Cold Weather Preparation and Operation." Unit 2 documents reviewed included the Unit 2 Updated Final Safety Analysis Report (UFSAR), the Unit 2 IPEEE, and N2-OP-102, "Meteorological Monitoring." The following areas were examined:

- Unit 1 Service Water and Diesel Fire Water Systems;
- Unit 1 Emergency Cooling System;
- Unit 2 Condensate Storage Tank; and
- Unit 2 Emergency Diesel Generator System.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdown. (71111.04 - 5 Samples)

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

- On October 25, the inspector performed a partial system walkdown on the Unit 2 reactor core isolation cooling (RCIC) system due to the high pressure core spray (HPCS) system being out-of-service during the Division III emergency diesel generator (EDG) overhaul. The walkdown included control room switch and indication verification, physical inspection, and partial verification of the system lineup. Procedure N2-OP-35, “Reactor Core Isolation Cooling,” was used for this review.

- On November 15, the inspector performed a partial system walkdown on the Unit 1 fire protection system due to the Unit 1 diesel fire pump being out-of-service for unplanned maintenance and the fire water system being cross-tied to Unit 2. The walkdown included control room switch and indication verification, physical inspection, and partial verification of the system lineup. Procedure N1-OP-21A, “Fire Protection System - Water,” was used for this review.

- On November 18, the inspector performed a partial system walkdown on the Unit 2 radiation monitor system due to an elevated fuel reliability index number showing the possible presence of a fuel leak. The walkdown included indication verification and physical inspection. Procedure N2-OP-79, “Radiation Monitoring,” was used for this review.

- On December 1, the inspector performed a partial system walkdown on the Unit 1, 103 EDG and emergency electrical distribution system due to the 102 EDG being inoperable for planned maintenance. Procedures N1-OP-45, “Emergency Diesel Generators” and N1-OP-33A, “115kV System,” were used for this review.

- On December 2, the inspector performed a partial system walkdown on the Unit 1 core spray loop 11 due to the 12 loop being inoperable for planned maintenance. Procedure N1-OP-2, “Core Spray System,” was used for this review.

Complete System Walkdown. (71111.04S - 1 Sample)

On December 22-29, the inspectors performed a full system walkdown of the Unit 2

HPCS system. Documents utilized in this inspection were the UFSAR, Operating Procedure N2-OP-33, “HPCS,” Valve Lineup Procedure N2-VLU-01, “Walkdown Order Valve Lineup and Valve Operations,” Attachment 33, N2-OP-33, “Walkdown Valve Lineup,” and N2-OP-72, “Standby and Emergency AC Distribution System.”

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope (71111.05Q - 10 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 FSAR and the Unit 2 UFSAR.

- Unit 1 Reactor Building 340 ft;
- Unit 1 Reactor Building 318 ft;
- Unit 1 Turbine Building 261 ft;
- Unit 1 EDG 102 and Emergency Switchgear Room 261 ft;
- Unit 1 EDG 103 and Emergency Switchgear Room 261 ft;
- Unit 2 Diesel Fire Pump Room;
- Unit 2 Division I Switchgear Room;
- Unit 2 Division II Switchgear Room;
- Unit 2 Division III Switchgear Room; and
- Unit 2 Division I and II Service Water Pump Rooms.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope (71111.06 - 2 external and 2 internal Samples)

The inspectors examined Unit 1 and Unit 2 risk significant areas to examine susceptibility to external flooding and verify that the licensee’s flooding mitigation plans and equipment were consistent with design requirements and the risk analysis assumptions. Unit 1 documents reviewed included the FSAR and the Unit 1 IPEEE. Unit 2 documents reviewed included the UFSAR and the Unit 2 IPEEE. In addition, EPIP-EPP-26, “Natural Hazard Preparation and Recovery,” was reviewed.

The inspectors performed a walkdown of Units 1 and 2 Intake Structures to examine susceptibility to internal flooding. Documents reviewed included the FSAR, UFSAR, and the Unit 1 and 2 Individual Plant Evaluations (IPEs).

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope (71111.07B - 3 Samples)

Based on risk significance and resident inspector input, the inspector selected the Unit 1 containment spray system heat exchangers and the Unit 2 division III EDG heat exchanger and the associated division III EDG switchgear room cooler for this biennial review.

For the selected heat exchangers, methods used by the licensee to ensure heat removal capabilities were reviewed and compared to commitments made in response to Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment.” Test methodology and results of heat exchanger performance testing were reviewed and verified to be consistent with accepted industry practices and guidance. Also, the inspector determined that test conditions were consistent with the chosen test method and that acceptance criteria were consistent with design basis values.

Although not selected as one of the heat exchangers for a detailed review, the inspector observed performance testing of a Unit 2 reactor building general area, elevation 261 foot, cooler (2HVR-UC414A). The test method and procedure were functionally the same as that for the division III EDG switchgear room cooler. Acceptance criteria were reviewed and found to be consistent with the heat exchanger specifications. Data requirements were consistent with industry guidance and provided sufficient information to allow the licensee to extrapolate heat exchanger performance to design conditions.

Additionally, the inspector reviewed methods for controlling biotic fouling and monitoring for zebra mussel growth to verify that they were implemented effectively.

The inspector walked down the Unit 1 and Unit 2 screen wells and the Unit 2 Division III EDG to assess the general material condition of the selected heat exchangers. Also, the inspector reviewed a sample of deviation event reports (DERs) related to the selected heat exchangers. This review was done to ensure that problems related to these components were appropriately identified, characterized, and corrected.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

1. Resident Inspector Quarterly Review (71111.11Q - 1 Sample and 71114.06 - 1 Sample)

The inspectors reviewed one licensed operator requalification training activity, to assess the licensee's training program effectiveness. The inspectors observed Unit 2 licensed operator simulator training on November 23. The inspectors reviewed performance in the areas of procedure use, self-checking and peer-checking, completion of critical tasks, and training performance objectives. Following the simulator training, the inspectors reviewed simulator fidelity through a sampling process. During the Unit 2 training, the inspectors evaluated emergency response organization (ERO) performance regarding initial and subsequent actions by licensed operators.

b. Findings

No findings of significance were identified.

a. Inspection Scope

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 (MC) 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)," as acceptance criteria, 10 CFR 55.46 Simulator Rule (sampling basis). These inspection activities were performed for both units.

The inspectors reviewed documentation of operating history since the last requalification program inspection. The inspectors also discussed facility operating events with the resident staff. Documents reviewed included NRC inspection reports and licensee Condition Reports that involved human performance and TS compliance issues. Specifically, the inspectors reviewed the following plant deviation/event reports (DERs):

- DER-NM-2003-2687; Inadequate operability determinations associated with 11 and #12 feedwater pump minimum flow valves (Unit 1);
- DER-NM-2003-2784; Unplanned entry into TS limiting condition for operations (LCO) 3.3.7.e during EDG 103 pre-start checks (Unit 1);
- DER-NM-2004-1898; RHS pump 1B started in shutdown cooling mode without a complete flowpath (Unit 2); and
- DER-NM-2004-3444; Required 8-hour notification to the NRC for HPCS unplanned inoperability was not submitted on time (Unit 2).

The Unit 1 examination consisted of both the biennial written exam and the annual operating exam. The Unit 2 exam consisted of the annual operating exam. The inspectors reviewed three licensed operator comprehensive biennial written exams and two cyclical quizzes administered at Unit 1 in 2004. The inspectors reviewed five sets of simulator scenarios and 10 job performance measures (JPM) administered at both units during this current exam cycle to ensure the quality of these exams met or exceeded the criteria established in the Examination Standards and 10 CFR 55.59.

The inspectors observed the administration of operating examinations to the Unit 1 “A” shift operating and staff crews. The operating examination consisted of two simulator scenarios for the operating crew and for the staff crew, and one set of eight (four in-plant and four control room) job performance measures administered to each individual. As part of the examination observation, the inspectors assessed the adequacy of licensee examination security measures.

The inspectors interviewed four evaluators, two training supervisors, three reactor operators (RO), and three senior reactor operators (SRO) for feedback regarding the implementation of the licensed operator requalification program. The inspectors also reviewed Operations Training Deviation Event Reports, Quality Assurance (QA) audits, Operations Training self-assessments, and recent plant and industry events to ensure that the training staff modified the program, when appropriate, to recommended changes.

The effectiveness of remedial training was assessed through the review of evaluation records for the past year, including four instances of evaluation failures at each unit.

Conformance with operator license conditions was verified by reviewing the following records:

- Attendance records for both units for the most recent year training cycle.
- Seven medical records from each unit to confirm all records were complete that restrictions noted by the doctor were reflected on the individual’s license and that the exams were given within 24 months of one another.
- Proficiency watch-standing and reactivation records. A sample of licensed operator watch-standing documentation, nine operators for Unit 1 and two for Unit 2, was reviewed for the current and prior quarter to verify currency and conformance with the requirements of 10 CFR 55.

The inspectors observed simulator performance during the conduct of the examinations, and reviewed simulator performance tests and discrepancy reports to verify compliance with the requirements of 10 CFR 55.46. Nine Mile is committed to the ANSI 3.5-1998 standard. The inspectors reviewed simulator configuration control and performance testing through interviews and the review of: facility simulator procedures; open and closed simulator condition reports and discrepancy reports; and the review of test results. Specifically, the following license operator requalification training (LORT)tests were reviewed:

Scenario-based tests:

- LORT Exam scenario O1-OPS-009-1DY-1-02;
- LORT Exam scenario O1-OPS-009-1DY-1-31;
- LORT Exam scenario O2-OPS-009-1DY-2-04; and
- LORT Exam scenario O2-OPS-009-1DY-2-25.

On December 22, the inspectors conducted an in-office review of licensee requalification exam results. For Unit 1, these results included both the biennial written and annual operating examinations. For Unit 2, these results included only the annual operating test (i.e., the comprehensive written exam was administered last year). The inspection assessed whether pass rates were consistent with the guidance of NRC MC 0609, Appendix I, "Operator Requalification Human Performance SDP." The inspectors verified that:

- Crew failure rate on the dynamic simulator was less than 20% for Unit 1 (failure rate was 0%), but was NOT less than 20% at Unit 2 (failure rate was 33%).

- Individual failure rate on the dynamic simulator test was less than or equal to 20% at both units (failure rate was 11% at Unit 1 and 13% at Unit 2).

- Individual failure rate on the walkthrough test JPMs was less than or equal to 20% at both units (failure rate was 0% at both units).

- Individual failure rate on the comprehensive biennial written exam was less than or equal to 20% at Unit 1 (failure rate was 0%).

- More than 75% of the individuals passed all portions of their exam at both units (89% of the individuals passed all portions of the exam at Unit 1 and 87% at Unit 2).

b. Findings

1. Crew Failure Rate on the Dynamic Simulator Portion of the Facility-Administered Annual Operating Examinations

Introduction. A finding of very low safety significance (Green) was identified at Unit 2, based on three of nine crews failing their facility-administered annual simulator examinations.

Description. During facility-administered annual operating testing of the licensed operators, licensee training staff evaluated crew performance on dynamic simulator scenarios using performance standards derived from NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." Facility results of crew performance showed that three of the nine crews evaluated (33%) did not pass their simulator exams.

Analysis. The inspector determined that the crew failures were a performance deficiency (PD) because operators are expected to operate the plant within acceptable standards of knowledge and abilities, as demonstrated through required periodic testing. Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for affecting the NRC's regulatory function and was not the result of any willful violation of NRC requirements or licensee procedures. The finding is more than minor because the PD affected at least the Reactor Safety/Mitigating Systems Cornerstone (and potentially Initiating Events and Barrier Integrity) objective and its related attribute on Human Performance (Human Error (Pre-Event and Post-Event)). Specifically, the finding reflected the potential inability of the crews to take appropriate safety-related actions in response to actual abnormal or emergency conditions while they were on-shift prior to the requalification testing. Since this is a more than minor requalification training issue, the perceived risk associated with the number of crews failing the annual operating tests is provided in the Simulator Operational Evaluation Matrix of NRC MC 0609, Appendix I, "Operator Requalification Human Performance SDP." The Matrix was entered using the number of crews that took the simulator test, nine, and the number of crews with UNSAT performance, three. Based on these numbers, the finding was characterized by the SDP as having very low safety significance (20 - 34% failure rate), or Green. The finding is of very low safety significance because the failures occurred during annual testing of the operators on the simulator, because there were no actual consequences to the failures, and because the crews were removed from watch standing duties, retrained, and re-evaluated before they were authorized to return to control room watches. FIN 05000410/2004005-01, Crew Failure Rate on the Dynamic Simulator Portion of the Facility-Administered Annual Operating Examinations

2. Failure of the Nine Mile Point Simulators to Correctly Demonstrate Expected Plant Response to Operator Input and to Transient Conditions

Introduction. Two examples of a self-revealing SDP Green Non-Cited Violations (NCV) of 10 CFR 55.46(c)(1), "Plant-referenced simulators," were identified for failure of Nine Mile Point simulators to correctly demonstrate the expected plant response to two separate events, one at each NMP unit. At Unit 1, the simulator failed to properly demonstrate expected plant response to operators placing the emergency condensers in service at approximately 65% power during manual initiation of the emergency cooling system (ECS) while conducting surveillance test procedure N1-ST-V19, "Emergency Cooling System - Heat Removal Capability Test at High Power," Revision 0. At Unit 2, the simulator failed to correctly demonstrate the expected plant response to operators injecting water into the reactor vessel when mitigating accident conditions while conducting emergency operating procedures (EOP) following a loss of coolant accident (LOCA) event.

Description. On January 9, 2004, a Nine Mile Point Unit 1 surveillance test procedure, N1-ST-V19, Revision 0, was aborted after three minutes into the test procedure by a licensed senior operator due to multiple unexpected reactor and plant responses including: (a) an increase in one area of the reactor core's neutron flux on local power range monitors (LPRMs) that caused related average power range monitor (APRM) channels #12 and #16 (indicative of asymmetric core flux tilt) to alarm high and subsequently initiate automatic actions to prevent control rod withdrawals; and (b) an increase in indicated core megawatt thermal (MW_{th}) power in excess of abort criteria established in the procedure; and a rapid closure of the main turbine bypass valves to maintain reactor pressure.

The inspectors noted that the licensee's assessment of this event (NMP 1 DER-NM-2004-87) identified that: (1) "... the response observed by operating crews during simulator validation of the procedure, and just-in-time (JIT) training, established expectations for plant nuclear and thermal-hydraulic response that was inconsistent with how the plant actually responded during the test." And (2) "... During the simulator performance of the surveillance test scenario (e.g., during JIT),... the crew observed minimal increase on the APRMs, a gradual decrease in MW_{th}, and the bypass valves modulating open 3-5% to maintain reactor pressure." Primarily, the licensed operator and senior operator negative training occurred due to insufficient simulator scope and fidelity of the reactor core and the reactor coolant system (RCS) including the emergency cooling system. This issue was self-revealing. Despite the negative training, which led operators to an incorrect understanding of expected nuclear and thermal hydraulic operating characteristics of the reactor, it was determined that operator actions during the event were appropriate and timely, reflecting that the operators and the procedure anticipated the unexpected. Following the reference plant event, simulation facility licensee staff modified the simulator's modeling of the reactor coolant system ECS response to correct discrepancies relating to condenser water level boil-down and make-up.

For the Unit 2 example, the inspectors reviewed DER-NM-2003-1857, "Improper Water Level Response During a DBA LOCA." The simulation facility had implemented a temporary software modification (e.g., a computer model substitute for inadequate, incorrect, and/or invalid response) for use during the conduct of licensed operating tests and/or during training to preclude operator actions required by EOPs when mitigating design basis accident (DBA) LOCA scenarios. The inspectors identified that some of the corrective actions taken in response to this DER did not comply with ANSI/ANS-3.5-1998. Contrary to that standard, Section 3.1, "Simulator Capabilities," the software model modification ensured that reactor vessel water level response never recovers from a DBA LOCA event irregardless of any operator action which could or may mitigate the accident (such as reestablishing electrical AC power sources and restarting needed pumps). The standard requires, among other things, that: (1) the scope of simulation shall be such that the operator is required to take the same action on the simulator to conduct an evolution as on the reference plant; and (2) that the response of the simulator resulting from operator action, no operator action, improper operation action, automatic reference unit controls, and inherent operating characteristics shall be realistic and shall not violate the physical laws of nature, such as conservation of mass, momentum, and energy. Specifically, the NMP 2 simulator shows that reactor vessel level stops increasing for an indefinite period of time at approximately 67% level (e.g., 2/3 core height) irrespective of the number and combination of emergency core cooling systems injecting into the reactor vessel during a loss of coolant accident, irrespective of any operator actions that may or could mitigate the event. Contrary to the simulator's response, the indicated level in the reference plant would be expected to continue to increase under accident conditions where the rate of addition, injection, and/or flooding into the reactor vessel exceeds the rate of removal of reactor coolant. The failure of the simulator to accurately reproduce or demonstrate the expected reference plant response has the potential to result in significant negative operator training.

Analysis. The inspectors determined that the failure of the Nine Mile Point Unit 1 and Unit 2 simulators to correctly replicate the plant response to the above-described events is a performance deficiency because NMP is expected to meet the requirements of 10 CFR 55.46(c)(1), "Plant-referenced simulators." Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements or NMP procedures. This finding is more than minor because it affects the Human Performance (human error) attribute of the Mitigating Systems cornerstone; in both cases the deviant conditions could mislead operators in a training evolution.

This finding was evaluated using the Operator Requalification Human Performance SDP (MC 0609 Appendix I) because it is a more than minor requalification training issue related to simulator fidelity. The SDP, Appendix I, Block #12, requires the inspector to determine if deviations between the plant and simulator could result in negative training or could have a negative impact on operator actions. "Negative Training" is defined, in ANSI/ANS 3.5-1998, "Nuclear Power Plant Simulators for Use in Operator Training and Examination," as "Training on a simulator whose configuration or performance leads the operator to incorrect response or understanding of the reference unit." The Office of Nuclear Reactor Regulation, (NRR) was requested to review and clarify the requirement that negative training could have occurred verses did occur. Based on the review, NRR determined that negative training did not have to occur but, there had to be a potential for negative training based on the difference between the plant-referenced simulator and reference plant. Therefore, based on this clarification, if differences between the simulator and plant could negatively impact operator actions or potentially result in negative training then the finding is Green.

In this regard, the Unit 1 plant-referenced simulator was known to be insufficient in scope and fidelity with regard to the reactor core and RCS models to accurately replicate nuclear and thermal-hydraulic operating characteristics. However, NMP 1 simulator training provided as late as January 8, 2004 (on the evening prior to performing the new test procedure on the reference plant) on the NMP Unit 1 simulator provided licensed operators with an incorrect understanding of the expected behavior and response of the reactor's nuclear and thermal-hydraulic characteristics as a result of adding significant amounts of cold water from upstream of the ECS loop condensate return isolation valve into a reactor recirculation loop suction flow path into the reactor vessel.

Also, in the Unit 2 case, the failure to correctly demonstrate or replicate the reference plant response to operator input from adding, injecting, and/or flooding water into the reactor vessel during an LOCA reduced the overall simulator fidelity and as a consequence, has the potential to result in negative operator training. Therefore, the answer to the Block #12 question is YES for both the Unit 1 and the Unit 2 events, which resulted in a finding of very low safety significance (Green). The finding is of very low safety significance (Green) because the simulator's uncorrected model discrepancy did not have an adverse impact on operator actions such that safety related equipment was made inoperable during normal operations or in response to a plant transient.

Enforcement. 10 CFR 55.46(c)(1) requires, in part, that "the simulator must demonstrate expected plant response to transient conditions." Contrary to this requirement, the Nine Mile Point Unit 1 plant-referenced simulator did not demonstrate expected plant response to the January 9, 2004, event involving a surveillance test for ECS - Emergency Condenser Heat Capacity Test at High Power. Specifically, the behavior of the reactor core as well as the RCS including the ECS emergency condensers on the simulator was significantly different from the reference plant performance. Also contrary to the requirement, the Nine Mile Point Unit 2 plant-referenced simulator did demonstrate the expected plant response to operator input from the addition of water to the reactor vessel during a LOCA event. The failure of the plant-referenced simulators to accurately replicate and model reactor and plant response to these events resulted in negative operator training. The failure to ensure that the simulator correctly replicates expected plant response to transient conditions is of very low safety significance and is being tracked for simulator fidelity correction in the licensees corrective action program under DER-NM-2004-730. This violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000220&410/2004005-02, Failure of the Nine Mile Point Unit 1 Plant-Referenced Simulator to Demonstrate Expected Plant Response to Operator Input and to Transient Conditions.

3. Acceptability or Suitability of Nine Mile Point Unit 1 and Unit 2 Simulator Scenario-Based-Tests (SBTs) For Meeting ANSI/ANS-3.5-1998 Performance Testing Criteria

10 CFR 55.4 states that the definition of "performance testing" means testing conducted to verify a simulation facility's performance as compared to actual or predicted reference plant performance. 10 CFR 55.46(c)(1) states that the plant reference simulator must demonstrate expected plant response to operator input and to normal, transient, and accident conditions to which the simulator has been designed to respond. 10 CFR 55.46 (d)(1) requires performance to provide continued assurance of simulator fidelity. To be consistent with the definition of "performance testing" in ANSI/ANS-3.5-1998 and the Commission's regulation, such testing must include a comparison of the results of integrated operation of the simulation facility to actual or predicted reference plant data.

In addition to the above, per Section 4.4.3.2, "Simulator Scenario Based Testing," of the ANSI/ANS-3.5-1998 standard, simulator scenario-based-tests (SBTs) need to demonstrate that the simulator is capable of being used to satisfy predetermined learning or examination objectives without exceptions, significant performance discrepancies, or deviation from the approved scenario sequence. Since simulator fidelity deficiencies can adversely affect the ability to meet training / learning objectives, SBTs must necessarily compare simulator performance to the actual or predicted performance of the plant.

The inspectors reviewed several SBTs. As described below, those reviews indicated that those tests did not compare and confirm the performance of simulator key parameters, automatic actions, and alarms against actual or predicted plant performance. In the absence of such comparisons, these tests did not meet ANSI/ANS-3.5-1998 requirements for performance tests. Because ANSI/ANS-3.5-1998 does not provide details regarding the extent of the comparison between the simulator and actual or predicted plant performance that is required during SBT, some confusion has developed regarding proper interpretation of the standard in this area. The NRC staff believes that the comparison, to be meaningful, must include key parameters, automatic actions, and alarms as referenced by Section 4.1.4, “Malfunctions,” of the standard. As advised by the NRC Operator Licensing Program Office, this item is unresolved pending anticipated enhancements to the ANSI/ANS-3.5 standard in this area and additional guidance or clarification/interpretation of existing guidance (e.g., revising Regulatory Guidance 1.149, “Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations,” Revision 3, October 2001).

The inspectors reviewed samples of NMP Unit 1 and Unit 2 simulator scenario-based-tests (SBTs) presented as ANSI/ANS-3.5-1998 performance tests. The simulation facility licensee uses exclusively NRC and/or facility developed operating tests (e.g., examination scenarios developed in accordance with guidance from NUREG-1021, Operator Licensing Examination Standards for Power Reactors, for the purpose of evaluating the performance of applicants for a license or licensed operators' requalification) as simulator SBT performance tests. As simulator performance tests, the scenario-based-tests did not sufficiently demonstrate that meaningful and adequate testing and documentation was conducted to verify the simulator's performance as compared to actual or predicted reference plant performance. ANSI/ANS-3.5-1998 Section 4.4.3, ”Simulator Performance Testing,” requires, among other things, that a record of the conduct of these tests, and a data comparison that the results meet reference unit data, shall be maintained. The simulator SBTs lacked required data comparisons, recording of tests results, and meaningful evaluations of tests results. The sampled SBTs included Attachment 10: “Scenario Validation Checklist” used by the licensee to ensure that the scenario can be used on the simulator for evaluating the performance of operators. However, the checklist relies heavily upon inferred or implied simulator performance from observations, for the most part, rather than a comparison to expected or predicted reference plant performance. The simulator SBT performance tests reviewed did not adequately identify specific key parameters, automatic actions, and/or alarms for comparison and evaluation to the reference unit expected or predicted response.

ANSI/ANS-3.5-1998 Section 4.1.4, “Malfunctions,” requires that it shall be demonstrated that simulator response during the conduct of malfunctions meet four specific acceptance criteria. These criteria would be appropriate for SBT and it is anticipated that they will be adopted for SBT in the upcoming revision to ANSI/ANS-3.5. NMP Unit 1 and Unit 2 simulator SBT malfunctions did not address three of the four required acceptance criteria of the standard. The criteria are: (a) any observable change in simulator parameters corresponds in direction to those expected from actual or best estimate response of the reference unit to the malfunction; (b) the simulator shall not fail to cause an alarm or automatic action if the reference unit would have caused an alarm or automatic action under identical circumstances; and, (c) the simulator shall not cause an alarm or automatic action if the reference unit would not cause an alarm or automatic action under identical circumstances.

Insufficient and inadequate SBT performance testing and documentation raise questions as to the adequacy or suitability of the license's operating test scenarios as simulator performance tests. The use of the licensee's checklist assertions do not provide an adequate and sufficient basis for demonstrating continued assurance of simulator fidelity. As noted above, the adequacy of the licensee's SBTs is unresolved, pending the clarification of ANSI standard and regulatory requirements.
URI 05000220&410/2004005-03, Acceptability or Suitability of Nine Mile Point Unit 1 and Unit 2 Simulator Scenario-Based-Tests (SBTs) For Meeting ANSI/ANS-3.5-1998 Performance Testing Criteria

4. Nine Mile Point Unit 2 Simulator Demonstration of Expected Plant Response to Operator Input and to Normal Evolutions (Reactor Startup, Reactor Shutdown, and Reactor Recovery After Trip from Power Operations) Using Only Operator Actions Normal to the Reference Unit

An NRC issue was identified for the potential failure of the Nine Mile Point Unit 2 simulator to correctly demonstrate the expected plant response to operator input and to normal conditions when conducting control rod withdrawal and/or control rod insertion control manipulations during reactor startup, reactor shutdown, and when approaching criticality using the reference plant procedures as applicable to normal evolutions.

The inspectors interviewed licensee simulator staff regarding its initial condition(s) development used on the NMP Unit 2 plant-referenced simulator. The inspectors found that contrary to the requirements of the ANSI/ANS-3.5-1998 standard, Section 3.1.3, "Normal Evolutions," the simulation facility licensee uses two different mathematical modeling changes rather than using only operator actions normal to the reference unit procedures when conducting normal evolutions for: (1) heat up from cold shutdown to hot standby; (2) unit shutdown from rated power to cold shutdown conditions; and (3) recovery to rated power after a reactor trip. Also, the licensee uses a pseudo-remote function to artificially change the reactor's reactivity with regard to control rod critical positions.

NMPNS Unit 2 reference plant does not have an "automatic" control rod withdraw or control rod insert design feature for the reactor manual control system. The reference plant reactor core reactivity is by design predetermined and cannot be adjusted during reactor operations under any conditions. Other factors (such as control rod worth, nuclear and thermal-hydraulic operating characteristics) that influence when a reactor achieves criticality cannot be artificially changed during the approach to critical in the reference reactor plant. Section 3.1.3, "Normal Evolutions," of the ANSI/ANS-3.5-1998 standard requires, among other things, that the simulator shall support the conduct of the reference unit evolutions using only operator action normal to the reference unit, including reactor startups and shutdowns, in a continuous manner without any mathematical model or initial condition changes. Additionally the standard requires that the response of the simulator resulting from operator action, no operator action, improper operator action, automatic reference unit controls, and inherent operating characteristics shall be realistic and shall not violate the physical laws of nature within the limits of the verification, validation, and performance testing criteria of Section 4, "Testing Requirements." This criteria is designed to ensure that no noticeable differences exist between the simulated systems when evaluated against the systems of the reference unit. Use of "mathematical model changes" instead of "only operator actions normal to the reference unit," without any validation testing to ensure the adequacy and accuracy of the mathematical models, fails to ensure that the simulator can correctly demonstrate repeatability with respect to time base relationships, sequences, durations, rates, and accelerations as required by Section 4.1, "Simulator Capabilities" criteria of the standard. Contrary to the standard, the inspectors found that the simulation facility licensee had conducted required normal evolution performance testing with regard to the Unit 2 plant-referenced simulator using mathematical model changes to preclude using only operator actions normal to the reference unit to automatically manipulate the movement of control rods. (e. g., allowing uncontrolled movement of control rods when this simulator feature is not in the reference plant and is not part of the design data for the control rod manual control system).

Incorrect generation of simulator initial condition sets, and the use of a remote function to effect reactivity changes, could impact operator actions on the reference plant as a result of licensed operators and senior operators being negatively trained on initial condition sets that were derived from an incorrect representation of the reactor manual control system. This item is unresolved pending the facility licensee's ability to demonstrate that the automatic rod programming and the remote changing of reactivity used by the licensee do not produce conditions in the simulator that vary from conditions operator would see at the reference unit. URI 05000410/2004-04, Nine Mile Point Unit 2 Simulator Demonstration of Expected Plant Response to Operator Input and to Normal Conditions (Reactor Startup, Reactor Shutdown, and Reactor Recovery After Reactor Trip from Power Operations) Using Only Operator Actions Normal to the Reference Unit.

1R12 Maintenance Effectiveness

a. Inspection Scope (71111.12Q - 2 Samples)

The inspectors reviewed the performance and condition history of two high safety significant systems, the Unit 1 diesel fire pump system and the Unit 2 residual heat removal system, to identify degraded or declining system performance or conditions. Reviews focused on: (1) proper maintenance rule (MR) scoping in accordance with 10 Code of Federal Regulations (CFR) 50.65(2); (2) characterization of failed structures, systems, and components (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 - 7 Samples)

The inspectors reviewed seven risk assessments and emergent work activities during this inspection period. For selected maintenance, work items or work orders 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. The following assessments/activities were reviewed:

- Diesel fire pump planned maintenance window (Unit 1);
- Emergency power transformer T101N inoperable due to failed cooling fan (Unit 1);
- EDG 102 inoperable due to hot spot identified in generator control circuitry (Unit 1);
- Excavation near transformers T101N and T101S (Unit 1);
- EDG 102 inoperable due to air leak on air receiver isolation valve (Unit 1);
- Reactor recirculation system flow control valve position detector (radial variable differential transformer, or RVDT) jumper installation (Unit 2); and
- Division III EDG six year preventive maintenance (Unit 2).

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-routine Evolutions and Events

a. Inspection Scope (71111.14 - 1 Sample)

On November 22, Unit 1 experienced a 0.5 gallon per minute leak from the reactor water cleanup (RWCU) system, outside of the drywell. A through wall leak had developed on a section of pipe connecting two shells in the regenerative heat exchanger. The leakage caused an increase in airborne radioactivity which resulted in operators ordering evacuation of the Reactor Building. The RWCU system was isolated and cooled down to stop the leakage. The inspectors proceeded to the Control Room in response to the evacuation announcement and observed operator actions to reduce reactor power, isolate the RWCU system, and restore power to rated. The inspectors reviewed EPIP-EPP-01, "Classification of Emergency Conditions at Unit 1," and the Emergency Action Limits matrix, along with various operating procedures, to assess operator performance.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15 - 4 Samples)

The inspectors reviewed four 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, (4) that the degraded systems remained operable. Procedure S-ODP-OPS-0116, “Operability Determinations,” was used for this review. Operability evaluations associated with the following issues were reviewed:

- On October 18, the inspectors reviewed the licensee’s evaluation of a water leak into the bearing oil reservoir on 13 condensate pump (Unit 1);

- On October 20, the inspectors reviewed the licensee’s evaluation of having an LPRM not in bypass that should have been due to spiking affecting APRM operability. Reference DER NM-2004-4767 (Unit 1);

C On December 15, the inspectors reviewed the licensee’s evaluation and management of the consequences of having declared valve BV-37-01, reactor vessel vent, inoperable within the scope of the in-service testing (IST) program, due to discovery of a non-environmentally qualified wire splice in the valve’s operating control circuit. This meant that the valve could not be considered available to perform its accident mitigation function. Reference DER NM-2004-5439 (Unit 1); and

C On October 26, the inspectors noted a draft DER (NM-2004-4589) which indicated that the Unit 2 jet pump curves used for the daily TS surveillance to verify jet pump flows are within 10 percent of each other had not been updated after the last refueling outage.

b. Findings

Introduction. A Green NCV of 10 CFR 50 Appendix B, Criterion XVI, “Corrective Action,” was identified in that the licensee failed to take prompt action to correct a condition adverse to quality. Data that is used to perform a daily TS-required jet pump surveillance was not updated after the last Unit 2 refueling outage, and once identified, action was not taken for 19 days.

Description. During review of the daily DER screening report on October 26, the inspectors noted DER NM-2004-4589, concerning a problem with surveillance procedure N2-OSP-LOG-D001, “Daily Checks Log.” Specifically, the graph of predicted jet pump loop flow versus flow control valve position, used to perform a daily TS surveillance to verify jet pump operability, had not been updated after the last refueling outage (RFO9). A new graph is to be generated after each refueling outage because core alterations can change the pressure differential across the core during operation, and thereby alter the relationship between core flow, jet pump flow, and recirculation loop flow. This had not been done, and the plant had been performing this portion of the daily TS surveillance using data from the previous operating cycle since startup from RFO9 approximately six months earlier. It should be noted that the surveillance consists of a verification of jet pump operability by two of three possible methods. Since this condition affected only one of the methods, it was still possible to successfully complete the surveillance.

The inspectors noted that, although this condition had been identified by engineering on October 7, the draft DER was not provided to the Operations Department for an operability assessment until October 26, 2004. Because this condition affected the performance of a TS-required surveillance, the inspectors concluded that the delay between the time of identification and the time at which action could be taken by the Operations Department to address the issue constituted a failure to take prompt action to correct the condition.

Analysis. The performance deficiency associated with this event was the failure to take prompt action to correct a condition that affected the ability of the Operations Department to verify the operability of safety-significant reactor vessel internal components (the jet pumps). The finding was greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the cornerstone attribute of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (in that it affects the ability to verify jet pump operability, and therefore affects the ability to verify the post-accident function of maintaining two-thirds core coverage). Additionally, the finding was consequential, in that, on October 3, use of the out-of-date graph produced an erroneous indication that the A jet pump loop flow was out of specification. The finding was determined to be of very low safety significance (Green) in accordance with Phase 1 of the Reactor Safety SDP because it was not a design or qualification deficiency that had been confirmed to result in a loss of function per Generic Letter 91-18, did not represent a loss of safety function, did not represent actual loss of safety function of a single train for greater than its TS allowed outage time, did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk-significant per 10 CFR 50.65 for greater than 24 hours, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. Furthermore, the out-of-date graph was nearly identical to the new graph over most of the range of operation, and, on the one occasion when it had produced an erroneous result (October 3), the daily TS surveillance had been satisfactorily completed using the two other methods of verifying jet pump operability.

The failure to promptly inform the Operations Department when a problem was identified that affected performance of the daily jet pump surveillance was an example of a cross-cutting issue in problem identification and resolution.

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that, "Measures shall be established to assure that conditions adverse to quality, such as . . . deficiencies . . . are promptly identified and corrected." Contrary to the above, following the identification on October 7, 2004, that the graph of predicted jet pump loop flow versus flow control valve position, used to perform a daily TS surveillance to verify jet pump operability, had not been updated as required after the 2004 refueling outage (April 2004), the deficiency was not corrected until October 26, 2004. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (DER NM-2004-4589), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000410/2004005-05, Failure to take Prompt Corrective Action for a Condition that Affected the Ability to Perform a TS Surveillance.

1R16 Operator Workarounds

a. Inspection Scope (71111.16 - 1 cumulative and 3 selected Samples)

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. The inspector looked for any cumulative effects of operator workarounds. NAI-REL-02, "Workaround Program," was used for this review.

Additionally, the inspectors selected three operator workarounds for a more detailed review:

- Workaround (WA) 04-04, Unit 1 operators have to manually control the steam jet air exhauster pressure;
- WA 04-20, Unit 1 circulating water system gate manipulation; and
- WA 00-03, Unit 2 shutdown confirmation feature of the rod worth minimizer does not work as designed.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope (71111.17A - 1 Sample)

The inspectors reviewed licensee efforts to repair a through wall leak which developed on the Unit 1 RWCU regenerative heat exchanger. An 8-inch long crack with a one-half inch break to the surface was repaired using manual overlay of weld material using ASME –504-2. The licensee also inspected an additional three welds and found no indications. The final weld was tested using an ASME, Section XI, Appendix 8, qualified ultrasonic testing (UT) technique. The expansion samples were also tested using a qualified technique.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope (71111.19 - 7 Samples)

The inspectors reviewed post-maintenance testing (PMT) procedures and associated testing activities for seven 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, and appropriate 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-RSP-12Q, “Instrument Channel Calibration High Radiation Reactor Building (RB) Ventilation Duct Monitor,” after replacement of the power supply (Unit 1);
- N1-ST-M4B, “EDG 103 EDG Operability Test,” after lube oil pump modifications (Unit 1);
- N1-ST-M4A, “EDG 102 EDG Operability Test,” after relay replacement (Unit 1);
- N1-PM-W9, “Fire Protection System-Weekly Operation of Fire Pumps,” for the diesel fire pump after cooling pump repairs (Unit 1);
- N1-ST-Q1A, “Core Spray Loop 111 Pump and Valve Operability Test,” after planned motor/pump preventive maintenance (PM) (Unit 1);
- N2-ISP-ADS-Q003, “Quarterly Functional Test and Calibration of the ADS Logic Timer Initiation Circuits,” performed after replacement of the Division I automatic depressurization system manual inhibit switch under WO 04-21548 (Unit 2); and
- PMT performed after the radial variable differential transformer (RVDT) jumper installation under WO 04-02150 (Unit 2).

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22 - 6 Samples)

The inspectors witnessed performance of surveillance test procedures and/or reviewed test data of selected risk significant SSCs to assess whether the testing satisfied TS, FSAR/UFSAR, and licensee procedure requirements, and to determine if the testing appropriately demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. The following surveillance tests were reviewed:

• N1-ST-Q13, Emergency Service Water Pump Operability Test (Unit 1);
• N1-ST-SA6, Drywell/Torus and Torus/RB Vacuum Reliefs Test (Unit 1);
• N1-ST-Q1B, Core Spray Loop 121 Pump and Valve Operability Test (Unit 1);
• N2-OSP-CSH-Q@002, HPCS Pump and Valve Operability and System Integrity Test (Unit 2);
C N2-OSP-ICS-Q@002, RCIC Pump and Valve Operability Test and System Integrity Test and ASME XI Functional Test (Unit 2); and
C N2-ISP-DER-M001, Monthly Functional Test of Primary Containment Drywell Floor and Equipment Drain Leak Rate Instrument Channels (Unit 2).

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23 - 1 Sample)

The inspectors reviewed a Unit 2 temporary modification which installed a jumper across the RCS flow control valve RCS*HYV17A selector switch to minimize the potential of developing a voltage differential across the switch. Reactor recirculation flow perturbations had developed which could be ameliorated by the installation of the jumper. The activity was conducted under WO 04-21050. The inspector observed the pre-evolution brief, plant power changes which were necessary to install the jumper, and activities associated with the jumper installation. The inspector observed restoration, pre-maintenance and post-maintenance test.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope (71114.04 - 1 Sample)

An in-office inspection that reviewed recent changes to the emergency plan (Revision 50) was conducted on December 15, 2004. The review verified the changes satisfied the standards of 10 CFR 50.54(q), 10 CFR 50.47(b), the requirements of 10 CFR 50 Appendix E, the intent of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," and that the changes did not decrease the effectiveness of the plan. These changes are subject to future NRC inspections to ensure that as a result of these changes the emergency plan continues to meet NRC regulations.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2PS2 Radioactive Materials Processing and Transportation

a. Inspection Scope (71122.02 - 6 Samples)

The inspector reviewed the solid radioactive waste system description in the FSAR and the recent radiological effluent release report for information on the types and amounts of radioactive waste disposed, and reviewed the scope of the licensee’s audit program to verify that it meets the requirements of 10 CFR 20.1101(c).

The inspector walked-down the liquid and solid radioactive waste processing systems to verify and assess that the current system configuration and operation agree with the descriptions contained in the FSAR and in the Process Control Program (PCP); reviewed the status of any radioactive waste process equipment that is not operational and/or is abandoned in place; verified that the changes were reviewed and documented in accordance with 10 CFR 50.59, as appropriate; and reviewed current processes for transferring radioactive waste resin and sludge discharges into shipping/disposal containers to determine if appropriate waste stream mixing and/or sampling procedures, and methodology for waste concentration averaging provide representative samples of the waste product for the purposes of waste classification as specified in 10 CFR 61.55 for waste disposal.

The inspector reviewed the radiochemical sample analysis results for each of the licensee’s radioactive waste streams; reviewed the licensee’s use of scaling factors and calculations used to account for difficult-to-measure radionuclides; verified that the licensee's program assures compliance with 10 CFR 61.55 and 10 CFR 61.56 as required by Appendix G of 10 CFR Part 20; and reviewed the licensee's program to ensure that the waste stream composition data accounts for changing operational parameters and thus remains valid between the annual or biennial sample analysis update.

The inspector observed shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness; verified that the requirements of any applicable transport cask Certificate of Compliance have been met; verified that the receiving licensee is authorized to receive the shipment packages; and observed radiation workers during the conduct of radioactive waste processing and radioactive material shipment preparation activities. The inspector determined that the shippers were knowledgeable of the shipping regulations and that shipping personnel demonstrate adequate skills to accomplish the package preparation requirements for public transport with respect to NRC Bulletin 79-19 and 49 CFR Part 172 Subpart H, and verified that the licensee's training program provides training to personnel responsible for the conduct of radioactive waste processing and radioactive material shipment preparation activities.

The inspector sampled non-excepted package shipment records and reviewed these records for compliance with NRC and DOT requirements.

The inspector reviewed the licensee's Licensee Event Reports (LERs), Special Reports, audits, State agency reports, and self assessments related to the radioactive material and transportation programs performed since the last inspection and determined that identified problems are entered into the corrective action program for resolution. The inspector also reviewed corrective action reports written against the radioactive material and shipping programs since the previous inspection.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

Annual Inspection. (71151 - 19 samples) The inspectors sampled licensee submittals for the performance indicators (PIs) listed below for the period from September 2003 through September 2004. 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; and
- 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 MR 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; and
- 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 MR 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 (RCS) Activity PI; and
- RCS 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 RCS activity and identified leak rate. The inspectors also observed the RCS leakage surveillance and observed a chemistry technician obtain and analyze an RCS activity sample. In addition, the inspectors 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.

b. Findings

No findings of significance were identified. Based on questions raised by the inspectors, the licensee plans to submit a frequently asked question (FAQ) to the NRC concerning an event associated with the Unplanned Power Changes per 7000 Critical Hours PI.

4OA2 Identification and Resolution of Problems

a. Inspection Scope (71152 - 1 Sample)

1. Unit 2 Rod Drive System (RDS) Suction Relief Valve (DER NM-2003-771)

On February 28, 2003, when the “B” RDS pump was started to perform PMT after replacement of the “B” RDS pump suction relief valve (2RDS-RV1B) and minimum flow line check valve (2RDS-V21B), the suction relief valve lifted and would not reseat. Immediate actions included securing the “B” RDS pump and isolating the suction side of the pump which stopped the flow of water from the relief valve into an equipment drain. Subsequent troubleshooting identified that, during switching operations of the RDS pumps, actuation of check valves on the discharge side of the pumps created pressure waves within the system that exceeded the relief valve set point. Additionally, due to slight differences in design and manufacture, the newly installed relief valve was more susceptible to the pressure waves than the original relief valve. Following research of applicable system and component specifications, and discussions with the system vendor, it was determined that the relief valve set point had been set to a lower value more than ten years previously. The final corrective action involved initiating and performing a design change to raise the relief valve set point to an appropriate setting within the limits of the system design and applicable codes.

This issue was selected for review due to the unique nature of the problem. The inspector reviewed the troubleshooting performed, appropriate design specifications, code requirements, and the design change performed to correct the problem. Also, the issue was discussed with system engineering personnel. The review was performed to verify:

- Completeness, accuracy, and timeliness of the problem identification and significance;
- Evaluation and disposition of operability and reportability issues;
- Consideration of extent of condition, generic implications, common cause, and previous occurrences;
- Classification and prioritization of the resolution of the problem commensurate with its safety significance;
- Identification of root and contributing causes of the problem;
- Identification of appropriate corrective actions; and
- Timely completion of corrective actions commensurate with the issue’s safety significance and verification that interim corrective actions and/or compensatory actions were identified and implemented to minimize the problem and/or mitigate its effects, until the implementation of the permanent corrective actions.

A complete listing of documents reviewed is included in the Attachment.

b. Findings

No findings of significance were identified and the inspectors identified no concerns with the corrective actions specified. Constellation personnel identified the root cause of the relief valve failure to be an incorrect set point and a susceptibility to pressure waves created during RDS pump switching operations. Corrective actions that have been completed include raising the relief valve set point to a value consistent with system design and applicable codes.

2. ASTM B43 Red Brass Piping Supplied in a Condition Inappropriate for Fluid System Application (DERs NM-2003-1052 and NM-2003-3996)

a. Inspection Scope (71152 - 1 Sample)

On March 12, 2003, while performing a spot check of installed instrument air system (IAS) piping at Unit 1, some IAS piping was found to exhibit the characteristics of being installed in the unannealed temper, contrary to NMP piping specifications. Spot checks were being performed as a result of a combined four IAS line failures at Unit 1 and 2. ASTM B43 Red Brass (Copper Alloy UNS No. C23000) piping is used at both Unit 1 and 2, principally in the IAS. Unannealed red brass piping is susceptible to stress corrosion cracking in the presence of an ammonia contaminant. All four IAS line failures have occurred in locations where unannealed red brass piping was used. Testing conducted on piping in the storeroom/warehouse found all of the piping was in the unannealed condition. Testing on some of the installed piping confirmed that unannealed piping had been installed in the IAS at both units. Corrective actions to date include obtaining properly annealed red brass piping for future use, disposition of the unannealed piping in the storeroom/warehouse, and revising procurement and material receipt procedures to clearly specify annealed red brass and to perform receipt testing to verify correct material is received. Additionally, at Unit 1, Constellation identified locations with unannealed red brass piping installed where the possibility of an ammonia contaminant existed. The piping in these areas was examined for indications of stress corrosion cracking. No indications were found.

This issue was selected for review due to the potential to impact IAS reliability at both units. The inspector reviewed the cause determination and held discussions with system engineering personnel. The review was performed to verify:

- Completeness, accuracy and timeliness of the problem identification and significance;
- Evaluation and disposition of operability/reportability issues;
- Consideration of extent of condition, generic implications, common cause, and previous occurrences;
- Classification and prioritization of the resolution of the problem commensurate with its safety significance;
- Identification of root and contributing causes of the problem;
- Identification of appropriate corrective actions; and,
- Timely completion of corrective actions commensurate with the issue’s safety

significance and verification that interim corrective actions and/or compensatory actions were identified and implemented to minimize the problem and/or mitigate its effects, until the implementation of the permanent corrective actions.

A complete listing of documents reviewed is included in the Attachment.

b. Findings

No findings of significance were identified. The inspector noted that the licensee identified locations of unannealed red brass piping in Unit 1, and determined its structural integrity and suitability for continued use, however, similar work remains to be done at Unit 2. The licensee's implementation of the corrective action program was acceptable, as the licensee's processes identified that progress to identify red brass piping in Unit 2 was not originally acceptable.

3. Continuous/Semi-Annual Resident Office Review

a. Inspection Scope (71152)

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 corrective action program. This review was accomplished by reviewing paper copies of each condition report, attending daily screening meetings, and accessing Constellation Energy's computerized database.

Semi-Annual Review

In an effort to identify trends where NMPNS personnel have not implemented effective corrective action to prevent recurrence of equipment performance issues, the inspectors conducted a screening review of all Deviation Event Reports (DERs) initiated since June 2004. Based upon that initial review, and the inspector's knowledge of the plant, several DERs that documented performance issues associated with cable splices at Unit 1, and the site fire protection program, were selected for additional follow-up. Through review of the DERs, and discussions with personnel in the engineering and operations departments, the inspector concluded that NMPNS personnel were aware that there had been performance issues in these areas, and that they had implemented corrective action to resolve the performance issues.

b. Findings

No findings of significance were identified.

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

Section 1R15 describes that the licensee failed to take prompt action to correct a condition adverse to quality that affected the ability of the Operations department to verify the operability of safety-significant reactor vessel internal components.

4OA6 Meetings, Including Exit

On January 14, 2005, the inspectors presented the inspection results to Mr. Tim O'Connor, and other members of licensee management. The licensee confirmed that proprietary information was not provided during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

T. DeSanto, Radiation Specialist
G. Detter, Manager, Security and EP Programs
T. Evans, CEG, Training Manager
C. Fisher, Maintenance Rule Coordinator
J. Gerber, ALARA Supervisor
R. Godley, Manager, Operations
T. Hogan, Radiation Protection Supervisor
B. Holston, Manager, Engineering Services
J. Jones, Director, Emergency Preparedness
A. Julka, CEG, Director, Q&PA
T. Kulczycky, Reliability Engineering
S. Leonard, CEG, GS Licensing
T. O'Connor, Plant General Manager
W. Paulhardt, Manager, Radiation Protection
G. Perkins, General Supervisor, Engineering Programs
J. Raby, Engineering Programs
J. Spina, Site Vice President
T. Syrell, Nuclear Regulatory Matters
D. Williams, Engineering Programs

NRC Personnel

W. Schmidt, Sr. Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000220&410/2004005-03	URI	Acceptability or Suitability of Nine Mile Point Unit 1 and Unit 2 Simulator Scenario-Based-Tests (SBTs) For Meeting ANSI/ANS-3.5-1998 Performance Testing Criteria
050000410/2004005-04	URI	Nine Mile Point Unit 2 Simulator Demonstration of Expected Plant Response to Operator Input and to Normal Conditions (Reactor Startup, Reactor Shutdown, and Reactor Recovery After Reactor Trip from Power Operations) Using Only Operator Actions Normal to the Reference Unit

Opened and Closed

05000410/2004005-01	FIN	Crew Failure Rate on the Dynamic Simulator Portion of the Facility-Administered Annual Operating Examinations
05000220&410/2004005-02	NCV	Failure of the Nine Mile Point Unit 1 Plant-Referenced Simulator to Demonstrate Expected Plant Response to Operator Input and to Transient Conditions
05000410/2004005-05	NCV	Failure to Take Prompt Corrective Action for a Condition that Affected the Ability to Perform a TS Surveillance

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R07: Heat Sink Performance

Deviation Event Reports

DER-NM-2003-867	DER-NM-2003-1180
DER-NM-2004-385	DER-NM-2004-4951
DER-NM-2004-4977	DER-NM-2004-5026
DER-NM-2004-5028	DER-NM-2004-5045

Design Basis Documentation
SDBD - 502, Rev 4, Service Water System
SDBD - 203, Rev 5, Containment Spray System

Other Documents
GAI-REL-04, Rev 01, Heat Exchanger Program
N2-TDP-REL-0104, Rev 00, GL 89-13 Service Water Systems Problems Affecting Safety Related Equipment Program Plan
Service Water System Health Reports, NMP 1, 9/03 - 2/04
Service Water System Health Reports, NMP 2, 3rd Quarter 2004
N2-CTP-SCT-0201, Rev 02, Service Water Chemical Treatment System
N1-CTP-V945, Rev 07, Service Water Zebra Mussel Treatment
N1-CTP-935, Rev 06, Operation of the Service Water Chemical Injection Skid
N1-MPM-080-410, Rev 3, Containment Spray Heat Exchanger Preventive Maintenance
N1-TTP-CTNSP-V001A, Rev 1, Containment Spray Heat Exchanger Heat Removal Capacity Test (HTX-80-33 (#121))
S14-93M006, Min Wall Evaluations - System 93
NER-2M-029, Rev 02, Heat Exchanger Tube Information
EGS-002, Impact of Plugging 19 Tubes in 2ECGS-EIC
N2-ESP-SWP-W790, Rev 08, Weekly Service Water Heater Current Test
S14-93-HX09 Containment Spray Heat Exchanger 111 and 121 Heat Removal Capacity Test Evaluation
SO-TORUS-M009, NMP 1 Torus Pool Heat-Up Analysis
12177-PID-11, NMP 2 Service Water System Piping and Instrumentation Diagram
N2-TTP-HVP-@001, Rev 5, Performance Evaluation Test for Diesel Generator Unit Cooler
N2-TTP-HVC-@102, Rev 3, Performance Evaluation Test for Unit Cooler 2HVC*UC102
N2-TTP-HVR-@414, Rev 04, Unit Cooler 2HVR-UC414A and 2HVR-UC414B Performance Evaluation Test

Section 1R11: Licensed Operator Requalification Program

DER-NM-2003-1857
DER-NM-1994-1019
DER-NM-1998-2440
DER-NM-2004-5297
DER-NM-2004-87

A Nine Mile Point Nuclear Station, Nuclear Training Procedure, NTP-TQS-504, Revision 19, Simulator Training and Evaluation, Effective 10/15/2004

Nine Mile Point Nuclear Station, Nuclear Training Procedure, NTP-TQS-506, Revision 17, Simulator Maintenance and Testing, Effective 10/15/2004

Nine Mile Point Nuclear Station, LLC, Unit 1 - Plant Reference Simulator Annual Report: ANSI/ANS 3.5 -1998 For The Year January 2002 - December 2002 (Year 3 of 4)

Nine Mile Point Nuclear Station, LLC, Unit 1 - Plant Reference Simulator Annual Report: ANSI/ANS 3.5 -1998 For The Year January 2003 - December 2003 (Year 4 of 4)

Nine Mile Point Nuclear Station, LLC, Unit 2 - Plant Reference Simulator Annual Report: ANSI/ANS 3.5 -1998 For The Year January 2002 - December 2002 (Year 3 of 4)

Nine Mile Point Nuclear Station, LLC, Unit 2 - Plant Reference Simulator Annual Report: ANSI/ANS 3.5 -1998 For The Year January 2003 - December 2003 (Year 1 of 4)

List of Open Simulator Discrepancies Unit 2, as of 11/12/04,

Unit 2 Simulator Exceptions, as of 11/10/04

List of Open Simulator Discrepancies Unit 1, as of 11/12/04,

Unit 1 Simulator Exceptions, as of 11/10/04

Simulator Exception Form, NMP 2 , DR# 2-2195,11/2/95, When suppression pool level drops below 195 ft, the RCIC RHR steam exhaust becomes uncovered and sc pressure should rise. The simulator SC model is not modeled to increase pressure in this case(s) - low training value vs. cost.

Simulator Exception Form, NMP 2, Change Tracking # PN2Y01EN012, 9/9/03, There are no automatic trips of any pump motors in the simulator due to the resultant effects from a loss of cooling water, (i.e., motor winding temps, bearing temps, etc.).

Simulator Exception Form, NMP 2, DR#3870, 4/7/03, Improper water level response during a DBS LOCA. ...A DBA LOCA should result in water level stabilizing at 2/3 core height plus some for the dynamic effects if the ECCS discharge pressure felt at the jet pump HP tap. Water level should not be recoverable above this level.

Simulator Discrepancy Report NMP1 # 3538, 2/3/2004, Based upon EC Capacity Test noted differences need to be addressed: EC boil down, plant takes approximately 27 minutes to boil down from 7.6 ft. to 4.0 ft.; in the simulator makeup capacity exceeds boil down. EC shell temperature rise takes approximately 10 minutes for EC shells to reach 212 DegF; simulator takes approximately three minutes. As temperature reaches 212 DegF, boiling effect on tank level indication changes by approximately 1 foot; simulator shows no effect.

Simulator Discrepancy Report NMP1 # 3557, 4/5/2004, EC initiation results in a delta level between loop 11 and 12 due to instrument tap locations - see DER 2004-376

Simulator Discrepancy Report NMP1 # 3580, 5/11/2004, During evaluation of ERV event at U1, evaluation showed that plant heat up of torus was faster than the simulator under same conditions. Need to evaluate containment spray flow and heat exchanger information.

Simulator Change Tracking Form # PC2-0197-00, NMP2, 7/29/2002, Provide Fuel bundles for NMP2 RF8 Ops Cycle 9. The new fuel bundle type (GE11 vs. GE14) will be determined during the modification process. Regardless of which type is chosen, the bundle enrichment will change slightly due to cycle eight core exposure history and the projected cycle nine energy requirements.

NMP 1 Operating Procedure, N1-OP-43A, Revision 13, Plant Start-Up, effective 8/30/2000.

NMP 2 Operating Procedure, N2-OP-101A, Revision 13, Plant Start-Up, effective 8/30/2000.

SDBD-204, Emergency Cooling System Design Basis Document, Rev 3, Nine Mile Point Nuclear Station, Unit 1, issued 6/17/03.

NMP1 Technical Support Surveillance Procedure, N1-ST-V16, Revision 01, Emergency Cooling System - Heat Removal Capability Test , performed on 4/22/2003.

Transient Tests ANSI/ANS 3.5, NMPNS Unit 2 Simulator Test Year 1 (2003) Panel of Experts Review 11/10/03.

NMP Simulator Scenario, 01-OPS-009-1DY-1-13, Rev 2, Complete Failure To Scram and Liquid Poison Injection, validated 10/16/03 [Example of examination scenario used as SBT].

NMP Simulator Scenario, 02-OPS-009-1DY-2-08, Rev 7, Inadvertent HPCS Injection with Feedwater Control Malfunction EHC Malfunction Causing a Turbine trip/Stuck Open SRV and Failure of Rods to Insert, validated 9/25/03.

NMPNS Generation Administration Procedure, GAP-OPS-05, revision 12, Reactivity Management, Effective date 09/05/2003.

Attachment three, Simulator Exceptions Sheet, NMP1, DR #1-1009, 11/9/93, “When changing reactor recirc flow to adjust power, an AGAF is needed to adjust APRMs in the simulator. This is not required at the plant.

Simulator Exception Form, NMP2, DR# PC2-0165-94, 4/25/97, Plant has two out of two logic for MSR High Level Trip of Main Turbine. Simulator logic is one out of one.

List of NMPNS Unit 1 Closed DRs (Period covering 11/01/03 to 11/01/04).

List of NMPNS Unit 1 Closed DRs (Period covering 11/01/03 to 11/01/04).

NMPNS Unit 1 Reactor Engineering Surveillance Procedure, N1-RESP-3, Revision 00, Reactivity Anomalies.

NMPNS Unit 2 Reactor Engineering Surveillance Procedure, N1-RESP-02, Revision 01, Reactivity Anomalies

NMPNS Unit 1 Reactor Engineering Surveillance Procedure, N1-RESP-11, Revision 01, In-Sequence Shutdown Margin Test.

NMPNS Unit 1 Reactor Engineering Surveillance Procedure, N1-RESP-10, Revision 00, Cold Critical Comparison and Shutdown Margin Test.

Design Change, N2-95-031 (SC2-0165-94) Conceptual Design Input / Add redundant level switches for 2DSM-LS70A/B.

Constellation Energy Nine Mile Point Nuclear Station, Operations Training Self-Assessment FSA 2004-64, “Simulator Effectiveness, November 4, 2004, Revision 1.

Section 1EP4: Emergency Plan and Implementing Procedures Changes

Site Emergency Plan, Revision 50

Section 2PS2: Radioactive Materials Processing and Transportation

Unit 1 Radwaste Process Control Program, Revision 7
Unit 2 Radwaste Process Control Program, Revision 6
Unit 1 FSAR Sections: X.B (reactor clean-up system); X.H (spent fuel storage pool filtering and cooling system); XII.A.2.2 (radioactive waste disposal system)
Unit 2 FSAR Sections: 5.4.8 (reactor water cleanup system); 9.1.3 (spent fuel pool cooling and cleanup system); 11.2 (radioactive liquid waste system)
Radioactive Material Shipments: 04-1091; 04-1108; 04-1177; 04-2052; 04-2038
2004-2005 Radwaste Operations Training and Qualification Plan
Radiation Protection Focused Self Assessment FSA-2003-03, Radioactive Shipments
Nuclear Quality Assurance Surveillance Report 03-0083-C, Radioactive Waste Processing and Shipping Programs
Quality Assurance Audit Report 02012, Radioactive Material Processing/Radiation Protection - ALARA Program Reviews
Deviation Event Reports: NM-2004-4925; NM-2004-4881

Section 40A2: Identification and Resolution of Problems

Deviation Event Report Instruction, NIP-ECA-01, Rev 34
RDS Pump Vendor Manual, N20241
RDS Vendor Manual, N20269
DCP-N2-03-057, RDS Relief Valve Set Point
B31.1 Power Piping Code
ASME Code Section VIII, Division 1, Sections UG-126 - UG 136
RDS Operating Procedure N2-OP-30, Rev 08
UFSAR Section 4.6
ASTM B43

DER NM-2004-4586
DER NM-2003-3996
DER NM-2003-1052
DER NM-2003-915
DER NM-2003-771
DER NM-2002-4481
Laboratory Report NMP Plant Air Line Failure, dated November 17, 2003

LIST OF ACRONYMS

ADAMS	agencywide documents access and management system
CFR	Code of Federal Regulations
DERs	deviation event reports
ECS	emergency cooling system
EDG	emergency diesel generator
EPIP	emergency plan implementing procedures
ERO	emergency response organization
FSAR	final safety analysis report
HPCS	high pressure core spray
IAS	instrument air system
IMC	inspection manual chapter
IPEEE	individual plant examination for external events
IR	inspection report
IST	inservice testing
JIT	just-in-time
JPMs	job performance measures
LCO	limiting conditions for operations
LER	licensee event report
LORT	licensed operator requalification training
MC	manual chapter
MR	maintenance rule
MW_{th}	megawatt thermal
NCV	non-cited violation
NEI	Nuclear Energy Institute
NMPNS	Nine Mile Point Nuclear Station
NRC	U.S. Nuclear Regulatory Commission
PARS	publically available records
PCP	process control program
PI	performance indicator
PI&R	problem identification and resolution
PM	preventive maintenance
PMT	post-maintenance testing
QA	quality assurance
RB	reactor building
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RDS	rod drive system
RHR	residual heat removal
RO	reactor operators
RVDT	radial variable differential transformer
RWCU	reactor water cleanup
SDP	significance determination process
SRO	senior reactor operators
SSCs	structures, systems, and components
TS	technical specification
URI	unresolved item
USFAR	updated final safety analysis report
UT	ultrasonic testing