U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-220/84-07

Docket No. 50-220

License No. DPR-63 Priority -- Category C

Licensee: Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Facility Name: Nine Mile Point Nuclear Station, Unit 1

Inspection At: Scriba, New York

Inspection Conducted: April 1, 1984 through May 31, 1984

Inspectors:

S. J. Collins for 8/22/84
S. D. Hudson, Senior Resident Inspector Date

S. J. Collins for 8/22/84
W. J. Lazarus, Project Engineer Date

Harry W. Kerch 8/17/84
H. W. Kerch, Mechanical Engineer Date

Approved by: S. J. Collins 8/22/84
S. J. Collins, Chief, Reactor Project Section No. 2C DPRP Date

Inspection Summary:
Inspection on April 1, 1984 through May 31, 1984 (Report No. 50-220/84-07)
Areas Inspected: Routine, onsite, regular, and backshift inspection by the resident inspector and two regional inspectors (148 hours). Areas inspected included: operational safety verification, follow-up on licensee identified items, allegation follow-up, refueling, physical security, plant tours, and maintenance activities.

Results: One violation was identified. (Failure to control tools on the refuel floor).

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DETAILS

1. Persons Contacted

    J. Aldrich, Supervisor, Operations
    W. Connolly, Supervisor, Q.A. Operations
    K. Dahlberg, Site Maintenance Superintendent
    W. Drews, Technical Superintendent
    F. Hawksley, Inservice Inspection Superintendent
    E. Leach, Superintendent of Chemistry and Radiation Management
    T. Perkins, General Superintendent, Nuclear Generation
    R. Raymond, Supervisor, Fire Protection
    T. Roman, Station Superintendent
    B. Taylor, Supervisor, Instrument and Control

    The inspectors also interviewed other licensee personnel during the course of the inspection including shift supervisors, administrative, operations, health physics, security, instrument and control, and contractor personnel.

2. Summary of Plant Activities

    During the inspection period, the plant coasted down from 94% to 81% power at end of its fuel cycle. On March 12, the licensee declared an Unusual Event due to a bomb threat received at the station switchboard. On March 16, the plant was shutdown for a scheduled refueling and maintenance outage.

3. Operational Safety Verification

    a. Control Room Observation

        Routinely throughout the inspection period, the inspector independently verified plant parameters and equipment availability of engineered safeguard features. The following items were observed:

        -- Proper control room manning and access control;

        -- Adherence to approved procedures for ongoing activities;

        -- Proper valve and breaker alignment of safety systems and emergency power sources;

        -- Shift turnover.

    b. Review of Logs and Operating Records

        The inspector reviewed the following logs and instructions for the period April 1-30, 1984 and May 1-31, 1984:

-- Control Room Log Book

-- Station Shift Supervisors' Log Book

-- Station Shift Supervisor's Instructions

The logs and instructions were reviewed to:

-- Obtain information on plant problems and operation;

-- Detect changes and trends in performance;

-- Detect possible conflicts with technical specifications or regulatory requirements;

-- Assess the effectiveness of the communications provided by the logs and instructions; and

-- Determine that the reporting requirements of technical specifications are met.

No violations were identified.

4. Allegation Followup

a. NRC Region I received an anonymous allegation in which the alleger indicated that he thought that the Reactor Operator requal exam grades, for the exams given in March, 1984 were abnormally high and that the exam questions/answers were passed to subsequent shifts which took the exam over a three day period. The inspector reviewed the results of the requal exams given on March 12, 13 and 14, 1984 and noted that the exam grades were considerably higher than those for the previous year. In an attempt to determine possible reasons for the higher exam grades, a comparison was made between the exams given on the three days. It was noted that three different exams were administered, however some exam questions were repeated on at least two out of three of the exams. Based on a review of the exam grades, there was no indication that those persons taking the exams on the second and third day were any better prepared than those taking the first exam, as the median grades appeared to be about the same. A comparison of the questions that appeared on the requal exams with the questions which had been administered during the previous quizzes for Cycle V (Reactor Theory), Cycle VI (Instrumentation and Controls), and Cycle VII (Plant Design and Safety Systems) was conducted. This comparison identified a high percentage of exam questions which were identical to questions that had been previously asked on quizzes, particularly in the Reactor Controls and I & C areas. Although quiz and exam questions are drawn from the same bank, the licensee was unaware of the high incidence of repetitive questions. The licensee agreed to reevaluate the manner in which questions are selected as a means to lessen the incidence of exact duplication of questions. Other factors which may have had an impact on the exam grades were operator sensitivity to recent use of NRC prepared requal exams, the availability of a question bank consisting of several hundred exam questions for the operators to study, and the change in emphasis in the "Procedures" section of the exam from administrative procedures on the previous exam, on which scores were generally lower, to operating procedures on this exam.

Based on the results of this review, no exam irregularities were identified.

b. On May 1, 1984, NRC Region I received an allegation concerning the use of untrained temporary employees as part of the fire brigade. NRC regulations require that a trained 5 man fire brigade be on site at all times. The inspector discussed the allegation with the station's Fire Protection Supervisor. Temporary employees are not being used. The licensee had considered hiring temporary employees for use as fire watches while repairs are performed on fire barrier penetrations but the licensee decided not to use them. Auxiliary operators from Unit 2 are being used to supplement each shift of fire brigade members as fire watches. During the current outage, the fire brigade has been split into two 12 hour shifts. Each shift consists of at least 13 trained firemen and 6 auxiliary operators.

After the current inspection period, the inspector learned that the licensee now plans to use temporary employees as fire watches starting June 10, 1984. The fire brigade members will return to a 40 hour work week. Forty temporary employees have been hired and trained specifically for use as fire watches during the repair of the fire barrier penetrations. These temporary employees will have no fire-fighting duties. Ten temporary employees will be assigned to each of the four rotating shifts. Five trained firemen will also be assigned to each shift as required by Technical Specifications.

No unacceptable conditions were identified.

c. On February 22, 1984, NRC Region I received an allegation regarding protective clothing that was contaminated when issued for use. The licensee uses protective clothing made of heavy cloth fabric. They are laundered to remove loose surface contamination prior to reuse. The licensee has established an administrative limit of 4000 counts/minute for fixed contamination that may be present on the clothing. The inspector witnessed the surveying of laundered clothing and determined that the licensee is properly controlling protective clothing. The inspector also independently measured the fix contamination of ten pieces of protective clothing which had been placed in the plant for use. All were found to be satisfactory.

No unacceptable conditions were identified.

5. Followup on Licensee Identified Items

a. On March 27, 1984, with the plant in the refueling mode, the licensee discovered a leak of approximately 15 drops/minute from a Control Rod Drive housing No.46-27 under the reactor vessel. The leak was reported to the NRC in accordance with 10 CFR 50.72. A slight leak (less than 1 drop/minute) was noted at CRD No. 14-11. After sections of the reactor vessel lower head insulation were removed, the licensee conducted a visual inpection of all CRD housings just below the reactor vessel. Seven additional CRD's were found to have rust on the outside of their housings indicating that these CRD's may have leaked in the past.

The licensee then conducted an underwater video taped inspection of selected CRD stub tubes. These stub tubes were furnace sensitized during the reactor vessel manufacturing process and therefore they have increased suspectibility to intergranular stress corrosion cracking. The licensee's sample of stub tubes for their detailed inspection included the 2 CRD housings that were leaking, the seven which had rust on their housings, and three that showed no indication of prior leakage. The results of this inspection are listed below:

HOUSING NO.	PRESENCE OF LEAKAGE	RESULTS OF TV INSPECTIONS
46-27	15 drops per minute leakage	Significant crack, 270 degrees in extent
14-11	Evidence of leakage (wetness) around housing-head annulus	Crack 200 degrees observed
30-07	Suspected of prior leakage	Tight crack approximately 330 degrees around circumference observed
34-07	Suspected of prior leakage	35 degrees long tight crack
42-39	Suspected of prior leakage	No evidence of cracks
38-31	Suspected of prior leakage	No evidence of cracks
34-39	Suspected of prior leakage	No evidence of cracks

10-27	Suspected of prior leakage	Light crack approximately 100 degrees long
46-39	Suspected of prior leakage	No evidence of cracks
34-35	No evidence of leakage	No evidence of cracks
10-23	No evidence of leakage	No evidence of cracks
10-31	No evidence of leakage	Light crack approximately 100 degrees long

Note: Observed cracks were generally horizontally oriented and approximately 1/2 to 1 inch below the elevation of the CRD housing to stub tube field weld.

On April 10, 1984, the licensee presented its findings and plans to roll the CRD housing to the reactor vessel at a meeting with representatives of the Office of Nuclear Reactor Regulation (NRR). NRR is evaluating the adequacy of the licensee actions. Their initial review indicates that the licensee's actions are acceptable. As part of its roll repair procedures, the licensee performed ultrasonic examination of the CRD housings to verify that the cracks in the stub tubes had not propagated into the CRD housing.

The licensee rolled nine CRD housings, then performed a hydrostatic test at 110% of normal operating pressure. No leakage was observed from the nine reactor vessel penetrations.

However, a small leak (6 drops/minute) was observed from another CRD housing that had not been previously inspected. This housing was then rolled into the reactor vessel. Another reactor vessel hydrostatic test will be performed prior to start-up. The licensee will submit its plans for future examination of the CRD housings to NRR by October 1, 1984.

b. On April 3, 1984, during an underwater video inspection, the licensee detected cracks in 10 of 12 Source Range Monitor and Intermediate Range Monitor (SRM/IRM) nuclear instrumentation dry tubes. On 5 dry tubes, the cracking was extensive, extending 180 degrees around the tube. In each case, the cracking was located in the spring plunger housing assembly. This is not part of reactor vessel pressure boundary. The licensee has elected to use the dry tubes without making any repairs until the next refueling outage. The inspector reviewed the licensee's safety evaluation No. 84-47. The licensee's analysis addressed the effects of the cracks on possible loose parts in the vessel, increased fuel channel wear, seismic performance, SRM/IRM operability, fuel loading and dry tube removal at a subsequent outage. It also considered the effects of flow induced vibration causing the cracks to progagate. It further postulated that if a crack was to propagate into the reactor pressure boundary portion of the dry tube during operation, the leak rate (.0014 square foot break) would not be significant to safety. The evaluation concluded that there was no safety significant effects due to operating with the existing cracks in the dry tubes. The inspector had no further questions in this area.

c. On May 7, 1984, during routine inservice inspection, the licensee identified intermittent surface indications on the core spray piping between the core spray system isolation valves. The indications were detected by ultrasonic and liquid penetrant methods. The indication was intermittent for a length of ten inches and varied in depth from .250" to .310". The indication was located 1" to 1-1/2" from weld SW49B.

The licensee had previously inspected all of the piping welds listed on piping isometric drawing No.40, figure 10-3, Rev. 0, for loop A and figure 10-4, Rev. 0 for loop B. The inspection had disclosed no defects and on discovery of these cracks he reinspected three sister pipe to tee welds in both loops to assure that no other indications had been overlooked. No additional indications were found.

The licensee has taken boat samples for further metallurigical examination. One of the samples will be sent to Brookhaven National Laboratory for an NRC independent examination.

The inspector reviewed the welding Procedure 8-8-BA-102, Revision 4, and the welding Procedure qualification used in the repair for compliance with the ASME Code, Section, XI, 1980 Winter addenda. He verified that the welders were qualified in accordance with the ASME Code, Section XI.

Review of the ultrasonic test (UT) data revealed that the UT report did not reflect the status of the examination, as either accept or reject, for the repair of weld SW49B. This report also accepted a condition referred to as dendritic properties for indication No. 1. The licensee did not have a procedure describing this condition or how it can be employed to disposition indications. The licensee has committed to developing a procedure to properly identify, document, and verify dendritic properties as they occur. This item is unresolved pending issuance of the procedure and review by the NRC (220/84-07-01).

Weld SW49B was reported as nonconforming by Nuclear Energy Services, the ISI contractor, because of base metal indications detected ultrasonically. The inspector attempted to determine how nonconforming conditions are processed when identified by the ISI program. The licensee does not issue a nonconformance report as prescribed by the Niagara Mohawk Quality Assurance Manual. These are transferred to a work request when further work is required to correct the nonconforming condition. However, if the condition is determined to be a "use-as-is" disposition, no work request is written, and there is no further review performed. The processing for use-as-is conditions identified under the ISI program is not well defined by current procedures. This item is unresolved pending the issue of definitive procedures and review by the NRC (220/84-07-02).

d. On May 13, 1984, during a routine inservice inspection, the licensee found an 88 inch long crack on the outside surface of a section of the high pressure feedwater piping. This portion of the feedwater system is also part of the High Pressure Coolant Injection system. The maximum measured depth of the crack is .28 inches. There was also a deformed pipe support directly above the crack.

The licensee also removed the insulation and examined nine welds in the high pressure feedwater piping. No other reportable indications were found. The inspector reviewed the licensee's safety evaluation No: 84-53. The licensee concluded that the crack appears to be a fabrication defect. A boat sample is being metallurgical analyzed by an independent lab to determine the cause of the crack. The licensee elected to replace the entire section of piping. He also analyzed the deformed pipe restraint (No.MK-30-P) and determined that the buckling of the vertical member of the restraint was due to the restraint's inability to allow for thermal growth of the system. The restraint was redesigned to provide horizontal restraint but allow for thermal growth vertically of the system. Prior to start-up, the inspector examined this portion of the system and verified that the piping had been replaced and the restraint modified.

No unacceptable conditions were identified.

6. Refueling Activities

The inspector reviewed the licensee's reload checkoff list and several of the completed surveillance tests performed in accordance with this checklist to verify that surveillance testing required by Technical Specifications had been completed prior to fuel handling operations. The inspector witnessed portions of the reloading operations from the control room and the refuel floor and verified that it was performed in accordance with approved procedures. The inspector also verified that the licensee's staffing during refueling was in accordance with Technical Specifications.

On May 1, 1984, the first day of refueling, the inspector noticed that Intermediate Range Monitor (IRM) #17 was indicating 15% while all other IRM's were indicating 0 to 2%. IRM's #11 and 16 were by-passed as permitted by Technical Specifications. The inspector questioned the operability of IRM #17. A minimum of three IRM's in each Reactor Protection System logic channel are required to be operable. The licensee restored IRM #16 to service, by-passed IRM #17 and investigated the cause of erroneous reading and its effect of the operability of the IRM. The cause was determined to be due to water inside the connector under the reactor vessel. The licensee stated that the IRM remained operable since the actual signal at the detector would still cause the indication to increase. Apparently this indication was present on the midnight shift when refueling operations began. The NRC is concerned that the licensee take prompt action to determine the operability of instruments monitoring safety-related parameters. In this particular instance there was no loss of safety function.

Later the same day, the inspector noticed that Source Range Monitor (SRM) #14 was by-passed during refueling. Technical Specification required that one SRM be operable in the core quadrant where fuel is being moved and that another SRM be operable in a adjacent quadrant. The licensed reactor operator monitoring refueling in the control room stated that there was no apparent reason for the SRM to be by-passed. He immediately unby-passed it. The licensee determined that the SRM was operable when by-passed since SRM's primary function is to allow for monitoring for an inadvertent criticality during refueling. This can still be performed when the SRM is by-passed. The SRM's do not provide an input to scram the reactor. The licensee issued an Operations Department night order to ensure that all personnel were made aware of these events that apparently malfunctioning instruments are given prompt attention, and that operable instruments should not be routinely by-passed. He stressed that a conservative approach should be used when evaluating a potential defective condition.

On April 11, 1984, an NRC inspector noticed that there were two pairs of binoculars on the refueling bridge while it was over the open reactor vessel. One pair was not logged on the Tool and Consumable Material Inventory Checklist as required by Fuel Handling Procedure FHP-2A, "Reactor Building Clean Room Work and Tool Control", Revision 2 dated September 11, 1979. This procedure established administrative controls to prevent foreign material from being inadvertently dropped into the reactor vessel. Technical Specification 6.8.1 requires in part that written procedures and administrative policies shall be established implemented, and maintained. The failure to fully implement Fuel Handling Procedure FHP-2A is a violation of Technical Specifications (220/84-07-03).

After the inspector informed the licensee of the violation, the licensee issued a night order reminding personnel of the requirements for tool control over the open reactor vessel. They also strengthened these controls by requiring the Refuel Floor Supervisor, who is a Senior Reactor

Operator, to verify the accuracy of the inventory checklist once per shift and formal documentation of the initial inspection of the bridge, reactor building crane, and the reactor cavity. These additional requirements were included in revision 3 of FHP-2A issued May 14, 1984.

7. Observation of Physical Security

The inspector made observations to verify that selected aspects of the plant's physical security system were in accordance with regulatory requirements, physical security plan and approved procedures. The following observations relating to physical security were made:

-- The security force was properly manned and appeared capable of performing their assigned functions.

-- Protected area barriers were intact - gates and doors closed and locked if not attended.

-- Isolation zones were free of visual obstructions and objects that could aid an intruder in penetrating the protected ares.

-- Persons and packages were checked prior to entry into the protected area.

-- Vehicles were properly authorized, searched and escorted or controlled within the protected area.

-- Persons within the protected area displayed photo badges, persons in vital areas were properly authorized, and persons requiring escort were properly escorted.

-- Compensatory mesures were implemented during periods of equipment failure.

No violations were identified.

8. Plant Tours

During the inspection period, the inspector made multiple tours of plant areas to make an independent assessment of equipment conditions, radiological conditions, safety and adherence to regulatory requirements. The following areas were among those inspected:

-- Turbine Building

-- Auxiliary Control Room

-- Vital Switchgear Rooms

-- Radwaste Area

-- Diesel Generator Rooms

-- Drywell

-- Reactor Building

The following items were observed or verif ed:

a. Radiation Protection:

-- Personnel monitoring was properly conducted.

-- Randomly selected radiation protection instruments were calibrated and operable.

-- Radiation Work Permit requirements were being followed.

-- Area surveys were properly conducted and the Radiation Work Permits were appropriate for the as-found conditions.

On April 17, 1984, the inspector found an individual's film badge, TLD, and dosimeter laying on a step-off pad in the Reactor Building. The general area radiation level in this area was less than 2 mrem/hr. The individual had apparently removed the items before he removed his protective clothing. The inspector contacted the individual and determined that he had exited the area about 25 minutes earlier and then proceeded directly out of the restricted area. The individual received no significant exposure (less than 1 mrem) while not wearing his dosimetry. The inspector returned the individual's dosimetry and reminded him of the requirement for wearing dosimetry in the restricted area.

Later the same day, the inspector noticed a small leak (15 drops/min) from a drain valve for the reactor recirculation system sample line (valve #110-129). The valve was required to be open for a local leak rate test which was in progress. The reactor water was dripping onto drywell elevation 237', directly where people were walking. The inspector informed the technician on duty, who immediately placed a drip pan under the water. Later he covered the open end of the drain valve with plastic sleeving to route the leakage to a floor drain. The inspector also discussed the event with the assistant Radiation Protection Supervisor. He stressed the need to control potential contamination sources in a Radiation Department night order.

b. Fire Protection:

-- Randomly selected fire extinguishers were accessible and inspected on schedule.

-- Fire doors were unobstructed and in their proper position.

-- Ignition sources and combustible materials were controlled in accordance with the licensee's approved procedures.

-- Appropriate fire watches or fire patrols were stationed when equipment was out of service.

c. Equipment Controls:

-- Jumper and equipment mark-ups did not conflict with Technical Specification requirments.

-- Conditions requiring the use of jumpers received prompt licensee attention.

-- Administrative controls for the use of jumpers and equipment mark-ups were properly implemented.

-- During refueling operations, the inspector examined the interior of all safety-related cabinets in the Auxiliary Control Room and all control panels in the Control Room to ensure that no jumpers or lifted leads were in effect that would affect the operation of safety-related equipment.

-- On May 25, 1984, the inspector noticed that the primary containment post LOCA vent blocking valve (#201.1-12) had a red tag on it which required the valve to be shut (red mark-up #RMU 3748) and a yellow "hold-out" tag on it which required the valve to be open. The valve was actually shut. The hold-out tag had just been hung the previous night in preparation for the containment integrated leak rate test (CILRT) and the individual hanging the tagging noted in the leak rate test procedure that the valve was shut. The licensee directed that hold-out tag be immediately moved. The system was later lined up for the CILRT after the red mark-up was cleared.

d. Vital Instrumentation

-- Selected instruments appeared functional and demonstrated parameters within Technical Specification Limiting Conditions for Operation.

9. Maintenance Activities

The inspector examined portions of various safety related maintenance activities. Through direct observation and review of records, he determined that:

-- These activities did not violate the limiting conditions for operation.

-- Required administrative approvals and tagouts were obtained prior to initiating the work.

-- Approved procedures were used or the activity was within the "skills of the trade".

-- Appropriate radiological controls were implemented.

-- Quality control inspections were conducted.

-- Post maintenance testing was performed.

During this inspection period, the following activities were examined:

- Control rod blade removal to support repair of the CRD stub tubes.
- Disassembly of CRD #71-361.
- Overhaul of Core Spray Topping Pump #12.
- Repair of the isolation valve for drywell high range pressure transmitter.
- Replace solenoid valve #39-05 and 06.

No violations were identified.

10. Exit Interview

At periodic internals throughout the reporting period, the inspector met with senior management to discuss the inspection scope and findings.