U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-410/87-38

Docket No. 50-410

License No. MPF-69

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

Facility Name: Nine Mile Point Nuclear Station, Unit 2

Inspection At: Scriba, New York

Inspection Conducted: October 12-15, 1987

Inspectors: L. J. Wink, Operations Engineer — 11/5/87 date

Approved by: D. J. Lange, Chief, BWR Section, DRS — 11-14-87 date

Inspection Summary: Inspection On October 12-15, 1987 (Report No. 50-410/87-38)

Areas Inspected: Routine, unannounced inspection by one region-based inspector of the overall power ascension test program including procedure review, test witnessing and test results evaluation and independent measurements and verifications.

Results: No violations were identified.

NOTE: For Acronyms Not Defined, Refer to NUREG-0544, "Handbook of Acronyms and Initialisms."

DETAILS

1.0 Persons Contacted

Niagara Mohawk Power Corporation

*R. Abbott, Station Superintendent
*C. Becknam, Manager Nuclear QA Operations
*J. Conway, Power Ascension Manager
*P. Eddy, Site Representative, New York State PSC
D. Helms, Lead Shift Test Supervisor
H. Pao, Shift Test Supervisor
*T. Perkins, General Superintendent Nuclear Generation
*A. Pinter, Licensing Engineer
B. Rudd, Shift Test Supervisor
R. Smith, Superintendent Operations
T. Tomlinson, Reactor Analyst
*W. Wambsgan, Assistant Superintendent Operations
*K. Zollitsch, Training Superintendent

NRC Personnel

*W. Cook, Senior Resident Inspector
C. Marschall, Resident Inspector
W. Schmidt, Resident Inspector

*Denotes those present at the exit meeting on October 15, 1987.

The inspector also contacted other members of the Licensee's Operations, Technical, Test and QA staffs.

2.0 Power Ascension Test Program (PATP)

2.1 References

- Regulatory Guide 1.68, Revision 2, August 1978, "Initial Test Program for Water Cooled Nuclear Power Plants."
- ANSI N18.7-1976, "Administrative Controls and Quality Assurance of Operations Phase of Nuclear Power Plants."

- Nine Mile Point Unit 2 (NMP-2) Technical Specifications, July 2, 1987.
- Nine Mile Point Unit 2 Final Safety Analysis Report (FSAR) Chapter 14, "Initial Test Program."
- Nine Mile Point Unit 2 Safety Evaluation Report.
- Nine Mile Point 2 AP-1.4, Startup Test Phase, Revision 5.

2.2 Overall power Ascension Test Program

The inspector held discussions with the power ascension manager (PAM), the lead startup test, design and analysis (STD&A) engineer and other members of the PATP staff to assess the status of testing, the test results evaluation process and the preparation and approval of test procedures. In addition, the inspector attended the daily power ascension management meetings.

At the beginning of the inspection period the unit was at 24% of rated power in Test Condition 2 with preparations in progress to conduct the loss of offsite power test. On October 13, 1987 the licensee successfully completed the loss of offsite power test (see paragraph 2.4 for discussion) and entered a four day outage to repair feedwater pump seals, a drywell vacuum breaker solenoid, RHR steam condensing mode pressure control valves and power ascension test instrumentation.

Following completion of outage activities, the Licensee plans to complete the remaining testing for test condition 2 (feedwater system and RHR steam condensing mode) and then commence testing in Test Condition 3.

2.3 Power Ascension Test Procedure Review

Scope

The procedures at Attachment A were reviewed for the attributes identified in Inspection Report No. 50-410/86-38, Section 4.3.

Discussion

The procedures reviewed were found to be acceptable. The procedure marked with an asterisk in Attachment A is a revision of a previously reviewed procedure.

Findings

No unacceptable conditions were identified.

2.4 Power Ascension Test Witnessing

Scope

The inspector witnessed the performance of the power ascension test discussed below. The performance of this test was witnessed to verify the attributes previously defined in Inspection Report No. 50-410/86-64, Section 2.3.

Discussion

N2-SUT-31-2, Loss of Turbine - Generator and Offsite Power

The performance of this test was also witnessed by the senior resident inspector and two resident inspectors (additional details may be found in Inspection Report No. 50-410/87-39). The purpose of the test was to demonstrate that all safety related equipment necessary to achieve a safe shutdown condition would operate properly with only on-site power available. The test also verifies that sufficient instrumentation and controls are available to monitor and maintain the plant in a safe shutdown condition.

On the day proceeding the scheduled test date, the inspector attended a pre-test briefing of operations personnel conducted at the training center. The personnel selected to participate in the test were the same as those who had conducted the cold shutdown loss of offsite power test on September 23, 1987. The briefing included the scope of the test, responsibilities of individual participants and changes made to the test procedure based on the cold shutdown test evaluation. Conditions were discussed which would require the termination of testing and contingent emergency actions were reviewed in the unlikely event that insufficient on-site power would be available.

On the day of the test, the inspector observed test preparations including the verification of prerequisites, the briefing of test engineers and the final on-shift briefing of Operations personnel. All plant systems were verified to be in their normal alignment with the unit at approximately 24% of rated thermal power.

At the command of the station shift supervisor the test was initiated at 1024 by tripping the main turbine generator and simultaneously opening two breakers in the Scriba switchyard to remove offsite power from station. A reactor scram occurred due to the tripping of the main turbine. All three EDGs started and automatically energized the emergency busses. Required loads were automatically sequenced on to the emergency busses as designed. Following confirmation of the scram and successful operation of the EDGs, the operator manually closed the MSIVs approximately 30 seconds following the test initiation. This pre-planned action was taken to protect the main turbine from a

potential loss of condenser vacuum and provided a bounding test demonstration by requiring the safety/relief valves (SRVs) to automatically limit the expected reactor pressure rise.

Following the closure of the MSIVs, decay heat caused a slow, steady rise in reactor pressure which was terminated by the automatic opening of 2 SRVs when reactor pressure reached 1072 psig. Peak reactor pressure was measured to be 1077 psig. Following the SRVs lifting, reactor water level was noted to be approximately + 145 inches (wide range) and stable. After confirmation of successful automatic operation of the SRVs, the operator took manual control of their operation and maintained reactor pressure between 950 psig and 1020 psig by opening single SRVs at approximately five minute intervals for the remainder of the test. The SRVs selected for manual operation were, by procedure, chosen to evenly distribute the thermal loading of the suppression pool.

At approximately nine minutes into the test the stub busses were energized and a CRD pump was manually started. The flow from the single CRD pump proved sufficient to makeup for the inventory losses through the SRVs and the minimum reactor water level during the test was approximately + 140 inches (Wide Range). Since this level was well above the normal initiation setpoint for both RCIC and HPCS, neither system was required during the test. Once stable water level was confirmed the operators manually initiated HPCS in the CST-TO-CST mode to demonstrate operability and provide a load for the HPCS EDG.

At 1056 the test was concluded following the required 30 minute demonstration and operators began to restore offsite power. RCIC was manually initiated and used to restore reactor water level to the normal band and as an alternate means of pressure control. At 1100 the normal station busses were energized from offsite power and preparation were begun to bring the unit to cold shutdown.

The inspector observed and evaluated overall shift crew performance during the conduct of the test and subsequent recovery actions. The actions of both test and operations personnel were judged to be excellent. All require actions were carried out in an expeditious and professional manner. The station shift supervisor maintained excellent overall control and insured good coordination and communication.

Findings

No unacceptable conditions were identified.

2.5 Power Ascension Tests Results Evaluation

Scope

The power ascension test results listed in Attachment B and discussed below were evaluated for the attributes identified in Inspection Report No. 50-410/86-64, Section 2.1.

The test results evaluated were in the process of being technically reviewed by the PATP staff and had not received a SORC review nor been accepted by the general superintendent. The inspector will verify the completion of this review and management acceptance of these test results during a future, routine inspection.

Discussion

N2-SUT-05-2, Control Rod Drive System

This test was performed in conjunction with the planned scram during the Loss of Turbine-Generator and Offsite Power Test, N2-SUT-31-2. The scram times of the four selected rods (coordinates 30-59, 54-43, 26-39 and 42-31) were measured and verified to be well within the acceptance criterion and technical specification limit of 7.0 seconds.

N2-SUT-11-2, LPRM Calibration

The initial calibration of the LPRMs was performed at 45.6% of rated thermal power. Three (3) of the 168 LPRMs were found to differ from their calculated value by more than the level 2 acceptance criterion of 10 %. This test exception was accepted 'As-is' pending recalibration of the LPRMs at a higher power level in test condition 3.

N2-SUT-22-2, Pressure Regulator

This test was performed at 44% of rated thermal power to demonstrate adequate response and stability with the main turbine control valves alone controlling the transient and with the main turbine bypass valves alone controlling the transient. All acceptance criteria were satisfied.

N2-SUT-27-2, Turbine Trip Within BPV Capacity

This test consisted of two parts: First, the capacity of the main turbine bypass valves is measured and verified to be greater than or equal to the value assumed in the FSAR and, second, a demonstration is performed to verify, that within measured bypass valve capacity, a main turbine trip will not cause a reactor scram.

A level 2 test exception was identified when the uncorrected data indicated that the bypass valve capacity was 2.96 MLb/hr which is less than the required value of 3.2 MLb/hr (25% of rated steam flow). Evaluation of the data is continuing and the inspector will review the resolution of this test exception in a future, routine inspection.

Using the measured value of bypass valve capacity, a successful demonstration of scram avoidance was performed by tripping the main turbine at a reactor power of 21.6% (Equivalent to 88.3% of measured bypass valve capacity).

N2-SUT-30-2, Reactor Recirculation Cavitation

The test successfully demonstrated that no cavitation exists in the Reactor Recirculation System at any condition exceeding the setpoint of the low feedwater flow cavitation interlock.

N2-SUT-31-2, Loss of Offsite Power

All acceptance criteria were satisfied. All required safety systems functioned properly without manual assistance. Proper instrument displays were verified and sufficient control was demonstrated to insure the safe shutdown of the unit without offsite power available.

During the review of the GETARS traces of the test, the inspector noted that the reactor water level instrumentation exhibited a similar "ringing" phenomenon to that observed at other recently licensed BWRs during their power ascension test programs. The phenomenon has been attributed to the extreme sensitivity of the Rosemont 1153 level transmitters to pressurization transients. The inspector discussed the phenomenon with licensee representative and indicated that the potential exists for unnecessary challenges to safety systems. The licensee representatives indicated that evaluation of possible corrective actions was already in progress. The inspector will follow the licensee's actions in this matter during a future, routine inspection.

N2-SUT-70-2, RWCU System

All acceptance criteria were satisfied for the system operating in its normal and blowdown modes.

Findings

No unacceptable conditions were identified.

3.0 QA Interface with the PATP

During the course of witnessing the loss of offsite power test, the inspector observed the QA engineer performing a surveillance of test activities. The inspector also noted, during the review of power ascension test procedures, that the procedures had been reviewed by QA prior to issuance.

No unacceptable conditions were noted.

4.0 Independent Measurements and Verifications

During the witnessing of the loss of offsite power test as discussed in paragraph 2.4, the inspector made multiple independent verifications of expected responses of the following systems: Reactor protection system, emergency diesel generators, service water system, standby gas treatment system, and safety/relief valves. In addition, during the evaluation of the results of power ascension test N2-SUT-05-2, Control Rod Drive System, the inspector measured the scram times of four selected control rods, using GETARS traces, and verified conformance with the test acceptance criterion and technical specifications. The inspector's measurements and verifications agreed with those made by the licensee.

No unacceptable conditions were noted.

5.0 Exit Interview

At the conclusion of the inspection on October 15, 1987, an exit meeting was held with licensee personnel (identified in Section 1.0) to discuss the inspection scope, findings and observations as detailed in this report. At no time during the inspection was written materials provided to the licensee by the inspector. Based on the NRC Region I review of this report and discussions held with licensee representatives during the inspection, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.

ATTACHMENT A

POWER ASCENSION TEST PROCEDURES REVIEWED

*N2-SUT-31-2	Loss of Turbine - Generator and Offsite Power, Revision 1, Approved October 12, 1987.
N2-SUT-22-6	Pressure Regulator Test Condition 6, Revision 0, approved March 4, 1987.
N2-SUT-29-6	Recirculation Testing, Revision 0, Approved February 13, 1987.
N2-SUT-30-6	Reactor Recirculation System, Revision 0, Approved February 13, 1987.
N2-SUT-33-6	Drywell Piping Vibration - Test Condition 6, Revision 1, Approved December 23, 1986.
N2-SUT-35-6	Recirculation System Flow Calibration, Revision 0, Approved March 30, 1987.
N2-SUT-79-6	Reactor Internals Vibration Measurement 100 Percent Load Line, Revision 1, approved October 5, 1987.
N2-SUT-81-6	Penetration Cooling Test Condition 6, Revision 1, Approved February 10, 1987.

ATTACHMENT B

POWER ASCENSION TEST RESULTS EVALUATED

N2-SUT-05-2	Control Rod Drive System, Revision 2, Completed October 13, 1987, in review.
N2-SUT-11-2	LPRM Calibration - Test Condition 2, Revision 1, Completed October 7, 1987, in review.
N2-SUT-22-2	Pressure Regulator Test Condition 2, Revision 2, Completed October 10, 1987, in review.
N2-SUT-27-2	Turbine Trip Within Bypass Capacity, Revision 2, completed October 11, 1987, in review.
N2-SUT-30-2	Reactor Recirculation System Cavitation Test, Revision 1, Completed October 10, 1987, in review.
N2-SUT-31-2	Loss of Turbine - Generator and Offsite Power, Rev. 1, Completed October 13, 1987, in Review.
N2-SUT-70-2	Reactor Water Cleanup System, Revision 1, Completed October 8, 1987, in review.