U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-423/86-14

Docket No. 50-423

License No. NPF-49

Licensee: Northeast Nuclear Energy Co.

P.O. Box 270

Hartford, Connecticut 06141-0207

Facility Name: Millstone Nuclear Power Station, Unit 3

Inspection At: Waterford, Conneticut

Inspection Conducted: April 14-25, 1986

Inspectors: J. Prell 5/21/86
J. Prell, Reactor Engineer date

Peter C. Wen 5/27/86
P. Wen, Reactor Engineer date

Approved by: D. Horek for 5/27/86
P. Eselgroth, Chief, Test Programs Section, OB, DRS date

Inspection Summary: Routine unannounced inspection conducted on April 14-25, 1986 (Report No. 50-423/86-14)

Areas Inspected: Startup test results review and startup test witnessing.

Results: No violations were identified.

DETAILS

1.0 Persons Contacted

Northeast Nuclear Energy Co.

* T. Cleary, Unit 3 Engineer, NNECO
* C. Clement, Maintenance Supervisor, NNECO
* G. Closius, QA/QC Supervisor, NNECO
\+ R. Enoch, I&C Supervisor, NNECO
* J. Harris, Engineering Supervisor, NNECO
* T. Lyons, Startup Engineer, NNECO
+* D. McDaniel, Reactor Engineer, NNECO
* D. Moore, Assistant Operations Supervisor, NNECO
* R. Rothgeb, Acting Maintenance Supervisor, NNECO
* M. Pearson, Assistant Operations Supervisor, NNECO

U.S. Nuclear Regulatory Commission

+* F.Casella, Resident Inspector
\+ J.T. Shedlosky, Senior Resident Inspector

* Denotes those present at mini exit on April 18, 1986.

\+ Denotes those present at exit meeting on April 24, 1986. The inspectors also interviewed other personnel during this inspection period.

2.0 Power Ascension Program

2.1 Startup Test Program

During this inspection period (April 14-24, 1986), the licensee completed the power ascension testing program. Tests were conducted at the 90% and 100% power levels. These tests included, but were not limited to power coefficient measurement, automatic steam generator level control test, 10% load swing test at 100% power, and generator trip test from 100% power. This last test completed the power ascension testing program. An initial review of test results indicate that the test results were as expected.

2.2 Startup Test Results Evaluation

Scope

The test data results from the tests listed in Appendix A were reviewed to verify that adequate testing had been accomplished. The results were also reviewed to verify that regulatory guidance and licensee commitments were satisfied and to ascertain whether uniform criteria were being applied in the evaluation of completed tests in order to assure their technical and administrative adequacy.

Discussion

The inspector reviewed the test results and verified the licensee's evaluation of test results by review of: test changes; test exceptions; test deficiencies, "as-run" copy of the test procedure; acceptance criteria; performance verification; recording of the conduct of tests; restoration of system to normal after the test; independent verification of critical steps or parameters; identification of personnel conducting and evaluating test data; and verification that the test results had been reviewed and approved by licensee management.

2.2.1 Loss of Power(20% Power) (Appendix 8030)

The plant was tripped from 16% power and all AC power to the inverters and battery chargers was removed for 2 hours to force battery operation. The purpose of the test was to verify that the diesels started with proper load sequencing and plant response. During the test the service water pump cubicles began flooding. This was due to the fact that the sump pumps were not operating because they are operated from a non-vital motor control center. During a loss of offsite power these motor control centers receive no power. The licensee wrote an UNSAT against this and is studying several alternative methods of solution. One solution, which is a temporary procedure change, requires the Operations Department to periodically monitor the cubicles during a loss of offsite power. If the water leve' rises to high, the cubicle is drained by opening the isolation doors.

Using the following formulas the inspector verified the licensees calculations that the turbine driven auxiliary feedwater pump room did not exceed the environmental limits for relative humidity during the test.

$$PH=Pwb-\frac{(P-Pwb)(tdb-twb)}{2830-1.44\ twb} \quad \text{and} \quad HR=\frac{PH}{Pdb}$$

P= atmospheric pressure in inches of Hg.
Pdb= dry bulb pressure in inches of Hg (from steam tables)
Pwb= wet bulb pressure in inches of Hg (from steam tables)
tdb= dry bulb temperature in degress Fahrenheit
twb= wet bulb temperature in degress Fahrenheit
PH= actual partial pressure of water vapor
HR= relative humidity

Twenty seven problems were identified during the course of this test. These problems and their resolution were reviewed. Based on this review the inspector had no further questions.

2.2.2 Main Steam Isolation Trip Valve Closure Test(Appendix 8037)

The purpose of this test was to verify the closure time of the MSIVs when simultaneously closed at 20% power and the ability of the automatic control systems to sustain a simultaneous closure. Recorder charts and data sheets were reviewed to see if the test results met test acceptance criteria and that proper administrative controls had been followed.

No problems were identified.

2.2.3 Steam Dump Control(Appendix 8013)

The purpose of this test was to verify proper response of the Tave Steam Dump Control System for both the plant trip and load rejection modes of operation. This test was run at approximately the 15% power level. The test was reviewed against FSAR requirements and licensee administrative control procedures. Data were reviewed to verify that all acceptance criteria were met. Question related to the test were satisfactorily answered by the licensee.

No problems were identified.

2.2.4 Reactor and Turbine Control(Appendix 8005)

The purpose of test was to establish the optimum Tave Program which would result in the highest possible steam pressure and optinum plant efficiency without exceeding turbine input pressures or maximum allowable Tave. Primary steady state system temperatures, steam pressures and reactor thermal power data were obtained for 0, 30, 50, 75, 90 and 100% power levels. These data were compared to predicted data for different power levels.

No problems were identified.

2.2.5 Power Distribution(SP-31003)

The procedure and method used by the licensee to verify that the plant is operating within the power distribution limits defined in TS were reviewed and discussed with cognizant licensee personnel. The incore flux map data taken by the Digital Flux Mapping System was analyzed by the plant computer using the licensee version of Westinghouse "INCORE" code. The licensee performed "INCORE" verification runs per test procedure 3-INT-2001, Computer Programs Test, Appendix 3R10, prior to entering the program into the plant computer for current cycle operation. The results of this verification were satisfactory.

The inspector independently reviewed the flux map taken at 100% power level on April 19, 1986. All parameters Fxy (Radial Peaking Factor), Fq(Z) (Heat Flux Hot Channel Factor), FΔH N (Nuclear Enthalpy Rise Hot Channel Factor), and QPTR (Quadrant Power Tilt Ratio) were within TS limits. The inspector also noted that all measured values of assembly power were in good agreement with predicated values.

No unacceptable conditions were identified.

2.2.6 Incore/Excore Axial Flux Difference(AFD) Calibration (Appendix 8028)

Incore/Excore AFD calibration was first performed at 75% power on March 27 and 28, 1986. Nine (9) axial flux differences obtained from 9 flux maps (2 full core and 7 quarter core flux maps) were analyzed and compared to responses of the excore detectors to develop a calibration curve for each power range detector. During data reduction (incore Δq vs. detector current plot), the licensee's reactor engineer noticed that N44 and N42 exhibited anomalous behavior. These included a significantly different dq/I slope for N44 and N42 upper and lower detectors responses. As documented in the NRC Inspection Report 50-423/86-08, the cause of this problem was attributed to water being inadvertantly introduced to the detector wells during the neutron shield tank test. N44 and N42 were subsequently replaced with spare detectors.

The Incore/Excore AFD calibration test was reperformed using four (4) additional flux maps data on April 14, 1986. The calibration curves from these four data points were consistent with calibration data generated from the first test (performed on March 27 and 28, 1986) for detectors N41 and N43.

Because the licensee experienced problems in Turbine EHC during the second Appendix 8028 test, only four flux maps could be obtained. Since detectors N41 and N43 showed consistent test results between the two Appendix 8028 tests, the calibration curves from this second test was thus applied to detectors N42 and N44.

During review of the licensee's 100% power flux map (flux map taken on April 19, 1986), the inspector noted that all power range AFD readings were consistent with the incore value with a deviation of only about 0.28%. This verified that the licensee Incore/Excore AFD calibration result is adequate.

No unacceptable conditions were identified.

2.2.7 RCS Flow Measurement(Appendix 8015)

This test was performed previously at 50% power. All 12 flow transmitters (from elbow flow taps) were calibrated at that time against the precision heat balance flow measurement result. The licensee reperformed this test at 90% power level on April 15, 1986. Test results essentially confirmed that the adjustment made previously was adequate. The indicated flows were either conservative with respect to Appendix 8015 test result or within instrument tolerance 0.5%.

The inspector had no further questions.

2.3 Startup Test Witnessing

The inspector witnessed the tests described below. The tests were reviewed against the attributes identified in inspection report 50-423/86-07 section 5.2.

2.3.1 10% Load Swing Test(Appendix 8022)

The inspector witnessed the 10% load swing test at 100% power performed on April 21, 1986. The purpose of the test was to verify proper plant transient response, including automatic control system performance. In a normal transient no manual intervention is required. The automatic control systems such as reactor rod control, steam generator level control, pressurizer pressure control, pressurizer level control, steam dump control, and feedwater pump speed control would bring plant conditions to steady state at a pre-determined power level. In this case however, it appeared that one of the feedwater regulating valve malfunctioned during the 10% load reduction test. The steam generator level experienced a larger than expected oscillation (> 5% from initial level). The operator correctly took manual control on steam generators 'B' and 'D' feed regulating valves and minimized the transient effects. As documented in the NRC inspection report 50-423/86-11, the steam generator level control also experienced unsteady behavior during normal power ascension from 54% to 65% power. The licensee I&C personnel conducted a special level control check on steam demand during the most recent plant startup (recovering from 75% power plateau test). However, the test result is inconclusive. At the exit meeting, a licensee representative stated that the steam generator level control problem will be continuously evaluated.

2.3.2 Generator Trip from 100% Power(Appendix 8032)

This test was performed on April 21, 1986, in accordance with test procedure Appendix 8032, "Generator Trip from 100% Power", during which the generator output breaker was manually opened from the control room. Opening of the generator output breaker caused a main turbine trip and subsequent reactor trip. Preliminary test results indicated that plant systems responded as designed, neither pressurizer safety valves nor steam generator safety valves lifted during the test. Performance of plant operators and test engineers appeared to be good. Operators were attentive, maintained their stations, monitored appropriate instrumentation, and reported important readings and alarms. The test procedure was followed completely. Since a reactor trip took place, the immediate actions of Emergency Operating Procedure E-0 "Reactor Trip and Safety Injection", were followed. A senior reactor operator read the procedure actions aloud and received formal responses from operators concerning completed actions. Communications during performance of post-trip actions were good. In general, annunciators were properly acknowledged.

The inspector conducted a plant walk-down following completion of the test. The inspector noted in the turbine building that two feed pump strainers were leaking. At the same time licensee personnel, also on the scene, immediately informed the control room operator. Due to loss of extraction steam following a reactor trip, a severe thermal transient appeared to cause the water leaking through both strainer manways. This problem was fixed through work order maintenance AWO 86-8377 and 86-8379 on April 21, 1986. No further leaks were identified during the subsequent plant startup.

3.0 QA/QC Interface

During performance of 10% load swing test at 100% power level, the inspector noted that QA/QC coverage was provided. A specialist in steam generator level control from NUSCO was also present to augment startup test coverage. During the 100% power reactor trip test, an additional QA auditor from NUSCO was present to witness the test. The licensee QA/QC coverage on startup testing during this inspection period appeared to be adequate.

4.0 Independent Calculation/Verifications

The inspector independently verified that the licensees calculations relating to the environmental limits requirements during the loss of power test were accurate (Section 2.2.1). The inspector also independently reviewed the 100% power flux map taken on April 19, 1986. The inspector verified that engineering and nuclear uncertainties as required by the TS were applied in the data summary. (Section 2.2.5)

5.0 Exit Meeting

A mini exit meeting was held on April 18, 1986 and a final exit meeting was held on April 24, 1986 to discuss the inspection scope and findings, as detailed in this report (see paragraph 1.0 for attendees). At no time was written material given to the licensee. The inspector determined that no proprietary information was utilized during this inspection.

APPENDIX A

TEST RESULTS REVIEW

3-INT-8000, Appendix 8005, Revision 0, Reactor and Turbine Control

3-INT-8000, Appendix 8013, Revision 0, Steam Dump Control

3-INT-8000, Appendix 8030, Revision 0, Loss of Power (20% Power)

3-INT-8000, Appendix 8037, Revision 0, Main Steam Isolation Trip Valve Closure Test

3-INT-8000, Appendix 8028, Incore/Excore AFD Calibration

3-INT-8000, Appendix 8015, RCS Flow Measurement