U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 88-18/88-18

Docket No. 50-220/50-410

License No. DPR-63/NPF-69

Licensee: Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Facility: Nine Mile Point, Units 1 and 2

Location: Scriba, New York

Dates: August 25, 1988 through October 3, 1988

Inspectors: W.A. Cook, Senior Resident Inspector
W.L. Schmidt, Resident Inspector
R.R. Temps, Resident Inspector
M. Banerjee, Project Engineer
A.G. Howe, Senior Operations Engineer

Approved by: Jon R. Johnson 10/21/88
J.R. Johnson, Chief, Reactor Projects Section 2C, DRP — Date

INSPECTION SUMMARY

Areas Inspected: Routine inspection by the resident inspectors of station activities including Unit 1 refueling outage activities and Unit 2 power operations, licensee action on previously identified items, plant tours, safety system walkdowns, surveillance testing reviews and allegation followup. This inspection involved 449 hours by the inspectors which included 42 hours of backshift inspection coverage.

Results: Two violations were identified for Unit 2. One violation involves failure of the licensee to take prompt and effective corrective action for an identified snubber surveillance requirement deficiency (see Section 1.2.a). Additional concerns in the area of corrective action are addressed in Section 2.2.b involving the PASS. The second violation identified licensee's failure to follow procedures and is discussed in Section 4.2. An unresolved item concerning the design basis of the core spray system at Unit 1 is discussed in Section 1.1.c. A second unresolved item involving Unit 1 ISI examinations is discussed in Section 1.1.e. A third unresolved item concerning the licensee's final review of Unit 2 Startup Testing Program data potentially impacted by the improper calibration of jet pump instrumentation is discussed in Section 7.

DETAILS

1. Review of Plant Events

1.1 UNIT 1

The unit remains in cold shutdown with the core off loaded. The following events occurred during the inspection period:

a. On September 8, a spike on the Unit 1 normal reactor building ventilation radiation monitor caused an automatic start of the emergency ventilation system. The apparent cause of the ventilation monitor spike was a cable short on an emergency condenser radiation monitor which shares the same power supply. All monitors fed by this power supply were observed to spike. The emergency ventilation system functioned properly and was subsequently secured. The inspector determined that the cable short was subsequently located and the cable replaced.

b. On September 15, the licensee discovered that the dampers on the Technical Support Center (TSC) heating and ventilation control (HVC) system were not functioning properly to maintain a positive pressure. The licensee made the appropriate notifications in accordance with the Emergency Plan and initiated troubleshooting and repair of the dampers. The inspector determined that the damper malfunction was caused by a damper motor failure. The motor was subsequently replaced and normal ventilation restored.

c. On September 16, the licensee reported that the unit may have been operated outside of its design basis for approximately 17 hours on November 10, 1987. The No. 12 core spray system had been declared inoperable due to a leak on a 1-inch pipe weld and a seven day Limiting Condition for Operation (LCO) was entered in accordance with Technical Specifications (TS). Detailed review by the NRC Safety System Functional Inspection (SSFI) Team of the 10 CFR 50 Appendix K Analysis and TS questioned whether the applicable TS LCO is able to ensure the unit is operated within the design basis with one core spray system inoperable.

The TS basis apparently assumed that one core spray system could ensure adequate core cooling/coverage in the event of an accident. The 1975 Appendix K Analysis, however, assumed that both core spray systems are operable to mitigate the consequences of a LOCA and TS were not revised to reflect this analysis. The licensee is performing further investigation and analysis with GE to verify the design basis. This item is unresolved pending further licensee investigation and review of licensee documentation by the NRC UNRESOLVED ITEM (50-220/88-18-01). Details of the NRC review, will be documented in NRC Headquarters Inspection Report No. 88-201.

d. On September 21, a meeting was held at the NRC Region I office with senior Niagara Mohawk management to discuss issues related to Unit 1 restart. Details of this meeting are documented in Meeting Report No. 50-220/88-322.

e. On August 16, the licensee's Quality Assurance (QA) Department issued a Stop Work Order (88-004) to their contractor Nuclear Energy Services (NES) for all examination activities associated with the Unit 1 Inservice Inspection (ISI) Program. The Stop Work Order was based upon multiple deficiencies identified by the QA Department which were characterized as practices contrary to the requirements of Appendix B to 10 CFR 50. These deficiencies are documented in licensee Corrective Action Requests (CARs) 88.2032, 88.2036, 88-2037, and several Surveillance Reports (SRs). The inspector conducted a cursory review of the above stated documents and met with licensee representatives to discuss the scope of Stop Work Order concerns.

Inspector's review of the issues that led to the stop work demonstrated that the QA Department, through their normal surveillance activities, was effectively monitoring and reviewing the work of the contractor. These surveillances led to examples being discovered where contractor personnel did not perform inspections of proper welds or components and used improperly calibrated equipment. In addition, the QA and Engineering Departments' review of the test results led to a high reject rate of the completed work. Based on discussions with the QA manager and the Engineering Department ISI Task Force manager, these types of concerns were being identified and corrected daily, but it was the increasing trend in their number that forced the Stop Work Order to be issued.

The licensee reviewed NES corrective measures and lifted the Stop Work Order on September 14, 1988. The specific ISI Program implementation concerns identified in Stop Work Order 88-004 and licensee corrective actions will be reviewed by NRC specialist inspectors in a future inspection and remain unresolved (50-220/88-18-02).

1.2 UNIT 2

The reactor operated at near 100% power from August 25 until September 2 when the unit entered a forced shutdown. The shutdown was initiated 15 hours into a 72 hour Limiting Condition for Operation for potentially missed snubber inspections (TS 4.7.5). While shutdown, the licensee effected repairs to the main generator stator cooling system which had a cooling water leak of approximately six gallons per hour and posed a potential personnel and equipment hazard. The unit was restarted on September 15 following repair of the stator cooling leak and resolution of the Technical Specification snubber inspection issues. The startup was conducted in the single recirculation loop mode due to the failure of the B recirculation loop discharge isolation valve to fully open. The

unit was manually scrammed on September 22 due a loss of reactor building closed loop cooling (RBCLC) and was again restarted in single loop operation on September 25. The unit operated at power until October 1 and then shutdown for a planned mid-cycle outage. The following highlights the events which occurred during this inspection period:

a. On September 2, the licensee informed the resident inspector that they had identified numerous non-safety related snubbers which had not been inspected in accordance with Technical Specification (TS) 4.7.5. This TS requires that snubbers be visually inspected after two months, but within twelve months of commencing power operations. This inspection includes all snubbers on safety related systems and those snubbers on non-safety related systems that, if they failed, could effect the operation of safety systems. Accordingly, the licensee entered the 72 hour Limiting Condition for Operations action statement for a unit shutdown and then 15 hours later conducted an orderly controlled unit shutdown.

   The inspectors determined that the Maintenance Department utilized a list of snubbers generated by the Engineering Department to conduct these visual examinations prior to July 29 (the end of the 12 month surveillance interval). In conjunction with the Engineering snubber list, the Maintenance Department used the Master Equipment List to assist in identifying, locating, and verifying the snubbers to be inspected. The inspectors determined that the Engineering snubber list was developed from the Preservice Inspection program by the Architect/Engineer. Added to this original list were non-safety related ASME Code continuation snubbers (piping snubbers outside primary containment up to the first anchor). Licensee Engineering representatives indicated that the original December 1987 snubbers list did not receive diligent Engineering Department review. The inspector subsequently determined that this snubber list was not formally controlled in accordance with any Nuclear Engineering and Licensing Department procedures.

   The inspector determined that in May 1988 the Mechanical Maintenance Supervisor requested Quality Assurance (QA) Department review the snubber inspection results to ensure compliance with the TS. Based on QA's review, Corrective Action Request (CAR) No. 88-1017 was generated on July 21. The CAR identified that the Engineering Department list of snubbers requiring visual inspection was incomplete. In addition, CAR 88-1017 specified that there was a concern for the technical basis for determining non-safety related snubbers whose failure would have an adverse effect on any safety related system and whether these snubbers had been included in the list.

   CAR 88-1017 required the Engineering staff to respond by July 27. The Engineering response to the CAR was received by QA on August 2, four days after the inspections should have been completed. This initial response was rejected by QA as not providing 100% assurance

that all the snubbers were identified and inspected per TS. QA required a revised response by August 19. This response was received by QA on August 23, and found acceptable on August 29. The response states that Engineering would complete a 100% verification by the end of the mid-cycle outage.

As a followup to the August 23 CAR response, on or about September 1, the Engineering department assigned a new engineer to assess the adequacy of the TS 4.7.5 snubber list. Initial review of the Technical Specification requirement prompted the engineer to immediately question if all non-safety related system snubbers were included on the list. Licensing and management were informed and concurred that a misinterpretation of the TS requirement had resulted in this group of snubbers being omitted from the original list.

The inspector determined that snubbers in the following categories had been omitted from the list as determined by a detailed licensee review: non-safety related system snubbers which could potentially impact safety systems should they fail (seismic 2 over 1 criteria); safety related snubbers because of transposition errors; safety related snubbers inadvertently missed by the Engineering staff; and multiple snubbers in the same piping node based on comparison between system walkdowns and documentation/calculation reviews.

A total of 148 snubbers were added to the list based on this review. Four snubbers were in safety related systems while the rest were non-safety related. Of these 148 snubbers, 92 required inspection prior to restart because they were not inspected per the original Engineering snubber list or inspected during the Power Ascension Testing Program. Credit could be taken for those snubbers visually inspected during the Power Ascension Test Program. Two of the 92 snubbers requiring inspection were safety related.

Based upon discussion with QA staff members, the inspector concluded that the QA department clearly had doubts as to the compliance with TS 4.7.5.b when CAR 88-1017 was written. The Engineering staff response to this CAR was neither timely nor adequate to prevent exceeding the TS 4.7.5 inspection time requirements. Site management either was not aware of the CAR or did not take prompt and appropriate action to expedite Engineering's resolution of the CAR. Failure to take prompt and effective corrective actions to prevent exceeding the TS 4.7.5 inspection requirements is a violation of Technical Specifications, NMPC Quality Assurance Program Topical Report, Section 16, and 10 CFR 50, Appendix B, Criterion XVI (50-410/88-18-01).

b. On September 12, while the unit was shutdown, the licensee identified a potential operability concern with the emergency starting circuits of the Division I and II Emergency Diesel

Generator (EDGs). The problem was identified during the ongoing update of the Failure Modes and Effects Analysis (FMEA) by Stone and Webster Engineering Corp. The specific concern involves a non-Class 1E relay and switch in the EDG local stop/start circuits which are optically isolated from the Class 1E emergency start/running relays. However, should one of these non-Class 1E components fail (assumed worst failure position) the Light Emitting Diode optical isolators would be actuated and the EDG would either fail to start or stop, if running, because of the existing Class 1E emergency start circuit design. The inspector determined that this potential failure was remedied by lifting the lead from the non-Class 1E relay and switch outputs and revising the EDG operating procedures to address this modification. The local stop capability was still available via a manually operated lever which secures fuel flow to the fuel rack. The inspector reviewed the temporary modification and 10 CFR 50.59 review and found no discrepancies. The inspector also verified appropriate compliance with the unit Technical Specifications until the EDGs' control circuitry could be permanently modified.

c. While shutdown for review of the TS snubber surveillance issue, the licensee performed additional routine surveillance testing. A problem with binding of the B recirculation loop discharge isolation valve (RCS-MOV18B, a 24-inch motor-operated gate valve) was identified on September 10. Preliminary investigation of the valve binding by General Electric Co., Niagara Mohawk and Anchor/Darling Valve Co. engineers indicated that there was probable stem galling which prevented the valve from opening greater than 30 percent.

   The licensee decided to defer repair of the valve and initiated a reactor startup with single loop operation on September 15. The B recirculation loop suction and discharge valves were closed and tagged and appropriate Technical Specifications complied with to initiate single loop power operation.

d. While operating in the single recirculation loop mode, control room operators were performing functional checks and observed that wide range reactor vessel level instruments were deviating by more than ten inches, the normally expected deviation during two loop operation. Information was subsequently provided by GE which explains the deviation as an affect of flow differences during single loop operation on the respective variable leg taps. Revised instrument deviation limits were established in accordance with the GE study. No deficiencies were noted in the licensee's evaluation of this operating phenomena.

e. On September 15, an automatic isolation of the reactor building ventilation and initiation of the B Standby Gas Treatment (SBGT) train occurred while troubleshooting the Above Refuel Floor Radiation Monitor (2HVR*RE14B). Technicians were investigating an equipment failure alarm on the monitor and were checking the fuses when the actuation occurred. All systems responded properly to the

actuation signal and were subsequently restored to the standby condition. The inspectors are monitoring licensee action to resolve this recurring problem with 2HVR*RE14B.

f. On September 20, the unit experienced two different Engineered Safety Feature (ESF) actuations. The first occurred when the Division I EDG was improperly secured following a surveillance test. When the operator went to secure the EDG at the local panel, he accidentally opened the Division I offsite power breaker rather than the switch for securing the EDG. This caused the EDG circuitry to sense a loss of offsite power for that division and it responded normally by going into its ESF lineup. All equipment operated as designed and no adverse impact on the plant resulted.

The second ESF actuation was as a result of a momentary loss of power to radiation monitor 2HVR*RE14B. When power was restored to the monitor, it caused a spike which caused the instrument to go into the alarm condition and resulted in the isolation of reactor building ventilation and actuation of the SBGT system. As noted above, similar problems of this type have occurred with this monitor and the licensee plans to perform in-depth troubleshooting during the October mid-cycle outage.

g. On September 15 the licensee determined that the unit had been operated outside its design bases for total core flow. A Region I specialist inspector was on site during the week of September 19 to review this event and licensee action. This review is documented in section 7 below.

h. On September 22, the reactor was manually scrammed due to a loss of reactor building closed loop cooling (RBCLC). The loss of RBCLC was caused when a service water (SW) alternate cooling outlet valve on a spent fuel cooling heat exchanger was opened. The heat exchanger was initially in a standby lineup (ie. RBCLC inlet open and outlet shut with the alternate SW inlet and outlet shut). The valve was cycled remotely from the control room as part of a post maintenance test. The valve was open for as little as two and one half minutes, but this resulted in short-circuited flow from the RBCLC system to the service water system. This condition caused RBCLC pumps and booster pumps to trip and standby pumps to start and trip due to low discharge pressure. The station shift supervisor ordered the plant scrammed when he observed that RBCLC flow could not be maintained.

The inspector determined that no appreciable increase was seen in recirculation pump temperature due to the loss of RBCLC. Reactor Water Cleanup (WCU) automatically isolated on a high temperature signal downstream of the nonregenative heat exchanger. Concurrently, a bearing on the operating RWCU pump failed. Water hammer was noted once the SW valve was closed and RBCLC repressurized. The licensee performed walkdowns on the RBCLC system and did not find any damage due to the water hammer.

Additionally, the licensee determined that the RBCLC water that was routed to the service water system posed no adverse environmental impact (RBCLC uses untreated fresh water). The inspector verified that the licensee has implemented corrective actions to prevent the inadvertent operation of system cross tie valves, including better review of plant impact prior to conducting Post-Maintenance Testing. The licensee reviewed system configurations and made valve position changes (including additional locking in some cases to prevent a similar event from happening.

i. On September 24, the licensee determined that a deficiency existed in the design of the control room emergency ventilation system. The deficiency could have allowed the exhaust damper to open permitting unfiltered and unmonitored air into the control room during operation in the emergency mode. The licensee has taken action to prevent this condition by tagging out the exhaust damper. Should this damper be required to operate to remove smoke from the control room, the tagout will be cleared.

j. On September 26, the licensee identified that the Class 1E control circuit for the automatic operation of five of six (pumps B,C,D,E & F) service water pump discharge isolation valves contained unqualified relays. The unqualified relays were discovered during field walkdowns of the valves' motor control centers following the receipt of NRC Information Notice No. 87-66, "Inappropriate Application of Commercial Grade Components", dated December 31, 1987. The Information Notice specifically identified the relays in question as Agastat 7000 series relays.

The licensee immediately declared the affected service water pumps inoperable and entered the applicable TS action statements. A one-for-one replacement with qualified Agastat E7000 series relays was promptly initiated in accordance with the licensee's safety-related work control program. The inspector noted no discrepancies with this process; however, he questioned the timeliness of licensee correction action with respect to receipt of the NRC Information Notice. The inspector will review this concern in a subsequent inspection.

The inspector verified that the appropriate NRC notifications were made per 10 CFR 50.72 via the Emergency Notification System for the events described above.

1.3 Above Average Summer Temperatures Station Impact

During the 1988 summer months, above average temperatures were experienced across the nation and in northern New York State. Lake Ontario water

temperatures consequently rose to the high seventies. Lake Ontario provides service water to both units and circulating water to Unit 1. Unit 2 circulating water is via a natural draft cooling tower with makeup from the service water system. In spite of the elevated service and circulating water temperatures no operating restrictions or condenser vacuum problems were experienced at Unit 2. Unit 1 remained in cold shutdown with the core off loaded during the summer months.

2. Followup on Previous Identified Items

2.1 Unit 1

a. In Inspection Report 50-220/88-17 the inspectors identified a concern for the large amount of excess materials stored in the spent fuel pool with respect to any potential impact this excess loading may have on the seismic qualification of the spent fuel storage racks. The licensee provided the inspector with a copy of a 10 CFR Part 21 Evaluation (Number P87-010) which looked specifically at the storage of control rod velocity limiters on top of the high density spent fuel racks. The evaluation concluded that although the condition is not a substantial safety hazard, the control rod velocity limiter could potentially cause damage to the fuel in a seismic event which would prevent its reuse. The report also stated that the staging of the rod velocity limiter on top of the fuel represented a poor housekeeping practice.

The report recommended the following:

1. The site pursue with Fuels and Engineering Departments alternative storage methods.

2. The site review administrative controls to prevent alteration or improper use of safety-related equipment and structures without proper evaluation of the safety significance.

3. Any spent fuel be examined prior to placement in the core.

Further, the licensee has stated that plans are being developed and a contract is out for bids to remove most of the excess materials from the spent fuel pool by the end of 1989. The NRC has no further question on this matter, but will continue to monitor licensee action related to the storage and removal of items from the spent fuel pool.

b. (Open) VIOLATION (50-220/87-21-03): ISI Program Deficiencies. On April 13, 1988 the licensee responded to the March 14, 1988 Severity Level III Notice of Violation and the associated $100,000 Civil Penalty. In the cover letter and the response overview, the licensee committed to improving management effectiveness, a reorganization of the Nuclear Division to increase the accountability of personnel, to strengthen the presence of Site Engineering, and to reorganize the ISI Program.

The licensee states the following in responding to the three specific violations:

1) The licensee admits to not having proper controls in place to ensure that identified flaw indications on safety systems were appropriately dispositioned prior to declaration of system operability and plant startup following the 1986 refueling outage. The licensee commits to having a program in place prior to startup from the current refueling outage to ensure that all ISI required examinations are complete.

2) The licensee admits to not identifying and promptly correcting conditions that are adverse to quality. The use of Deficiency Corrective Action (DCA) reports is recognized as a weakness since these documents did not meet the requirements for resolving nonconforming conditions. The licensee discontinued the use of DCAs and has committed to the following: processing of Occurrence Reports (OR) when nonconforming conditions are identified that could effect system operability; reevaluation of all dispositioned DCAs generated during the 1986 refueling outage; and review of the First Ten Year Interval of the ISI Plan and completion of any additional or missed inspections during the 1988 refueling outage.

3) The licensee admits to not having reported nonconforming conditions to the Operations Department for their evaluation with respect to system operability. The licensee has revised the NonConformance Report (NCR) process to ensure that, if an OR is needed, an OR is written. The licensee committed to issuing procedure changes to ensure that Engineering and Licensing personnel understand the requirements for reporting system operability concerns to the Operations Department.

The licensee's dispositioning of DCAs (open item 50-220/87-21-06) has been partially inspected as documented in Inspection Report No. 50-220/88-09 and remains open. This violation remains open pending further review by the inspector of the corrective actions taken by the licensee.

c. (Closed) UNRESOLVED ITEM (50-220/84-14-12): Post accident sampling system (PASS) solenoid operated valves (SOVs) and associated limit switches scheduled for replacement during the week of August 27, 1984. The inspector reviewed work packages for replacement of solenoid valves (SOV) for IV-122-03, BV-122-04, BV-122-05 and the limit switches associated with IV-122-03. The SOVs were replaced with ASCO model numbers 206-381-2RU and NP 8321A5E respectively

and the limit switches for IV-122-03 were replaced with NAMCO model number EA 180-31302. These replacements were installed in August of 1984. Limit switches for BV-122-04 and BV-122-05 were not replaced and qualification documentation was prepared for them.

The inspector reviewed qualification documentation for the subject SOVs and limit switches including qualification file review checklists, environmental assessment reports, vendor qualification reports, and environmental qualification summary evaluation forms. These documents qualified the subject equipment to a radiation only harsh environment, as the licensee concluded that the equipment was not needed for a high energy line break in the reactor building outside primary containment.

The inspector questioned the basis for not considering thermal aging of the Buna-N ring and boot seal in the micro-switch limit switches. The licensee produced a telephone conversation record with the vendor which indicated that the Buna-N seals are for providing protection against dust, rain, falling dirt and humidity and that degradation of Buna-N, to some extent, would not affect the function of the switch if installed in a clean environment. The limit switches are located in the emergency condenser return valve cubicle. The inspector noted during system walkdowns that poor housekeeping practices were evident in this area. So much dust and debris collected on certain equipment that their nameplates were not visible.

The licensee reassessed their earlier evaluation and provided a thermal aging calculation for the Buna-N seals used in the subject limit switches. The calculation utilized a revised ambient temperature profile for the area based on measurements taken from 1984 through 1987. The licensee plans to incorporate the calculation in the EQ file. The revised ambient temperature profiles are currently being reviewed by the licensee for necessary EQ file update. The licensee stated that no adverse impact to equipment qualification is anticipated. The inspector had no further questions. This item is closed.

d. (Closed) UNRESOLVED ITEM (50-220/84-14-13): Environmental qualification of safety related control switch RMS-122-03D. During a team inspection of the licensee's implementation of the post accident sampling system, the licensee did not provide environmental qualification data for control switch RMS-122-03D which was identified to be environmentally qualified. The licensee subsequently determined that the switch is located in the control room, which is a mild environment, and not within the scope of the

EQ rule. The switch was procured for the environmental conditions of the control room. The inspector verified the location of the switch on the control room Panel F, reviewed procurement documentation and had no further concerns. This item is closed.

2.2 Unit 2

a. (Closed) UNRESOLVED ITEM (50-410/87-45-06): On August 15, the licensee's Quality Assurance Department lifted the two Stop Work Orders (88-001 and 88-002) dealing with control of commercial grade items. The first Stop Work Order was issued on January 19, 1988, to prevent issuance of items that were procured as commercial grade from General Electric for Unit 2. The second was issued on February 1, 1988, to prevent the issuance of all commercially purchased items for both units pending improved control and dedication processes.

These actions were taken because QA had found that commercial grade parts had not been properly dedicated prior to issuance and application in safety-related systems. Subsequently, these parts were identified and dedicated. In order to allow issuance of parts after the Stop Work Orders were in affect, QA personnel were stationed at the storeroom. These inspectors provided independent verification that any commercial grade part had adequately been dedicated prior to issuance.

As of the date that the Stop Work Orders were lifted, all commercial grade item purchase orders had been reviewed. Nonconformance Reports have been issued to Materials Engineering to ensure that the required dedications are performed on the in-stock commercial grade items. The inspector verified that in-stock items have been segregated and tagged to prevent use without dedications. This item is closed based on the inspector review of the licensee's corrective actions.

b. (Open) INSPECTOR FOLLOW ITEM (50-410/87-22-10): Item was open pending licensee action to provide a method for restoring power to the Post Accident Sampling System (PASS) containment isolation valves to ensure sampling capability in the event of a loss of Division I or II power to these solenoid operated valves (SOVs). Procedure N2-EMP-GEN-518, " Temporary Restoration of Power for Post Accident Sampling", was written to install temporary jumpers to energize Div I SOVs from the power supply for DIV II SOVs in the event of a loss of DIV I power, and vice versa for the loss of DIV II power.

The inspector reviewed this procedure and found that it was adequate for the purpose of installing the jumpers and energizing the effected SOVs. However, when the inspector reviewed PASS sampling procedure N2-CSP-13, " Chemical Post Accident Assessment at Unit 2", he found that even with the temporary jumpers installed certain PASS samples would be unobtainable due to other SOVs in the sample path which would be deenergized and closed upon loss of DIV I or II power. Additionally, the inspector identified six (6) SOVs on system print 12177-PID-106A-7 which had their power supplies mislabeled.

The licensee's actions to ensure PASS sampling capability in the event of a loss of DIV I or II power are inadequate and it appears that no effort was made to verify that the problem with loss of power to SOVs identified in the original NRC concern did not exist elsewhere in the PASS system. Also, it appears that no attempt was made to verify that with the temporary jumpers installed, that the PASS sampling procedure would work to allow the various PASS samples to be collected.

The inspector's concerns were discussed with the licensee's Engineering Department on September 12, 1988, yet on September 15, the licensee issued a letter (NMP2L 1164) stating in summary that Niagara Mohawk had taken action to satisfy the NRC concerns in Open Item 50-410/87-22-10 and that they considered the item closable. This indicates that a problem still exists with senior management being informed in a timely manner of problems identified by or to members of their staff.

This item remains open pending additional review by the licensee of the problems identified to them by the NRC and consequent corrective actions.

3. Plant Inspection Tours

During this reporting period, the inspectors made tours of the Unit 1 and 2 control rooms and accessible plant areas to monitor station activities and to make an independent assessment of equipment status, radiological conditions, safety and adherence to regulatory requirements. The following were observed:

3.1 Unit 1

As noted in section 2.1.c above, housekeeping in the area of the emergency condensers was observed to be below standard.

3.2 Unit 2

No discrepancies were noted.

4. Surveillance and Maintenance Review

The inspectors observed portions of the testing listed below to verify that the test instrumentation was properly calibrated, approved procedures were used, the work was performed by qualified personnel, limiting conditions for operations were met, and the system was correctly restored following the testing.

4.1 Unit 1

On August 1, 1988, the licensee initiated a Corrective Action Request (88-2033) based upon a routine QA Department surveillance of Inservice Inspection snubber inspections. The QA surveillance identified several cases of conflicting ISI examination results where the differences represented a change in acceptability of the snubbers examined. In addition, certain examination results were found to be unsatisfactory although not identified as such by the qualified examiner. Lastly, the QA surveillance identified examinations which were inappropriately conducted (ie. visual examinations performed by a qualified, but not technically certified inspector and taken credit for by the technically qualified inspector).

A summary of the licensee's corrective action for this event include the following:

1. Complete and independent investigation of the circumstance involving this issue by the Maintenance and Security Departments.

2. Individual involved will have ASME XI VT 3/4 certification and N45.2.6 certification withdrawn.

3. Previously performed snubber inspections by this individual will be reviewed for evidence of document changes or inadequacy.

4. A complete retraining of all Maintenance Department VT 3/4 inspectors will be conducted.

5. All 1988 refueling outage VT 3/4 snubber inspections per the Unit 1 ISI Plan will be reperformed.

6. Disciplinary action against the individual.

The inspector discussed this event with station management and concluded that corrective actions taken, to date, appear to be appropriate. The inspector will review the licensee's final investigation results in a subsequent inspection period. The adequacy of the reinspection efforts will be reviewed by specialist inspections in conjunction with followup to violation 50-220/87-21-03.

4.2 Unit 2

On September 15, the unit was taken critical in the single loop mode of operation. Unit TS require that several of the protective system setpoints be adjusted to more conservative values when operating in the single loop mode. Two surveillance procedures for accomplishing this were reviewed by the inspector. Procedures reviewed were N2-ISP-RMC-@101, RBM Flow Biased Trip Point Adjustment for Single Loop Operation, and N2-ISP-NMS-@101, APRM Flow Biased Trip Point Adjustments for Single Loop Operation.

The inspector verified that all data was properly recorded and sign offs made in procedure N2-ISP-NMS-@101, but questioned some of the data and sign offs made in procedure N2-ISP-RMC-@101. The inspector found that step 9.1 of the procedure was checked off and signed indicating that all "as left" data obtained and recorded in step 8.2.23 met the criteria of the checklist/data sheet provided for step 8.2.23. However, the inspector found that the "as left" trip values for the Rod Block Backup Trip (channels A and B) were not recorded and reference to a note made instead. The reference basically stated that the values could not be checked due to the trip clamp potentiometer being set at a lower value.

The inspector questioned the licensee as to how the sign off was made for data which was not collected. The licensee's response was that the sign off should not have been made, and in accordance with step 6.1 of the procedure, the SSS and I&C Supervisor should have been informed that the procedure could not be completed as written. The inspector determined that the sequencing of the procedure prevented the "as left" value for the Rod Block Backup Trip from being obtained and recorded at step 8.2.23. The "as left" setpoint was verified to be correct earlier in the procedure, but was not recorded. The failure to record the "as left" data as required by step 8.2.23, and to notify the SSS and I&C Supervisor when a procedural inadequacy existed as required by step 6.1, is a violation of procedure N2-ISP-RMC-@101 and Technical Specification 6.8 (50-410/88-18-02).

The inspector verified that a temporary change to the procedure was subsequently issued which corrected the sequencing error. The I&C technicians reperformed appropriate portions of the procedure to verify that the "as left" settings for the Rod Block Backup trip were correct.

5. Safety System Operability Verification

On a sample basis, the inspectors directly examined portions of selected safety system trains to verify that the systems were properly aligned in the standby mode. The following systems were examined:

5.1 Unit 1

-- Core Spray
-- Emergency Power Supplies

5.2 Unit 2

-- Reactor Building Closed Loop Cooling
-- Residual Heat Removal

No violations were identified.

6. Allegation Followup

During the inspection period, the inspectors conducted interviews and inspections in response to an allegation presented to the NRC. The inspector and licensee actions are noted below:

6.1 Unit 1

Allegation No. RI-88-A-0069

On July 14, 1988, the resident inspector's office received a telephone call from a representative of the New York State Department of Law concerning alleged painting of stainless steel piping at Nine Mile Point (NMP) Unit 1. The state representative stated that his information was from a person who supposedly worked at NMP Unit 1 and had concerns for the improper painting of stainless steel piping in the drywell and torus areas. The painter was specifically concerned for halide migration from the paint to the stainless steel and potential adverse effects. When questioned for more specific information, the state representative indicated he had no further information nor could he contact the alleger for further information as he did not have the alleger's name or address.

Based upon the information provided, there is insufficient basis or information to conduct a detailed inspector followup. The painting of stainless steel pipe is not inappropriate in all applications and halide migration is not a necessary consequence of painting stainless steel. However, during routine inspection tours, the inspectors will be mindful of the potential concern and investigate any questionable painting of stainless steel.

By letter dated August 18, 1988, the NRC informed the New York State Representative of the followup action to his concern. This allegation is closed.

6.2 Unit 2

Allegation No. RI-88-A-0082

On July 29, 1988, the resident inspector's office was contacted by a station employee alleging instances of both sexual and racial harassment in the work place. Specific examples of harassment and intimidation were detailed by the individual for inspector followup. The employee was concerned that job performance was suffering and stated that station management was aware of many of the instances which were communicated to the inspector.

An Allegation Panel meeting was convened in the Region I office to determine the course of action for followup of these alleged instances of harassment. Although the NRC has no specific regulatory authority in the area of sexual or racial harassment, the NRC is concerned for the safe and professional working environment in a nuclear power plant. Accordingly, and with the concurrence of the Allegation Panel, the inspectors discussed the alleged instances of harassment with station and corporate management.

The inspector determined that some of the examples stated by the alleger had, in fact, occurred and were either being dealt with or have been appropriately resolved by Niagara Mohawk management and the individual(s) involved. The inspectors reemphasized their concern for the potential adverse impact of problems of this nature on the safe and professional working environment. The inspectors concluded, based upon these discussions with Niagara Mohawk management, that the licensee was aware of and appropriately handling the alleger's concerns and that the licensee, at several levels of management, was particularly sensitive to this issue and the need to preclude further problems in this area. This allegation is closed.

7. Follow-up of Improper Calibration of Jet Pumps - Unit 2

7.1. Overview

During reviews of the flow calibration procedure on September 15, 1988, the licensee identified that the flow coefficient used in the calibration calculations for two of the four double tap (calibrated) jet pumps was in error. This error resulted in an actual core flow of approximately 104.5 percent when indicated core flow was 100 percent. The error was caused when design data for flow constants determined during laboratory testing was improperly transposed as it was entered into the computer that performs the flow calculations. The error resulted in operation outside the FSAR accident analysis which uses 102.5 percent flow as one of its initial conditions.

The error was discovered during a startup of the unit. When informed of the problem the acting Station Superintendent placed a hold on power changes until the situation was understood and adequate corrective actions implemented. Corrective actions taken by the licensee in order to resume the startup included the following:

-- rerun the jetpump calibration program using the correct input data.

-- recalibrate the jetpump loop flow summer cards and the total drive flow summer cards based on the revised calculations.

-- revise the jetpump and APRM flow operability curves for single loop operation.

The licensee also performed the following corrective actions as soon as practical after commencing power increases:

-- Revise the process computer jet pump flow constants and arrays that compare core flow to drive flow for single loop operation.

-- Adjust the total drive flow limiter.

The licensee also started evaluations of the technical aspects, safety related concerns and startup test effects that were impacted by the calibration error. These evaluations are discussed below.

7.2. Findings and Conclusions

A special inspection was conducted to review the appropriate safety related issues, to assess the adequacy of the licensee's corrective actions, to evaluate the impact on surveillance testing validity, and startup testing validity.

a. Safety Related Issues

The safety related issues reviewed were the effects on the validity of the fuel Minimum Critical Power Ratio (MCPR) thermal limit, the effects of increased flow on core loading and vibration of vessel internals, and the adequacy of the flow related scram and rod block instrumentation.

1) MCPR

Initial design basis analysis was performed for 102 percent of rated core flow. The licensee has recently completed an increased core flow analysis that concludes the plant can operate at up to 105 percent of rated core flow without requiring any changes to current operating limits. This analysis has not been reviewed by the NRC,

but the licensee plans to submit this analysis for review. The process computer computation of the fuel power shape is slightly affected by the flow error, but the calculation of the peak fuel bundle nodal powers that are compared with these limits are not significantly affected. Since the flow input to the process computer was lower than the actual flow, the calculated critical bundle power was lower than the actual critical bundle power. Thus, for flow conditions up to 105 percent of rated, there were conservative calculations of the bundle critical powers and no changes in the MCPR operating limits.

A flow transient occurred in April, 1988 which resulted in an indicated flow of greater than 100 percent. Actual flow has been recalculated based on the recent corrected flow indication data and was determined to have been 105.9 percent of rated. This flow is beyond the increased flow analysis performed by the licensee for core flows up to 105 percent of rated. The licensee has determined a maximum possible change of 0.01 in the MCPR operating limit for this flow. As already stated, the critical power computations were conservative. In addition, the operating history of the core has shown a margin of 0.15 between the operating values for MCPR and the MCPR operating limit. Therefore, the MCPR operating and transient limits were not approached during operation of the reactor with flow at greater than 100 percent of rated.

2) Reactor Vessel Internals Vibration and Loading

During startup testing, internals vibration data was taken at what was actually 104.5 percent of rated core flow and the measured vibration levels were within the acceptance criteria. The increased core flow analysis is applicable to 106 percent of rated core flow for vibration and loading. The previous operation and the flow excursion of April 1988 are within the bounds of the analysis. Thus, vibration and loading limits were not exceeded.

3) Flow Related Scram and Rod Block Instrumentation

Setpoints for the flow biased APRM thermal trip and the APRM and RBM rodblock were affected by the calibration error since rated drive flow to reach rated core flow was determined with erroneous core flow indication. In effect, rated drive flow was determined to be higher than what was actually needed and resulted in establishing flow biased setpoints that were lower than the Technical Specification (TS) requirements. The maximum setpoint clamp is not affected by the calibration error. The APRM flow unit upscale rod block is affected in a nonconservative manner. Allowable TS limits are 108 percent flow nominal and 111 percent flow allowable. The function would not have occurred until 113.94 percent flow was

reached. Exceeding this limit does not represent a safety concern because the increased flow analysis provides an allowable value of 114 percent, no credit is taken in any FSAR Chapter 15 analysis for this feature, and the purpose of the feature is to provide a rod block in the event of a flow unit upscale failure that would disable the flow biased rod blocks.

b. Corrective Actions

The immediate corrective actions taken by the licensee (as indicated above) were appropriate for the short term because they took action to assess the situation before any power changes were made, determined the scope of the problem and initiated the appropriate corrective actions. For long term corrective actions the licensee has committed to the following actions to be completed before the end of the fall outage which is scheduled to start on October 1, 1988:

-- revise the two loop operating map, the jet pump operability curves, and the APRM flow operability curves.

-- develop and implement a permanent plant test procedure for core flow calibration which has adequate procedure controls for checking data entry and calculation and reperform the flow calibration on subsequent restart review and upgrade procedural controls associated with the offline evaluation of thermal limits via the backup core limits evaluation program

-- develop procedure controls for the entry of data to a computer and for the control of design data. The proposed corrective actions are adequate to address the causes and relevant effects of the error. The effectiveness of these corrective actions will be reviewed during subsequent inspections.

c. Impact on Surveillance Testing Validity

The validity of the jetpump surveillance testing was not affected by the calibration error because the test is designed to identify any changes from baseline data and the calibration error would not mask this information. The curves for the tests have been revised to reflect the correct data.

d. Impact on Startup Testing Validity

The impact on startup testing validity has not been fully assessed by the licensee. The effects on the tests vary from (1) no impact to (2) core flow was a specified parameter for data gathering to (3) core flow was a parameter for which acceptance criteria needed to be satisfied. For the core flow determination procedure (SU-35), the flow calculations have

been analytically reperformed in order to take the initial corrective actions. These recalculations provided test results that were closer to the predicted values of the initial analysis. Based on these recalculations and an initial Engineering staff review, the overall impact should be small with some data revisions, but no changes in overall test result acceptability. The licensee has committed to perform a detailed evaluation of the startup test program results and to identify corrective actions as soon as practicable. Review of this information is considered unresolved (50-410/88-18-03). No violations or deviations were identified by the inspector. The proposed licensee actions were found to be adequate to correct the problem. The effectiveness of these actions will be reviewed in future inspections.

8. Assurance of Quality

The licensee findings resulting from their update of the Failure Modes and Effect Analysis concerning the EDG control circuit and the control room emergency ventilation system were found to be sound and properly handled. Alert watchstanding by the Unit 2 control room operators to identify the reactor vessel level instrument divergence was contrasted by the events involving the improper securing of an EDG and the loss of RBCLC leading to a reactor scram. The Quality Assurance Department activities in three areas were noteworthy: action taken at Unit 1 to place a Stop Work Order on the ISI contractor; the determination that the Unit 2 TS snubber list was not complete; and, the determination that Unit 1 snubber inspections were inadequate. The issuance of the Unit 1 Stop Work Order and the Unit 2 snubber surveillance list discrepancies indicate poor Engineering Department oversight and support of site activities. The failure of station and corporate management to properly pursue resolution of the Unit 2 snubber surveillance list resulting in a TS surveillance violation, is another example of a previously identified concern for licensee's ineffective corrective action. In contrast, licensee action upon discovery of the core flow calibrations being inaccurate was conservative and proper. However, this does not change the fact that the data input error should have been caught via an independent verification when it was entered or during Startup Testing review.

9. Exit Meetings

At periodic intervals and at the conclusion of the inspection, meetings were held with senior station management to discuss the scope and findings of this inspection. Based on the NRC Region I review of this report and discussions held with licensee representatives, it was determined that this report does not contain Safeguards or 10 CFR 2.790 information.