U.S. NUCLEAR REGULATORY COMMISSION
REGION I

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Report No. 50-220/85-09

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: May 1 to June 30, 1985

Inspector: S. D. Hudson, Senior Resident Inspector

Approved By: James C. Linville Jr. 7/19/85
J. C. Linville, Chief, Reactor Projects Section No. 2C, DRP date

Inspection Summary: Inspection on May 1 to June 30, 1985 (Report No. 50-220/85-09)
Areas Inspected: Routine inspection by the resident inspector (132 hours). Areas inspected included: licensee action on previous inspection findings, operational safety verification, physical security, plant tours, safety system verification, Licensee Event Reports, review of licensee identified event, potential for overpressurization of ECCS, and TMI Action Plan Items.

Results: One violation was identified. The apparent cause of this violation was lack of control over mechanical jumpers (paragraph 9). Another concern involved the failure to establish procedures to implement new Technical Specification surveillance requirements prior to implementation of the License Amendment (paragraph 8).

DETAILS

1. Persons Contacted

J. Aldrich, Supervisor, Operations
W. Connelly, Supervisor, Q.A. Operations
T. Roman, Station Superintendent

The inspector 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

The plant conducted normal power operations through the inspection period. Four major power reductions to 50% power were performed during May to plug leaking tubes in the main condenser.

3. Licensee Action on Previous Inspection Findings

(Closed) Inspection Follow-up Item (84-25-01): Allegation No. 84-A-0080. On May 25, 1984, NRC, Region I received a letter from a former plant employee alleging various violations of the plant's radiation protection procedures and intimidation when he tried to enforce the procedures. Specifically he stated that: (1) he was told to overlook the mistakes of his union brothers, (2) film badges were destroyed to prevent an individual from exceeding their radiation limit, and (3) incorrect names, exit times, and total dose received were entered on the Radiation Work Permits. The allegations were based on the individuals experience while working as an access technician on the turbine floor during the spring 1984 refueling outage. The NRC requested that the licensee investigate and evaluate these allegations.

The inspector reviewed the licensee's investigation report. Although allegations No. 1 and 3 were partially substantiated, the licensee found that there was no significant hazard to the public health and safety or to the safety of the plant personnel. Allegation No. 2 could not be substantiated. Additionally the licensee's existing procedures would ensure that a representative dose was assigned to an individual in the event a film badge was missing.

The alleger's principal concern was that he was told to overlook minor violations of radiation protection procedures. There was apparently some misunderstanding between the alleger and some co-workers who had told him that minor violations should be identified and corrected but not documented. This was based on a fear that a plant worker would "get into trouble" with

management if a report was filed. The licensee had a procedure for identifying and documenting violations of this type. The licensee acknowledged that minor violations may only require a verbal discussion with the individuals involved. Junior technicians are encouraged to seek the guidance of more experienced technicians in determining the safety significance of observed deficiencies.

The licensee has since revised Radiation Protection Procedure S-RP-5, "Radiation and Radioactive Contamination Control," Revision 2 to clarify its procedure for self auditing of radiation protection procedures. Observed violations should be documented, then forwarded to the Radiation Protection Supervisor who decides what action should be taken. Reports of minor violations may be cancelled by the management representative if an investigation determines that the error was not willful. The reports are to be used as a management tool to correct unsafe conditions, not to punish individuals for minor errors.

4. Followup on Licensee Identified Event

On May 7, 1985, during the performance of a quarterly surveillance test, slight leakage of lake water was observed from the water box to tube sheet weld on containment spray heat exchanger No. 111. This heat exchanger was declared inoperable in accordance with Technical Specification 3.2.6a. On May 8, 1985, similar leakage was found during testing on heat exchanger No. 122. This heat exchanger was also declared inoperable. Technical Specification 3.3.7.c requires that the components be restored to an operable status within 7 days or that the reactor be shut down. The two remaining heat exchangers were operationally checked and no leakage was found. Dye penetrant testing revealed additional indications in the water box to tube sheet weld on all 4 heat exchangers.

In accordance with Technical Specification 3.6.2a.,the licensee requested relief from the requirements of the ASME code until a permanent repair could be performed. As a temporary repair, the licensee welded straps from the water box to the heat exchanger shell to ensure the structural integrity of the heat exchanger. At the request of the NRC staff, the licensee conducted dye penetrant testing of the two tube sheet to shell welds on one heat exchanger. The inspector witnessed the testing to verify that it was properly performed. No unacceptable indications were found on the containment spray side of this heat exchanger. On May 15, 1985, the Office of Nuclear Reactor Regulation issued a safety evaluation which authorized the licensee to consider the heat exchangers operable with the temporary repair completed until the 1986 refueling outage. The licensee was also required to increase the frequency of the operational surveillance test to monthly and to declare the heat exchangers inoperable if leakage increased to 7 1/2 gpm.

The inspector examined each heat exchanger and verified that the straps have been installed per the design drawings.

No unacceptable conditions were identified.

5. 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;

-- Reactor control panel instrumentation and recorder traces;

-- Reactor protection system instruments to determine that the required channels are operable;

-- Stack gas monitor recorder traces;

-- Core thermal limits; and

-- Shift turnover.

b. Review of Logs and Operating Records

The inspector reviewed the following logs and instructions:

-- Control Room Log Book

-- Station Shift Supervisor's Log Book

-- Station Shift Supervisor's Instructions

-- Reactor Operation Log Book

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.

6. 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, the 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 area.

-- 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 an escort were properly escorted.

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

No violations were identified.

7. Plant Tours

During the inspection period, the inspector made frequent 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

-- Cable Spreading Room

-- Diesel Generator Rooms

-- Reactor Building

The following items were observed or verified:

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.

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 requirements.

-- Conditions requiring the use of jumpers received prompt licensee attention except as noted in paragraph 8 for mechanical jumpers.

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

d. Vital Instrumentation:

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

e. Radioactive Waste System Controls:

-- Gaseous releases were monitored and recorded.

-- No unexpected gaseous releases occurred.

f. Housekeeping:

-- Plant housekeeping and cleanliness were in accordance with approved licensee programs.

8. Review of Licensee Event Reports (LER's)

The LER's submitted to NRC, Region I were reviewed to determine whether the details were clearly reported, including accuracy of the description of the cause and adequacy of the corrective action. The inspector also determined whether the assessment of potential safety consequences had been properly evaluated, whether generic implications were indicated, whether the event warranted on site follow-up and whether the reporting requirements of 10 CFR 50.73 had been met.

During this inspection period, the following LER's were reviewed:

LER No.	Event Date	Subject
85-03	March 4, 1985	Reactor scram during surveillance testing
This event is discussed in detail in Inspection Report 85-04.		
84-05	March 4, 1985	HPCI initiation due to high reactor water level
85-05	April 16, 1985	Reactor scram due to malfunction of the turbine pressure control system
This event is discussed in detail in Inspection Report 85-04.		
85-06	April 19, 1985	Initiation of Reactor Building Emergency Ventilation due to spike on radiation monitor.
85-07	April 24, 1985	Initiation of Reactor Building Emergency Ventilation System due to maintenance error.
This event is discussed in detail in Inspection Report 85-04.		
85-09	May 22, 1985	Missed surveillance tests

The two missed surveillance tests were discovered by an internal licensee review. The tests were immediately conducted and the instruments were found to meet their acceptance criteria. The licensee has proposed to implement additional administrative controls by September 1, 1985 to prevent reoccurrence. These actions will be examined during a future inspection. (50-220/85-09-01).

9. Safety System Operability Verification

On a sampling basis, the inspector directly examined selected safety system trains to verify that the systems were properly aligned in the standby mode. This examination included:

-- Verification that each accessible valve in the flow path is in the correct position by either visual observation of the valve or remote position indication.

-- Verification that power supply breakers are aligned for components that must actuate upon receipt of an initiation signal.

-- Visual inspection of the major components for leakage, proper lubrication, cooling water supply, and other general conditions that might prevent fulfillment of their functional requirements.

-- Verification by observation that instrumentation essential to system actuation or performance was operational.

During this inspection period, the following systems were examined:

-- Emergency Diesel Generators

-- Control Rod Drive Hydraulic Control Units

On June 18, 1985, the inspector noticed that the threaded caps on the vent lines for each of 129 the control rod drive withdraw lines were removed and the vent paths were connected together by tygon tubing. The manual vent valve in each vent path was closed. The tubing was apparently installed as a flush path during the 1982-1983 recirculation piping replacement outage and not removed after the outage. The missing cap on the vent line does not cause loss of control rod drive system function or a violation of primary containment integrity. It does, however, reduce the redundancy provided to maintain primary containment integrity. The vent valve and the cap are the isolation boundary for primary containment on each of the 129 control rod drive withdraw lines.

The licensee apparently failed to establish adequate controls of mechanical jumpers since the tygon tubing was not removed after the outage. The licensee's existing Administrative Procedure AP-3.3.2, "Placement of Jumpers or Blocks or Lifting of Leads" applies only to electrical jumpers. Technical Specification 6.8.1 requires that written procedures and administrative policies shall be established implemented and maintained that meet or exceed the requirements of Section 5.1 and 5.3 of ANSI N18.7-1972. ANSI N18.7-1972, Section 5.3.5 (3), requires, in part, that instructions shall be included for returning equipment to its normal operating status. All jumpers shall be controlled. The failure to establish administrative controls for mechanical jumpers is a violation of Technical Specification 6.8.1. (50-220/85-09-02)

10. Licensee Action on TMI Action Plan Items

The TMI Action Plan Requirements (NUREG 0737) required specific actions to improve operations and emergency activities as a result of the lessons learned from the review of the accident at Three Mile Island Nuclear Power Plant. The item number is given in NUREG 0737.

Item II.K.3.16.B. - Challenges and Failures of Relief Valves.

In a letter, C.V. Mangan to D.B. Vassello, dated June 8, 1984, to the Office of Nuclear Reactor Regulation (NRR) the licensee stated that the type of relief valves installed at the station (Dresser electromatic) have a history of failures that is substantially better than those (Target Rock) commonly used on Boiling Water Reactors. No further modifications were necessary. In a letter, D. B. Vassello to B. G. Hooten, dated November 14, 1984, NRR stated that the licensee's position was acceptable. This item is closed.

Item II.K.3.19. - Interlock Recirculation Pump Isolation Valves.

In letters, T. E. Lempges to D. G. Eisenhut, dated February 9, 1981, the licensee proposed the installation of reactor level instrumentation that would not be dependent on the position of the recirculation loop isolation valve position. Therefore no modification was required for the recirculation loop isolation valve logic. In a letter, P. J. Polk to D. P. Dise, dated February 12, 1982, NRR concluded that this approach was acceptable. The licensee also stated that this reactor level instrumentation would also satisfy the requirements of item II.F.2., Instrumentation for Inadequate Core Cooling. These items remain open pending further review of the installation of this equipment.

Item II.K.3.28 - Qualification of Accumulators on Automatic Depressurization System Valves.

The inspector verified that no accumulators are used as part of the ADS system at Nine Mile Point, Unit 1. No action is required by the licensee. This item is closed.

11. Review of the Potential for Overpressurization of Emergency Core Cooling System

The possibility of overpressurizing an Emergency Core Cooling System (ECCS), as a result of component failure or personnel error, has been a long-standing concern of the NRC. Recent experience at several facilities has resulted in overpressurizing the low pressure portion of an ECCS system. The inspector examined the existing design features and administrative controls that are provided to minimize the potential for such an overpressurization. From discussions with senior station personnel, the inspector determined that there has never been an actual or partial overpressurization event of this type at Nine Mile Point, Unit 1.

DESIGN:

At Nine Mile Point, Unit 1, the two identified core spray systems and their associated keep full systems are the only systems which could possibly be overpressurized by normal reactor pressure (1034 psig). The core spray system, from the suction of the topping pump to the isolation check valve, is designed for 465 psig. From and including the check valve to the reactor vessel, it is designed for 1200 psig. The design of each core spray system from the reactor vessel to the low pressure portion of the system consists of two motor-operated isolation valves (MOVS's) inside the drywell arranged in a parallel flow path; a MOV and a check valve outside the drywell. No testable check valves are used in this or any other system. The inside MOV's are normally shut. The outside MOV is normally open with the power for the motor operator deenergized. Therefore the check valve serves as the isolation valve outside containment.

The keep full system supplies condensate water at 150 psig downstream of the core spray check valve. It is isolated from the condensate system by two check valves in series. This portion of the condensate system is designed for 600 psig.

The following physical features provide protection against overpressurizing the low pressure portion of the system. A low reactor pressure signal (365 psig) in conjunction with low low reactor level or high drywell pressure signals the valves to open after an accident. An interlock prevents manually opening the inside and outside MOV's at the same time. A relief valve set at 320 psig is provided to protect the low pressure portion of the system.

SURVEILLANCE:

Each MOV is stroke tested once per quarter as required by Technical Specification 4.1.4.c to verify operability and determine stroke time. The sensors for the automatic initiation logic are tested monthly as required by Technical Specification 4.6.2d. During this testing, the isolation valves are not actually opened.

The check valve in each core spray system and the check valves in their keep full systems are periodically leak rate tested as required by Technical Specification 4.2.7.1a. These Technical Specifications were part of an Order issued on April 20, 1981 and provide additional assurance of the integrity of these check valves. This test is performed each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance. The limit is 5 gpm.

The relief valve on the low pressure portion of the core spray system is not currently periodically tested to verify its set point. The valve on each side of the core spray system was tested during the 1984 refueling outage as part of other maintenance activities. The licensee is preparing a procedure to ensure that the relief valve is periodically checked. This procedure will be in place by the 1986 refueling outage. The interlock, which ensures that the inside and outside MOV's cannot be opened at the same time, is also not periodically tested. It was last tested after installing a fire protection modification during the 1984 refueling outage. It performed correctly. There is currently no requirement for the periodic testing of this interlock.

MAINTENANCE:

The inspector reviewed all work requests for 1983 and 1984 for the core spray system. All components were correctly classified as safety related and therefore received appropriate quality control inspections. The MOV's appear to have required no corrective maintenance during this period. The motor operators of several of the MOV's were replaced to meet the requirements for environment qualification of electrical equipment. The inspector determined that the valves had been properly tested after this modification. The core spray check valves each initially failed their leak rate testing in April 1984. After corrective maintenance, each valve was satisfactorily tested. Existing plant procedures require two man verification that components are correctly restored to service.

CONCLUSION:

Existing equipment design, testing requirements, and administrative controls appear to provide adequate assurance to prevent overpressurization of low pressure portions of ECCS system. Additional protection would be provided by periodic testing of the MOV interlock.

12. Exit Interview

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

Based on the NRC Region I review of this report it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.