May 11, 2000

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, 2nd Floor
P.O. Box 63
Lycoming, NY 13093

SUBJECT: **NRC INTEGRATED INSPECTION REPORT NOS. 05000220/2000-001 AND 05000410/2000-001**

Dear Mr. Mueller:

This report transmits the findings of safety inspections conducted by NRC inspectors at the Nine Mile Point Nuclear Station, Units 1 and 2, from February 13, 2000 to April 1, 2000. At the conclusion of the inspection, the findings were discussed with members of your staff.

Overall, the conduct of operations at the Nine Mile Point Nuclear Station reflected an acceptable safety focus. We noted that your staff identified a number of weaknesses associated with station procedures and test controls which contributed to plant problems this inspection period. NMPC corrective actions and management focus in these areas were appropriate.

Based on the results of this inspection, the NRC has determined that three Severity Level IV violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1.a of the Enforcement Policy. The NCVs are described in the subject inspection report and involved an inadequate maintenance procedure which contributed to the improper adjustment of a recirculation pump speed controller at Unit 1, an inadequate maintenance procedure for reassembly of a Unit 2 emergency diesel generator output breaker potential transformer assembly, and an inadequate procedure which contributed to the trip of service water pumps at Unit 2. If you contest these violations or their severity level, you should provide a response within 30 days of the date of this inspection report, with 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, and the NRC Resident Inspector at the Nine Mile Point facility.

In accordance with 10CFR2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available to the public.

Sincerely,

/RA by
William A. Cook Acting For/

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

Docket Nos. 05000220, 05000410
License Nos. DPR-63, NPF-69

Enclosure: NRC Inspection Report Nos. 05000220/2000-001 and 05000410/2000-001

cc w/encl:
G. Wilson, Esquire
M. Wetterhahn, Winston and Strawn
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.
F. Valentino, President, New York State Energy Research
  and Development Authority
J. Spath, Program Director, New York State Energy Research
  and Development Authority
T. Judson, Central NY Citizens Awareness Network

Distribution w/encl: **(VIA E-MAIL)**
H. Miller, RA/J. Wiggins, DRA (1)
M. Evans, DRP
W. Cook, DRP
D. Vito, ORA
G. Hunegs - Nine Mile Point
J. Shea, RI EDO Coordinator
E. Adensam, NRR
P. Tam, NRR
W. Scott, NRR
J. Wilcox, NRR
R. Junod, DRP
Inspection Program Branch (IPAS)
DOCDESK
Region I Docket Room (with concurrences)
Nuclear Safety Information Center (NSIC)

DOCUMENT NAME: G:\BRANCH1\\NMPSTUFF\NMP2000-001.WPD
To receive a copy, indicate in the box: "**C**" = Copy w/ attachment/enclosure "**E**" = Copy w/ attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP	E	RI/DRP	E
NAME	GHunegs/WAC for		MEvans/WAC for	
DATE	05/11/00		05/11/00	

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket/Report Nos.:	05000220/2000-001 05000410/2000-001
License Nos.:	DPR-63 NPF-69
Licensee:	Niagara Mohawk Power Corporation P. O. Box 63 Lycoming, NY 13093
Facility:	Nine Mile Point, Units 1 and 2
Location:	Scriba, New York
Dates:	February 13, 2000 - April 1, 2000
Inspectors:	G. Hunegs, Senior Resident Inspector R. Fernandes, Resident Inspector B. Fuller, Resident Inspector J. Carrasco, Reactor Engineer J. Jang, Senior Health Physicist J. Furia, Senior Health Physicist L. Scholl, Senior Reactor Inspector F. Arner, Reactor Engineer
Approved by:	Michele G. Evans, Chief Projects Branch 1 Division of Reactor Projects

EXECUTIVE SUMMARY

Nine Mile Point Units 1 and 2
05000220/2000-001 & 05000410/2000-001
February 13, 2000 to April 1, 2000

This inspection report included aspects of licensee operations, engineering, maintenance, and plant support. The report covered a seven-week period of resident inspection. In addition, the results of inspections of the radiation protection program, the radiological environmental monitoring and meteorological monitoring programs, and the Unit 2 inservice inspection program were also included in this inspection report.

Operations

On March 3, 2000, during a normal Unit 2 reactor shutdown, control room operators inserted a manual reactor scram from 28 percent power due to the potential loss of the operating feedwater pump. Operators appropriately placed the plant in a stable condition and their actions were safe and conservative. (O1.2)

On March 5, 2000, an operator error while performing a surveillance test resulted in running the number 112 containment spray pump with no discharge flow path at Unit 1. The event had no safety consequence and the pump was not damaged. The error represented a lapse in individual and crew performance. (O1.3)

Maintenance

During the February 2000, Unit 1 maintenance outage, the numbers 13 and 14 reactor recirculation pump mechanical seals were replaced to correct degraded seal performance. Inspection of the old seals indicated that intrusion of foreign material into the seals, combined with normal shaft motion, had accelerated wear of the seals resulting in increased leakage. The NMPC review and resolution of the seal degradation was effective. (M1.1)

On March 2, 2000, at Unit 1, an inadequate maintenance procedure and personnel work practices contributed to the improper adjustment of a recirculation pump speed controller. The improper adjustment resulted in the unexpected reduction of recirculation flow and power of approximately 10 percent. The failure to ensure that an adequate maintenance procedure was prepared and used to perform the work was a non-cited violation. (M1.2)

On March 15, 2000, a 4160 VAC over-voltage condition occurred during the performance of emergency diesel generator loss of off-site power testing. The cause was determined to have been an inadequate procedure which resulted in improper re-assembly of the emergency diesel generator output breaker potential transformer assembly. The failure to have an adequate maintenance procedure for this work activity was a non-cited violation. NMPC performed adequate evaluations of equipment affected by the over-voltage condition, and replaced equipment that was damaged or degraded. (M1.3)

NMPC identified inadequate maintenance planning and procedure pre-requisites contributed to the March 17 and 22, 2000, events during restoration from the Division I electrical bus maintenance outage. The procedural inadequacy was a non-cited violation. NMPC corrective actions for these events were appropriate. (M1.4)

NMPC appropriately implemented the structural weld overlay repair to the feedwater nozzle N4D to safe-end weld, following the prescribed regulatory requirements established in the applicable Code Cases of the ASME Boiler and Pressure Vessel Code and per the NMPC Inservice Inspection Program. Overall Inservice Inspection Program implementation at Unit 2 was acceptable. (M2.1)

Engineering

On March 3, 2000, during Unit 2 scram recovery, the reactor core isolation cooling system was manually started and subsequently automatically tripped on low suction pressure. The low suction pressure was caused by a water hammer due to the discharge piping being partially drained. NMPC identified that system design provided no allowance to keep the system full and appropriately corrected this design deficiency prior to unit restart. (E1.1)

The design documents and technical evaluations for Unit 2 design change N2-98-017, "WCS Appendix R High/Low Pressure Interface," were adequate. 10 CFR 50.59 applicability reviews and safety evaluations were performed in accordance with the guidelines specified in the station procedures. (E.8.1)

Plant Support

Generally effective controls for access to radiologically significant areas and for maintaining occupational exposures as low as is reasonably achievable were established for the Unit 2 refueling outage. (R1.1)

NMPC maintained adequate radioactive liquid and gaseous effluent control programs. The Offsite Dose Calculation Manual contained sufficient specifications and instructions to acceptably implement and maintain the radioactive liquid and gaseous effluent control programs. (R1.2)

The Unit 1calibration programs for effluent radiation monitoring systems, hydrogen monitoring system, and flow rate measurement devices met the technical specification (TS) requirements. The Unit 2 calibration programs for hydrogen monitoring system and flow rate measurement devices met the TS requirements. The Unit 2 radiation monitoring system (RMS) calibration program exceeded TS requirements. The Unit 2 RMS System Engineer performed detailed evaluations for all calibration results and tracked the conversion factors and linearity. (R2.1)

NMPC maintained and implemented an effective routine surveillance test program for air cleaning systems and implemented an effective program for maintaining negative pressure in the radiologically controlled areas (reactor, radwaste, and turbine buildings). (R2.2)

NMPC's Quality Assurance Surveillance Audit and self-assessment programs for effluent control were effectively implemented. The Quality Control program to validate analytical results for radioactive liquid and gaseous effluent control was effective. (R7.1)

TABLE OF CONTENTS

Table of Contents (cont'd)

ATTACHMENTS

Report Details

Summary of Plant Status

Nine Mile Point Unit 1 (Unit 1) began this inspection period at full power with four reactor recirculation loops in operation. On February 22, 2000, one of the four operating reactor recirculation pumps tripped due to an electrical fault resulting in three-loop operation. Unit 1 was shutdown on February 25, 2000, to repair the recirculation pumps. On March 1, 2000, the unit was restarted and was returned to full power on March 2, 2000. Unit 1 ended the inspection report period at full power in five-loop operation.

Nine Mile Point Unit 2 (Unit 2) began this inspection period at 100 percent power. On March 3, 2000, the plant was shutdown to perform a scheduled refueling outage. Unit 2 remained shutdown through the end of the period.

I. Operations

O1 Conduct of Operations [1]

O1.1 General Comments (71707)

Using NRC Inspection Procedure 71707, the resident inspectors conducted frequent reviews of ongoing plant operations. The reviews included tours of accessible areas of both units, verification of engineered safeguards features (ESF) system operability, verification of adequate control room and shift staffing, verification that the units were operated in conformance with Technical Specifications (TSs), and verification that logs and records accurately identified equipment status or deficiencies. In general, the conduct of operations was professional and safety-conscious.

O1.2 Manual Reactor Shutdown (scram) (Unit 2)

a. Inspection Scope (71707)

On March 3, 2000, at 2:17 p.m., while performing a normal plant shutdown, Unit 2 was manually scrammed from 28 percent power. The inspector observed portions of the scram recovery process. The inspector also reviewed the operator logs, post-scram review documentation, and the sequence of events. Additionally, the event was discussed with Unit 2 operations and management personnel.

b. Observations and Findings

On March 3, operators were performing a normal plant shutdown for a refueling outage, when an auxiliary operator reported that the operating feedwater pump had a casing steam leak. The control room supervisor elected to manually scram the reactor due to the potential for loss of the only operating feedwater pump. The operator’s decision was determined to have been a safe and conservative action. During the event, the

[1] Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics. The NRC inspection manual procedure or temporary instruction that was used as inspection guidance is listed for each applicable report section.

inspector noted good command and control. Scram recovery was well controlled. NMPC was continuing to investigate the cause of the feedwater pump casing leak at the conclusion of the inspection.

Plant systems response was as designed. The reactor core isolation cooling (RCIC) system was manually initiated in anticipation of a loss of feedwater flow. Subsequently, the RCIC system automatically tripped on low suction pressure. The licensee determined that the low suction pressure trip was due to partial voiding in the discharge piping and the resultant water hammer pressure transient upon RCIC system initiation (see section E1.1). Following the scram, the feedwater pump was used to maintain reactor vessel water level. There were no emergency core cooling system (ECCS) automatic initiation signals received during the transient. Following the scram, the main generator tripped on reverse power, as designed. A fast transfer of house loads occurred and no electrical system abnormalities were observed. Excluding the problems with the RCIC system, operators executed a routine scram recovery and placed the plant in a stable condition.

In accordance with 10 CFR 50.72, the control room staff made appropriate notifications for the March 3, 2000, reactor protection system (RPS) and ESF actuations, (Event No. 36753).

c. Conclusions

On March 3, 2000, during a normal Unit 2 reactor shutdown, control room operators inserted a manual reactor scram from 28 percent power due to the potential loss of the operating feedwater pump. Operators appropriately placed the plant in a stable condition and their actions were safe and conservative.

O1.3 Operation of Containment Spray Pump with No Discharge Flowpath (Unit 1)

a. Inspection Scope (71707)

On March 5, 2000, containment spray (CS) pump No. 112 was started and operated for approximately 2 ½ minutes with no discharge flowpath. The inspectors interviewed operations personnel, reviewed the operations procedure, and reviewed the root cause analysis and deviation/event report (DER).

b. Observations and Findings

During the performance of quarterly surveillance N1-ST-Q6C, Containment Spray Pump Test, the responsible control room operator failed to open the bypass valve to provide a discharge path for the CS pump, as required by the procedure. The alignment error was not detected until after the pump was started and high discharge pressure coupled with no flow was noted. The CS pump was secured and the surveillance test was suspended. The CS pump had run for approximately 2 ½ minutes without a discharge flowpath. After determining that the valve lineup was incorrect, the operators satisfactorily performed the surveillance test, using concurrent verification of each test procedure step. DER 1-2000-0733 was initiated to enter the event into NMPC's corrective action program. This minor procedural adherence error was not subject to formal enforcement action.

NMPC's root cause analysis identified a number of operator performance shortcomings, in addition to the obvious procedural adherence and inattention to detail error. For example, the control room staff did not notify operations management and engineering personnel of the event until after the surveillance had been reperformed satisfactorily. Engineering personnel subsequently reviewed the test results and vibration data, and found no evidence of pump degradation. Additionally, the responsible operator did not utilize self- or peer-checking during the test.

c. Conclusions

On March 5, 2000, an operator error while performing a surveillance test resulted in running the number 112 containment spray pump with no discharge flow path at Unit 1. The event had no safety consequence and the pump was not damaged. The error represented a lapse in individual and crew performance.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Recirculation Pump Mechanical Seal Degradation (Unit 1)

a. Inspection Scope (62707)

On November 16, 1999, the No. 14 reactor recirculation pump (RRP) developed indications of abnormal mechanical seal operation. The pump was secured and the loop was isolated. No. 13 RRP developed similar indications of abnormal seal operation over the period November 1999 to February 2000. On February 25, NMPC shut down Unit 1 to repair an electrical fault on No. 15 RRP and to repair the degraded seals on Nos. 13 and 14 RRP.

The inspectors reviewed the RRP operating history, the DERs for the seal failures and discussed the seal failure mechanisms with mechanical maintenance and engineering personnel.

b. Observations and Findings

The No. 14 RRP mechanical seals were replaced in October 1999. On November 16, 1999, shortly after restart from the October outage, the seals exhibited signs of failure which resulted in NMPC's decision to secure the pump and isolate the recirculation loop. The No. 13 RRP mechanical seals had been installed approximately seven and one-half years ago and had performed normally until November 1999, when inter-seal pressure had begun to decrease slowly. The seal performance was still acceptable in February 2000. NMPC decided to replace the seals on Nos. 13 and 14 RRPs as part of the forced outage to repair the electrical fault with the No. 15 RRP.

NMPC determined that the cause of the degradation of the No. 14 RRP seal was attributed to a brief period of operation at elevated speed, which may have thermally distorted the seal stator. The distortion of the stator allowed water to leak past the back face, carrying debris which became lodged on the stator surface. The debris acted to erode the seal, increasing seal leakage. Normal shaft motions during pump operation added to the increased leakage.

The cause of the degradation of the No. 13 RRP seal was attributed to gradual erosion of the seal by debris which infiltrated the seal during normal operation. The debris identified during the seal inspections was described as particulate with a color similar to iron oxide. Unit 1 does not have seal injection for the recirculation pumps, as the seal water is supplied directly from the recirculation system. Therefore, any particulate present in the reactor coolant can contaminate the seals. Particulate is present in the coolant from normal wear and corrosion of reactor components and may also be present as a result of maintenance performed on the reactor systems. The last refueling outage for Unit 1 involved in-vessel machining and abrasion of the reactor shroud. The particulate generated as a result of those operations was removed via filtration systems installed near the machining equipment. NMPC postulated that the capture efficiency of the filtration equipment was not adequate to prevent escape of some particulate into the coolant. NMPC plans to replace all of the RRP mechanical seals on a periodic basis during future refueling outages.

c. Conclusions

During the February 2000, Unit 1 maintenance outage, the numbers 13 and 14 reactor recirculation pump mechanical seals were replaced to correct degraded seal performance. Inspection of the old seals indicated that intrusion of foreign material into the seals, combined with normal shaft motion, had accelerated wear of the seals resulting in increased leakage. The NMPC review and resolution of the seal degradation was effective.

M1.2 Unexpected Reduction in Recirculation Pump Flow During Maintenance (Unit 1)

a. Inspection Scope (62707)

On March 2, 2000, at Unit 1, the No. 11 RRP flow decreased unexpectedly during adjustment of the pump speed controller electrical stops. The inspector reviewed the DER, licensee root cause analysis, and interviewed operations and maintenance personnel.

b. Observations and Findings

The reactor recirculation pumps have both mechanical and electric stops to limit maximum pump speed and core flow. During previous reactor operation with a recirculation loop isolated, the stops on the four operating pumps had been adjusted to allow full core flow. After the restoration of the plant to five-loop operation, NMPC restored the RRP stops to their normal position.

The adjustment of the electrical stops required manipulating a potentiometer. During the adjustment, the responsible instrumentation and control (I&C) technician assumed the potentiometer was of the multi-turn variety and directed the operator to turn the potentiometer two full turns clockwise. The potentiometer turned only one-quarter of a turn before hitting its stop. Rather than stop and inform his supervisor of the unexpected response, the I&C technician assumed the potentiometer had been turned in the wrong direction. The technician directed that the potentiometer be returned to its original position and then turned slowly counter-clockwise. This improper adjustment resulted in a reduction in reactor recirculation flow and power of approximately 10 percent. The control room operators immediately stopped any further adjustments pending an investigation of this event.

The licensee's investigation identified that the maintenance procedure did not contain information on potentiometer differences or provide guidance with regard to which direction the potentiometer was to be turned to obtain the desired adjustment of the electrical stop. The failure to provide an adequate work procedure is a violation of 10 CFR 50 Appendix B Criterion V, "Instructions, Procedures, and Drawings." This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the Enforcement Policy **(NCV 05000220-001-01)**. This procedural adequacy violation is in the licensee's corrective action program as DER 1-2000-0688.

c. <u>Conclusions</u>

On March 2, 2000, an inadequate maintenance procedure and poor personnel work practices contributed to the improper adjustment of a Unit 1 recirculation pump speed controller. The improper adjustment resulted in the unexpected reduction of recirculation flow and power of approximately 10 percent. The failure to ensure that an adequate maintenance procedure was prepared and used to perform the work was a non-cited violation.

M1.3 <u>Division I Emergency Diesel Generator (EDG) Overvoltage (Unit 2)</u>

a. <u>Inspection Scope (62707)</u>

On March 15, 2000, at 3:20 pm, during the performance of procedure N2-OSP-EGS-R003, Diesel Generator Loss of Off-site Power with no Emergency Core Cooling System (ECCS), the Division I EDG output voltage increased to 5332 volts alternating current (VAC) rather that the expected 4160 VAC. The inspectors reviewed the DER and root cause analysis; interviewed operations, engineering, and maintenance personnel; and inspected the EDG switchgear.

b. Observations and Findings

Procedure N2-OSP-EGS-R003, simulates a loss of off-site power to the Division 1 electrical buses. After the test conditions were established, the loss of off-site power signal was inserted and, as expected, the alternate feeder breaker to the Division I switchgear (2ENS*SWG101) opened and bus loads were shed. The Division I EDG automatically started and energized the 2ENS*SWG101 bus. Control room operators observed that the 1A service water pump sequenced onto the bus and service water non-essential isolation valves closed, as designed. Operators also observed that the 2ENS*SWG101 control panel bus voltage meter was upscale high (>5250VAC). The EDG was tripped locally using the overspeed trip mechanism. It was later determined that this 4160 VAC bus voltage reached 5332 VAC for approximately 130 seconds.

NMPC personnel investigated the diesel control circuitry and identified that an intermittent open circuit in the potential transformer (PT) secondary (which acts as a feedback circuit to the diesel voltage regulator) caused the high voltage condition. With the open circuit, the voltage regulator sensed zero bus voltage and attempted to raise voltage to the desired 4160 VAC. The open circuit resulted from poor electrical contact on the PT fuse carriage due to inadequate maintenance.

The PT fuse carriage is hinged in the cubicle with a linkage attached to the cubicle door which tilts the carriage upward as the door is opened. As the carriage tilts upward, the primary fuse fingers are disengaged from the primary stabs and the feedback circuit auxiliary fingers disengage from secondary contacts mounted on the floor of the cubicle. The inadequate fuse carriage engagement resulted from a loosely connected linkage assembly which did not provide enough downward force upon closing the cubicle door to fully engage the PT fuse carriage and auxiliary contacts.

The inspector determined that the PT fuse carriage is disconnected as part of the normal personnel and equipment protection markup boundary for maintenance and testing. As part of the markup restoration, the PT drawer linkage assembly must be reconnected and the cubicle door closed to place the PT back into the feedback circuit. Feedback circuit continuity is verified by the markup restoration procedure only. The Division I EDG was successfully run four times prior to this failure during test N2-OSP-EGS-R003. NMPC personnel postulated that vibration in the switchgear coupled with the loose linkage allowed the carriage to become unseated.

The inspector noted that NMPC engineering and maintenance personnel performed walkdown and elementary diagram reviews of Division I 4160, 600, and 120VAC systems to identify equipment that was energized and potentially degraded by the over-voltage condition. Some 120VAC relays and components were replaced, either to correct damaged equipment or as a preemptive replacement. NMPC concluded that Division I equipment exposed to over-voltage was acceptable for operation in Modes 4 and 5. The Division I diesel generator was subsequently tested for proper voltage and speed control with no anomalies noted.

The NMPC root cause analysis identified several weaknesses in the procedure and test control. Included in the licensee identified weaknesses were: 1) the pre-job briefings did not meet NMPC management expectations. Specifically, this test was not treated as a heightened level of awareness (HLA) activity, pre-job briefing checklists were not used, and the pre-job briefing did not cover the contingency of tripping the EDG locally; 2) Operator recognition of the actual over-voltage condition was delayed because they thought it was a meter problem, until they smelled overheated electrical components; 3) The restoration procedure for the PT markup did not provide sufficiently detailed instructions for reassembly of the PT linkage; and, 4) The restoration procedure allowed electricians to cycle the PT cubicle door to obtain a satisfactory continuity check. Licensee follow-up identified that the initial readings were outside the acceptable range and the electricians opened and closed the door three subsequent times to wipe the contacts and obtain a satisfactory reading.

The licensee concluded from their investigation that the breaker drawer linkage assembly had excessive play in its length enabling it to be readily shortened (thus reducing its force on the fuse carriage). This was the result of improper maintenance, related to inadequate procedural guidance. The failure to provide an adequate work procedure is a violation of 10 CFR 50 Appendix B Criterion V, “Instructions, Procedures, and Drawings.” This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the Enforcement Policy (**NCV 05000410-001-02)**. This procedural adequacy violation is in the licensee’s corrective action program as DER 2-2000-1004.

c. Conclusions

On March 15, 2000, a 4160 VAC over-voltage condition occurred during the performance of emergency diesel generator loss of off-site power testing. The cause was determined to have been an inadequate procedure which resulted in improper re-assembly of the emergency diesel generator output breaker potential transformer assembly. The failure to have an adequate maintenance procedure for this work activity was a non-cited violation. NMPC performed adequate evaluations of equipment affected by the over-voltage condition, and replaced equipment that was damaged or degraded.

M1.4 Service Water System Non-Essential Load Isolation and Pump Trip (Unit 2)

a. Inspection Scope (62707)

During restoration from the Division I 4160 VAC electrical bus planned maintenance outage, non-essential load isolations and service water system pump trips occurred (reference Emergency Notification System Event Nos. 36808 and 36823, dated March 17 and 22, respectively). The inspectors reviewed the licensee’s response to these events and the corrective action taken.

b. Observations and Findings

On March 17, while restoring plant systems from the Division I electrical bus outage per procedure N2-PM-@12, Attachment 2, the two Division II service water pumps (2SWP*P1B and P1D) tripped. The licensee's investigation determined that the pumps tripped on low flow due to the closure of the isolation valves to the non-essential service water system loads. The non-essential loads isolation valves closed due a failed relay in the Division I service water pump control logic. NMPC determined that the relay in the logic circuit had failed as a result of the Division I 1A EDG overvoltage event (see Section M1.3) and had gone undetected by the licensee's event follow-up and troubleshooting of affected electrical systems. As a corrective action, all similar relays in the Division I electrical system were replaced.

The second non-essential loads isolation event occurred on March 22, when the electrical jumper used for the electrical bus maintenance failed to maintain contact within the service water pump logic circuit and caused a fuse to blow. The maintenance staff replaced the blown fuse and operators properly restored the service water system non-essential loads header.

NMPC identified that the procedures being used to remove and restore the Division I electrical systems from service did not appropriately specify a service water system pump (SWP) configuration or proper flow requirements as a pre-requisite. This contributed to the March 17 isolation of non-essential service water system loads and the tripping of the Division II service water pumps. The failure to provide an adequate procedure to control maintenance activities is a violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings." This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the Enforcement Policy (**NCV 05000410-001-03)**. This procedural adequacy violation is in the licensee's corrective action program as DER 2-2000-1004.

A third non-essential service water system loads isolation and pump trip event occurred following the conclusion of the inspection period (reference Event No. 36862, dated April 4, 2000) and will be reviewed in a subsequent report. The inspectors determined that NMPC's March 17 and 22 notifications per 10 CFR 50.72 were appropriate and timely and that the service water system response to the associated equipment problems was as expected.

c. Conclusions

NMPC identified inadequate maintenance planning and procedure pre-requisites contributed to the March 17 and 22, 2000, events during restoration from the Division I electrical bus maintenance outage. The procedural inadequacy was a non-cited violation. NMPC corrective actions for these events were appropriate.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Inservice Inspection Program Implementation and Weld Overlay Repair of Feedwater Nozzle to Safe-End Weld Review (Unit 2)

a Inspection Scope (73753)

The inspector reviewed the NMPC Inservice Inspection (ISI) activities scheduled for the Unit 2 refueling outage. The review included NMPC's repair of a reactor pressure vessel feedwater nozzle (N4D) to safe-end weld. The inspector also focused on the licensee's adherence to the applicable ASME Code Cases.

d. Findings and Observations

In 1990, NMPC examined the N4D weld using the first generation of Smart System; no relevant indications were detected. During the 1995 refueling outage, General Electric Nuclear Energy (GENE) reported a planar flaw initiating from the inside surface located in the alloy 182 (Inconel) buttering on the safe-end side of the N4D weld. The flaw was evaluated in accordance with the requirements established in the ASME Code Section XI, Article, IWB-3500, and found to be acceptable. During the 1998 refueling outage, and while performing ultrasonic (UT) examinations, NMPC identified a flaw indication on the reactor pressure vessel (RPV) side of the N4D nozzle to safe-end weld and noted that an indication was located in the Inconel material. This indication was reported to have a maximum depth of 0.29 inches and a circumferential length of 5.3 inches. This exceeded the allowable size specified in Article IWB-3514.3 of Section XI of the ASME Code. Based on these results, NMPC performed further evaluation under Article IWA-3600 (fracture mechanics) which demonstrated the acceptability of the as-found condition for continuing service. The inspector verified that the results of the fracture mechanics evaluations were submitted to the NRC. On June 25, 1998, the NRC issued a Safety Evaluation Report (SER) and concluded that NMPC had performed the flaw evaluation in accordance with the procedures and acceptance criteria established for austenitic piping per paragraph IWB-3640 of the ASME Code.

During the refueling outage, NMPC noted an extension of the previously identified N4D weld indication. Accordingly, NMPC submitted a proposed alternative for the contingency repair of reactor pressure vessel nozzles, following the guidance of Generic Letter (GL) 88-01. The inspector observed that NMPC appropriately applied a full structural weld overlay repair to N4D weld 2RPV-KB20.

The inspector examined the pertinent drawings, plans, and field preparations, and observed that the welding procedure specification (WPS) and the procedure qualification record (PQR) for the weld overlay repair to the ND4 safe-end weld were acceptable. The WPS provided the required essentials, supplementary essentials, and non-essential variables, as prescribed in the ASME Code, Section IX, QW-256. The associated PQR that qualified the WPS listed the actual variables used in making the test sample and listed the test results and the welder performance qualification. The inspector also determined that the NDE personnel engaged in this repair were appropriately qualified.

c. Conclusion

NMPC appropriately implemented the structural weld overlay repair to the feedwater nozzle N4D to safe-end weld, following the prescribed regulatory requirements established in the applicable Code Cases of the ASME Boiler and Pressure Vessel Code and per the NMPC Inservice Inspection Program. Overall Inservice Inspection Program implementation at Unit 2 was acceptable.

III. Engineering

E1 Conduct of Engineering

E.1.1 Reactor Core Isolation Cooling Turbine Trip (Unit 2)

a. Inspection Scope (37551)

On March 3, 2000, while performing a normal reactor shutdown, the reactor was manually scrammed from 28 percent power due to a potential loss of the operating feedwater pump. Shortly after the scram, the operations staff manually initiated the RCIC system to ensure water supply to the reactor in the event of the loss of feedwater. The RCIC turbine momentarily started and then tripped on low suction pressure. The inspectors reviewed the event and NMPC's corrective actions and performed a system walkdown.

b. Observations and Findings

By reviewing information provided by the transient analysis recorder, NMPC was able to determine that the RCIC turbine tripped because of low pressure in the suction side of the pump. Further, NMPC determined that this occurred because of a pressure wave in the piping caused by a water hammer event. The licensee postulated that a portion of the discharge piping was empty and upon pump start and acceleration, a large amount of water moved through the empty piping and when it reached a bend in the piping, caused a rapid pressure increase. NMPC concluded that this pressure wave traveled back through the system to the suction side of the pump and, when it was deflected a second time at the opposite end of the system, caused the suction pressure to decrease enough to trip the turbine.

NMPC determined that the discharge piping may have drained through two leak paths. The piping is connected to the residual heat removal system and the check valve isolating the two sections of piping may have allowed the volume of water in the RCIC discharge piping to drain out over a period of time. In addition, in December 1998, NMPC identified that 2ICS*MOV126, the outboard injection valve, had a valve body to bonnet leak of approximately seven drops per minute, due to a previously damaged pressure seal. An engineering support analysis was completed at that time and concluded that the valve remained operable. However, the potential adverse affect of the leakage from the RCIC system injection piping was not identified at that time. The inspector concluded that this was a missed opportunity.

NMPC performed a review of the event to ensure that no damage was done to the RCIC system. This review included walkdowns of the piping and piping supports. NMPC discovered that the suction pressure gauge was damaged, replaced it, and performed instrument calibrations for the remainder of the system. The inspectors performed independent walkdowns of the piping system, including the drywell, and did not identify any additional damage or concerns.

NMPC corrective actions included performing a root cause analysis of the event and developing a solution to prevent recurrence. NMPC determined that the cause of the event was an inadequate design, in that, there was no allowance to keep the system full. The design analysis assumed that the steam leakage past the inboard testable check valve would condense and maintain the discharge piping downstream of the injection valve full of water. Contrary to this design assumption, there was no leakage or method to check for leakage and condensation. NMPC elected to implement a modification to continuously keep the injection piping full by installing a “keep-full” system. In addition, changes were implemented to the pump protective circuitry and a time delay was added to the pump low suction pressure trip circuit.

c. Conclusions

On March 3, 2000, during Unit 2 scram recovery, the reactor core isolation cooling system was manually started and subsequently automatically tripped on low suction pressure. The low suction pressure was caused by a water hammer due to the discharge piping being partially drained. NMPC identified that system design provided no allowance to keep the system full and appropriately corrected this design deficiency prior to unit restart.

E8 Miscellaneous Engineering Issues

E8.1 Reactor Water Cleanup System (WCS) Modification (Unit 2)

a. Scope (37551)

The inspector performed an in-office review of documentation associated with Unit 2 modification N2-98-017, “WCS Appendix R High/Low Pressure Interface.” The inspection included a review of associated applicability reviews and safety evaluations performed in accordance with the station 10 CFR 50.59 safety evaluation program.

b. Observations and Findings

In 1997 the licensee identified a design deficiency associated with high/low pressure interface valves in the reactor water cleanup system. The deficiency involved the potential for multiple fire-induced faults in a single fire area to cause the spurious opening of redundant interface valves. The failure of redundant valves could result in a loss of coolant accident. Plant modification N2-98-017 was developed to correct the deficiency by adding isolation devices for the affected valves. Detailed design information for this modification was included in design document changes (DDCs) 2E11868 and 2E11869.

The inspector's review of design documents associated with this modification included the initial applicability reviews and safety evaluations performed by the licensee to ensure compliance with the requirements of 10 CFR 50.59, "Changes, Tests and Experiments." The inspector noted that the initial applicability review (AR 25029) was completed on May 20, 1999, and concluded that a 10 CFR 50.59 safety evaluation (SE) was required. Revision zero of the safety evaluation (SE 99-062) was approved on June 25, 1999, and the SE was subsequently revised on July 20, 1999 (Revision one). Revision one corrected errors to the original safety evaluation and the associated licensing document change request.

The inspector subsequently reviewed the licensee's evaluation of changes to the design documents that were made following the performance of the initial AR. For these changes, Section 3.2.8 of procedure NIP-SEV-01, "Applicability Reviews and Safety Evaluations," requires that "changes to proposed activities following completion of an AR shall be evaluated for impact to the AR. If necessary, the AR shall be: 1) Re-performed on a new AR; or 2) Amended by initialing and dating changes, re-signing, initialing and re-reviewing the AR."

The inspector reviewed DDC 2E11869 which was changed on October 22, 1999, by the issuance of Revision A, to allow the use of Unit 1 cable in Unit 2. Revision A to the DDC included a technical justification for the change and concluded that the two cable types were essentially identical in physical and electrical characteristics, as well as conformance with applicable codes and standards.

Based on this review, the inspector questioned whether an additional applicability review was performed or if the original AR was amended as a result of revising DDC 2E11869. The inspector was informed that engineering revised the design change control form for the modification to document the issuance of Revision A to the DDC and at that time determined that the issuance of a new AR or amending the original AR was not necessary. This conclusion by engineering was based on the technical evaluation that determined that the Unit 1 cables were nearly identical to the Unit 2 cables since the cables met the necessary technical requirements and Unit 2 USAR commitments (including the flame test) and the only differences in the cables were minor items such as cable markings.

The inspector noted that step 3.2.8 of procedure NIP-SEV-01 does not include specific criteria for determining when a new AR must be performed or when the original AR must be amended following changes to an activity. As a result, the decision by design engineering to not issue a new AR or amend the existing AR was within the guidance provided by the procedure. The inspector also found that the technical evaluation for the change was adequate to support a conclusion that the revision did not implement a change to the facility as described in the USAR. Therefore, no additional 10 CFR 50.59 safety evaluation was required.

c. Conclusions

The design documents and technical evaluations for Unit 2 design change N2-98-017, "WCS Appendix R High/Low Pressure Interface," were adequate. 10 CFR 50.59

applicability reviews and safety evaluations were performed in accordance with the guidelines specified in the station procedures.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiological Controls During Outage (Unit 2)

a. Inspection Scope (83750)

A health physics inspection during the Unit 2 refueling outage was conducted. Areas of inspection focus were based on the following regulatory requirements from 10 CFR Part 20:

20.1101	Radiation Protection Program
20.1601	Control of Access to High Radiation Areas
20.1602	Control of Access to Very High Radiation Areas
20.1902	Posting Requirements
20.1904	Labeling Containers
20.2103	Records of Surveys

Special focus during this inspection was on radiation work controls for the Unit 2 refueling outage (2RF07).

b. Observations and Findings

For 2RF07, NMPC established an exposure goal of not more than 171.716 person-rem. Through the first two weeks of the outage, exposures were approximately 10 person-rem less than projected. In general, in-service inspections (ISI) were revealing better conditions than previously encountered in past outages, resulting in fewer test failures, and fewer repairs. Since the NMPC outage exposure goal assumed an ISI test failure rate of 7% (11 person-rem), much of the exposure savings observed was attributed to this reduced test failure rate. The only significant scope reduction which had occurred was that the reactor vessel noble metal addition was postponed. The only significant outage scope addition made was a weld overlay for a feedwater nozzle. This scope addition was emergent work based on weld inspection results. The weld overlay was expected to add approximately 2.3 person-rem to the exposure total.

Prior to the start of the outage, the Unit 2 as low as is reasonable achievable (ALARA) group issued the “Nine Mile Point Unit 2 2000 Refueling Pre-Outage ALARA Report.” This document outlined the major activities scheduled for 2RF07, together with the associated radiation work permits (RWPs), dose goals, and for major jobs (greater than 1 person-rem total exposure), a summary of work and the ALARA controls to be implemented.

In support of the outage, 25 contractor radiation protection (RP) technicians were hired, and 10 RP technicians were borrowed from Unit 1. Adequate RP technician staffing was available for the scope of work being performed in the radiologically controlled areas (RCAs).

Work being performed in the drywell, a posted high radiation area, was observed. Appropriate pre-job briefings, including discussions on area dose rates, low dose waiting areas, and responses to alarms were provided to the workers. Alarm set points for the electronic dosimeters worn by the workers were appropriately established based on work scope and the expected area dose rates. Local informational postings were generally useful in identifying higher and lower dose rate areas within larger posted areas.

Observation of NMPC planning efforts to address diving operations in the reactor cavity were also made. Prior to cavity flood-up, a strippable coating was applied to the cavity walls, to aid in cavity decontamination following drain-down. Following cavity flood-up, the coating had become partially dislodged. Detailed plans for the removal and recovery of the damaged coating were developed to minimize exposure and to ensure that divers were kept far away from the fuel located in both the reactor vessel and in the spent fuel pool. Since the flooded cavity had dose rates near the fuel of greater than 500 rads per hour, this evolution was appropriately viewed as entry into a very high radiation area, and appropriate access controls were developed to limit and control activities.

Several minor radiological challenges were effectively resolved by the radiation protection staff. These included one incident where a worker cut his hand while working in the drywell (a posted contaminated area), and significantly elevated radon levels in the RCA due to ventilation system problems. This resulted in numerous personnel contamination alarms at the main access control point.

c. Conclusions

Generally effective controls for access to radiologically significant areas and for maintaining occupational exposures as low as is reasonably achievable were established for the Unit 2 refueling outage.

R1.2 Implementation of the Radioactive Liquid and Gaseous Effluent Control Programs

a. Inspection Scope (84750-01)

The inspection consisted of the following areas for both units: (1) review of radioactive liquid and gaseous effluent release permits; (2) review of selected effluent control procedures; (3) review of the 1998 and 1999 Semi-annual Radioactive Effluent Reports; (4) review of the Offsite Dose Calculation Manual (ODCM); and (5) review of overall effluent program implementation.

The inspection included tours of: (1) the control room; (2) selected radioactive gas and liquid processing facilities and equipment; and (3) effluent radiation monitoring systems (RMS).

b. Observations and Finding

All Technical Specification (TS) and ODCM required effluent radiation monitors and air cleaning systems were operable during this inspection. The radioactive liquid and gaseous effluent release permits were complete as required by the TS/ODCM.

The ODCM provided appropriate descriptions of the sampling and analysis programs for quantifying radioactive liquid and gaseous effluent activities and for calculating projected doses to the public. All necessary parameters, such as effluent radiation monitor setpoint calculation methodologies and site-specific dilution factors, were listed.

The 1997, 1998, and the first half of 1999 Semiannual Radioactive Effluent Reports provided complete data indicating total released radioactivity for liquid and gaseous effluents and projected maximum individual doses resulting from radioactive airborne and liquid effluents. Projected doses to the public were well below the TS/ODCM limits with no anomalous measurements, omissions, or adverse trends.

c. Conclusions

NMPC maintained adequate radioactive liquid and gaseous effluent control programs. The Offsite Dose Calculation Manual contained sufficient specifications and instructions to acceptably implement and maintain the radioactive liquid and gaseous effluent control programs.

R2 Status of RP&C Facilities and Equipment

R2.1 Calibration of Effluent Radiation Monitoring Systems (RMS)

a. Inspection Scope (84750-01)

The inspector reviewed the most recent calibration results for the following list of effluent RMS and flow rate measurement devices. The inspector also reviewed calibration results for the explosive gas (hydrogen) monitoring systems.

Unit 1: RMS

- Liquid Radwaste Effluent Radiation Monitor
- Service Water Effluent Radiation Monitor
- Stack Noble Gas Effluent Monitors (Low and High Ranges)
- Condenser Air Ejector Noble Gas Monitor
- Emergency Condenser System Noble Gas Monitor

Unit 1: Flow Rate Measuring Device

- Stack Gas Flow Rate Measuring Device
- Liquid Radwaste Effluent Line Flow Rate Measurement Device
- Discharge Canal Flow Rate Measurement Device

Unit 1 : Explosive Gas Monitor

- Main Condenser Offgas Treatment System Hydrogen Monitor

Unit 2: RMS

- Liquid Radwaste Effluent Radiation Monitor
- Service Water Effluent Radiation Monitor
- Cooling Tower Blowdown Line Monitor
- Radwaste/Reactor Building Vent Monitors (Low and High)
- Main Stack Gaseous Effluent Monitors (Low and High Range)

Unit 2: Flow Rate Measuring Device

- Cooling Tower Flow Rate Calibration
- Unit 2 Main Stack Air Flow Rate Calibration

Unit 2 : Explosive Gas Monitor

- Main Condenser Offgas Treatment System Hydrogen Monitor

b. Observations and Findings

Unit 1

The I&C, Chemistry, and Radiation Protection departments had the responsibility to perform electronic and radiological calibrations for the RMS. The I&C Department also had the responsibility to perform hydrogen monitor and flow rate measurement device calibrations. All reviewed calibration data were within the licensee's acceptance criteria. The Unit 1calibration programs for effluent radiation monitoring systems, hydrogen monitoring system, and flow rate measurement devices met the TS requirements.

Unit 2

The Radiation Protection Department had the responsibility to perform electronic and radiological calibration for all Unit 2 RMS. All reviewed calibration results, including hydrogen monitoring system and flow rate measurement devices, were within the licensee's acceptance criteria. The Unit 2 RMS calibration program exceeded TS requirements. The RMS System Engineer performed detailed evaluations of calibration results and tracked the conversion factors and the linearity for Unit 2.

c. Conclusions

The Unit 1calibration programs for the effluent radiation monitoring systems, hydrogen monitoring system, and flow rate measurement devices met the TS requirements.

The Unit 2 calibration programs for the hydrogen monitoring system and flow rate measurement devices met the TS requirements. The Unit 2 RMS calibration program exceeded TS requirements. The Unit 2 RMS System Engineer performed detailed evaluations for all calibration results and tracked the conversion factors and linearity.

R2.2 Surveillance Tests for Air Cleaning and Ventilation Systems

a. Inspection Scope (84750-01)

The inspector reviewed the licensee's most recent surveillance test results (visual inspection, high efficiency particulate (HEPA) and charcoal systems leak tests, air capacity test, iodine collection efficiency test, and delta pressure test) for the following systems:

Unit 1

- Control Room Air Treatment System; and
- Reactor Building Emergency Ventilation System.

Unit 2

- Standby Gas Treatment System; and
- Control Room Outdoor Air Special Filter Train System.

The inspector reviewed the licensee’s response to NRC Generic Letter 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal. The inspector also reviewed records of delta pressure relative to the atmosphere for the reactor, radwaste, and turbine buildings.

b. Observations and Findings

All surveillance test results were within the TS/administrative acceptance criteria. The response letter to NRC Generic Letter 99-02 was acceptable. The reactor, radwaste, and turbine buildings were maintained at the negative pressure required by TS.

c. Conclusions

NMPC maintained and implemented an effective routine surveillance test program for air cleaning systems and implemented an effective program for maintaining negative pressure in the radiologically controlled areas (reactor, radwaste, and turbine buildings).

R7 Quality Assurance (QA) in RP&C Activities

R7.1 Review of QA Audits and QC Program

a. Inspection Scope (84750-01)

The inspection consisted of the following areas for both units:

(1) review of the 1999 QA Audit No. 99015, Environmental Protection, Radioactive Effluents, and Radioactive Material Processing; and
(2) review of the implementation of the radioactivity measurement laboratory quality control (QC) program for in-plant and a contractor laboratories.

b. Observations and Findings

The scope and depth of the 1999 QA audit were acceptable. The audit identified minor weaknesses in the area of radioactive liquid and gaseous effluent control programs. None of the findings were assessed to have regulatory significance. NMPC continued the self-assessment process to enhance the radioactive liquid and gaseous effluent control programs.

The QC program consisted of measurements of spiked samples through a vendor-supplied service. No significant discrepancies were evident from QC data for inter-laboratory and intra-laboratory comparisons. When minor discrepancies were found, effective resolutions were determined and implemented.

c. Conclusions

NMPC's QA Surveillance Audit and self-assessment programs for effluent control were effectively implemented. The QC program to validate analytical results for radioactive liquid and gaseous effluent control was effective.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of the licensee management on April 17, 2000. The licensee acknowledged the findings presented.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Niagara Mohawk Power Corporation

R. Abbott, VP Nuclear Engineering
D. Barcomb, Radiation Protection Manager,Unit 2
J. Conway, VP Nuclear Generation
L. Hopkins, Unit 1 Plant Manager
J. Mueller, Senior VP and Chief Nuclear Officer
M. Peckham, Unit 2 Plant Manager
V. Schuman, Radiation Protection Manager, Unit 1
C. Terry, VP Quality Assurance, Nuclear

INSPECTION PROCEDURES USED

IP 37550	Engineering
IP 37551	On-Site Engineering
IP 61726	Surveillance Observations
IP 62707	Maintenance Observations
IP 71707	Plant Operations
IP 71750	Plant Support
IP 73753	Inservice Inspection
IP 83750	Occupational Radiation Exposure
IP 84750-01	Radioactive Waste Treatment, and Effluent and Environmental Monitoring

ITEMS OPENED, CLOSED, AND UPDATED

OPENED AND CLOSED

05000220/2000-001-01	NCV	Inadequate maintenance procedure which contributed to the improper adjustment of a recirculation pump speed controller potentiometer.
05000410/2000-001-02	NCV	Inadequate maintenance procedure for reassembly of a mechanism associated with the Unit 2 emergency diesel generator potential transformer.
05000410/2000-001-03	NCV	Inadequate test procedure which resulted in the trip of a service water pump on low flow.

LIST OF ACRONYMS USED

AR	Applicability Review
ALARA	As Low As Reasonably Achievable
CFR	Code of Federal Regulations
CS	Containment Spray
CRS	Control Room Supervisor
DDCs	Design Document Changes
DER	Deviation/Event Report
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
ESF	Engineered Safeguards Feature
HEPA	High Efficiency Particulate
HLA	Heightened Level of Awareness
I&C	Instrumentation and Control
ISI	Inservice Inspection
NCV	Non-Cited Violation
NMPC	Niagara Mohawk Power Corporation
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
QA	Quality Assurance
QC	Quality Control
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
RMS	Radiation Monitoring System
RP	Radiation Protection
RP&C	Radiological Protection and Chemistry
RPS	Reactor Protection System
RRP	Reactor Recirculation Pump
RWP	Radiation Work Permit
SE	Safety Evaluation
SWP	Service Water Pump
TS	Technical Specification
USAR	Updated Safety Analysis Report
Unit 1	Nine Mile Point Unit 1
Unit 2	Nine Mile Point Unit 2
VAC	Volts Alternating Current
WCS	Water Cleanup System
2RFO7	Unit 2 Refueling Outage