November 5, 2003

Mr. Peter E. Katz
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P.O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000220/2003005 and 05000410/2003005

Dear Mr. Katz:

On September 27, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection of your Nine Mile Point Nuclear Station, Units 1 and 2. The enclosed integrated inspection report documents the inspection findings which were discussed on October 10, 2003, with Mr. L. Hopkins and other members of your staff.

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

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

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year '02, and the remaining inspection activities for Nine Mile Point were completed in May 2003. The NRC will continue to monitor overall safeguards and security controls at Nine Mile Point.

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

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000220/2003005 and 05000410/2003005
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.:	50-220, 50-410
License Nos.:	DPR-63, NPF-69
Report No.:	05000220/2003005 and 05000410/2003005
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:	June 29, 2003 - September 27, 2003
Inspectors:	G. Hunegs, Senior Resident Inspector B. Fuller, Resident Inspector E. Knutson, Resident Inspector J. Caruso, Senior Operations Engineer L. Cheung, Senior Reactor Inspector C. Colantoni, Reactor Engineer E. Gray, Senior Reactor Inspector J. Jang, Senior Health Physicist G. Johnson, Operations Engineer J. Noggle, Senior Health Physicist T. O'Hara, Reactor Inspector
Approved by:	James M. Trapp, Chief Projects Branch 1 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000220/2003005, 05000410/2003005; 06/29/2003 - 09/27/2003; Nine Mile Point, Units 1 and 2; Permanent Plant Modifications; and Other Activities.

This report covered a 13-week period of inspection by resident inspectors and announced inspections by eight region-based inspectors. Two Green findings were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process," (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding when the number 12 core spray (CS) keep-full system was taken out of service for maintenance without determining the effect of its removal on the operability of the CS train number 12.

  The finding is greater than minor because it is associated with the configuration control attribute of the mitigating system cornerstone and adversely affects the cornerstone objective. Specifically, the reliability of the 12 CS train was reduced due to the increased susceptibility for water hammer that would potentially cause piping damage and affect the capability of the 12 CS train to respond to an initiating event. The finding is of very low safety significance, because it is not a design or qualification deficiency and it does not represent an actual loss of the CS safety function or of a single CS train that contributes to internal or external event (e.g., seismic, fire, flooding, or severe weather) core damage accident sequences. Additionally, there was no evidence of significant draining of the number 12 CS train piping during the time period that the keep-full system was removed from service.

  A contributing cause of the finding was related to the human performance cross-cutting area. Operators removed a core spray keep-full subsystem from service without determining the effect of its removal on the core spray system. (Section 1R17)

- Green. The inspectors identified a non-cited violation of technical specification (TS) 6.4.1.b because the licensee did not develop and validate an emergency operating procedure (EOP) to reflect current plant design. Specifically, EOP-2 "Reactor Pressure Vessel Control Flowchart's" did not direct the operators to bypass the high pressure coolant injection (HPCI) mode feedwater flow control valve low pump discharge pressure interlock to allow the use of the condensate system following a HPCI failure.

  The finding is greater than minor because it is associated with the Mitigating Systems cornerstone attribute of procedure quality and affected the associated cornerstone objective of ensuring the capability of the condensate system, a preferred low pressure injection water source, to 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 and it did not represent an actual loss of the low pressure injection safety function or of a single low pressure injection train that contributes to internal or external events (e.g., seismic, fire, flooding, or severe weather) core damage accident sequences. (Section 4OA5)

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Nine Mile Point Unit 1 (Unit 1) began the inspection period at 100 percent power. On July 2, 2003, an unplanned power reduction to approximately 82 percent occurred due to a trip of a reactor recirculation pump (RRP) 14. The RRP was returned to service on July 3, 2003, with full power operation resumed on July 4, 2003. On August 14, 2003, an automatic reactor scram occurred due to offsite power grid disturbances. The unit was returned to service on August 19. On August 21, power was reduced to 82 percent for unplanned maintenance on RRP 13. Unit 1 operated at 100 percent power for the remainder of the inspection period.

Nine Mile Point Unit 2 (Unit 2) began the inspection period at 100 percent power. On July 24, 2003, Unit 2 experienced a plant transient when a feedwater control circuit power supply failed and the plant automatically scrammed because of an oscillation power range monitor trip signal. The unit was returned to service on August 3. On August 14, 2003, an automatic reactor scram occurred due to offsite power grid disturbances. The unit was returned to service on August 17 and operated at 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

This inspection activity represented two samples. The inspectors completed one sample by examining two risk significant systems to verify that design features and operating procedures support operation of these systems during periods of hot weather. Documents reviewed included the Unit 1 Final Safety Analysis Report (FSAR) and the applicable System Design Basis Documents. The systems that were examined were:

- Service water system (Unit 1)
- Emergency diesel generator raw water system (Unit 1)

The inspectors completed one sample by reviewing one site specific weather related condition. The inspectors reviewed site preparations for inclement weather and high winds prior to Hurricane Isabel and performed plant walkdowns for selected structures, systems and components to evaluate the readiness for impending adverse weather conditons. The inspectors reviewed N2-SOP-90 “Natural Events,” N1-SOP-10 ”High Winds,” N1-OP-64 “Meteorological Monitoring” and N2-OP-102 “Meteorological Monitoring” in order to compare their observations to procedures for inclement weather.

b. Findings

No findings of significance were identified

1R02 Evaluation of Changes, Tests or Experiments (71111.02)

a. Inspection Scope

The inspectors reviewed eight selected safety evaluations associated with design changes that were completed during the past two years at the Nine Mile Point Station (NMPS). The selected safety evaluations were associated with the initiating event, mitigating systems and the barrier integrity cornerstones. These were reviewed to verify that changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR) were evaluated and documented in accordance with 10 Code of Federal Regulations (CFR) 50.59. The inspector also verified that the safety issues pertinent to the changes were properly resolved or adequately addressed. These safety evaluations were selected based on the safety significance of the associated design changes and the risk to structures, systems and components.

The inspectors also reviewed 18 screen-out and equivalency evaluations, for changes, tests and experiments for which the licensee determined that safety evaluations were not required. This review was performed to verify that the licensee's threshold for performing safety evaluations was consistent with 10 CFR 50.59.

This inspection activity represented 26 samples.

The listing of the safety evaluations, screen-outs and equivalency evaluations reviewed is provided in Attachment 1.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q)

a. Inspection Scope

The inspectors performed five partial system walkdowns. The walkdowns were conducted to verify system and component alignment and to note any discrepancies that would impact system operability. This inspection activity represented five samples.

- The inspectors selected the Unit 1 containment spray system to conduct a partial system walkdown based on its safety significance. The walkdown included a physical inspection and switch verification. Operating procedure N1-OP-14, "Containment Spray System," was used for this review.

- The inspectors selected the Unit 1 liquid poison system to conduct a partial system walkdown based on its safety significance. The walkdown included a physical inspection and switch verification. N1-OP-12, “Liquid Poison System,” was used for this review.

- The inspectors selected the Unit 2, division I uninterruptible power supply (UPS) inverter 2VBA*UPS2A to conduct a partial system walkdown, when the division 2 UPS inverter 2VBA*UPS2B was out of service for unplanned maintenance. The walkdown included a physical inspection and switch verification of the UPS. N2-OP-71D, “Uninterruptible Power Supplies (UPS)” was used for this review.

- The inspectors selected the Unit 2 high pressure core spray system to conduct a partial system walkdown, when the reactor core isolation cooling (RCIC) system was out of service for planned maintenance. The walkdown included a physical inspection and switch verification. N2-OSP-M001, “HPCS Discharge Piping Fill and Valve Line-up Verification” was used for this review.

- The inspectors selected the Unit 2 RCIC system in a normally inaccessible area inside the drywell dome. The inspectors evaluated the condition of piping flanges, pipe supports and valves. Various system drawings were used for this review.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors walked down accessible portions of fire areas described below to assess the licensee’s control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers and any related compensatory measures. The condition of fire detection devices, the readiness of the sprinkler fire suppression systems and the fire doors were also inspected against industry standards. In addition, the fire protection features were inspected, including the ventilation system fire dampers, structural steel fire proofing, and electrical penetration seals. Reference material reviewed for installed features included the Unit 1 FSAR and the Unit 2 UFSAR. This inspection activity represented nine samples. The following plant areas were inspected:

- Screenwell intake structure (Unit 1)
- Emergency diesel generator (EDG) rooms (Unit 1)
- EDG power board rooms (Unit 1)
- Containment spray corner rooms (Unit 1)
- Feedwater heater bays (Unit 2)
- Normal switchgear building 261 foot elevation (Unit 2)

- Normal switchgear building 237 foot elevation (Unit 2)
- Control building 237 foot elevation (Unit 2)
- Control building 244 foot elevation (Unit 2)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors completed one sample by reviewing procedures for coping with external flooding and inspected flood protection barriers. Documents reviewed during this inspection included the Unit 1 FSAR, the Unit 2 UFSAR, the Unit 1 and 2 Individual Plant Examination for External Events, Operating Procedures N1-OP-64 and N2-OP-102, “Meteorological Monitoring,” and Emergency Plan Implementing Procedure EPIP-EPP-26, “Natural Hazard Preparation and Recovery.”

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspectors completed one annual review sample by reviewing the Unit 1 containment spray raw water system. The inspectors examined the resolution of heat exchanger performance issues to verify that the heat exchanger performance is acceptable with the allowable raw water temperature specified in the FSAR. The inspectors reviewed the engineering support analysis for DER 1-2003-0867. Other documents reviewed included the FSAR and the containment spray system design basis document.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

The inspectors reviewed a licensed operator requalification training activity which included procedure 71114.06, "Drill Evaluation," simulator-based training evolution, to assess the licensee's training program effectiveness. The inspectors observed Unit 2 licensed operator simulator training on September 8, 2003. The inspectors reviewed performance in the areas of procedure use, self and peer-checking, completion of critical tasks, and training performance objectives. Following the simulator training, the inspectors observed the crew debrief and critique, and reviewed simulator fidelity through a sampling process. The inspectors evaluated emergency response organization performance regarding initial and subsequent actions by licensed operators. This inspection activity represented one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

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

- The Unit 1 control rod drive (CRD) system was selected for review based on several emergent failures of the CRD pumps during the refueling outage.

- The Unit 1 vacuum relief system was selected for review based on several emergent relief valve failures.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

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

The following work items/WOs were reviewed:

- The loss of 14 RRP on July 2 resulted in a power reduction to 82 percent (Unit 1)
- Assessed the effect of planned maintenance on 12 CS keep-full system on operability of the CS system (Unit 1). This issue is further discussed in section 1R17 of this report.
- Assessed the effect of emergent maintenance on 11 CRD pump (Unit 1). The pump was declared inoperable on August 5 due to rising vibration, and the maintenance isolation required that keep-full to the emergency cooling system be removed from service.
- Emergent work on August 21 to clean the 13 RRP exciter collector area due to oil contamination and carbon dust buildup (Unit 1)
- WO-03-11474, Division 2 UPS inverter 2VBA*UPS2B repair (Unit 2)
- WO-03-05858, Replace EDG102 circulating lube oil pump pressure control valve (Unit 1)
- WO-03-01138-00, Generator stator water cooling pump 2GMC-P1B (Unit 2)
- WO-03-05277-00, Division II EDG air starting system (Unit 2)

This inspection activity represented eight samples.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions and Events (71111.14)

The inspectors reviewed personnel performance for three transient operations, two of which are documented below and one which is documented in Section 4OA3 of this inspection report. This inspection activity represented three samples.

Reactor Scram Due to Electric Grid Disturbance (Unit 1)

a. Inspection Scope

On August 14, 2003, at 1611, Unit 1 automatically scrammed from 100 percent power. The cause of the reactor scram was a turbine trip due to a large disturbance in the electric grid. No maintenance or operational activities that could have initiated the event were in progress at the time of the scram. All control rods inserted as designed. Turbine bypass valves and electromatic relief valves (ERVs) opened automatically to relieve reactor pressure. Operators subsequently operated ERVs and the emergency condensers to control reactor pressure. The CRD injection system was used to provide a source of make-up water to aid in controlling reactor vessel water level.

As a result of the grid disturbance, undervoltage conditions occurred on the emergency buses which resulted in each of the two EDGs automatically starting and powering their associated emergency buses as designed. Post scram, non-safety related loads initially remained powered by off-site 115 kV (kilovolt) lines and experienced significant voltage transients. These voltage transients eventually resulted in the loss of all RRPs and circulating water pumps causing the loss of the main condenser. Although there was never a complete loss of both 115 kV off-site lines, the voltage fluctuations were such that off-site power was considered to be unstable.

Because of the grid instability, an unusual event (UE) was declared at 1633. The EDGs continued to power the emergency buses until the grid was determined to be stable. Subsequently, the electrical lineup was returned to normal and the UE was terminated at 0120 on August 15, 2003.

The inspectors responded to the site and observed control room operator response including use of emergency operating procedures and implementation of the emergency plan. As part of the follow up on this event, the inspectors reviewed plant chart recorders, compared procedure requirements to observations of operator performance and held discussions with plant personnel regarding operator control of critical plant parameters.

Reactor Scram Due to Electric Grid Disturbance (Unit 2)

On August 14, 2003, at 1611, Nine Mile Point Unit 2 automatically scrammed from approximately 100 percent power. No maintenance or operational activities that could have initiated the event were in progress at the time of the scram. The scram signal was from a turbine control valve fast closure trip. All control rods inserted as designed. The turbine bypass valves were used to control reactor pressure. The safety relief valves were not challenged by this transient. Reactor water level initially lowered to 157 inches due to the increase in reactor pressure and level transient (shrink) immediately following the scram. Reactor vessel water level then rose rapidly due to the feedwater level control valves locked at 55 percent open due to the loss of electrical power to the valve actuator. The operator manually tripped the B feed pump in anticipation of a level 8 trip signal. The C feed pump was left running and preparations were made to start RCIC. The C feed pump tripped on level 8. RCIC was started and used to control level in the 160 - 200 inch band. The minimum reactor level observed was 157 inches after the reactor scram with a subsequent maximum of 203 inches.

As a result of the grid disturbance, undervoltage conditions occurred on each of three emergency buses which resulted in each of the three divisional EDGs automatically starting and powering their associated emergency buses as designed.

Although both off-site 115 kV lines remained energized, because of the grid instability, a UE was declared at 1700. The UE was terminated at 0734 on August 15, 2003 after off-site power was restored to all three emergency buses.

There were no automatic emergency core cooling system (ECCS) actuations during the transient. Operators started all low pressure ECCS pumps in accordance with procedure, running them on minimum -flow to allow starting additional service water pumps.

The inspectors responded to the site and observed control room operator response including use of emergency operating procedures and implementation of the emergency plan. As part of the follow up to this event, the inspectors reviewed plant chart recorders, compared procedure requirements to observations of operator performance and held discussions with plant personnel regarding operator control of critical plant parameters.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- NIP-ECA-01, Deviation/Event Reports

- GAP-OPS-02, Administration of Operations, Revision 19
- S-ODP-OPS-0116, Operability Determinations

The following issues and licensee documents were reviewed:

- NM-2003-2868, LS-44.2-38, channel 12 scram dump volume(SDV) did not trip as required. This level switch provides the SDV high level scram function (Unit 1)
- NM-2003-2920, CRD flow controller became erratic while controlling in automatic. The inspector reviewed the use of a dedicated operator for manual control of the CRD flow controller to maintain the CRD system operable while not in automatic control (Unit 1)
- NM-2003-3337, Expectation for not declaring control rods inoperable during hydraulic control unit (HCU) instrumentation calibration not defined (Unit 1)
- NM-2003-3994, A small RRP 12 control circuit transient caused a small increase in recirculation flow. This resulted in a corresponding 1-3 megawatt thermal increase in reactor power. The inspector reviewed the acceptability of continued operation at 100 percent power prior to correction of this recurrent problem (Unit 1)
- NM-2003-3522, Reserve B transformer load tap changer failed during plant electrical transient (Unit 2)
- NM-2003-3756, post maintenance testing of diesel fuel oil transfer pump (Unit 2)
- NM-2003-3464, Uninterruptible power supply (UPS) 2B SCR failure (Unit 2)
- NM-2003-3264, Force due to reactor core isolation cooing piping misalignment not evaluated in the pipe stress calculation (Unit 2)

This inspection activity represented eight samples.

b. <u>Findings</u>

No findings of significance were identified.

1R17 <u>Permanent Plant Modifications</u> (71111.17A and B)

a. <u>Inspection Scope</u>

The inspectors reviewed twelve risk-significant plant modification packages selected from among the design changes that were completed within the past two years at both Units 1 and 2. The review was to verify that: (1) the design bases, licensing bases, and performance capability of risk significant structures, systems or components had not been degraded through modifications; and, (2) modifications performed during increased risk configurations did not place the plant in an unsafe condition.

The selected plant modifications were distributed among initiating event, mitigating systems, and barrier integrity cornerstones. For these selected modifications, the inspectors reviewed the design inputs, assumptions, and design calculations, such as instrument set-point, instrument uncertainty calculations, to determine the design adequacy. The inspectors also reviewed field change notices that were issued during the installation to confirm that the problems associated with the installation were adequately resolved. In addition, the inspectors reviewed the post-modification testing, functional testing, and instrument calibration records to determine readiness for operations. Finally, the inspectors reviewed the affected procedures, drawings, design basis documents, and UFSAR sections to verify that the affected documents were appropriately updated.

For the accessible components associated with the modifications, the inspectors also walked-down the systems to detect possible abnormal installation conditions.

This inspection activity represented twelve samples.

The listing of the modifications reviewed is provided in Attachment 1.

In addition, the resident inspectors completed the annual review by reviewing design change N2-03-082, “Replace K611-2FWSN34, 2FWS-PWRS1A, AND 2FWS-PWRS1B with New Redundant Power Supplies.” This design improvement was implemented as a result of the July 24, 2003, scram.

b. Findings

Introduction. A Green finding (FIN) was identified when the number 12 core spray (CS) keep-full system was taken out of service for maintenance without determining the effect of its removal on the operability of the number 12 CS train.

Description. The inspectors identified that prior to taking the keep-full system of the number 12 CS system train out of service for maintenance, on July 31, 2003, the licensee did not perform an operability determination. On July 31, 2003, with Unit 1 operating at 100% power, the licensee isolated and tagged out the keep-full system for the number 12 CS system. This was done to repair leaking threads on a pressure indicator. The keep-full system was tagged out for 6 hours and 55 minutes. The importance of the keep-full system to the successful operation of the CS system is recognized in the Unit 1 FSAR, which states that the keep-full system operation will be continuous during normal operation.

Although during the time that the keep-full system was isolated for maintenance on July 31, 2003, there was no data available to determine the condition of the core spray piping, the licensee stated that the inboard isolation valves leak, as documented in DER 2003-4515, which should prevent a void. The licensee documented that, based on the above and engineering judgement, the CS system was operable. DER 2003-3334 noted that maintaining the piping between the inboard isolation valves and outboard check valves full of water is an important consideration for the CS system to fulfill its safety mission and that controls need to be implemented to assure the piping is maintained full of water. Corrective actions state that for the future, until an analysis is completed, if the keep-full system is inoperable, the core spray system shall be declared inoperable.

The purpose of the keep-full system is to keep the CS injection piping full of water and thus free of air or water vapor, which could cause water hammer upon CS initiation. The occurrence of water hammer in the CS injection piping could cause damage to the piping and possibly disable the system preventing the system from performing its intended accident mitigation functions.

The CS system consists of two separate and independent CS loops to prevent overheating of the fuel following a postulated loss-of-coolant accident (LOCA). Each loop has redundant, active components within itself. This system is designed to accommodate the range of LOCAs from the smallest up to the largest line break as discussed in Section XV of the Unit 1 FSAR. In addition, the CS system provides a source of water supply to the shutdown cooling containment isolation valves to meet 10 CFR 50, Appendix J waterseal requirements.

The CS systems were modified, in 1981, in accordance with a US NRC Safety Evaluation (SER Supporting Amendment Number 44 to Facility Operating License Number DPR-63 Niagara Mohawk Power Corporation Nine Mile Point Nuclear Station, Unit 1, Docket Number 50-220.) This safety evaluation states: “The licensee’s letter of March 2, 1981 proposed installing a keep-full system, a high point vent, and associated containment piping, valves and control circuitry. The keep-full system maintains a water inventory in the CS piping during normal operation. This system should prevent any water hammer if the CS system were activated.”

Analysis. The performance deficiency associated with this event is that an operability determination was not performed for CS with the keep-full system out of service. The finding is greater than minor, because it is associated with the configuration control attribute of the mitigating system cornerstone and adversely affects the cornerstone objective. Specifically, the reliability of the 12 CS train was reduced due to the increased susceptibility for water hammer that would potentially cause piping damage and affect the capability of the 12 CS train to respond to an initiating event.

Using the Phase I of the Reactor Safety SDP the finding is determined to be of very low safety significance, (Green), because it is not a design or qualification deficiency and it does not represent an actual loss of the CS safety function or of a single CS train that contributes to internal or external event (e.g., seismic, fire, flooding, or severe weather) core damage accident sequences. Also, there was no evidence of significant draining of the number 12 CS train piping during the time period that the keep-full system was removed from service.

A contributing cause of the finding is related to the human performance cross-cutting area. Operators removed a core spray keep-full system from service without determining the effect of its removal on the core spay system. This performance deficiency was entered into the corrective action program (DER NM-2003-3334).

Enforcement. No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance because administrative procedures on operability determinations are not specifically included in the regulatory basis for the TS that governs procedure requirements and the allowed outage time for the CS system was not exceeded. **(FIN 05000220/2003-05-01, Operability Determination Not Performed for CS With Keep-Full System Out of Service)**

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- WO03-05283-00 Div II EDG fuel oil transfer pump breaker preventative maintenance (Unit 2)
- WO 03-11474-01 Load test for UPS 2B after silicone controlled rectifier SCR replacement (Unit 2)
- WO 03-10876 for the RCIC injection line following installation of new fasteners, realignment of the piping run, and adjustment of the hangers and seismic restraints (Unit 2)
- Reviewed PMT for the RCIC injection check valve that failed its local leak rate test due to a scratched seat/disc (Unit 2)
- Observed operation of the Division II EDG that was performed as PMT following six EDG instrument calibrations, performed in accordance with N2-IPM-GEN-@001, “Safety Related Loop Calibration” (Unit 2)

This inspection activity represented five samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- N1-ST-Q6B, containment spray system loop 121 quarterly operability test (Unit 1)
- N1-ST-Q1B, CS 121 pump, valve and shutdown cooling water seal check valve operability test (Unit 1)
- N1-IPM-029-008, flange and wide range vessel level calibration, to verify that unsuccessful completion of level transmitter 36-35 calibration did not render the high pressure coolant injection system 11 inoperable (Unit 1)
- N2-OSP-RHS-Q@006, RHR loop C pump and valve operability test (Unit 2)
- N2-OSP-ICS-Q@002, RCIC pump and valve operability test (Unit 2)
- N2-OP-29, reactor recirculation system HPU subloop swap (Unit 2)

This inspection activity represented six samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed three temporary plant modifications to determine whether the temporary changes adversely affected system or support system availability; or adversely affected a function important to plant safety. The inspectors reviewed the associated system design bases, including the FSAR, TS and assessed the adequacy of the safety determination screening and evaluations. The inspectors also assessed configuration control of the temporary changes by reviewing selected drawings and procedures to verify whether appropriate updates had been made. The inspectors compared the actual installations to the temporary modification documents to determine whether the implemented changes were consistent with the approved documented modification. The inspectors reviewed the post-installation test results to verify whether the actual impact of the temporary changes had been adequately demonstrated by the test. The following temporary modifications were reviewed:

- The inspector selected Unit 2 temporary modification N2-03-018, which installed a clamp ring on 2FWS- LV10C and injected leak sealing compound to stop a steam leak. LV10C is the level control valve for the ‘C’ reactor feed pump. The valve had a steam leak at the body to bonnet joint.

- The inspector reviewed Unit 1 temporary modification N1-03-170, which moved the scram discharge volume high level scram input to Reactor Protection system (RPS) channel 12 from the normal float switch (which had failed) to the high-high level alarm float switch.

- The inspector observed installation of a temporary power supply in parallel with the one that failed and initiated the transient that led to the Unit 2 scram on July 24. The temporary modification was accomplished in accordance with WO 03-10799-00.

This inspection activity represented three samples.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety and Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspectors reviewed two work activities in exposure significant areas.

- On August 27, 2003, workers were replacing seals on the Unit 2 reactor water cleanup 1B pump. The inspectors conducted a radiation survey of the work area and temporary lead shield evaluation with respect to documented licensee survey records, electronic dosimeter setpoints specified on radiation work permit number 203089 and temporary shielding package number 01-12034-00. Audio and video remote monitoring of the work activities by radiation protection technicians was reviewed with respect to the high radiation area requirements specified in TS 6.12.

- On August 27, 2003, workers were removing the fuel preparation machine from the Unit 2 spent fuel pool. This work activity was initially a high radiation area work activity with underwater hydrolasing specified to reduce radiation levels as the fuel preparation machine was removed from the spent fuel pool. The inspectors observed continuous radiation protection technician radiation and contamination monitoring of the equipment during removal as required by radiation work permit number 203801 and TS 6.12 requirements.

Walkdowns of the Unit 1 and Unit 2 spent fuel pools were performed to verify that there were adequate physical controls for highly activated reactor hardware in storage.

Walkdowns of the Unit 1 and Unit 2 reactor and turbine buildings were conducted during August 26-28, 2003, to verify the radiological postings using a radiation survey instrument and to verify the adequacy of locked high radiation areas by challenging the locked doors and gates. On August 29, 2003, an inventory of high and very high radiation area keys was conducted at both the Unit 1 and Unit 2 radiological controlled area access points with respect to procedure 5-RAP-RPP-0801 Rev. 15, “High Radiation Area Monitoring and Control” and TS 6.12 requirements.

There were no performance indicator incidents for review and no internal exposures >50 mrem committed effective dose equivalent during 2003.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (71122.03)

a. Inspection Scope

The inspector reviewed the following documents at the following locations to evaluate the effectiveness of the licensee’s Radiological Environmental Monitoring Program (REMP).

- Nine Mile Point Units 1 and 2
- Contractor Laboratory (JAF Environmental Laboratory, Fulton, NY)
- Aquatic and Terrestrial Sampling Contractor, EA Engineering Science and Technology, Oswego, NY.

The REMP requirements are specified in the TSs / Offsite Dose Calculation Manual (TS/ODCM).

J. A. FitzPatrick Environmental Laboratory

- Analytical results for 2003 REMP samples
- Calibration results for gamma, alpha/beta, and tritium measurement instruments
- Implementation of the quality control (QC) program
- Review of the 2002/2003 gamma, alpha/beta, and tritium QC charts
- Implementation of the interlaboratory and intralaboratory comparisons
- Implementation of the environmental thermoluminescent dosimeters (TLDs) program
- Associated sampling and analytical REMP procedures
- Analysis of total uncertainties associated with routine radioassays performed by the JAF Environmental Laboratory (by Corporate Nuclear Northeast)

Nine Mile Point Units 1 and 2

- NMP ODCMs (Unit 1 Revision 23, October 29, 2002 and Unit 2, Revision 23, December 19, 2002) and technical justifications for ODCM changes
- 2002 annual REMP report
- TS/ODCM air samplers calibration results
- Meteorological monitoring instruments for wind direction, wind speed, and temperature for the primary and backup towers calibration results
- 2002/2003 meteorological monitoring data recovery statistics
- 2002 NQA audit report (audit report number 02013) and 2003 quarterly assessment reports (report nos. 03-1Q and 03-2Q) for the REMP/ODCM implementations.

EA Engineering Science and Technology

- Aquatic and terrestrial sampling procedures
- Land use census (for residence and garden for 2003) procedure and the 2002 results

The inspector also toured and observed the following activities to evaluate the effectiveness of the licensee's REMP.

- Observation for the operability of meteorological monitoring instruments at the tower and the control room
- Observation at the licensee's analytical laboratory's activities, JAF Environmental Laboratory
- Observation for air iodine/particulate sampling techniques
- Walkdown for determining whether all air samplers, milk farms, and 25% TLDs were located as described in the ODCM (including control and indicator stations) and for determining the equipment material condition.

Radioactive Material Control Program

a. Inspection Scope

The inspector reviewed the following documents to ensure that the licensee met the requirements specified in the licensee's program for the unrestricted release of material from the radiologically controlled area (RCA):

- Calibration results for the radiation monitoring instrumentation small articles monitor, (SAM-9), including the (a) alarm setting, (b) response to the alarm, and (c) the sensitivity;
- The licensee's criteria for the survey and release of potentially contaminated material using a gamma spectroscopy (calibration efficiency for bulk sample analyses);
- RCA methods used for control, survey, and release
- Bulk sample analyses associated procedures and records of detection
- SAM-9 use at RCA access points

The review was against criteria contained in 10CFR20, NRC Circular 81-07, NRC Information Notice 85-92, NUREG/CR-5569, Health Position Data Base (Positions 221 and 250), and the licensee's procedures.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

40A2 Identification and Resolution of Problems (71152)

a. Inspection Scope

The inspector reviewed calendar year 2002-2003 DERs associated with the REMP to evaluate the effectiveness of the licensee's problem identification and resolution processes in the areas of the REMP. The listing of the DERs reviewed is provided in Attachment A-2.

The inspector reviewed three quality assurance (QA) audit reports (02012, 03-1Q, and 03-2Q) of the radiation protection program during 2003, and 3 DERs (2003-2785, 2003-2902, 2003-3364) that were initiated from June through August 2003 and were associated with the occupational radiation safety cornerstone. The purpose of the review was to evaluate the licensee's effectiveness at properly identifying, characterizing, investigating and resolving problems in implementing the licensee's radiation protection program.

The inspectors reviewed DER action documents associated with 10 CFR 50.59 issues and plant modification issues to ensure that the licensee was identifying, evaluating, and correcting problems associated with these areas and that the corrective actions for the issues were appropriate. The inspectors also reviewed a QA audit and self-assessments related to 10 CFR 50.59 and plant modification activities at the Nine Mile Nuclear Power Plant. The listing of the DERs reviewed is provided in Attachment A-2.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

1.0 (Closed) Licensee Event Report (LER) 50-410/2003-001, Oscillation Power Range Scram Due to Power and Flow Perturbations Resulting From a Power Supply Failure

On July 24, 2003 at 5:50 a.m., a power supply failed resulting in power and flow perturbations which lead to an automatic reactor scram. Unit 2 was at 100 percent power when the power supply failed and was at 45 percent power at the time of the scram. The failed power supply affected main steam flow instrumentation, feed water level control and reactor recirculation flow control. The oscillation power range monitor (OPRM) generated the scram signal when power oscillations met the trip criteria. The plant OPRM system monitors core thermal hydraulic instabilities and analyzes the data. The July 24 transient included a flow runback and recirculation pump downshift with plant parameters settling at about 45 percent rated power and 28 percent rated flow. This condition was in the upper left corner of the power/flow map which is typically susceptible to thermal hydraulic oscillations. The scram resulted from an actual instability event and the OPRM detected the power oscillation and a trip signal was generated as designed.

The inspectors reviewed the LER and DER 2003-3227 to verify that the cause of the July 24, 2003, Unit 2 event was identified and that the corrective actions were reasonable. At the time of the event, the inspectors responded to the control room, reviewed operator logs, plant computer data, and the post scram review report. The inspectors reviewed operator performance to determine if operator response was timely and appropriate. The inspector verified that all control rods fully inserted and that no emergency core cooling systems actuated or should have actuated. In addition, the inspectors reviewed a vendor provided instability event evaluation. The power supply failed because of age related failure of internal components. The licensee's corrective actions listed in the LER were reasonable. The design did not represent a design deficiency. This LER is closed.

2.0 (Closed) Unresolved Item (URI) 50-410/2003-05-02, Request for Enforcement Discretion Related to Inverters (Unit 2)

a. Inspection Scope

On August 12, 2003, the Division II uninterruptible power supply (UPS) inverter 2 B failed and the licensee entered TS LCO 3.8.7 which required that the Unit be placed in hot shutdown within 12 hours and cold shutdown within 24 hours. Troubleshooting revealed a loose gate lead on a SCR. The time to complete the troubleshooting and repairs would require exceeding the TS allowed outage time and consequently the licensee requested enforcement discretion (ED). ED was granted and repairs were completed on August 13 at 5:08 A.M. In a letter dated August 18, 2003, the NRC noted that in a letter dated August 14, 2003, NMP requested that the exercise discretion not to enforce compliance with the actions required by TS limiting condition for operation (LCO) 3.8.7, "Inverters - Operating, " for Unit 2 be granted. The licensee's letter documented information previously discussed with the NRC in a telephone conference at 5:30 p.m. on August 12, 2003. NRC inspectors verified the information submitted to the NRC in the licensee's oral and written request and that compensatory actions had been completed. Inspectors reviewed the request to ensure that it was consistent with NRC policy and guidance. Although the event constituted a violation of TS, it was not the result of a licensee performance deficiency and therefore was not evaluated as a potential finding. URI 50-410/2003-05-02, Request for Enforcement Discretion Related to Inverters. This URI is closed.

b. Findings

No findings of significance were identified.

4OA4 Cross Cutting Aspects of Findings

Section 1R17 describes an operator performance deficiency that was a contributing cause to a finding. Specifically, operators removed the number 12 CS keep-full system from service without determining the effect of its removal on the operability of the number 12 CS train.

4OA5 Other Activities

1.0 (Closed) URI 50-220/2002-06-01: EOP-2 did not provide instruction to bypass the HPCI discharge pressure interlock to allow low pressure condensate injection.

a. Inspection Scope

The inspectors reviewed the URI, associated DER 2002-4256 and licensee corrective actions.

b. Findings

Introduction. A Green non-cited violation of TS 6.4.1.b because the licensee did not develop and validate an Emergency Operating Procedure (EOP) to reflect current plant design. Specifically, EOP-2 did not direct the operators to bypass the High Pressure Coolant Injection (HPCI) mode feedwater flow control valves (FCVs) low pump discharge pressure interlock to allow the use of the condensate system as one of the multiple methods of recovering reactor vessel water level following a HPCI failure.

Description. NRC examiners identified the EOP-2 HPCI interlock during a recent Initial license operator examination and issued URI 50-220/2002-06-01. The EOP-2 HPCI interlock would close the FCVs if HPCI pump discharge pressure was less than 900 psig, as it would be following failure of both HPCI (reactor feedwater) pumps. Following a HPCI failure, the EOPs direct the operators to depressurize the reactor and inject with low pressure sources of water, including the condensate system (preferred due to water quality). The procedures did not indicate that the interlock would not be bypassed to "restore and maintain RPV Water Level."

The licensee initiated DER 2002-4256 to document this problem within their corrective action program and revised EOP-2 to include provisions to bypass the interlock if/when the condensate system is needed for RPV make-up. In addition they have trained/re-trained their operators on the HPCI interlock and the steps necessary to bypass the interlock in order to be able to use the condensate system should no feedwater pumps be available.

Analysis. The performance deficiency was a failure to develop and validate EOPs that satisfied TS 6.4.1.b. Specifically, EOP-2 did not address a hardware interlock that needed to be bypassed in order to use the condensate system to "restore and maintain RPV water level above 61 inches."

The performance deficiency was more than minor because it was associated with the Mitigating systems cornerstone attribute of procedure quality and affected the associated cornerstone objective of ensuring the capability of the condensate system, a low pressure, preferred injection water source, to respond to initiating events to prevent undesirable consequences. Using the Phase I of the Reactor Safety SDP the finding was of very low safety significance, Green, because it was not a design or qualification deficiency and it did not represent an actual loss of the low pressure injection safety function or of a single low pressure injection train that contributes to internal or external event (e.g., seismic, fire, flooding, or severe weather) core damage accident sequences.

Enforcement. TS 6.4.1.b requires that the licensee provide procedures including "The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33." Generic Letter 82-33 required licensees to submit a Procedures Generation Package (PGP). By letter dated March 1, 1984, with revisions in supplementary letters dated April 18, 1986 and March 3, 1987 Niagara Mohawk Power Corporation (the licensee) submitted the PGP for Nine Mile Point Unit 1. In its submittal of March 3, 1987 the licensee described the EOP Evaluation Criteria. As part of this evaluation criteria, under the General

Concept section the licensee was to ensure that the EOPs had “compatibility with plant hardware.”

As part of the HPCI design basis (dating back at least to 1974) an interlock is provided that closes the feedwater regulating valves if discharge pressure of the feedwater pumps is below 990 psig. The interlock was intended to preclude starting a feedwater pump with the reactor at reduced pressure and the feedwater regulating valve(s) open. Under these conditions the feedwater pump could(would) be in a runout condition.

This design (hardware) feature precludes the use of the condensate system following a HPCI initiation (<53" water level) and loss of feedwater pumps. Since the condensate system is the preferred system to “restore and maintain reactor level above 53 inches” this interlock inhibits the use of a “preferred” system. In implementing the GE Generic Guidelines this interlock was not addressed in the Plant-Specific Technical Guidelines, and subsequently was not addressed in the EOPs. In addition, the impact of the hardware on the EOPs was not identified during EOP validation as described in the March 3, 1987 submittal.

Contrary to the above requirements, the HPCI interlock was not addressed in N1-EOP-2. This is a violation of TS 6.4.1.b.

However, because of the very low safety significance and because the licensee included the issue in the corrective action program (DER 2002-4256) the issue is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000220/2003005-03, Failure to Provide for Bypassing the HPCI Interlock in EOP-2.**

2.0 The Inspector Review of World Association of Nuclear Operators Assessment Report

The inspector reviewed the World Association of Nuclear Operators (WANO) peer review which was conducted during the weeks of September 16 and 23, 2002.

4OA6 Meetings, Including Exit

On October 10, 2003, the inspectors presented the inspection results to Mr. L. Hopkins, Plant General Manager, Nine Mile Point, and other members of licensee management. The licensee acknowledged the findings and confirmed that proprietary information was not provided during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

R. Dean, General Supervisor, Design Engineering
G. Detter, Manager, Support Services
J. Gerber, ALARA Supervisor
L. Hopkins, Plant General Manager
J. Jones, Supervisor, Emergency Preparedness
M. Navin, Manager, Site Operations
B. Holston, Manager, Engineering Services
W. Paulhardt, Radiation Protection Manager
B. Randall, General Supervisor, System Engineering
C. Terry, Manager, Quality and Performance Assessment
D. Wolniak, General Supervisor, Licensing

NRC Personnel

W. Schmidt, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000220/2003005-01	FIN	Operability Determination Not Performed for CS With Keep-Full System Out of Service (Section 1R17)
05000410/2003005-02	URI	Request for Enforcement Discretion Related to Inverters (Section 4OA3)
05000220/2003005-03	NCV	Failure to Provide for Bypassing the HPCI Interlock in EOP-2 (Section 4OA5)

Closed

05000220/2002006-01	URI	EOP-2 Did Not Provide Instruction to Bypass the HPCI Discharge Pressure Interlock to Allow Low Pressure Condensate Injection (Section 4OA5)

05000410/2003-001	LER	Oscillation Power Range Scram Due to Power and Flow Perturbations Resulting From a Power Supply Failure (Section 4OA3)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 1R17: Deviation/Event Reports

NM-1999-2526	NM-1999-425	NM-1999-1245	NM-1999-2469
NM-1999-3333	NM-1996-1992	NM-1997-710	NM-2000-1224
NM-2000-3586	NM-2000-3587	NM-2000-4028	NM-1005-1690
NM-2001-1580	NM-2001-1034	NM-2001-257	NM-2001-389
NM-2001-981	NM-2002-917	NM-2002-1201	NM-2002-1915
NM-2002-2057	NM-2002-4112	NM-2002-5205	NM-2003-3334
NM-2003-3492*	NM-2003-3510*		

(Note: “*” indicates inspection related).

Section 2PS3: Deviation/Event Reports for Routine REMP

NM-2002-572	NM-2002-2371	NM-2002-2827	NM-2002-3542
NM-2002-5172	NM-2002-5373	NM-2003-809	NM-2003-3021
NM-2003-3101	NM-2003-3106	NM-2003-3191	NM-2003-3204
NM-2003-3340	NM-2003-2565		

Section 2PS3: Deviation/Event Reports for Meteorological Monitoring Program

NM-2002-887	NM-2002-980	NM-2002-2979	NM-2003-248
NM-2003-251	NM-2003-1717	NM-2003-2048	

Section 2PS3: Deviation/Event Reports for Radioactive Material Control Program

NM-2002-1629	NM-2002-3680	NM-2003-2524	NM-2003-3316
NM-2003-3317	NM-2002-3521	NM-2003-3269	

Section 1R17: Procedures Reviewed

NP-SEV-01, Rev. 10 Applicability Reviews and Safety Evaluations
NAI-SEV-02, Rev. 00 Guidelines for Preparing Safety Evaluations
NAI-SEV-01, Rev. 00 Guidelines for Preparing Applicability Reviews
NAI-DSE-01, Rev.00 10 CFR 50.59 Resource Manual
DDC 2A000024Rev.A Appendix C Seismic Test Procedure NumberSP98P0930/1
DDC 2A000024Rev.A Appendix D Test Procedure NumberTP98P0930/1
DDC 2A000024, Rev.A Appendix E Sequencing Document
N1-ST-Q1B, Rev. 9 Nine Mile Point Nuclear Station Unit 1 Surveillance Test Procedure, CS-121 Pump, Valve, and SDC Water Seal Check Valve Operability Test
N1-ST-Q1A, Rev. 9 Nine Mile Point Nuclear Station Unit 1 Surveillance Test Procedure, CS-111 Pump, Valve, and SDC Water Seal Check Valve Operability Test
N1-ST-M6, Rev. 11 Niagara Mohawk Power Corporation Nine Mile Point Nuclear Station Unit 1 Surveillance Test Procedure CS Keep-full System Verification Test
NIP-CON-01 Design Configuration Control Process
NIP-CON-01, Rev 5 Design and Configuration Control
GAP-PSH-03, Rev 7 Control of On-line Work Activities
GAP-OPS-117, Rev 00 Integrated Risk Management
S-ODP-OPS-0116, Rev 02 Operability Determinations

Section 1R17: Drawings Reviewed

124-789-02	3/2 Way Solenoid Valve
124-790-02	3/2 Way Solenoid Valve
C-1820-C	Emergency Diesel Generator (EDG) Number 102 Starting Air, Cooling Water, Lube Oil and Fuel
C-18007-C	Reactor CS
C-18003-C	Condensate Flow
PID-31A-16, Unit 2	Residual Removal System P&ID, Unit 2
C-18002-C, Unit 1	Main Steam & High Pressure Turbine P&ID, Sheet 1
C-19859-C, Unit 1	Reactor Protection System Elementary Wiring Diagram, Channels 11 & 121 Auto Depressurization - CS, Sheets 18, 18A, 24, and 24A

Section 1R17: Plant Modifications Reviewed

N2-93-002	(PN2Y93MX002), Reactor Stability - Oscillation Power Range Monitor
N1-56-143	RWCU System Fill Connections
N1-99-013	Replace NMP1 EDG RAW Water Heat Exchanger
N1-98-023	Service Water Modifications
N2-98-009	Weld Overlay for Feed Water Nozzle N4D

N2-96-052	Replace SSPV's
N1-01-160	Diesel Fire Pump Intertie to Emergency Service Water and Emergency Diesel Cooling Water Systems
N1-01-050	Replace Eagle Timers in ADS System
N1-56-100	Replacement of Emergency Condenser Timer Relays
N1-01-203	RC Networks for Main Steam Line Break Relays
SC2-0144-93	Deletion of RPS & MSIV Input From Main Steam Radiation Monitors
N2-01-204	Check Valve Internals Modification 2RHS*AOV39A/B

Section 1R02: 10 CFR 50.59 Safety Evaluations Reviewed

SE 2000-006	(From Modification N1-98-023 Service Water Modifications)
SE 2000-080	Parallel Operation of Reactor Recirculation and Shutdown Cooling Pumps
SE 2001-059	Control of RPS Shorting Links
SE 2000-014	(From Modification N1-99-025); Addition of Overpressure Protection of CS System Containment Penetrations X-13A and X-14
99-042	Pressure Regulator Design Deficiency and Mitigating Requirements
SE 96-049	(From Modification SC2-0017-95 Redesign CWS Water Box Vent to Allow Venting from a Low Radiation Dose Area)
2002-026	Internal Modification and Removal of Remote Testing and Position Indication from 2RHS*AOV39A/B
2000-075	Deletion of RPS & MSIV Input From Main Steam Radiation Monitors

Section 1R02: 10 CFR 50.59 Applicability Determinations (Screen outs) Reviewed

N1-99-013	Replace NMP1 EDG RAW Water Heat Exchanger
N1-56-143	RWCU System Fill Connections
N2-02-209	Revise 48-2HVCA01 & 48-HVCB01 Setpoints (Control Room HVAC)
N2-01-120	LPCS Pump 2CSL*P1 Low Discharge Pressure Alarm Set Point Change
N2-96-052	Original SSPV's no Longer Available
N1-01-160	Diesel Fire Pump Intertie to Emergency Service Water and Emergency Diesel Generator Cooling Water Systems
N1-01-050	Replace Eagle Timers in ADS System
N1-56-100	Replacement of Emergency Condenser Timer Relays
N1-01-203	RC Networks for Main Steam Line Break Relays
N2-99-006	DDC 2M11882
N2-00-021	DDC 2M11861A, 2RHS*SOV126 Valve Replacement
N2-03-046	2ABD-V8 replacement valve, Equivalency Evaluation
N2-03-011	2LWS-EV2, Repair of Evaporator Tank, Equivalency Evaluation
N1-01-096	Traveling screens, Equivalency Evaluation EE00421
NI-02-105	Replace PSV 210-87, Equivalency Evaluation EE00355
N1-02-147	Recirc Pump Replacement, Equivalency Evaluation EE00424
N1-03-085	CRD Pump Motor Replacement, Equivalency Evaluation EE00593
N1-03-196	CRD Pump Replacement, Equivalency Evaluation EE00693

Section 1R17: Calculations Reviewed

S15-79-F002	Replacement EDG HX Flow Resistance Effect on RAW Water Pump
S15-79-HTX03	Replacement EDG RAW Water Heat Exchanger Design
S13.1-100F007	Hydraulic Analysis of Diesel Fire Pump Supply to ESW number 11 and Emergency Diesel Cooling Water Systems
S14-40P010	Pipe Stress CS GL-96-06 Modification at Valves IV-40-01& IV-40-10

Section 1R17: Additional Documents Reviewed

Nine Mile Point Unit 1	Docket Number 50-220, DPR-63, TAC Number MB3109 Submittal of Revision 17 to the Nine Mile Unit 1 FSAR (Updated), Including Changes to the QA Program Description, and the 10 CFR 50.59 Evaluation Summary Report
Nine Mile Point Unit 1	Safety Evaluation "Full Power Testing Capability by Addition of a Keep Full System" (1980)
Nine Mile Point Unit 2	Docket Number 50-410, NPF-69 Submittal of Revision 15 to the NMPS, LLC., Unit 2 Updated Final Safety Analysis Report and the 10 CFR 50.59 Evaluation Summary Report (TAC Number MB6536)

SQ Operations Acceptance Form SQAI s-0321
Ultrasonic Test Report; 2-6.05-00-0058; N4D RC Feedwater Nozzle
Ultrasonic Test Report; 2-3.00-00-0122; N4D RC Feedwater Nozzle
Ultrasonic Test Report; 2-3.00-00-0123; N4D RC Feedwater Nozzle
Liquid Penetrant Exam Sheet; 2-3.00-00-0111; N4D RC Feedwater Nozzle
GE Nuclear Energy, Examination Summary Sheet, Nine Mile Point Unit 2 -RF07, Rpt. Number 146
EPRI Review of 2RPV-KB20OL(N4D) Feedwater Nozzle to Safe End Weld, 4/3/2000
SER Supporting Amendment Number 44 to Facility Operating License Number DPR-63 Niagara Mohawk Power Corporation Nine Mile Point Nuclear Station, Unit 1, Docket Number 50-220.
Nine Mile Point Unit 1, TSs
Nine Mile Point Unit 1, FSAR
Nine Mile Point Unit 1, Design Basis Document, CS System
Nine Mile Point Unit 2, UFSAR
Nine Mile Point Design Document Change 1M00943B for Safety Evaluation Number 2000-014
Information Notice 91-50 Supplement 1, Water Hammer Events Since 1991
Information Notice 85-76, Recent Water Hammer Events
SDBD-302, Reactor Protection System Design Basis Document, Revision 2
SDBD-301, ADS Design Basis Document, Revision 2
Spadstimer 2-1, Determine Setpoint allowance for ADS Timers, Revision 1

LIST OF ACRONYMS

CFR	Code of Federal Regulations
CRD	control rod drive
CS	core spray
DERs	deviation event reports
EDG	emergency diesel generator
ERV	electromatic relief valves
FIN	finding
FSAR	final safety analysis report
HPCI	high pressure coolant injection
IR	inspection report
kV	kilovolt
LER	licensee event report
LOCA	loss of coolant accident
NCV	non-cited violation
NMPC	Niagara Mohawk Power Corporation
NMP1	Nine Mile Point Unit 1
NMPS	Nine Mile Point Station
NRC	U.S. Nuclear Regulatory Commission
PMT	post-maintenance testing
QA	quality assurance
QC	quality control
RCA	radiologically controlled area
RCIC	reactor core isolation cooling
RPV	reactor pressure vessel
SDP	significance determination process
SDV	scram dump volume
SCR	silicone controlled rectifier
SE	safety evaluation
SIL	safety information letter
SSCs	structures, systems, and components
TS	technical specifications
UE	unusual event
UFSAR	updated final safety analysis report
WOs	work orders