U. S. NUCLEAR REGULATORY COMMISSION
REGION I

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Report No. 50-220/85-04

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: March 1 to April 30, 1985

Inspectors: S. D. Hudson 5/22/85
S. D. Hudson, Senior Resident Inspector Date
A. J. Luptak, Resident Inspector

Approved by: Samuel J. Collins for 5/22/85
J. C. Linville, Chief, Reactor Date
Project Section No. 2C, DRP

Inspection Summary:

Inspection on March 1 to April 30, 1985 (Report No. 50-220/85-04)

Areas Inspected: Routine inspection by the resident inspectors (235 hours). Areas inspected included: operational safety verification, physical security, plant tours, safety system verification, surveillance testing, maintenance activities, Licensee Events Reports, review of licensee identified events and review of licensee response to safety concerns.

Results: No violations were identified in the areas examined.

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 operated at full power throughout the inspection period except for reactor scrams on March 4 and April 16, 1985.

On April 11, 1985 the licensee reduced power to about 77% as a precautionary measure when a leak developed in the main generator stator water cooling system. Loss of the stator water cooling would have initiated an automatic turbine runback. Stator water cooling make up flow maintained the level in the system until the leak was isolated. Subsquently, the licensee reduced power to 46% to inspect for condenser tube leaks. After plugging 8 tubes in the main condenser, the plant was restored to full power late on April 11, 1985.

3. Followup on Licensee Identified Events

a. On March 4, 1985 at 2:34 p.m., the reactor tripped from 100% power. All systems responded as required. The cause of the scram was technician error while attempting to vent a reactor vessel low-low level transmitter in preparation for a routine surveillance test. Although this transmitter does not produce a direct reactor trip signal, the transmitter is located on a common manifold with the reactor vessel low level transmitters which do generate reactor protection system inputs. Apparently, the technician cracked open the isolation valve for the transmitter under test instead of the bypass valve. This caused a hydraulic transient in the sensing lines for the level transmitters which produced the low level scram signal. The actual reactor water level was normal at the time of the scram.

   The inspector observed the operators' response to the scram from the control room. All actions were in accordance with approved procedures and were executed promptly thus minimizing the time required to recover from the scram. The inspector also attended the meeting of the Site Operations Review Committee which reviewed the cause of the scram and actions necessary to restart the plant. The reactor was taken critical at 3:05 a.m. on March 5, 1985. The inspector reviewed the start-up check list to verify that all prerequisites for the start-up had been completed. He also witnessed portions of the

reactor power increase and loading of the main turbine. No unsatisfactory conditions were identified. The inspector also verified that this event was reported as required by 10 CFR 50.72.

To help prevent future scrams during surveillance testing, the licensee plans to add a precaution in the surveillance procedure. The licensee's actions will be reviewed during a future inspection. (50-220/85-04-01)

b. On April 16, 1985 at 2:51 p.m., the plant tripped from 100% power due to high neutron flux. The power increase to 120% was caused by an increase in reactor pressure due to a malfunction in the turbine pressure control system. Maximum reactor pressure during the transient was 1050 psig. Normal reactor pressure is 1034 psig. All systems functioned as required except for one station service electrical bus (Power Board #11) which failed to automatically transfer to off-site pwer. Operators manually transferred Power Board #11 to off-site power. The control room became very crowded and somewhat noisy immediately following the scram. The shift supervisor eventually cleared the excess personnel from the control room.

   The licensee's post-scram analysis did not find a specific cause for the malfunction in the turbine pressure control system. A strip chart recorder has been temporarily installed to monitor the performance of some components in the pressure control system. A dirty relay which prevented the transfer of Power Board #11 was cleaned and tested prior to start-up. The licensee also plans to modify the computer generated post trip logs to better aid in the post scram analysis. Various chart recoders of reactor pressure, power, turbine control valve position, etc. were more helpful in this analysis. The inspector witnessed the post scram response in the control room, and portions of the reactor start-up, heat-up and turbine loading and determined that they were in accordance with approved procedures. He reviewed the licensee post scram analysis and determined that it was completed as required. This event was reported to the NRC as required by 10 CFR 50.72.

c. On April 24, 1985, the Reactor Building Emergency Ventilation system automatically initiated when a fuse blew in a 24VDC power supply to the Reactor Building Ventilation Radiation Monitor, causing the radiation monitor to fail up-scale. The cause of the blown fuse was due to maintenance on another radiation detector which shares this common power supply. The Reactor Building Ventilation system functioned as designed. After the fuse was replaced, the system was returned to normal line-up. This event was reported to the NRC as required by 10 CFR 50.72

4. 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 Operating 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. 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 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.

On March 15, 1985 the inspector noted two administrative errors in maintaining the Equipment Status Log (ESL). One system had not been entered in the log after it had been removed from service on March 13, 1985 while another indicated a pump was out of service when in fact, it had been returned to service. Review of the previous months entries revealed no other anomalies. Discussions were held with the Operations Supervisor, who acknowledged the problem and discussed planned changes in the ESL system. A night order was issued to ensure that the ESL is accurately maintained.

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.

7. 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:

-- Control Room Emergency Ventilation

-- Reactor Building Emergency Ventilation

-- 125 VDC Batteries and Battery Boards

The inspectors noted that the breaker labelled "CO2 heater" was not in the "on" position as required by Operating Procedure N1-Op-47A. The licensee's review determined that the breaker was really a spare and was improperly labelled. The label was corrected and the procedure was immediately changed to show the actual condition.

No violations were identified.

8. Surveillance Testing

The inspector witnessed the performance of selected surveillance to verify that:

--Surveillance procedures conform to technical specification requirements and have been properly approved.

--Test instrumentation is calibrated.

--Limiting conditions for operations for removing equipment from service are met.

--Surveillance schedule is met.

--Test results met technical specification requirements.

--Appropriate corrective action is initiated, if necessary.

--Equipment is properly restored to service following the test.

The following tests were included in this review:

-- ST-C9 "Control Room Emergency Ventilation Flow and HEPA/Charcoal DP Operability Test" performed on #12 Control Room Emergency Ventilation Fan on April 8, 1985.

-- ST-M8 "Emergency Ventilation Operability Test" performed on March 26, 1985.

-- RTP-23, "Routine Calibration of Reactor Building Ventilation Monitor" performed on April 19, 1985.

No violations were identified.

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 as appropriate.

-- Post-maintenance testing was performed.

During this inspection period the following Environmental Qualification modifications (10 CFR 50.49) were examined:

-- Replacement of connectors of the solenoids for the Emergency Condenser steam line drain valve.

-- Replacement of the limit torque motors on the torus Air Vent and Purge Isolation Valve and the Drywell Nitrogen Vent and Purge Outside Isolation Valve.

No violations were identified.

10. 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 was reviewed:

LER No.	EVENT DATE	SUBJECT
85-02	February 14, 1985	Inoperable Halon System in Auxiliary Control Room

During a surveillance test, the licensee found 3 fire dampers inoperable in the Auxiliary Control Room, but failed to declare the Halon System inoperable and post a continuous fire watch as required by Technical Specifications. The condition was discovered eight days later during a supervisory review. During this time, fire detectors for this area remained in service. This event is also discussed in Inspection Report 85-02.

The inspector had no further questions in this area.

11. Review of Licensee's Response to Safety Concerns

a. In response to an event at an operating BWR/4 which resulted in a crack in the vent header in the torus, GE issued a Service Information Letter (SIL) to recommend actions that could be taken to prevent this type of event and help ensure containment integrity. The crack was attributed to brittle fracture caused by the injection of cold nitrogen into the torus during inerting. The licensee responded to GE SIL 402 by having a consultant evaluate the design and operation of the containment inerting system as described in recommendations 1 and 2 of GE SIL 402.

   The consultant determined the design of the inerting system was adequate with one exception. In reviewing the containment makeup and atmosphere dilution system the consultant noted that a failure of a single temperature controller could cause the loss of electric heat. A low temperature alarm would occur; however, there is no automatic means of shutting off nitrogen flow. The licensee is reviewing a modification to install automatic nitrogen shut off capacilities in this system.

   The consultant's review of operating and maintenance history and the operating and maintenance programs for the inerting systems indicated no history of significant problems.

Recommendation 3, to test the drywell/wetwell bypass leakage, was not implemented by the licensee based upon review of chart recordings of drywell and wetwell pressure during the past several years showing no indications of anomalies.

Recommendations 4 and 5, to inspect nitrogen injection lines and the containment respectively, were implemented by the licensee and no cracks were found.

The inspector's review indicated the licensee has adequately implemented the recommendations of GE SIL 402, with the exception of number 3, for which it has appropriate justification. The inspector had no further questions.

b. In November 1983, the NRC issued Inspection and Enforcement (IE) Information Notice 83-75 to alert licensees of two cases of improper control rod movement due to misuse of the Rod Worth Minimizer. (RWM) On April 17, 1984, Institute of Nuclear Power Operations also issued Significant Operating Event Report (SOER) 84-2 addressing this topic. The inspector reviewed both documents to determine if the licensee had adequately addressed this safety concern.

The inspector reviewed an NMPC internal memos dated April 10, 1984, August 23, 1984, and October 2, 1984 and found that the licensee had evaluated each of the recommendations of the SOER. The inspector also reviewed selected operator procedures to ensure that appropriate use of the RWM, scram timing switches, notch override and continuous withdraw switches are addressed. Through discussions with licensee personnel, the inspector learned that these topics are part of operator requalification training program. The inspector had no further questions on this area.

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.