January 24, 2002

Mr. John T. Conway
Site Vice President
Nine Mile Point Nuclear Station, LLC
P.O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION - NRC INSPECTION REPORT 50-220/01-10, 50-410/01-10

Dear Mr. Conway:

On December 29, 2001, the NRC completed an inspection of your Nine Mile Point Nuclear Station, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on January 11, 2002, with you and other members of your staff.

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

Based upon the results of this inspection, the inspectors identified two issues of very low safety significance (GREEN). The issues were determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these issues as Non-cited Violations (NCVs), consistent with Section VI.A.1 of the NRC Enforcement Policy, issued on May 1, 2000, (65FR25368). If you contest the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at the Nine Mile Point Nuclear Station.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories, and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of your response to these advisories and your ability to respond to terrorist attacks with the capabilities of the current design basis threat (DBT). From these audits, the NRC has concluded that your security program is adequate at this time.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publically Available Records (PARS) component of NRC's document management system (ADAMS). ADAMS is accessible from the NRC web site at http://www.nrc.gov/reading-rm.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Michele G. Evans, Chief
Projects Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report 50-220/01-10, 50-410/01-10

Attachment 1 - Supplemental Information

cc w/encl:
M. Wallace, President, Constellation Generation Group
G. Wilson, Esquire
M. Wetterhahn, Esquire, Winston and Strawn
J. M. Petro, Jr., Esquire, Counsel, Constellation Power Source, Inc.
J. Rettberg, New York State Electric and Gas Corporation
P. Eddy, Electric Division, Department of Public Service, State of New York
C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
J. Vinquist, MATS, Inc.
W. Flynn, President, New York State Energy Research and Development Authority
J. Spath, Program Director, New York State Energy Research and Development Authority
Supervisor, Town of Scriba
C. Adrienne Rhodes, Chairman and Executive Director
T. Judson, Central NY Citizens Awareness Network
R. L. Wenderlich, Senior Constellation Nuclear Officer

Distribution w/encl:
H. Miller, RA/J. Wiggins, DRA (1)
M. Evans, DRP
T. Bergman, RI EDO Coordinator
E. Adensam, NRR (ridsnrrdlpmlpdi)

P. Tam, PM, NRR
D. Skay, PM, NRR (Backup)
G. Hunegs, SRI - Nine Mile Point
W. Cook, DRP
S. Barr, DRP
R. Junod, DRP
Region I Docket Room (with concurrences)

DOCUMENT NAME: G:\BRANCH1\NMPSTUFF\NMP0110.WPD
After declaring this document “An Official Agency Record” it **will** be released to the Public. **To receive a copy of this document, indicate in the box:** "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP		RI/DRP	
NAME	GHunegs/GH		MEvans/MGE	
DATE	01/24/02		01/24/02	

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket Nos: 50-220, 50-410

License Nos: DPR-63, NPF-69

Report Nos: 50-220/01-10, 50-410/01-10

Licensee: Nine Mile Point Nuclear Station, LLC (NMPNS)

Facility: Nine Mile Point, Units 1 and 2

Location: P. O. Box 63
Lycoming, NY 13093

Dates: November 11, 2001 - December 29, 2001

Inspectors: G. Hunegs, Senior Resident Inspector
B. Fuller, Resident Inspector
R. Fernandes, Resident Inspector
F. Arner, Reactor Inspector
S. Barr, Senior Project Engineer
T. Burns, Reactor Inspector
P. Kaufman, Senior Reactor Inspector
J. Noggle, Senior Health Physicist
J. Williams, Senior Operations Engineer

Approved by: Michele G. Evans, Chief
Projects Branch 1
Division of Reactor Projects

Summary of Findings

IR 05000220-01-10, IR 05000410-01-10, on 11/11-12/29/2001; Nine Mile Point Nuclear Station, LLC; Nine Mile Point, Units 1 & 2. Resident Inspector Report

This inspection was conducted by resident inspectors and six region-based inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using IMC 0609, "Significance Determination Process," (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating System

Green. A non-cited violation of 10CFR50, Appendix B, Criterion XVI, Corrective Actions, for inadequate corrective actions related to a deviation and event report (DER), which adversely affected the licensee's ability to assess and trend the operability of the residual heat removal system (RHS) heat exchangers. The inspector determined that the corrective actions documented as completed in the DER, had not been performed. This issue was greater than minor because the licensee's ability to accurately assess the RHS heat exchangers performance and to conservatively identify their potentially degrading performance was negatively impacted and never actually resolved. The issue was of very low safety significance based on the actual results of the performance testing indicating the RHS heat exchangers had always remained operable. (Section 1R07)

Green. A non-cited violation of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, was identified for the failure to incorporate and implement specific operator actions documented in approved engineering support analysis 2M01-03 to monitor, control, and maintain reactor building temperature; specifically the emergency core cooling system, reactor core isolation cooling system and motor control center room temperatures within a specified temperature range when normal reactor building ventilation is isolated during a loss of coolant accident. This finding was determined to be of very low safety significance based on a Phase 1 Significance Determination Process (SDP) review because severe winter temperature conditions had not occurred since October, 2001, when it was identified that procedural changes were required to support operability of equipment. Therefore, the safety function of mitigating equipment and the secondary containment was not impacted. (Section 1R17)

B. Licensee Identified Violations

Violations of very low significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in section 4OA7 of this report.

Report Details

SUMMARY OF PLANT STATUS

Nine Mile Point Unit 1 (Unit 1) operated at 100 percent power throughout the inspection period.

Nine Mile Point Unit 2 (Unit 2) began the inspection period at 100 percent power. On December 2, Unit 2 operators inserted a manual scram from 75 percent power due to lowering reactor water level and imminent low level scram. Lowering reactor water level was a result of an electrical fault causing a trip of the "A" feedwater pump motor. Unit 2 was started up and returned to power operation on December 6, and full power was reached on December 7. On December 15, Unit 2 operators inserted a manual scram from 60 percent power due to drywell unidentified leak rate increasing. The reactor coolant system leakage was identified and the leak was repaired. Unit 2 was started up on December 17 and returned to power operation on December 18. Full power was reached on December 19 and remained there through the end of the inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspector conducted a review of Unit 1 actions to ensure protection of mitigating systems from cold weather effects. The inspector reviewed the Unit 1 cold weather preparation check list (N1-PM-A5) and selected several items on the list and verified that they were successfully completed. The inspector performed a walkdown of the Unit 1 screen house, turbine building and reactor building as part of the inspection for cold weather protection.

The inspector conducted a review of Unit 2 actions to ensure protection of mitigating systems from adverse weather effects. The inspector verified that the preventive maintenance (PM) for the service water (SW) intake bar rack heaters was current and that the weekly checks were being performed. Procedures reviewed included N2-ESP-SW-W790, Weekly SW Heater Current Test and N2 -ESP-SW-R79, Refueling Cycle SW Heater Resistance Test. In addition, the inspector reviewed the PM surveillance test data bank for any outstanding PM activities for heat trace systems. The inspector reviewed the cold weather preparation check list and selected several items on the list and verified that they were successfully completed. The inspector performed a walkdown of the Unit 2 service water intake structure, service water pump rooms, and the emergency diesel generator equipment rooms as part of the inspection for adverse weather protection.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed safety evaluations associated with mitigating systems, initiating events, and barrier integrity cornerstones to verify that changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR) were reviewed and documented in accordance with 10CFR50.59. Safety evaluations were selected based upon the safety significance of the changes and the risk to structures, systems and components.

The inspectors also reviewed applicability reviews (10CFR50.59 safety screens) for changes, tests and experiments for which the licensee determined that a safety evaluation was not required. This review was performed to verify that the licensees' threshold for performing safety evaluations was consistent with 10CFR50.59.

The inspectors reviewed a sample of deviation and event reports (DERs) documenting problems identified by the licensee in their corrective action program related to safety evaluations to verify the effectiveness of corrective actions.

A listing of the 10CFR50.59 safety evaluations, safety screens, and DERs reviewed is provided in Attachment 1.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Equipment Alignment

a. Inspection Scope

The inspector selected the Unit 2 division I emergency diesel generator (EDG) system to walkdown, while emergent work was conducted on the division II EDG. The walkdown included a control room switch line-up verification, an EDG system walk down, and the review of open work orders, DERs, and the EDG system health report.

The inspector utilized operating procedures and plant drawings to perform a partial system line-up of the Unit 2 reactor core isolation cooling (RCIC) system to verify correct system lineup. The walkdown included panel switch position, local valve position verification in the RCIC pump room, valve room, suction valve pit and other areas of the reactor building.

The inspector selected the Unit 1 containment spray 121 loop to walkdown, while pump replacement was performed for the 122 loop. The review included a control room switch line-up verification, a field walkdown, and the review of open work orders and DERs.

The inspector selected the Unit 1 103 EDG system to walkdown, while emergent work was conducted to replace the EDG 102 raw water pump. The walkdown included a control room switch line-up verification, an EDG system walkdown, and the review of open work orders and DERs.

b. Findings

No findings of significance were identified.

.2 Complete Equipment Alignment

a. Inspection Scope

The inspector performed a complete walkdown of the Unit 2 residual heat removal (RHS) system. The inspector utilized the individual plant examination to identify the risk significant equipment to inspect and procedures to review. The inspector reviewed the system health report and maintenance rule status . The inspector further reviewed recently completed surveillance test results and current open work orders to verify system operability and material condition. In order to verify that the licensee was identifying equipment alignment problems at an appropriate threshold, the inspector sampled the licensee corrective action program records related to the RHS system. The inspector performed a walkdown of all accessible portions of the RHS system, using plant drawings and procedures, and checked for proper valve position and material condition of system components. A tour of the Unit 2 control room was made to verify that the system switch line-up was in accordance with plant procedures.

b. Findings

No finding of significance were identified.

1R05 Fire Protection

.1 Routine Inspection

a. Inspection Scope

The inspectors conducted walkdowns of the fire areas to determine if there was adequate control of transient combustibles and ignition sources. The condition of fire detection devices, the readiness of the sprinkler fire suppression systems and the fire doors were also inspected against industry standards. In addition, the passive fire protection features were inspected, including the ventilation system fire dampers, structural steel fire proofing, and electrical penetration seals. The following plant areas were inspected:

- Fire area 8 West and North electric cable tunnels (Unit 2)
- Fire area 18 South cable tunnel (Unit 2)
- Control room (Unit 2)
- Relay room (Unit 2)

b. Findings

No findings of significance were identified.

.2 Annual Observation of a Fire Brigade Drill

a. Inspection Scope

On November 20, 2001, the inspector observed a fire brigade drill in Unit 1 to evaluate the readiness of Nine Mile Point Nuclear Station's (NMPNS) personnel to fight fires. The following aspects were evaluated:

- Protective clothing /turnout gear use
- Self-contained breathing apparatus use
- Fire hose capabilities and employment
- Fire team methodology, employment, and communications
- Smoke removal operations

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the licensee's flood mitigation equipment, such as watertight barriers, drainage and pumping systems, to ensure that the equipment was capable of meeting design requirements. The inspectors reviewed the UFSAR and IPE to identify those areas with the potential to be affected by internal flooding. Selected areas were walked down to inspect penetration seals, adequacy of watertight doors and the valves which were credited for isolation of flood areas within the reactor building. In addition, the inspector reviewed operating procedures, alarm response procedures, and emergency operating procedures. For those areas where operator actions are credited, the inspector verified that the procedures for coping with flooding, could reasonably be used to achieve the desired actions. In addition, the inspector reviewed the most recent surveillance test results for the Unit 2 emergency core cooling system (ECCS) pump room flood alarm instrumentation.

The following areas were selected:

- ECCS pump rooms (Unit 2)
- Turbine building and reactor building basement areas (Unit 2)
- Screen house (Unit 1)
- ECCS pump rooms (corner rooms) (Unit 1)

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspector reviewed the past performance and the recent inspection of the Unit 1 EDG 102 heat exchanger due to recently identified degradation in the flow supplied by the 102 raw water pump (PMP-79-53) as documented in DER NM-2001-5495. The inspector reviewed calculation S15-79-HX-01, Rev 02, EDGCW Hx Thermal Performance, which determined the minimum required raw water flow rate for varying lake temperatures. The inspector assessed the potential for common cause degradation of heat exchangers and reviewed the results of surveillance flow rate testing for the EDG 103 raw water system (N1-ST-Q25). A review was made of deficiency reports for performance problems related to the heat exchangers.

The inspector reviewed the past performance testing of the Unit 2 1A and 1B residual heat removal (RHS) system heat exchangers, specifically the most recent performance results of NMPNS procedure N2-TTP-RHS-4Y001, Residual Heat Removal System Heat Exchangers Performance Monitoring, for each RHS heat exchanger. The inspector also reviewed recent DERs related to the RHS heat exchangers in order to verify that the licensee had entered significant heat exchanger problems in their corrective action program and that appropriate corrective actions had been taken.

b. Findings

Green. A non-cited violation of 10CFR50, Appendix B, Criterion XVI, Corrective Actions, for inadequate corrective actions related to a DER, which adversely affected the licensee's ability to assess and trend the operability of the RHS heat exchangers.

The inspector reviewed two DERs which the licensee had generated concerning procedure N2-TTP-RHS-4Y001. These were DER-NM-2000-4181, which identified that the procedure was resulting in thermal performances of greater than 130% for the heat exchangers, which was rendering the action levels of the procedure (at 90% and at 75%) non-conservative; and DER-NM-2000-3552, which identified discrepancies in the engineering assumptions used in the procedure, also causing the acceptance criteria in the procedure to be non-conservative. The licensee determined the corrective actions required for the two DERs to be similar and closed DER -4181 by reference to DER -

3552. The licensee identified six corrective actions for DER -3552. One immediate action was to raise the procedure action level to a thermal performance level of 105%, thereby making the procedure more conservative until longer-term corrective actions could be taken. Five corrective actions were developed to investigate the heat exchanger design issues and resolve procedure calculations and discrepancies. DER - 3552 was closed by the licensee on November 22, 2000, noting that all corrective actions for the DER had been completed as of June 5, 2000.

The inspector reviewed the status of the corrective actions and determined that none of the corrective actions, including the proposed immediate corrective action of raising the action level to 105%, had ever been performed. This issue was greater than minor because the licensee's ability to accurately assess the RHS heat exchangers performance and to conservatively identify their potentially degrading performance was negatively impacted and never actually resolved. The issue was of very low safety significance based on the actual results of the performance testing indicating the heat exchangers had always remained operable. The licensee's failure to implement the corrective actions in the closed DER was a violation of 10CFR50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants, specifically criterion XVI, Corrective Action, which requires that conditions adverse to quality, such as deficiencies and nonconformances , be promptly identified and corrected. However, because of the very low safety significance and because the issue has been re-entered in the licensee's corrective action program, it is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 50-410/2001-010-01)**. This issue is in the licensee's corrective action program under DER-NM-2001-5590.

1R11 Licensed Operator Requalification

.1 Requalification Exam Results Review

a. Inspection Scope

A review was conducted of licensee requalification exam results for the biennial testing cycle. The inspection assessed whether pass rates were consistent with the guidance of NUREG-1021, Revision 8, Operator Licensing Examination Standards for Power Reactors and NRC Manual Chapter 0609, Appendix I, Operator Requalification Human Performance Significance Determination Process (SDP).

The inspector verified that:

- Crew pass rate was greater than 80%. (Pass rate was 87.5% for Unit 1 and 100% for Unit 2)
- Individual pass rate on the written exam was greater than 80%. (Pass rate was 98% for Unit 2. No written exam was given this year for Unit 1)

- Individual pass rate on the walk-through was greater than 80%. (Pass rate was 100% for both units)

- More than 75% of the individuals passed all portions of the exam. (100% of the individuals at Unit 1 and 98% of the individuals at Unit 2 passed all portions of the exam)

b. Findings

No findings of significance were identified.

.2 Observation of Operator Requalification Training Activities

a. Inspection Scope

The inspectors reviewed the licensed operator requalification training activities to assess the licensee's training program effectiveness. The inspectors observed Unit 1 licensed operator simulator training on December 11 (Crew D) and December 13 (Crew E). The inspectors reviewed performance in the areas of procedure use, self and peer-checking, completion of critical tasks, and training performance objectives. Following the simulator exercise, the inspectors observed the crew debrief and critique and reviewed simulator fidelity through a sampling process.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed performance based problems involving selected in-scope structures, systems, and components (SSCs) to assess the effectiveness of the maintenance program. Reviews focused on: (1) proper maintenance rule scoping, in accordance with 10 CFR 50.65; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10 CFR 50.65 (a)(1) and (a)(2) classifications; and, (5) the appropriateness of performance criteria for SSCs classified as (a)(2), and goals and corrective actions for SSCs classified as (a)(1). The inspectors reviewed the licensee's system scoping documents and system health reports. The following DERs were reviewed:

- NM-2001-4654, Turbine building roof vent failures (Unit 1)
- NM-2001-4830, Main steam line high flow instrument channel (Unit 2)
- NM-2001-2625, Division I emergency diesel generator ventilation dampers not modulating (Unit 2)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

For selected maintenance work orders (WOs), the inspectors evaluated: (1) the effectiveness of the risk assessments performed before the maintenance activities were conducted; (2) risk management control activities; (3) the necessary steps taken to plan and control resultant emergent work tasks; and, (4) the overall adequacy of identification and resolution of emergent work and the associated maintenance risk assessments. The following WOs were reviewed:

- WO 01-07284, “D” source range monitor (SRM) repair (Unit 2)
- WO 01-13430, Feedwater low flow control valve 2 FWS-LV55B (Unit 2)
- WO 01-13957-01, Rod drive control system transponder replacement (Unit 2)
- WO 01-07698-02, EDG 102 raw water pump replacement (Unit 1)

c. Findings

No findings of significance were identified.

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

.1 Manual Reactor Scram due to Loss of Feedwater Pump (Unit 2)

a. Inspection Scope

On December 2, 2001, Unit 2 was manually scrammed from 75 percent power. The reactor was manually scrammed in response to lowering reactor water level and an imminent low level scram. Lowering reactor water level was a result of an electrical fault and trip of the “A” feedwater pump motor and subsequent failure of the “B” reactor recirculation flow control valve hydraulic power unit power supply which prevented the “B” flow control valve from the expected runback. The inspector reported to the site and reviewed operator response to the transient and human and equipment performance. There were no emergency core cooling system actuations and all control rods fully inserted. The licensee’s corrective actions were documented in DER 2-2001-5609. The event was determined to not have been risk significant.

b. Findings

No findings of significance were identified.

.2 Manual Reactor Scram due to Unidentified Reactor Coolant System Leakage (Unit 2)

a. Inspection Scope

On December 15, 2001, at 4:32 a.m. the Unit 2 drywell particulate radiation monitors alarmed. Drywell (DW) floor drain leakage was observed to rise from 0.22 gallons per minute (GPM) to 0.4 GPM. The leakage remained the same until approximately 5:00 p.m. when it rose to 1.67 GPM. At 7:00 p.m. a plant shutdown was commenced before reaching any of the leakage limits specified in the Technical Specification limiting condition for operation (LCO) for reactor coolant leakage. TS 3.4.5, RCS Operational Leakage, requires the plant to be in hot shutdown in 12 hours when RCS leakage increases greater than two GPM within the previous 24 hour period or unidentified leak rate increases greater than five GPM. At 8:40 p.m., at approximately 60 percent power, the leak rate had risen in a step change to 5.87 GPM and at 8:46 p.m. a manual scram was inserted. The inspector reviewed TS, and operator response to the RCS leakage. The control room staff was effective in controlling reactor pressure vessel (RPV) level. A briefing was performed for the evolution in an effective manner. There were no ECCS system actuations. All control rods fully inserted, and all systems functioned as per design. The plant was stabilized in a pressure band of 400 to 500 psig for drywell inspection to identify the source of reactor coolant system leakage. An inspection of the drywell identified packing leakage from the “A” reactor recirculation pump discharge isolation valve, 2 RCS*MOV18A, as the cause. The licensee corrective actions were documented in DER 2001-5894. The event was determined to not have been risk significant.

b. Findings

No findings of significance were identified.

.3 Review of Licensee Event Reports

a. Inspection Scope

The inspectors reviewed selected Licensee Event Reports (LERs) to ensure that licensee staff actions taken in response to the events were in accordance with station procedures and regulatory requirements. The inspectors reviewed the licensee’s analysis of the event and associated corrective actions to ensure that appropriate measures were implemented to address any personnel performance concerns and that equipment problems were adequately resolved to prevent a recurrence of the identified problems. The following LERs were reviewed by the inspectors:

LER 50-410/2001-04, “Reactor Scram Due to Inadequate Main Steam Isolation Valve Surveillance Procedure.” The details of this event, including enforcement, were discussed in NRC inspection report No. 01-08. This LER is closed.

LER 50-220/2001-01, “Reactor Scram Due to Failure of Generator Protective Relay.” The details of this event were described in NRC inspection report No. 01-07. On August 22, 2001, Unit 1 scrammed from 100 percent power due to a generator trip. The cause of the generator trip was the faulty actuation of the negative phase sequence current relay in response to a grid perturbation. The relay malfunction was due to a design flaw. All control rods inserted on the scram. Main steam isolation valves (MSIVs) automatically closed due to the mode switch being in “Run” with reactor pressure at 850 psig. MSIVs were reopened six minutes after closing. The event was determined to be not risk significant. NMPNS noted that in 1987, General Electric (GE) had issued Service Advisory Letter (SAL) 189.1 to document the relay design flaw. Corrective actions included replacing the relay with one of a new design, briefing of operating crews on positioning the mode switch to “Shutdown” immediately after a scram and a review of GE SALs to evaluate potential design flaws. This LER is closed.

LER 50-220/2001-02, “115 kilovolt Line 4 Inoperable due to Inadequate Analysis of Design Change.” Line 4 was declared inoperable on September 7, 2001, when NMPNS determined that Line 4 could not provide the power required for a LOCA with Line 1 out of service, as required by Technical Specifications. The event was determined to have very low risk significance. On September 12, 2001, Line 4 was returned to operable following tap changes to the reserve station service transformers and procedure changes to ensure Line 4 voltage is maintained above minimum values. The cause of the inoperability of Line 4 was inadequate review of a design change which raised the degraded voltage relay (DVR) setpoints which initiate transfer of Unit 1 ECCS loads to the emergency diesel generators upon loss of off-site power. The capability of the 115kV lines to maintain voltage above the new DVR was not thoroughly evaluated. This LER is closed.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability evaluations affecting risk significant mitigating systems, to assess: (1) the technical adequacy of the evaluation; (2) whether continued system operability evaluations were warranted; (3) whether other existing degraded systems adversely impacted the affected system or compensatory measures; (4) where compensatory measures were used, whether the measures were appropriate and properly controlled; and, (5) the degraded systems impact on TS limiting condition for operations. The following licensee documents were reviewed:

- DER 2001-5652 Backup radial variable differential transformer indicator cable frayed (Unit 2)
- DER 2001-5495 Flow from EDG 102 raw water pump was less than required flow (Unit 1)

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors selected and reviewed a sample of permanent modifications of Nine Mile Point Units 1 and 2. The modifications were selected from the population of design changes completed since 1999 based on risk insights from the probabilistic risk assessment and the potential for impacting reactor safety cornerstones. The modifications involved safety related piping and components, electrical power systems, and changes to plant operating procedures.

Review of selected portions of the modification packages included the safety evaluation screening forms, 10CFR50.59 safety evaluations, design calculations, set point changes, and results of post-modification testing. Where appropriate, the inspectors discussed the scope and extent of the modifications, technical aspects of the changes, and implementation of the changes with the responsible engineering personnel.

In addition, the inspectors reviewed a sample of deviation event reports documenting problems identified by the licensee related to plant modifications in order to verify the effectiveness of the licensee's corrective actions.

A listing of the modifications and deviation event reports reviewed by the inspectors is provided in attachment 1.

b. Findings

Green. A non-cited violation of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, was identified for the failure to incorporate and implement the operator actions documented in approved engineering support analysis (ESA) 2M01-03 to monitor, control, and maintain reactor building temperature, specifically the ECCS, reactor core isolation cooling system (RCIC) and motor control center (MCC) room temperatures, within a specified temperature range when normal Reactor Building ventilation is isolated, such as a loss of coolant accident (LOCA). This finding was determined to be of very low safety significance (Green) because severe winter temperature conditions had not occurred since October, 2001, when it was identified that procedural changes were required to support operability of equipment. Therefore, the safety function of mitigating equipment and the secondary containment was not impacted.

In 1987, a design change was implemented that changed the control logic for the ECCS/RCIC room unit coolers from thermostatic to automatic start of all ECCS/RCIC room unit coolers on a LOCA signal to ensure the secondary containment drawdown time and capability is maintained within Technical Specifications. In accordance with the American Society Mechanical Engineers (ASME) Section III, NC 2300 code requirement, the containment penetrations and Class I and II piping must be above the metal nil-ductility temperature plus a margin specified by the ASME Section III code. This translated into minimum room temperature requirements varying from 55°F to

70°F, depending on which components are in the room. Operations procedure, N2-OP-52, Reactor Building Ventilation, was subsequently revised in 1987 to incorporate nil-ductility temperature guidance. This required operators, during non-LOCA situations as well as postulated LOCA conditions, to periodically turn unit coolers on and off when the lake is cold because the cooling capacity could exceed the heat load in the room for a system in a standby condition.

In October 1999, the licensee issued DER 1999-3478 and initiated design change N2-00-022 to restore thermostatic control to ECCS/RCIC room unit coolers. The purpose of this proposed design change was to eliminate an operator workaround related to the auto starting of all ECCS/RCIC room unit coolers on a LOCA signal, which requires operators to monitor and control ECCS/RCIC room, piping, and containment penetration material temperatures within a specific temperature range for the prevention of non-ductile fracture as specified in General Design Criteria (GDC) 51 of 10CFR50, Appendix A and (ASME) Section III, NC 2300.

During a subsequent secondary containment post-LOCA analyses performed in October 2001, in support of the proposed thermostatic control design change, the licensee identified that the current operating procedures may not be adequate to maintain the Reactor Building area temperatures within design basis requirements without excessive cycling of the ECCS and ICS room unit coolers. In some cases, the number of unit cooler cycles would be above their analyzed design limits. The analyses also indicated that there would be pressure control disturbances in the reactor building due to unit cooler cycling. This could cause the pressure to drop below the Technical Specification required ≥.25 inch water gauge vacuum, causing secondary containment to lose the required vacuum during post LOCA conditions. During this inspection, DER-NM-2001-5873 was initiated to determine if these conditions would have prevented the design function of ECCS, RCIC, and standby gas treatment systems.

On October 9, 2001, detailed operator guidance was developed in ESA-2M01-03 in accordance with engineering support analysis procedures (NEP-ECA-01) to ensure operability during various service water temperature conditions. These operator actions were to be incorporated into operational procedures to support system, component, and equipment operability by controlling the ECCS/RCIC room coolers and room temperatures based upon operators monitoring service water temperatures. The licensee believed that these operator actions had been incorporated into operations procedure, N2-OP-52, Reactor Building Ventilation, in October 2001. However, the inspectors found that the procedure had not been revised, even though the corrective actions to accomplish this task were found to be closed without incorporating the operator guidance into the procedure. The licensee entered this deficiency into their corrective action program as DER NM-2001-5842. Contingency actions documented in ESA-2M01-03 were implemented on December 12, 2001 and the reactor building ventilation procedure, N2-OP-52, was revised on December 14, 2001 to incorporate the operator guidance.

The inspectors determined the failure to incorporate the instructions into procedures was more than minor because the issue had a credible impact on safety. Specifically, during periods of severe winter temperature conditions, low service water temperatures had the potential during a postulated LOCA condition to adversely impact; (1) the operability of the unit coolers based on excessive cycling, (2) the design margin of safety related piping systems and containment penetration materials, and (3) the ability to maintain the required $\geq$.25 inch water gauge vacuum in secondary containment.

The failure to incorporate the required instructions into procedures was determined to be of very low safety significance (Green) based on a Phase 1 Significance Determination Process (SDP) review because severe winter conditions had not occurred since October, 2001, when it was identified by the licensee that procedural changes were required to support system, component, and equipment operability. Therefore, the safety function of mitigating equipment and secondary containment was not impacted during this inspection. 10 CFR 50 Appendix B, Criterion V, requires that activities affecting quality shall be prescribed by documented instructions and procedures of a type appropriate to the circumstances. Contrary to this, required operator actions which had been determined through an operability analyses, had not been prescribed in appropriate operating procedures. The inspectors determined this to be a non-cited violation of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings for failure to establish procedures, consistent with Section VI.A 1 of the NRC Enforcement Policy. This issue has been entered into the licensee’s corrective action program as DER NM-2001-5842. **(NCV 50-410/2001/010-02).**

1R19 Post-Maintenance Testing

a. Inspection Scope

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

- WO 01-10646, 2 RCS*MOV18A repack (Unit 2)
- WO 01-13975, 2FWS*LV55B repair (Unit 2)
- N2-ISP-CPS-002, Drywell and Suppression Chamber Purge System Exhaust Isolation Valve Leak Check (Unit 2)
- WO 01-13957-01, Rod drive control system transponder replacement (Unit 2)
- EDG102 Raw water pump N1-ST-Q25 after pump replacement (Unit 1)
- Containment spray 122 N1-ST-Q6D after pump replacement (Unit 1)
- WO 01-05236-00, emergency condenser vent block valve 05-05 (Unit 1)
- WO 01-07156-04, reactor level transmitter (LT-36-04C) replacement (Unit 1)

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed performance of surveillance test procedures and reviewed test data of selected risk significant SSCs to assess whether the SSCs satisfied Technical Specifications, Updated Final Safety Analysis Report (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 test was witnessed:

- N2-OP-101A, Plant Startup Attachment 2, Primary Containment Pre-Startup Check (Unit 2)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspector reviewed temporary change package (TCP) N2-01-198, Temporary Defeat Recirc Pump 1B trip on 2 RCS*MOV18B Less Than 90 Percent Open and De-energize 2 RCS*MOV18B. Due to a problem with the position indication on the "B" recirculation pump discharge blocking valve, the licensee implemented a temporary change to prevent the "B" recirculation pump from inadvertently tripping while at power. The pump control circuitry is such that if the discharge blocking valve is less than 90 percent open based on its position indication system, the pump will trip. To minimize the risk of an unplanned transient, the licensee disabled the interlock. The inspector reviewed the FSAR, TS, emergency and normal operating procedures to verify that the installation was consistent with the TCP.

The inspector reviewed action request (ACR) 4184, which was installed on October 22, to disable the main turbine high vibration trip. The inspector noted that N2-OP-21, Main Turbine, describes operation with the high vibration trip disabled. The inspector reviewed NIP-CON-01, Design and Configuration Control Process, and noted that temporary alterations are allowed when they are identified and controlled in technical procedures.

c. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector reviewed calibration methods and documentation of current calibrations of the Shepherd Models 89, 28-5, 142-10 calibrators; the high and low well calibrators; and the Tech Ops Area Radiation Monitor (ARM) portable calibrator. These calibrators were used for the calibration of radiation monitoring instrumentation.

The inspector observed and reviewed the calibration of the Unit 2 reactor building refueling floor gas radiation monitor (No.107) with respect to industry standards and procedural requirements methods. The inspector also reviewed the calibration and associated calibration records, for the following permanent in-plant instruments, relative to requirements contained in applicable calibration procedures:

- Unit 1 and 2 refuel platform high range radiation monitor
- Unit 1 and 2 control room ARM

- Unit 1 inner traverse in-core probe room ARM
- Unit 1 refuel floor ARMs (3)
- Unit 1 and 2 drywell high range gamma monitor
- Unit 1 drywell continuous air monitor
- Unit 2 drywell atmosphere monitor
- Unit 2 reactor building 240 foot elevation ARM
- Unit 2 below refuel floor air monitor

Portable health physics survey instrument calibration methods and selected in-use instrument calibration documents were also reviewed for the following radiation survey instruments, contamination survey instruments, personnel electronic dosimeters, and air sample counting instruments:

- Eberline 6112B teletectors (2)
- Eberline RO-2/2A ion chambers (9)
- Eberline AMS-3 continuous air monitor (1)
- Bicron Frisktech contamination monitors (2)
- Victoreen VAMP area radiation monitor (1)
- Eberline PNR-4 neutron radiation monitor (1)
- NNC Friskall personnel contamination monitors (2)
- Small Article Monitors (2)
- Digidose 100 electronic dosimeters (10)
- Eberline BC-4 beta counters (2)
- Eberline SAC-4 alpha counters (2)
- a high purity germanium gamma counter (1)

Emergency Plan specified self contained breathing apparatus (SCBA) locations in both Unit 1 and Unit 2 control rooms, the Unit 1 Administration Building 261 foot elevation, the Unit 2 access passage 261 foot elevation, and the Unit 1 SCBA air compressor room were visited and the SCBA units and spare breathing air bottles were examined for operability and licensee inspection history. The current control room shift staffing roster was utilized to review all on-shift control room operators for currency of SCBA use qualifications.

Condition reports with respect to radiation monitoring instrumentation or emergency SCBA use were reviewed from January 1, 2001 through December 11, 2001.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator (PI) Verification
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Occupational Radiation Safety, Safeguards

a. Inspection Scope

The inspectors reviewed the following 2001 PI data for the first through third quarter for Units 1 and 2:

- Unplanned scrams per 7000 critical hours
- Scrams with loss of normal heat removal
- Unplanned power changes
- Safety system unavailability
- Safety system functional failures
- Reactor coolant system activity
- Reactor coolant system leakage

The inspector reviewed implementation of the licensee's occupational exposure control effectiveness PI program to verify that occurrences meeting the criteria specified in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 1, were identified and reported as occurrences. Specifically, the inspector reviewed DERs and other pertinent documents, for occurrences involving locked high radiation areas, very high radiation areas, and unplanned personnel exposures covering the fourth quarter 2000 through the third quarter 2001, against the specified criteria. The inspector reviewed the licensee's programs for gathering and submitting data for the Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment Performance Indicators. The review included the licensee's tracking and trending reports, personnel interviews and security event reports for the PI data submitted from the first quarter of 2001 through the third quarter of 2001.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution

This inspection report documents two issues where corrective actions which were documented as completed in a DER had not been performed. In section 1R07, the NRC identified that corrective actions which impacted the ability to accurately assess the RHS heat exchanger performance were not implemented. In section 1R17, the NRC identified that corrective actions to change procedures associated with maintaining reactor building temperature were not implemented. The causal relationship of these errors is that the corrective actions, documented as completed, have not been implemented. This issue is in the licensee's corrective action program under DER NM-2001-5590.

4OA3 Event Follow-up

(Closed) LER 50-410/1999-17, "Drywell floor and equipment drain tank fill rate monitoring systems inoperable." The details of this event were described in NRC inspection report No. 99-07. On September 2, 1999, Unit 2 entered TS 3.4.3.1.d, "Reactor Coolant Leakage Detection System," limiting condition for operation (LCO) which required that a shutdown be initiated within 24 hours, if either the drywell floor drain tank fill rate monitoring system or the drywell equipment drain tank fill rate monitoring system could not be made operable. Following troubleshooting and repair activities on September 3, 1999, the licensee requested and received verbal enforcement discretion for the TS 3.4.3.1 24 hour allowed outage time (reference licensee letter, dated September 3, 1999, and Notice of Enforcement Discretion for the Licensee Regarding Nine Mile Point Unit 2, No. 99-1-005, dated September 8, 1999). This LER is closed.

(Closed) LER 50-410/1999-06, Supplement 1, "Inadequate surveillance of automatic depressurization nitrogen supply system isolation valves." The details of this event and the original LER were described in NRC inspection report No. 99-09. This supplement extended the completion date of a corrective action by two months. This LER is closed.

4OA5 Other

(Closed) Inspector Followup Item (IFI) 50-220/1999-04-03: Computer security for 3D-Monicore. 3D-Monicore is a system of computer programs designed to monitor and predict core parameters. In March 1999, inadvertent processing of in-core flux data resulted in thermal limits being exceeded at Unit 1. NMPNS documented the event in LER 50-220/1999-03, which was previously reviewed and closed. NMPNS determined that the cause was inadequate computer system security on the 3D-Monicore system. The system was not protected, in that the design allowed data to be processed without authorization from uncontrolled locations. The inspector reviewed corrective actions as described in DER 1-1999-0837 and applicable procedures, and discussed the event with the reactor engineering staff. The 3D-Monicore workstations have been modified to disable certain operational commands. User account and password information were reviewed to ensure that user accounts on the system were correct. This item is closed.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. Conway, Site Vice President, and other members of licensee management at the conclusion of the inspection on January 11, 2002. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations:

The following findings of very low significance were identified by the licensee and are violations of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-cited Violations.

NCV Tracking Number	Requirement Licensee Failed to Meet
050-220/2001-010-03	10CFR50, Appendix B, Criterion III, Design Control, for failure to verify the adequacy of design for the modification to the degraded voltage relays associated with LER 50-220/2001-02. (Reference section 1R14.3)

ATTACHMENT 1

a. Key Points of Contact

Licensee

J. Conway, Site Vice President
R. Dean, Manager Unit 2 Engineering
L. Hopkins, Unit 1 Plant General Manager
B. Montgomery, General Manager Nuclear Engineering
M. Peckham, Unit 2 Plant General Manager
B. Randall, Manager Unit 1 Engineering
D. Wolniak, Manager, Licensing

NRC

J. Trapp, Senior Reactor Analyst

b. List of Items Opened, Closed and Discussed

Opened and Closed

50-410/2001-010-01	NCV	10CFR50, Appendix B, Criterion XVI, Corrective Action, for the failure to implement the corrective actions in a closed DER 2000-3552 associated with the RHS heat exchanger.
50-410/2001-010-02	NCV	10CFR50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, for failure to establish procedures associated with reactor building equipment cooling.
50-220/2001-010-03	NCV	10CFR50, Appendix B, Criterion III, Design Control, for failure to verify the adequacy of design for the modification to the degraded voltage relays associated with LER 50-220/2001-02.

Closed:

50-410/1999-17	LER	Drywell floor and equipment drain tank fill rate monitoring systems inoperable.
50-410/1999-06-01	LER	Inadequate surveillance of automatic depressurization nitrogen supply system isolation valves.
50-220/1999-04-0	IFI	Computer security for 3D-Monicore.
50-220/2001-01	LER	Reactor scram due to failure of generator protective relay.

50-220/2001-02	LER	115 kilovolt Line 4 inoperable due to inadequate analysis of design change.
50-410/2001-04	LER	Reactor scram due to inadequate main steam isolation valve surveillance procedure.

c. List of Documents Reviewed

Procedures

Calibration of the High and Low Level Instrument Calibration Wells, S-RTP-14, Rev. 4
Operation and Calibration of the Shepherd Irradiator Range Model 89 and Model 28 Single Source Beam Calibration System, N2-RTP-164, Rev. 1
Operation and Calibration of the Eberline Multiple Source Gamma Calibrator - Model 1000B, S-RTP-3
Operation and Calibration of the Shepherd Model 142 Panoramic Irradiator, S-RTP-94, Rev. 2
Control and Issue of Radiation Protection Instruments, S-RAP-RPP-102
Operation and Calibration of the Teletector, S-RTP-16, Rev. 16
Operation and Calibration of the Eberline Ion Chamber Model RO-2/RO-2A Beta Gamma Dose Rate Instrument, S-RTP-52, Rev. 5
Operation and Calibration of the NNC Friskall IIa/IIb/III-20 Whole Body Contamination Monitors, S-RTP-122, Rev. 4
Channel Calibration Test of the DRMS Area Radiation Monitor with G-M Detectors, N2-RTP-111, Rev. 5
Instrument Channel Calibration of High Radiation Reactor Building Ventilation Duct Radiation Monitors, N1-RSP-120, Rev. 6
Core Spray Operability With The Suppression Chamber Dewatered, N1-ST-V2

10 CFR 50.59 Safety Evaluations

94-080	GE Core Shroud Repair Design
SE 2000-059	One Hour Completion Time For Actions (TRM 3.3.9 DOC L.2)
SE 2000-054	RCIC Time Delay Relays Setpoints For Steam Line Isolation
SE 2000-078	WCS Pump Seal Filtration
SE 2000-018	HPCI Trips-Phase 2
SE 2000-024	Emergency Operating Procedure Revision-Primary Containment

<u>10CFR50.59 Safety Screens</u>

N1-01-031	Core Spray Check Valves CKV-40-03,CKV-40-13
N1-ST-V2	Core Spray Operability With Suppression Chamber Dewatered
N1-PM-C3	Electric And Diesel Fire Pump Performance Tests
N2-1PM-RDS-3Y001	Scram Discharge Volume Level
NEP-DES-22	Seismic/Dynamic Qualification
N2-99-006	MSIV Actuator Air Pilot Control Valve Replacement
AR 60560	SWP Valve Packing Glands, Studs And Nuts
AR 60473	Applicability Review RHR Vent Valves Equivalency Evaluation
AR 60607	Applicability Review Replacement Valve Core Spray
AR 43412	Applicability Review Soft Seat Insert To Disc Seat Surface
AR 43387	Applicability Review Shroud Head Bolt Reduction
N1-MPM-ICS-V452	ICS Turbine And Accessories
N1-01-016	RWCU Low-Low Flow System Isolation Delay Timer

<u>Plant Modifications</u>

<u>Barrier Integrity</u>

N1-00-026	Shroud Head Bolt Reduction And Replacement Contingency
N2-00-022	ECCS Rooms Low Temperature Modification
N2-01-122	Downcomer Vacuum Breaker Limit Switches

<u>Mitigation Systems</u>

2F02349	Replacement Of Velan Valves With Edward Valves
2F02341	Equivalency Justification Velan Valve Replacement
2F02346	SWP Valve Packing Glands
1M01120	Add Soft Insert To Disk Check Valve CKV-40-03
N1-00-022	Repeat Trips HPCI Pumps following scram
N1-00-001	Revise Low Suction Setpoints For PS-51-71 & 72
N1-01-016	RWCU Low-Low Flow System Isolation Delay Timer

<u>Deviation Event Reports</u>

NM-2001-5863
1-2001-1103

d. List of Acronyms

ARM	Area Radiation Monitor
ASME	American Society Mechanical Engineers
DBT	Design Basis Threat
DER	Deficiency/Event Report
DVR	Degraded Voltage Relay
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
ESA	Engineering Support Analysis
GDC	General Design Criteria
GE	General Electric
GPM	Gallons Per Minute
ICS	Isolation Cooling System
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MCC	Motor Control Center
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NMPNS	Nine Mile Point Nuclear Station
NRC	Nuclear Regulatory Commission
PI	Performance Indicator
PM	Preventive Maintenance
RCIC	Reactor Core Isolation Cooling
RHS	Residual Heat Removal
RPV	Reactor Pressure Vessel
SCBA	Self-Contained Breathing Apparatus
SDP	Significance Determination Process
SSC	Structures, Systems and Components
SW	Service Water
TCP	Temporary Change Package
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
Unit 1	Nine Mile Point Unit 1
Unit 2	Nine Mile Point Unit 2
WO	Work Order