U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-423/86-07

Docket No. 50-423

License No. NPF-49 Priority -- Category --

Licensee: Northeast Nuclear Energy Co.

P. O. Box 270

Hartford, Connecticut 06141-0270

Facility Name: Millstone Nuclear Power Station - Unit 3

Inspection At: Waterford, Connecticut

Inspection Conducted: January 19 - February 14, 1986

Inspectors: James A. Prell 3/11/86
James A. Prell, Reactor Engineer date

Peter C. Wen 3/10/86
Peter Wen, Reactor Engineer date

Approved by: for Larry E. Briggs 3/12/86
Peter Eselgroth, Chief, Test Program Section, OB, DRS date

Inspection Summary: Routine unannounced inspection conducted on January 19 - February 14, 1986 (Report No. 50-423/86-07)

Areas Inspected: Startup program review, post core hot functional test witnessing and test result review, initial criticality and low power physics testing, power ascension program review and test witnessing, and review of licensee actions on previous inspection findings. The inspection involved 225 hours on site by two region based inspectors.

Results: No violations were identified.

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DETAILS

1.0 Persons Contacted

R. Bradley, Startup Engineer, NNECO
*K. Burton, Operations Supervisor, NNECO
*J. Crockett, MP-3 Unit Superintendent, NNECO
E. Fries, Startup Engineer, NNECO
N. Hulme, Startup Engineer, NNECO
*J. Jensen, QA Specialist, NNECO
T. Kulterman, Senior Engineer, NUSCO
J. Langon, Plant Engineer, NNECO
T. Lyon, Startup Test Engineer, NNECO
D. Miller, Startup Manager, NNECO
D. Moure, Assistant Operations Supervisor, NNECO
M. Pearson, Assistant Operations Supervisor, NNECO
*W. Richter, Assistant Startup Supervisor, NNECO
A. Stengal, Startup Test Engineer, NNECO
C. Wooten, Startup Test Engineer, NNECO
*D. McDaniel, Reactor Engineer, NNECO

U.S. Nuclear Regulatory Commission

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

*Denotes those present at the exit meeting on February 14, 1986.

2.0 Licensee Action on Previous Inspection Findings

(Closed) IFI (50-423/85-71-16) The technical manual and precautions in SP 3622.5, relating to the Turbine Driven Auxiliary Feedwater Pumps, (TDAFWP) caution the operator to exercise the manual speed adjusting knob following turbine operation to prevent system failure. A review of the procedure related to TDAFWP operation found that they did not adequately delineate this precaution nor did they require independent verification that the governor was reset to its standby position.

The inspector verified that Revision 7 to OP 3322 requires the operator to exercise the manual speed adjusting knob and provide independent verification of valve lineup. This item is closed.

(CLOSED) UNR (50-423/85-71-17) The following items were identified related to SP 3622 and Op 3322:

1) OP 3322, step 4.2 refers to TS Figure 3.7. Figure 3-7 is not included in the TS.

2) SP 3622.6 requires operation of the turbine driven AFW Pump at the minimum governor speed for 2 minutes. OP 3322 state that the AFW pump should not be operated below 1500 rpm to "ensure proper lubrication"

3) OP 3322 requires manual lubrication of the AFW pumps bearings if the pumps have not been operated within the previous 30 days. The procedures did not however provide lubrication instructions.

The inspector verified that the licensee has deleted the reference in OP 3322 to TS Figure 3.7 and replaced it with reference to TS 3.7.1.3.

The minimum governor speed setting for the turbine driven AFW pump is 1700 rpm, which is well above the 1500 rpm minimum speed stated in OP 3322. Also a standby D.C. Oil Lubrication pump would start on a low pressure signal to "ensure proper lubrication" if the rpms went below 1500 rpm.

A letter from the manufacturer now indicates that manual lubrication is required for AFW pumps bearings if they have not been operated within the previous 40 days versus 30 days. Technical Specification surveillance requirement 4.7.1.2.1 requires each AFW pump to be demonstrated OPERABLE at least once per 31 days by developing a differential pressure of 1460 psid across each motor-driven pump and across the steam-turbine driven pump. Thus all pumps would have to be operated at least once every 39 days (31 days + 25%), which is one day short of 40 days maximum. Thus making the manual lubrication problem moot.

Based on the above this item is closed.

(CLOSED) UNR (50-423/85-71-21) The precautions and restoration sections of SP 3646A.3, "Diesel Generator Interdependent Test", both listed "None", which appeared inappropriate. Also the procedure did not include provisions for independent verification of equipment restoration following testing.

The reason there was no precautions or restoration section for SP 3646A.3 is that the SP references OP 3346A for implementation. The inspector verified that OP 3346A has precautionary statements and restores the system. The SP data sheet now incorporates provisions for independent verification of system restoration.

Based on the above this item is closed.

(OPEN) UNR (50-423/85-71-22) Various problems were identified with the draft procedure SP 3646A.6, "Offsite Power Transfer Operability Test." However, the problems identified by the inspector refer to testing of relays associated with the ESF system. Thus SP 3646A.4 "Engineered Safety Systems Integrated Test" appears to be the procedure of concern.

SP 3646A.4 is still in draft form. This surveillance however is only to be done during refueling operations. The initial surveillance was satisfied by preoperational test 3-INT-2004, "ESF Without Loss of Power".

Based on the above this item remains unresolved pending review of the final approved surveillance procedure SP 3646A.4.

(CLOSED) CAT Finding - Violation (50-423/85-04-03): This item pertains to the welds identified as nonconforming to the procurement specifications related to some of the installed tanks, pressure vessels, and heat exchangers. The licensee's field quality control representatives reinspected all fillet welds identified as "Undersized" for equipment nozzles and supports for the subject QA Category I tanks, pressure vessels, and heat exchangers. Subsequently, the nonconformance was evaluated for actual stress loading and reworked, as required, under SWEC's QA program, to meet the ASME Code for minimum weld size. To preclude similar incidents from happening in the future, the SWEC QA department instituted a comprehensive weld inspection training program for QA/QC personnel, including project QA inspectors. In addition, the SWEC QA division, through a memorandum to all district managers, stressed the company's commitment to proper fillet weld inspection. A review of the documentation and field verification supported the adequacy of the licensee's corrective action against this deficiency.

Based on the above, the inspector determined that the licensee's action is complete. This item is closed.

(CLOSED) UNR (50-423/85-34-04) This item pertains to effluent monitoring sensitivity data. The inspector reviewed the Kaman summary report and detector calibration data for the particulate and gas radiation monitoring systems. The detector is certified to have the characteristics noted in the referenced documents. Calibration is based on deposited activities on a copper disc using beta efficiencies of a proportional counting system. The beta and gamma scintillation detectors used were first calibrated using the standard factory procedure KNP 18-60 revision E.

No violations were identified.

This item is closed.

3. Post Core Hot Functional Test

3.1 Test Witnessing

A various times during the inspection period, the inspectors witnessed testing in progress on a sampling basis and evaluated most portions of the Post Core Hot Functional Test. The tests witnessed included:

-- 3-INT-5000, Appendix 5001, Shutdown Margin;

-- 3-INT-5000, Appendix 5004, Rod Control Slaver Cycle and CRDM Timing;

-- 3-INT-5000, Appendix 5006, RCS Leak Detection;

-- 3-INT-5000, Appendix 5007, Pressurizer Heaters and Spray;

-- 3-INT-5000, Appendix 5008, Rod Drop Testing;

-- 3-INT-5000, Appendix 5011, Movable In-Core Detector Operation;

-- 3-INT-5000, Appendix 5015, Digital Rod Position Indication Operational Test;

-- 3-INT-5000, Appendix 5018, Rod Control Operational Test;

Tests were observed for the following areas:

-- Tests were conducted in accordance with the approved test procedures; change to the procedures were made in accordance with the administrative procedure.

-- Prior to performing each test, briefing with the test crew and operation personnel were conducted and the briefing was adequate.

-- Test prerequisites and initial conditions were met.

-- Operator actions were correct.

-- Summary analysis was made upon completion of each test.

All test results as verified by inspector direct observations indicated that overall test acceptance criteria have been met or proper test deficiencies were documented and followed up by the licensee.

The inspector also attended some of PORC/JTG (Plant Operation Review Committee/Joint Test Group) meetings which approved test procedure change and final test result acceptance. The inspector noted that in all instances the PORC/JTG approval process was formal, thorough, and deliberate.

3.2 Test Results Review

Those Post Core Hot Functional Test results identified in Appendix A were reviewed by the inspector to verify that:

-- test changes were approved and implemented in accordance with administrative procedures;

-- changes did not impact the basic objectives of the test;

-- test deficiencies and exceptions were properly identified, resolved, and resolution accepted;

-- the cognizant engineering group had evaluated the test results and signified that testing demonstrated design conditions were met; and,

-- test results compared with established acceptance criteria or were properly resolved.

Details relating to some of those test results reviewed are described below.

3.2.1 RCS Leak Detection (Appendix 5006)

The Millstone Unit 3 Technical Specifications (TS) require that surveillance be performed at least once per 72 hours using water inventory balance method to determine leakage from the reactor coolant system (RCS). The plant computer RCS leakage calculation program, SP 3J3 is designed to fulfill this requirement. Surveillance procedure SP 3601F.6, RCS Leak Test, delineates calculational steps and is used as a backup when the plant computer is not available. The purpose of this RCS Leak Detection test (Appendix 5006) was to verify that a known 1 gpm leak rate could be detected by the RCS Leakage Calculation Program SP 3J3 and surveillance procedure SP 3601F.6.

Baseline Leak Rate Test

Prior to performing superimposed leak rate test, a baseline RCS leak rate was measured per program SP 3J3 and surveillance procedure SP 3601F.6 on January 22, 1986. The inspector performed independent calculations using an NRC-developed leak rate computer program NUREG-1107, "RCSLK9: Reactor Coolant System Leak Rate Determination for PWRs", to verify the licensee's calculation. These comparisons are:

Test Date	Leak Rate (LR) (gpm)	Licensee Calculation Pgrm 3J3	Hand Cal.	Inspector Cal.
1-22-85 (0630-1030)	Identified LR	0.723	0.74	0.73
	Unidentified LR	0.362	0.73	0.36

Based on the same set of initial and final data input, the inspector's calculation agreed closely with the licensee's program SP 3J3 calculation. The licensee's RCS leakage water inventory calculation methodology appeared to be adequate.

However, from test results review, the inspector identified the following items were either not included or not clearly spelled out in the licensee's procedure.

-- The large variation of unidentified leak rate observed between the computer program SP 3J3 and surveillance procedure hand calculation (0.362 gpm vs. 0.73 gpm) was primarily due to different values of initial pressurizer level used in the calculation. Instantaneous values at beginning and end of test were used in the licensee's calculation. Since the computer point data and the hand calculation input data were taken at slightly different times, system variation and/or instrument uncertainty might have contributed to this deviated inputs. The large uncertainty of RCS leakage calculation caused by this system variation could have an adverse impact on calculation results especially when the plant is experiencing some known leakage.

-- Level changes in Containment Drains Transfer Tank (CDTT), Primary Drains Transfer Tank (PDTT), Pressurizer Relief Tank (PRT) and Accumulator Tank are counted as an identified leakage. The unidentified leak rate is calculated by subtracting the identified leak rate from the total RCS leak rate. The TS limit (TS 3.4.6.2) for the unidentified leakrate is 1 gpm. Thus, incorrect determination of identified leak rate would affect the adequacy of the unidentified leak rate calculation. The inspector noticed that the level change in these tanks, especially the PDTT, could come from various sources other than RCS. However, the surveillance procedure does not clearly indicate that leakage from RCS can only be considered for identified leakage.

In addition, during the test, computer point GWS-F84 (VCT Divert flow to BRS) was observed having noise spikes which caused an erroneous computer entry of the VCT dump flow. The computer program SP 3J3 requires manual adjustment of the calculated leak rate under these circumstances.

These items are collectively an unresolved item pending further review by the inspector (50-423/86-07-01).

Superimposed Leak Rate Test

At the end of 4 hours baseline leak rate test, a known leak rate of approximately 1 gpm was induced by opening valve 3CHS-V800 (drain to auxiliary building sump). The Superimposed Leak Rate Test was subsequently performed for 2 hours. The licensee calculated leak rate results along with the inspector independently calculated results are compared below:

Test Date	Leak Rate (LR) (gpm)	Licensee Calculation Pgrm 3J3	Hand Cal.	Inspector Cal.
1-22-85	Identified LR	0.654	0.61	0.73
(1156-1401)	Unidentified LR	1.625	1.76	1.54

The change in the RCS leak test unidentified leakage is:

Licensee Calculation		
Program 3J3	Hand Calculation	Inspector Calculation
1.263	1.03	1.18

This comparison indicated that the licensee's computer program performed adequately well. The superimposed leak rate as determined by the licensee's computer program was 1.263 gpm (1.625 - 0.362) which did not meet test acceptance criteria of 1 gpm ± 9%. This discrepancy was discussed during JTG test result review. The relatively large uncertainty associated with this test was attributed to the Controlatron flow instrument reading (3CHS-FI 391) which monitored the induced flow rate. During the test, this flow meter reading was shown to fluctuate from 0.98 gpm to 1.17 gpm. The test result was subsequently accepted by the JTG.

3.2.2 Pressurizer Heaters and Spray (Appendix 5007)

Pressurizer heaters and spray effectiveness was checked on January 21, 1986. The inspector noted the following test results:

-- Pressurizer pressure response to opening of both normal spray valves (PCV 455B and PCV 455C) was within the test limits as recommended by the vendor (Westinghouse)

-- Pressurizer pressure response to energization of all pressurizer heaters was also within the test limits. However, the total heater (Groups A through E) capacity (1703.69KW) was found slightly off the test acceptance criteria of 1710-1890 KW

TS require that Group A and Group B heaters, which are supplied by emergency power, each has a capacity of at least 175 KW. The measured heater capacities for Group A and Group B heaters was 329.9 KW and 330 KW, respectively. Based on the satisfactory results for the pressurizer pressure response and TS requirements, the JTG accepted the test results.

-- Constant flow was established through each pressurizer spray line to minimize thermal shock when spray actuation was needed. However, due to leakage of both normal spray valves, the bypass valves V15 and V59 were set at 1/16 turn which caused the proportion heater (Group C) to run at about 80% demand. This exceeded the vendor recommended value of 50%. The licensee and vendor are in the process of evaluating these test identified problems.

3.2.3 Rod Drop Testing (Appendix 5008)

The control rod drop time measurement was performed in accordance with test procedure 3-INT-5000, Appendix 5008, Component Rod Drop Testing. A computer timing system (Auto Rod Drop Test System) was set and calibrated prior to its use in measuring the rod drop time. All 61 control rods drop times were measured at cold no flow, cold full flow, and hot full flow conditions. Under cold test conditions an additional three rod drop time measurements were performed for those control rods whose rod drop times fell outside the two sigma (standard deviation) limit of the drop time for all control rods. While under the hot full flow test conditions, the fastest and slowest rods were dropped ten additional times in addition to three additional rod drop tests for those rods that fell outside the two sigma limit.

All test results were consistent and acceptable and within the TS limit of 2.2 seconds.

4.0 Low Power Physics Tests

4.1 Procedure Review

Scope

The approved test procedures listed below were reviewed for technical and administrative adequacy and to verify that test planning satisfies regulatory guidance and licensee commitments.

-- 3-INT-7000, Appendix 7004, Revision 1, "Isothermal Temperature Coefficient"

-- 3-INT-7000, Appendix 7005, Revision 0, "RCCA or Bank Worth Measurement"

-- 3-INT-7000, Appendix 7006, Revision 0, "Natural Circulation"

Discussion

The procedures were 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 independent verification of critical steps or parameters, and quality control and assurance involvement.

Findings

The review indicated that the procedures are 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 these procedures.

4.2 Test Witnessing and Data Review

4.2.1 Low Power Physics Testing

At various times during the inspection period, the inspectors witnessed most portions of Low Power Physics Tests (LPPT). The tests witnessed include:

-- initial criticality;

-- reactivity computer checkout;

-- boron endpoint measurement;

-- isothermal temperature coefficient measurement;

-- rod worth measurement; and

-- flux mapping.

Tests were observed for the following areas:

-- Low Power Physics Tests were conducted in accordance with the approved test procedure, 3-INT-7000, "Low Power Physics Tests," Revision 1.

-- Prior to performing each test, briefings with the test crew and operation personnel were conducted and the briefing was adequate.

-- Test prerequisites and initial conditions were met.

-- Summary analysis was made upon completion of each test.

The major events during LPPT were as follows:

January 23	2200	Reactor Critical
January 24	0200	Completed Hot Zero Power Testing Range Determination. The range of neutron flux (reactivity computer picoammeter) was determined to be 1.6E-8 to 1.6E-7 amps.
	0420	Completed Reactivity Computer Checkout Comparisons of predicted and measured reactivities based on doubling time measurement were acceptable.
	0715	Commenced ARO Boron Endpoint Measurement. Measured Endpoint Boron Concentration was 1570.6 ppm. The predicted value was 1566 ppm.
	1004	Commenced ITC (ARO) measurement
	1030	Obtained first set of cooldown/heatup ITC values
	1400	Operations personnel found the RCS leakage source (Seal Filter Vent Valve) and isolated it. The inspector noted that at about 1330, a reactor operator was able to identify the RCS leakage based on control room panel indications.
	1560	Obtained additional set of ITC (ARO) data. The averaged ITC's from heatup and cooldown agreed within ±1 pcm/°F. The measured ITC (ARO) value of -1.03 pcm/°F agreed well with the predicted value of -1.69 pcm/°F. However, the corresponding MTC result of (+)0.92 pcm/°F exceeded the TS 3.1.1.3 a limit. The licensee reactor engineer was fully aware of the plant conditions and corresponding TS 3.1.1.3 a LCO requirements.

	2100	Commenced Control Bank (CB) D Rod Worth Measurement (Dilution Method)
	2219	Completed CB D Rod Worth Measurement
		The measured CB D rod worth of 619.5 pcm was within the predicted range of 593 ±59 pcm.
January 25	0020	Commenced Control Bank D Inserted Measurement
	0255	Measured Endpoint Boron Concentration (D IN) was 1516.7 ppm. The predicted value was 1499 ppm.
	0435	Commenced ITC (D IN) Measurement - Cooldown
	0447	Commenced ITC (D IN) Measurement - Heatup. The measured cooldown ITC was identical to the heatup ITC. The measured ITC value of -2.5 pcm/°F was within the predicted range of -3.24 ±3 pcm/°F.
	0602	Commenced CB C Rod Worth Measurement (Dilution Method)
	0807	Completed CB C Rod Worth Measurement
		The measured CB C rod worth of 1223 pcm was within the predicted range of 1254 ± 125 pcm.
	0830	Commenced CB C&D Inserted Measurement
		Measured Endpoint Boron Concentration (C&D IN) was 1384 ppm. This value was within the predicted range of 1357 ± 136 ppm.
	0935	Commenced ITC (C&D IN) Measurement The measured ITC VALUE OF -6.07 pcm/°F was within the predicted range of -6.52 ± 3 pcm/°F. Also, the average ITC's from heatup and cooldown agreed within ± 1 pcm/°F.

1100	Commenced Dilution for Rod Worth Measurement of CB A & B
1704	The measured Endpoint Boron Concentration (A+B+C+D IN) was 1115 ppm. This value was within the predicted range of 1086 ± 109 ppm.
	The measured CB B rod worth of 1239.5 pcm was within the predicted range of 1208 ± 121 pcm.
	The measured CB A rod worth of 1216.3 pcm was also within the predicted range of 1239 ± 124 pcm.
1746	Manually tripped the Reactor per TS 4.10.1.2 requirement (In preparation for All-Rods-In Testing)
2000	Started up by pulling CB B
2016	Manually tripped the reactor per Test Procedure 7000, Step 7.12.3.4, also in preparation for All-Rods-In Testing.
2025	Commenced Reactor Startup
2102	Reactor Critical with CB A at 40 steps
2144	Commenced Shutdown Bank E (SE) rod worth measurement
2206	Completed SE rod worth measurement
	The mesaured SE rod worth of 185.7 pcm was within the predicted range of 188 ± 19 pcm.
2217	Commenced Shutdown Bank D (SD) rod worth measurement
2313	Completed SD rod worth measurement The measured SD rod worth of 547.8 pcm was within the predicted range of 526 ± 53 pcm.
2316	Commenced Shutdown Bank C (SC) rod worth measurement

January 26	0052	Completed SC rod worth measurement The measured SC rod worth of 679.6 pcm was within the predicted range of 655 ± 66 pcm
	0130	Measured Single Rod F-2 Rod Worth for information purpose.
	0330	Measured N-1 total rod worth with rod F-2 Stuck Out The measured value of 7925.7 pcm was within the predicted range of 7571 ± 757 pcm.
	0845	Commenced All-Rods-In With F-2 Stuck Out Boron Endpoint Measurement
	0911	Completed All-Rods-In with F-2 Stuck Out Boron Endpoint Measurement The measured value of 765.2 ppm was within the predicted range of 725 ± 73 ppm.
	0913	Manually tripped the reactor per test procedure 7000 step 7.16.
	2257	Reactor Critical
	2238	Commenced control bank rod worth measurement in overlap
January 27	0656	Completed control bank rod worth measurement in overlap (Boration Method) The inspector noted that the measured total overlapping control bank rod worth of 4365.6 pcm was consistent with the previously measured sum (4298.3 pcm) of
January 28		LPPT temporarily being hold due to difficulty in getting secondary side chemistry in specification.

January 29	0313	Resumed LPPT. Started flux mapping with rod configuration at Rod Insertion Limit. The measured incore tilt of 1.006 was less than the design criterion of ≤ 1.02
	0600	Measured a single rod D-12 rod worth from Hot Zero Power Rod Insertion Limit.
		The measured value of 396.6 pcm was within the safety analysis assumed value of 780 pcm.
	1046	Started flux mapping with rod configuration at Rod Insertion Limit with D-12 fully out to simulate ejected rod.
		The incore program picked up this asymmetric rod configuration with calculated maximum $F_Q(Z) = 6.4757$ which was less than FSAR assumed value of 11.5
January 30	0630	Started flux mapping with rod configuration of Control Bank D In
		The measured incore tilt (1.023) was within FSAR/Safety Review Criteria of 1.04, however, was slightly in excess of the design limit of 1.02. The licensee reactor engineer evaluated this situation and attributed the cause to the Xenon left from the previous D-12 ejected rod worth/flux mapping test.
		A six pass symmetric thimble map taken at approximately 48 hours later showed almost 0 tilt. This confirmed the licensee reactor engineer's evaluation.
	103[illegible]	Started flux mapping with All-Rods-Out rod configuration.
		The measured incore tilt was the same as the Control Bank D In case.
		The measured $F_{\Delta H}$ in both Control Bank D In and ARO cases were acceptable.
	1135	Completed flux mapping and LPPT.

4.2.2 Low Power Physics Test Results Review

The inspector independently verified that the predicted values and acceptance criteria were obtained from "The Nuclear Design and Core Physics Characteristics of the Millstone Generating Station Unit 3 Cycle 1," WCAP-10791, Revision 1. The LPPT results are summarized below:

	Endpoint Boron (ppm)		Rod Worth (pcm)		ITC (pcm/°F)	
Test Cond.	Meas.	Predicted	Meas.	Predicted	Meas.	Predicted
ARO	1570.6	1566	-	-	-1.03	-1.69
D IN	1516.7	1499	619.5	593	-2.50	-3.24
D+C IN	1384	1357	1223	1254	-6.07	-6.52
D+C+B IN	-	-	1239.5	1208	-	-
D+C+B+A IN	1115	1086	1216.3	1239	-	-
D+C+B+A+ S_E IN	-	-	185.7	188	-	-
D+C+B+A+ S_E+S_D IN	-	-	547.8	526	-	-
D+C+B+A+ S_E +S_D+S_C IN	-	-	679.o	655	-	-
N-1 Rod Worth with F-2 Stuck Out	765.2	725	7925.7	7571	-	-
Control Banks Over-lapping (Boration Method)	-	-	4365.6	4294	-	-
Rod D-12 Worth from RIL	-	-	396.6	491	-	-

All measured values were close to and within the analytically predicted ranges with the exception of D-12 ejected rod worth measurements from RIL. In this case the measured rod worth of 396.6 pcm was within the safety analysis assumed value of 780 pcm. Although the licensee did not measure the Shutdown Banks A and B rod

worth in fuel vendor recommended rod configuration (Table A.4, WCAP-10791), instead an N-1 configuration with Rod F-2 stuck out was performed. This configuration provided a direct comparison of measured rod worth (7925.7 pcm) against the value assumed in the FSAR analysis (6814 pcm). The measured rod worth confirmed the FSAR analysis value.

Following the discovery of the positive MTC value at ARO.HZP conditions, the licensee took correct actions including establishing administrative restrictions on Rod Withdrawal Limits and Boron Concentration Limits. The special report on this subject was submitted to NRC Region I (Letter from W. D. Romberg (NNECO) to T. E. Murley (NRC), dated February 3, 1986) in accordance with TS 3.1.1.3 requirement. The inspector verified that the operation personnel were aware of these restrictions and that the associated curve was located in the control room curve log and was being used.

The inspector reviewed the low power flux maps and noted that the predicted power distribution was generally agreed with the predicted values. No unacceptable power distributions were identified.

4.2.3 Conclusion

Low Power Physics Test was accomplished in accordance with approved procedures, data were acceptable, and test objectives were met.

Licensee performance during approach to criticality and subsequent LPPT was deliberate, and carefully controlled. TS surveillance requirements associated with the special test exceptions during LPPT were correctly addressed in the controlling procedure 3-INT-7000, "Lower Power Physics Tests," and adequately performed. Licensee management was responsive to inspector observations. Problems identified during the test such as positive MTC value at all-rods-out condition were disseminated to the appropriate groups and corrective actions were implemented.

5.0 Power Ascension Tests

5.1 Procedure Review

The approved test procedures listed in Attachment B were reviewed for technical and administrative adequacy and to verify that test planning satisfies regulatory guidance and licensee commitments.

Discussion

The procedures were examined for: management review and approval; procedure format, clarity of stated test objectives; prerequisities; 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 procedures are consistent with regulatory requirements, guidance, and with the licensee's commitments. No discrepancies or unacceptable conditions were identified. Questions raised by the inspector to the licensee relating to the procedures were satisfactorily answered.

It was noted that the licensee is planning on seeking from the NRC full credit for two R.G.1.68 required tests: the ejected rod test at above 10% full power and the dropped rod test at 50% full power. This request will be based on the fact of these tests have been adequately demonstrated at prototype plants similar to MP-3. In a letter to NEU from Westinghouse dated January 20, 1985, Westinghouse concurred with NEU and stated that "The cost in time and the probability of subsequent undesirable power distributions resulting from these tests lead to the conclusion that they should be dropped from the Millstone Nuclear Power Station Unit 3 test program." Westinghouse supported this statement with a table showing the number of operating plants similar in design to MP-3 which had successfully completed the test. The licensee committed in the FSAR Table 14.2-2 item number 30, to perform these two tests and will therefore be seeking a change to the FSAR from NRC allowing deletion of them.

This is an unresolved item (50-423/86-07-02).

5.2 Test Witnessing

Portions of the following Power Ascension Test and the entirety of the Natural Circulation Test were witnessed by the inspectors. This included the initial test preparation and test restoration as well as witnessing the actual test.

-- 3-INT-7000, Appendix 7006, "Natural Circulation" Revision 0

-- 3-INT-8000, Appendix 8018, "Automatic Steam Generator Level Control", Revision 0, Step 7.1 - Testing at Low Power.

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

These tests were witnessed for the following attributes:

1. Appropriate procedure revision was available and in use by all crew members.

2. Minimum crew requirements were met.

3. All test prerequisites and initial conditions were met and/or those which were waived were reviewed/approved in accordance with procedure/technical specification (TS) requirements.

4. Test equipment required by the procedure was calibrated and in service.

5. Test data equipment required by the procedure was calibrated to a common time base.

6. Test was performed as required by a technically adequate procedure.

7. Crew actions appeared to be correct and timely during the performance of the test. Coordination was adequate.

8. Quick summary analysis was made to assure proper plant response to the test.

9. All data was collected for final analysis by the proper personnel.

10. Overall test acceptance criteria were met.

11. The licensee's preliminary test evaluation was consistent with inspector's observation.

12. Adherence to TS requirements was maintained for those tests which affect TS LCOs.

The following discussion pertains to these tests.

5.2.1 <u>Natural Circulation Test (Appendix 7006)</u>

It took approximately ten minutes to establish natural circulation after the reactor coolant pumps had been tripped. Natural circulation took place when ΔT reached approximately

37°F (T_{HOT} ~ 594°F and T_{COLD} ~ 557°F). There appeared to be close correlation between the incore thermocouple readings and T_{HOT}. The majority of incore thermocouple readings ranged between 596°F and 602°F.

After the RCS had stabilized on natural circulation, the letdown/charging system was initiated to see what affect it had on cooling down the primary coolant. It was discovered that the RCS began cooling down at a rate of approximately 1°F every four minutes. Questions raised by the inspector concerning the results of the test were satisfactorily answered by the licensee.

5.2.2 Steam Generator Water Level Control Test (Appendix 8018)

This test was first performed on February 10, 1986. During the test, I&C personnel inadvertently shorted the steam generator level control system circuits which caused all feedwater regulating bypass valves to close. This resulted in a S/G low level reactor trip. The test was resumed on February 11, 1986. The inspector noted that a lesson learned from an earlier reactor trip which occurred on February 7, 1986, was incorporated. Auctioneered high nuclear flux was fed to the bypass control circuits to position the bypass valves during low power automatic S/G level control. Following the February 7, 1986 reactor trip, the bypass valve response was determined to be too fast. Currently a gain adjustment factor of 2.44% valve opening/1% NI power was set in lieu of 8% valve opening/1% NI power.

Preliminary test results indicated that the automatic S/G level control was stable. No unacceptable conditions were identified.

5.2.3 Steam Dump Test (Appendix 8013)

The steam dump test was performed in accordance with procedure Appendix 8013, "Steam Dump Control," Revision 0. This test was originally planned to be performed at 0% power. Test Change No. 3 modified the test procedure to be performed at 15% power level. There appeared, however, to be a lack of proper review for Test Change No. 3. As a result, during initial performance of Step 7.4, Plant Trip Controller Response Test, the procedure was found to difficult to follow. The test engineer quickly identified the problem and corrected the situation through Test Change No. 6. which was further reviewed by the PORC/JTG. Upon PORC/JTG approval, the test was resumed and completed without incident.

Preliminary test results indicated that the test objectives were met and test data were acceptable.

6.0 Independent Calculation

During the review of Post Core Hot Functional Test, Appendix 5009 - Pre-critical RCS Flow Measurement, the inspector verified that the calculations and math used to convert percentage of flow to inches of water (ΔP) and to calculated flow rates were correct.

Also, as described in Section 3.2.1 detailed RCS leak rate calculation was performed using NRC-developed leak rate computer program NUREG-1107, "RCSLK9: Reactor Coolant System Leak Rate Determination for PWRs", to verify the licensee's calculation.

7.0 Quality Assurance /Quality Control

The site QA/QC organization, manpower and planned activties to cover the startup test program was described in the NRC inspection report 85-57. During this inspection period, the inspectors noted that QC inspectors were actively following startup program tests. Appropriate surveillance reports were issued.

No unacceptable conditions were identified.

8.0 Plant Tours

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

9.0 Exit Meeting

An exit meeting was held on February 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

POST CORE HOT FUNCTIONAL TEST RESULTS REVIEWED

-- Appendix 5004, Rod Control Slave Cycler and CRDM Timing

-- Appendix 5006, RCS Leak Detection

-- Appendix 5007, Pressurizer Heaters and Spray

-- Appendix 5008, Rod Drop Testing

-- Appendix 5009, Precritical RCS Flow Measurement

-- Appendix 5011, Moveable In-Core Detector Operation

-- Appendix 5018, Rod Control Operational Test

APPENDIX B

POWER ASCENSION TEST PROCEDURES REVIEWED

-- 3-INT-8000, Power Ascension Test, Revision 0

-- 3-INT-8000, Appendix 8001, Calorimetric, Revision 0

-- 3-INT-8000, Appendix 8002, Nuclear Instrumentation Operational Alignment Verification, Revision 0

-- 3-INT-8000, Appendix 8003, Calibration of Steam Flow and Feedwater Flow Instrumentation at Power, Revision 0

-- 3-INT-8000, Appendix 8004, Operational Alignment of Process Temperature Instrumentation, Revision 0

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

-- 3-INT-8000, Appendix 8007, Radiation Monitoring System, Revision 0

-- 3-INT-8000, Appendix 8008, Ventilation System Operability, Revision 0

-- 3-INT-8000, Appendix 8009, Plant Chemistry, Revision 0

-- 3-INT-8000, Appendix 8010, Neutron Shield Tank Cooling Test, Revision 0

-- 3-INT-8000, Appendix 8011, Containment Penetration Temperature Monitoring, Revision 0

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

-- 3-INT-8000, Appendix 8015, RCS Flow Measurement, Revision 0

-- 3-INT-8000, Appendix 8016, Turbine Overspeed, Revision 0

-- 3-INT-8000, Appendix 8017, Automatic Reactor Control, Revision 0

-- 3-INT-8000, Appendix 8018, Automatic Steam Generator Level Control, Revision 0

-- 3-INT-8000, Appendix 8019, Turbine Plant Component Cooling Water System Balancing, Revision 0

-- 3-INT-8000, Appendix 8020, Power Coefficient, Revision 0

-- 3-INT-8000, Appendix 8022, 10% Load Swing Tests, Revision 0

-- 3-INT-8000, Appendix 8023, Reactor Trip/Shutdown Outside Control Room, Revision 0

-- 3-INT-8000, Appendix 8026, Large Load Reduction, Revision 0

-- 3-INT-8000, Appendix 8029, Pipe Fluid Transient Vibration Testing, Revision 0

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

-- 3-INT-8000, Appendix 8031, Reactor Coolant System Boron Measurement, Revision 0

-- 3-INT-8000, Appendix 8034, Thermal Expansion Restraint, Revision 0

-- 3-INT-8000, Appendix 8035, Loose Parts Monitoring System, Revision 0

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