U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-423/86-09

Docket No. 50-423

License No. NPF-49

Licensee: Northeast Nuclear Energy Company

P. O. Box 270

Hartford, Connecticut 06141-0270

Facility Name: Millstone Nuclear Power Station, Unit 3

Inspection At: Waterford, Connecticut

Inspection Conducted: February 18-March 14, 1986

Inspectors: J. Prell — 4/29/86
J. Prell, Reactor Engineer — date

Peter C. Wen — 4/30/86
P. Wen, Reactor Engineer — date

Approved by: A. W. Eselgroth — 5/2/86
P. Eselgroth, Chief, Test Programs Section, OB, DRS — date

Inspection Summary:
Routine Unannounced Inspection Conducted On February 18-March 14, 1986 (Report No. 50-423/86-09)

Areas Inspected: Startup program review, power ascension test procedure review, test results review, test witnessing, and review of licensee actions on previous findings.

Results: One violation was identified (failure to implement the defined QC surveillance program).

DETAILS

1. Persons Contacted

*G. Clossius, QA Supervisor, NNECO
+J. Crockett, MP-3 Unit Superintendent, NNECO
P. Finck, Startup Engineer
E. Fries, Startup Engineer, NNECO
*J. Jensen, QA Specialist, NNECO
E. Laware, QA Engineer, NNECO
C. Libby, Operations QA Supervisor, NUSCO
*+D. McDaniel, Reactor Engineer
D. Miller, Startup Manager, NNECO
D. Moore, Assistant Operations Supervisor, NNECO
M. Pearson, Assistant Operations Supervisor, NNECO
+W. Richter, Assistant Startup Supervisor, NNECO

U.S. Nuclear Regulatory Commission

+F. Casella, Resident Inspector
T. Rebelowski, Senior Resident Inspector
+T. Shedlosky, Senior Resident Inspector

+Denotes those present at mini exit meeting on February 28, 1986
*Denotes those present at the exit meeting on March 14, 1986. The inspectors also interviewed other personnel during this inspection period.

2.0 Licensee Action of Previous Inspection Findings

(Open) Unresolved Item (50-423/85-34-03) pertaining to qualification of the containment High Range Radiation Monitor, Kaman Model KDI-1000 and Mineral Insulation Cable Assembly.

-- Qualification of Model KDI-1000 High Range Containment Area Radiation Detector and Mineral Insulation Cable Assembly. Report No. 46-0036-001, Revision A.

-- Qualification Report for Model KDI-1000 High Range Containment Area Radiation Detector and Mineral Insulation Cable System. Report No. 460036-002, Revision 2.

-- Installed Specification for Mineral Triaxial Cable Penetration Assembly. Report No. 460036-002, Revision A.

-- Installation Specification for Mineral Insulation (MI) Cables for High Range In-Containment Area Monitors.

In reviewing the above documents the inspector concluded that additional test data needs to be made available for review in order to conclude that the Containment High Range Radiation Monitor, Kaman Model KDI-1000 and MI Cable Assembly, will function as required in a LOCA environment.

The basis for this determination is the numerous deviations, anomalies and failure reports (DAFR) identified in the qualification report No. 460036-002, Revision 2 which reduced acceptance criteria for problem areas.

For example:

-- DAFR-9, -10, -13 reduced torque cycle requirements in an attempt to correct relaxation problems of torqued connectors.

-- DAFR-14 deletes the voltage withstand test of signal cables as a pre-seismic functional test.

-- DAFR-16 requires the chemical spray to be applied only during non-superheated conditions.

-- The thermal transients behavior effect inside the cable during rapid temperature transients add a false signal to the radiation signal.

-- There are numerous references to test failures attributed to moisture intrusion/contamination due to a relaxation of torqued connectors which compromises the seal.

-- The critical assembly/handling requirements for connector/cables, requires a clean room type facility when servicing or calibrating the monitoring system. This is not found inside containment.

-- The MI cables are subject to rupture from mishandling, surface scratches and breaking of the sheath due to excessive bending.

The licensee has experienced considerable difficulties in installing a functional Kaman High Range Containment Monitor due to problems in finding an acceptable length of MI cable assembly. As a result, the licensee has received an NRC exception to use an acceptable shortened cable assembly with one Kaman monitor installed at a lower containment elevation. Two other monitors (General Atomic) have been located inside containment at the proper elevation.

This item remains open pending receipt of conclusive environmental test data that ensures equipment availability in compliance with the requirements of NUREG-0737, Table II F.1-3.

3.0 Startup Test Program

3.1 Procedure Review

Scope

The following approved power ascension test procedure was reviewed for technical and administrative adequacy and to verify that test planning satisfies regulatory guidance and licensee commitments:

3-INT-8000 Appendix 8032, Revision 0, Generator Trip From 100% Power.

Discussion

The above procedure was examined for: management review and approval; procedure format; clarity of stated test objectives; prerequisites; environmental conditions; acceptance criteria; source of acceptance criteria; references; initial conditions; attainment of test objectives; test performance documentation and verification; degree of detail for test instructions; restoration of system to normal after testing; identification of test personnel; evaluation of test data; independent verification of critical steps or parameters and quality control and assurance involvement.

Findings

The review indicated that the procedure was consistent with regulatory requirements, guidance, and with the Licensee's commitments. No discrepancies or unacceptable conditions were identified. The inspector had no further questions on this procedure.

3.2 Test Witnessing

Scope

The following power ascension tests were witnessed to verify the licensee's conformance to regulatory and procedural requirements, to observe the performance of the operating staff, and to ascertain the adequacy of test program records, including preliminary evaluation of test results:

3-INT-8000 Appendix 8023, Revision 0, Reactor Trip and Shutdown From Outside the Control Room.

3-INT-8000 Appendix 8006, Revision 0, Secondary Plant Performance Test.

Discussion

The above tests were witnessed to verify that; the operating crew was using the most current test procedure, minimum crew requirements were met, all test prerequisites and initial conditions were met, required test equipment was properly calibrated, the implementing procedures were technically adequate to perform the test, crew actions during the test were correct and timely, a quick analysis and evaluation was made to assure proper plant response to the test, all test data were collected for analysis by the appropriate personnel, overall test acceptance requirements were met and adherence to TS requirements was met for any LCO's affected by the test.

FINDINGS

-- Appendix 8023, "Reactor Trip and Shutdown from Outside the Control Room" - The inspector verified that the test satisfied R.G. 1.68 Appendix A, paragraph 5.d.d and FSAR requirements. Credit was taken for a test performed during the pre-core hot functional test program which demonstrated that the plant could be placed from a hot standby to a hot shutdown condition. Therefore, this portion of the test was not conducted.

The inspector observed the pretest briefings of both the Test crew, located at the Auxiliary Shutdown Panel, and the Normal crew located in the Control Room. The briefing of the Test crew included a step by step review of the procedure, identification of the various controls and predicted plant indications and alerting the crew to actions to be taken in the event unsafe conditions developed during the test. The Normal crew was briefed on the purpose of test and the actions they could and could not perform during the test.

It was verified that communications were established and maintained during the test between the Auxiliary Shutdown Panel (ASP) and the Control Room, Control Transfer Panels (orange and purple trains), the Auxiliary Building and the Engineer Safeguards Features Building.

The reactor was manually tripped from 15% power with the CRDM breaker located in the Auxiliary Building. Emergency Operating Procedure EOP 3503 "Shutdown Outside Control Room," Revision 2 was used to shutdown the reactor and transfer control to the ASP. It was verified that the test operators were able to maintain Tave at 557°F for over 1/2 hour from the ASP and then transfer control of the plant back to the Control Room. No problems were identified with the test, test results or performance of the operating crew. A problem was identified with QC involvement with this test. See paragraph 4.0 "QA/QC Interface" for details.

-- Appendix 8006, "Secondary Plant Performance Test". This test is performed at various power levels - 30%, 40%, 50%, 75%, 90% and 100% - in order to obtain base line data and identify any secondary plant problems. The inspector witnessed this test at the 30% power level. It was verified that the latest procedures were being used, the test crew was knowledgeable of the program and their responsibilities and the test data was being reviewed and evaluated against expected results.

No problems were identified.

3.3 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 if 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; QA inspection records; 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.

FINDINGS

Turbine Overspeed Test (Appendix 8016)

The capability of the mechanical overspeed trip device for the main turbine generator was tested on February 15, 1986. Three (3) actual trips occurred at 1962, 1963, 1963 rpm, respectively. This test was accepted as satisfactory since the measured values were within test procedure established criterion of ≤1998 rpm. The Backup Overspeed Trip (BOST) Normal Mode is set approximately 0.5% above the mechanical overspeed trip setting, to provide additional protection against turbine overspeed during plant normal operation. The BOST electronic circuit was also tested at 105% rated speed prior to performing mechanical overspeed trip test. The test result was also satisfactory.

Reactor Trip/Shutdown Outside Control Room (Appendix 8023)

The objectives of this test was to demonstrate the following:

a. Verification that the unit can be safely shutdown from outside the control room using the Auxiliary Shutdown Panel.

b. Verification that the unit can be maintained in a hot standby condition from outside the control room.

c. Verification that the unit can be cooled down to the hot shutdown condition from outside the control room.

Through test results review and test witnessing, the inspector noted the licensee has demonstrated the remote shutdown capability and met the above test objectives with the exception of item c.

The cold shutdown capability (item c) was demonstrated earlier during preoperational test, 3-INT-3000, Appendix 3014, "Remote Shutdown with Cooldown" on November 1, 1985.

The remote shutdown capability test was accepted as satisfactory by the JTG.

Automatic Reactor Control (Appendix 8017)

The performance of the automatic control system in maintaining reactor coolant average temperature within programmed value was verified per test procedure 3-INT-8000, Appendix 8017 on February 17, 1986. The inspector independently reviewed test data and noted that automatic reactor control system responded well during the test. Tavg returned to programmed value within specified oscillation amplitude, and no unstable response was observed. The corresponding system responses such as pressurizer pressure, pressurizer level and steam generator level responded well with no unusual behavior noticed.

During performance of this test, Step 5.5 required the feedwater pump speed control system to be set per 3-INT-3000, Appendix 3010. A test engineer misread Appendix 3010 to be Appendix 8010. Since Appendix 8010, Neutron Shield Tank Testing, has nothing to do with this test, Step 5.5 was deleted during the test. The inspector informed the licensee of this discrepancy. The test engineer re-reviewed the test result and determined that this test change did not have adverse impact on the test outcome, since Appendix 3010 had already been completed. The inspector concurred with the licensee's assessment and had no further questions.

Power Coefficient (Appendix 8020)

The doppler-only power coefficient measurements are to be obtained at 30, 50, 75, 90 and 100 percent power. The measurement at 30% power level was performed on February 17, 1986.

Since it is difficult to directly measure the reactivity change due to fuel temperature change during a power change, an indirect measurement technique was utilized. This technique involved measuring the primary side responses such as the change in core average temperature (ΔTAVG) and the corresponding change in RCS loop parameters with respect to a small change in turbine load. The doppler-only power coefficient was then calculated from this set of data and isothermal temperature coefficient information as previously derived from Zero Power Physics Test.

All test results, including five sets of power swing data, met the test acceptance criteria.

The inspector noted that an assumption of $dTm=dT_f$ (Where T_m is moderator temperature and T_f is average fuel temperature) was used in the licensee's measuring methodology for the doppler-only power coefficient derivation. The validity of this assumption was discussed with licensee reactor engineer and consultation provided by a NRR Core Physics specialist. This assumption is valid when there is no power change. From the test results and information provided by the fuel vendor (Westinghouse), it appeared that this assumption is also valid under test condition with minimum power change.

The inspector had no further questions.

Natural Circulation Test (Appendix 7006)

The test witness and preliminary test result review was documented in the Inspection Report 86-07. Although the licensee completed this test and demonstrated that plant core heat can be removed satisfactorily by using natural circulation, the detailed test results had not been thoroughly analyzed. From the test result review, the inspector noted that PORV lifted 3 times during the initial stage of the natural circulation test. Plant behavior and lessons learned from this test will be evaluated by the licensee and incorporated in the licensee operator training program and possible procedure enhancement.

The licensee management agreed that this detailed evaluation and incorporation into operator's training will be completed by May 1, 1986. This is an unresolved item (50-423/86-09-01).

RCS Flow Coastdown (Appendix 5017)

The purpose of this test was to verify that the RCS responded as designed to a partial and complete loss of forced reactor coolant flow. With four reactor coolant pumps (RCPs) operating, partial loss of forced reactor coolant flow was established by tripping one RCP. Complete loss of reactor coolant flow was established by simultaneously tripping all four RCPs.

The inspector verified that low flow alarm times, control rod drop times, core flows and low flow alarm values met FSAR Chapter 15 Table 15.3-1 requirements. This was determined by reviewing the strip chart recordings and the sequence of events computer printouts for reactor coolant loop flow, RCP alarm, reactor trip times, and low flow alarms. It was also verified that the licensee had correctly translated the information from the strip chart recorders and computer print-out to the data sheets and, by performing independent calculations, that the final results were correct.

Post Core Hot Functional Tests UNSATS

A review of the disposition of all UNSATs identified during the Post Core Hot Functional Test Program was made. These UNSAT's, identified in Appendix B, were reviewed for conformance with administrative requirements and proper disposition of problems. No problems were identified.

4.0 QA/QC INTERFACE

While observing the performance of 3-INT-8000 Appendix 8023, Reactor Trip/Shutdown Outside the Control Room, the inspector questioned a QC inspector also witnessing the test as to his involvement with the test. From these discussions the following information was determined:

-- Although the QC inspector stated that he was there to verify operator compliance to EOP 3503, he had not yet reviewed EOP3503 or Appendix 8023 nor did he have a copy of the procedures with him.

-- The QC inspector had not been briefed by his management on what to look for or expect during the test. This was later verified by his immediate supervisor.

-- The QC inspector had not reviewed the FSAR or RG 1.68 requirements pertaining to this test although he was aware he should do so.

-- The QC inspector was not aware of what was taking place as evidenced by the fact he did not know from what power level the reactor had been tripped.

The NUSCO QA and NNECO QC departments share responsibility for performing surveillances of startup tests. This responsibility is loosely coordinated with NUSCO QA having primary responsibility. During this particular test, representatives from both organizations were present. The auditor from the QA section appeared knowledgeable of the test procedure and what was taking place.

The above occurrence of an inspector being assigned an inspection task for which that inspector had not been trained or briefed constitutes a violation (50-423/86-09-02).

At the exit meeting, the licensee QA representative agreed that:

-- More specific guidance will be provided to QC inspectors who will cover the remaining startup tests, as to which areas/criteria to look into.

-- NUSCO QA will continue to provide surveillances during the startup test program.

The following NUSCO Operations QA surveillance reports were reviewed to determine the adequacy of OQA's involvement with the Startup Test Program:

NUSCO OQA SURVEILLANCE NO.	TITLE
TC 3950	Digital Rod Position Indication
TC 3960	Boron Endpoint Measurements
TC 3960A	Boron Endpoint Measurements
TC 3961	Initial Criticality
TC 3961A	Initial Criticality
TC 3968	Natural Circulation
TC 3972	Preparations for Power Ascension Testing
TC 3976	Power Ascension Test
TC 3986	Integrated Plant Testing
TC 3986A	Integrated Plant Testing
TC 3986B	Integrated Plant Testing

The surveillance appeared to be thorough with good follow-up of identified concerns. No problems were identified.

5.0 Independent Calculations

3-INT-5000 Appendix 5010, "RTD Bypass Loop Verification" obtains data which is used to calculate the hot leg and cold leg flow and transport times through the RTD Bypass Lines. Using the formulas:

$$F_{H_1} = \frac{Ft}{(1+ \frac{Fc}{F_H})} \qquad Fc' = F_t - F_{H1}$$

Fc' = Calculated cold leg flow

F_{H_1}= Calculated hot leg flow

F_t= Measured Total flow

Fc = Measured cold leg flow
F_H = Measured hot leg flow

The inspector verified the flows and transport time for the hot and cold leg RTD bypass lines.

Power coefficient measurement requires lengthy data reduction. The inspector independently verified that the predicted reactor physics parameters were correctly taken from the nuclear design reference. The inspector also performed an independent calculation and confirmed that the first power swing case data were being correctly reduced.

6.0 Plant Tours

The inspector made several tours of the facility during the course of the inspection. This included tours of the control building and control room. A review of the work in progress, security, cleanliness and housekeeping was made.

7.0 Exit Meeting

An exit meeting was held on March 14, 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 DATA REVIEWED

TEST NUMBER	TITLE
3-INT-5000 Appendix 5001	Shutdown Margin
3-INT-5000 Appendix 5010	RTD Bypass Loop Verification
3-INT-5000 Appendix 5015	Digital Rod Position Indication Operation Test
3-INT-5000 Appendix 5016	Loose Parts Monitoring
3-INT-5000 Appendix 5017	RCS Flow Coastdown
3-INT-5000 Appendix 5031	Chemical and Volume Control System
3-INT-6000	Initial Criticality
3-INT-7000 Appendix 7006	Natural Circulation
3-INT-8000 Appendix 8023	Reactor Trip/Shutdown Outside Control Room
3-INT-8000 Appendix 8016	Turbine Overspeed Test
3-INT-8000 Appendix 8017	Automatic Reactor Control
3-INT-8000 Appendix 8020	Power Coefficient

APPENDIX B

POST CORE HOT FUNCTIONAL TEST UNSATS REVIEWED

TEST PROCEDURE	UNSAT #'s
3-INT-5000, Anpendix 5001	7497
3-INT-5000, Appendix 5002	7471
3-INT-5000, Appendix 5004	7341, 7342
3-INT-5000, Appendix 5006	7495, 7492, 7493
3-INT-5000, Appendix 5007	7485, 7486, 7489, 7496
3-INT-5000, Appendix 5009	7466
3-INT-5000, Appendix 5015	7487
3-INT-5000, Appendix 5016	7475, 7479
3-INT-5000, Appendix 5017	7504, 7510
3-INT-5000, Appendix 5031	7472, 7473, 7474, 7476 7477, 7478, 7484, 7488 7490, 7491, 7499
3-INT-5000, Appendix 5033	7378, 7417, 7420