U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-410/87-21

Docket No. 50-410

License No. NPF-54

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: June 8-19, 1987

Inspectors: L. J. Wink FOR 7/8/87
M. Evans, Reactor Engineer, DRS date

L. J. Wink 7/8/87
L. Wink, Reactor Engineer, DRS date

Approved by: P. W. Eselgroth 7/9/87
Peter W. Eselgroth, Chief, Test Programs Section, Division of Reactor Safety date

Inspection Summary: Inspection on June 8-19, 1987 (Report No. 50-410/87-21)

Areas Inspected: Routine, unannounced inspection by two region based inspectors of overall power ascension test program including procedure reviews, test witnessing and results evaluation, followup of events, QA/QC interfaces and independent verifications.

Results: No violations were identified.

NOTE: For acronyms not defined, refer to NUREG-0544 "Handbook of Acronyms and Initalisms."

DETAILS

1.0 Persons Contacted

Niagara Mohawk Power Corporation

*R. Abbot, Station Superintendent
C. Beckham, Quality Assurance Manager, Operations
G. Carlisle, Lead STD&A Engineer
M. Colomb, Station Shift Supervisor
*J. Conway, Power Ascension Manager
J. Harris, Shift Test Supervisor
M. Jones, Operations Superintendent
K. Korcz, Licensing Engineer, Unit 2
P. McKenna, BOP Test Engineer
T. Perkins, General Superintendent
*A. Pinter, Site Licensing Engineer
L. Prunotto, Lead Senior Structural Engineer, Unit 2
P. Wilde, Supervisor, QA Surveillance, Unit 2

Other NRC Personnel

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

*Denotes those present at the exit meeting on June 19, 1987.

The inspector also contacted other members of the licensee's operations, technical, test and QA staff.

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 N18.7-1987 "Administrative Controls and Quality Assurance for Operations Phase of Nuclear Power Plants."
- Nine Mile Point Unit 2 (NMP-2) Technical Specifications, Revision 0, October 31, 1986.
- 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 Unit 2 AP-1.4, Startup Test Phase, Revision 3

2.2 Overall Power Ascension Test Program

The inspector held discussions with the Power Ascension Manager (PAM), the Lead Startup, Design and Analysis (STD&D) Engineer and other members of the PATP staff to assess the status of low power 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 and Site Operations Review Committee (SORC) meetings involving the PATP.

During this inspection the unit experienced two unplanned scrams. The first occurred at 2056 on June 12, 1987, as a result of a Intermediate Range Monitor (IRM), high neutron flux trip. The power spike was due to the rapid increase of feedwater flow to the reactor vessel which resulted from the full opening of the "A" feedwater pump high pressure/low flow control valve. The valve failed full open when the valve position feedback linkage vibrated loose and detached from the valve. All systems responded as designed during the event. The licensee's review indicated that the maximum power achieved during the transient was approximately 4.2% of rated.

On June 15, 1987, at 1133, with the reactor at 910 psig, a second reactor scram occurred as a result of a Redundant Reactivity Control System (RRCS) actuation of Alternative Rod Insertion (ARI).

At the time of the event, surveillance procedure N2-ISP-ISC-M009, RRCS-High Reactor Pressure Channel Functional Test, was in progress on Division II, Channel "B" and it was in the tripped condition. An apparently spurious, Division II, Channel "A" low reactor water level trip was noted, which satisfied the logic and caused an ARI. The ARI vented the scram air header and the control rods drifted into the core, shutting down the reactor. The filling of the Scram Discharge Volume (SDV) generated a reactor scram on high level approximately 41 seconds following the ARI.

The inspector reviewed the process computer sequence-of-events log and control room strip chart recorder traces and concluded that all systems had functioned as designed. The inspector also agreed with the licensee's conclusion that the Division II, Channel "A" low reactor water level trip was spurious. On June 16, 1987, the SORC met to review the ARI initiation and subsequent investigation. The investigation concluded that the spurious low reactor water level trip was not hydraulic in origin, since other instruments, sharing common reference and variable legs with the RRCS level transmitter, showed no unusual level transient. Review of the process computer alarms revealed several alarms received during RRCS testing, both before the trip and during a re-performance of the surveillance after the event. These alarms were traced to random, short duration (1 msec) pulses in the RRCS electronics. The origin of these pulses could not be determined and their duration was too short to cause the trip signal to seal-in (20 msec).

To allow continuation of the low power test program while troubleshooting continued on the RRCS, the SORC directed that a safety evaluation be performed to allow the bypassing of RRCS below 5% of rated power. On June 18, 1987, the inspector attended the SORC meeting to review Safety Evaluation 87-083, Bypass RRCS up to 5% Thermal Power. Following an extensive review of the safety implications of the proposed bypass the SORC approved the safety evaluation with minor modifications. The inspector was satisfied that the proposed actions did not present a safety concern. The final resolution of the troubleshooting of RRCS will be reviewed during a subsequent routine inspection.

At the conclusion of this inspection preparation were underway to restart and resume low power testing.

2.3 Power Ascension Test Procedure Review

Scope

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

Findings

The procedures reviewed were found to be acceptable. No deficiencies 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-5-HU, Control Rod Drive System-Scram Testing of Selected Rods

This test was performed on June 11, 1987, at a reactor pressure of 600 psig.

The inspector observed the scram tests performed on control rods 22-31, 30-31, 38-15, and 38-47. The overall test crew performance and interface with operations personnel was satisfactory. The inspector observed pre-scram data taking, initiation of testing for each rod and data reduction following the testing of each rod. The test results were well within acceptance criteria limits.

Findings

No deficiencies were identified.

2.5 Power Ascension Tests Results Evaluation

Scope

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

Discussion

N2-SUT-5-HU, Control Rod Drive System - Scram Testing of Selected Rods

This test was still in progress during the inspection. The inspector evaluated the results of those portions of the test completed to date.

The rods selected for monitoring during heatup were 22-31, 30-31, 38-15, and 38-47. Section 6.2 of the test was performed on June 11, 1987, and involved individual scrams of the selected rods at a reactor pressure of 600 psig. Section 6.3 of these test was performed on June 15, 1987, and involved individual scrams of the selected rods at a reactor pressure of 800 psig. The inspector reviewed the test data and independently calculated the scram times of the rods using GETARS traces. The inspector confirmed that the acceptance criterion (scram time to notch position 05 less than or equal to 7 seconds) was satisfied for the selected rods at both reactor pressures. The slowest rod was 38-47, with a time to notch position 05 of 3.273 seconds at 800 psig.

N2-SUT-78-HU, BOP System Expansion

The test was still in progress during the inspection. The inspector had previously evaluated the results of this test and the test exceptions identified up to a reactor temperature of 350°F (140 psig) during Inspection No. 50-410/87-16. The current review involved the results obtained during subsequent testing up to rated reactor temperature. The review included test exceptions (TEs) identified and the corrective actions taken to resolve them.

On June 14, 1987, the licensee identified a test exception (TE#13) involving thirteen (13) pipe segments whose movement had exceeded the Level 1 acceptance criteria for thermal growth and ten (10) pipe segment whose movement had exceeded the Level 2 acceptance criteria, at rated reactor temperature. An engineering evaluation was performed to assess the impact of these deficiencies and the resulting pipe stresses and nozzle loadings. On June 16, 1987, the inspector attended a Site Operations Review Committee (SORC) meeting to review the deficiencies and the engineering assessment of their

significance. The SORC reviewed Site Services Memorandum (SSM) N062-0060, which indicated that, in all cases, the pipe stresses and nozzle loading were within acceptance limits and that all identified deficiencies were acceptable "as-is". The Lead Senior Structural Engineer discussed each of the identified problem areas and the basis for the determination of acceptability. Following the presentation, the SORC accepted the engineering evaluation and authorized the lifting of the Level 1 hold condition and the resumption of testing.

The inspector also reviewed six additional test exceptions (TE #10, 11, 12, 14, 15, and 16) and verified that appropriate resolutions had been identified.

Findings

No unacceptable conditions were identified during this review.

3.0 QA/QC Interfaces with the PATP

The inspector reviewed three QA surveillance reports during this inspection. Two reports covered performance and test results review of startup test N2-SUT-12-HU, APRM Calibration, while the other covered test results review of startup test N2-SUT-10-HU, IRM Performance. The inspector noted that each QA surveillance report included checklists which detailed critical attributes to be monitored. All QA inspector concerns identified were adequately resolved.

The inspector also observed QA surveillance coverage of the testing activities discussed in paragraph 2.4.

Findings

No unacceptable conditions were identified.

4.0 Independent Measurements and Verifications

The inspector independently calculated the scram times of individual rods using GETARS traces and verified conformance with acceptance criteria during the witnessing, as discussed in paragraph 2.4, and the evaluation of test results, as discussed in paragraph 2.5, of power ascension test N2-SUT-5-HU, Control Rod Drive System-Scram Testing of Selected Rods. The times calculated by the inspector agreed with these determined by the licensee.

5.0 Exit Interview

At the conclusion of the inspection on June 19, 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 were 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-1-3,	Chemical and Radiochemical 45% to 75% Power Testing, Revision 2, May 22, 1987
N2-SUT-02-3,	Radiation Measurements - TC3, Revision 1, November 3, 1986
N2-SUT-11-3,	LPRM Calibration - TC3, Revision 1, October 29, 1986
N2-SUT-12-3,	APRM Calibration - TC3, Revision 1, February 11, 1987
N2-SUT-16-3,	Selected Process Temperatures and Water Level Measurements Test Condition 3, Revision 1, December 23, 1986
N2-SUT-17-2,	System Expansion - Test Condition 2, Revision 1, December 23, 1986
N2-SUT-17-3,	Systems Expansion - Test Condition 3, Revision 1, December 23, 1986
N2-SUT-19-3,	Core Performance - Test Condition 3, Revision 1, April 16, 1987
N2-SUT-22-3,	Pressure Regulator - Test Condition 3, Revision 0, March 4, 1987
N2-SUT-23-3,	Feedwater System - TC3, Revision 1, February 18, 1987
N2-SUT-25-3,	Main Stream Isolation Valves - Test Condition 3, Revision 0, May 6, 1987
N2-SUT-29-3,	Recirculation Testing Revision 0, February 12, 1987
N2-SUT-30-3,	Reactor Recirculation System, Revision 0, February 12, 1987
N2-SUT-33-3,	Drywell Piping Vibration Test - Test Condition 3, Revision 1, December 23, 1986
N2-SUT-35-3,	Recirculation System Flow Calibration, Revision 0, April 1, 1987
N2-SUT-74-3,	Offgas System - TC3, Revision 1, February 12, 1987
N2-SUT-76-3,	ESF Area Cooling System - Test Condition 3, Revision 0, August 5, 1986
N2-SUT-77-2,	BOP and Small Bore Piping Vibration, Revision 0, October 20, 1986
N2-SUT-77-3,	BOP and Small Bore Piping Vibration, Revision 0, October 6, 1986
N2-SUT-79-3,	Reactor Internals Vibration Measurements 60 Percent Load Line, Revision 0, July 27, 1986
N2-SUT-80-3,	Emergency Recirculation Ventilation - TC3, Revision 0, July 17, 1986