U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-443/86-28

Docket No. 50-443

License No. CPPR-135

Licensee: Public Service of New Hampshire
P.O. Box 330
Manchester , New Hampshire 03105

Facility Name: Seabrook Station, Unit 1

Inspection At: Seabrook, New Hampshire

Inspection Conducted: May 12-22, 1986

Inspectors:	David Wallace David Wallace, Reactor Engineer	6/24/86 date
	Norman Blumberg Norman Blumberg, Lead Reactor Engineer	6/24/86 date
	S Chaudhary Suresh Chaudhary, Lead Reactor Engineer	6/24/86 date
Approved by:	Jon R Johnson Jon Johnson, Chief, Operational Program Section, Operations Branch, DRS	6/20/86 date

Inspection Summary: Routine, unannounced inspection conducted on May 12-22, 1986(Report No. 50-443/86-28).

Areas Inspected: Routine unannounced inspection of licensee action on previous inspection findings, construction deficiency reports, and an IE Circular; operating programs and procedures; licensee control of NRC action items, commitments, and technical specification changes; and, reactor engineering procedures. The inspection was conducted onsite by three region-based inspectors.

Results: No violations were identified. A concern regarding inaccuracies in reactor engineering procedures is described in paragraph 5.2.

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DETAILS

1.0 Person Contacted

Public Service of New Hampshire

*Abely, D.	Maintenance Department Supervisor, New Hampshire Yankee (NHY)
*Conti, D.	Firefighter Supervisor (NHY)
DeLoach, R.J.	Yankee Nuclear Services Division Project Manager, Yankee Atomic Electric Company (YAEC)
Dickson, W.	Technical Services, Lead Mechanical Engineer, NHY
Gelineaux, G.	Computer Development Supervisor, NHY
Gritto, J.	Assistant Operational Manager, NHY
Guillette, R.	Assistant Project Construction Quality Assurance Manager, NHY
*Gurney, P.	Reactor Engineering Department Supervisor, NHY
*Kingston, G.	Compliance Manager, NHY
*Kline, G.	Technical Services Manager, NHY
*Malone, J.	Operations Administration Supervisor, NHY
Martel, R.	Surveillance Test Coordinator, NHY
*Middleton, W.	Quality Assurance Staff Engineer, YAEC
*Murphy, T.	Instrumentation and Controls (I&C) Department, NHY
*Sanchez, V.	Site Licensing Supervisor, YAEC
*Singleton, J.	Assistant Quality Assurance Manager, YAEC
Small, E.	Document Control Supervisor, NHY
Stetzer, T.	Startup Engineer, NHY
*Trump, E.	Senior Fire Protection Engineer, NHY
*Walsh, G.	Operations Manager, NHY

U.S. Nuclear Regulatory Commission

*Barkley, R.	Reactor Engineer, (Seabrook Station)
Cerne, A.	Senior Resident Inspector
Elsasser T.	Chief, Reactor Projects, Section 3C
Ruscitto, D.	Resident Inspector

*Denotes those present at exit meeting conducted on May 22, 1986.

2.0 Licensee Action on Previous Inspection Findings

(Closed) Unresolved Item, 85-15-01: Cables in solid bottom trays had loose and/or missing "Kellem Grip" cable supports in riser trays. The licensee responded by initiating a reinspection of all such trays. The results of the inspection were documented in Nonconformance Reports (NCRs). The licensee had identified loose or missing "Kellem Grip" cable supports, and has completed the corrective action. This item is closed.

(Closed) Violation, 84-07-08: A number of bolts fastening structural steel members in the containment building were found to be insufficiently torqued. Because the structural connections in steel were designed as friction connections a lack of sufficient clamping torque could result in joint slippage. The licensee initiated a bolt torque verification effort to determine the extent of this problem by verifying existing torque in all double knife joints, and a random sample of all other types of joints. The subsequent report disclosed that a total of twenty-one (21) joints failed to meet the minimum torque requirement of 425 ft-lbs. However, further engineering calculations using actual mechanical loads showed that the joints were acceptable for intended design loads. The cause of the problem was determined to be mostly in double knife joints; because, the initial clamping by torquring two bolts relaxed when all other bolts were tightened. The possibility of this problem reoccurring is precluded because the double knife joint design in structural connections have not been used except for 123 joints evaluated in this effort. This item is closed.

(Closed) Construction Deficiency Report, 85-00-06: The deficiency involved potential damage to Brown Boveri 480V circuit breaker auxiliary switch wiring when the circuit breaker was racked out to full disconnect position with compartment door closed. In this position, the control wire insulation on the auxiliary switch might have been cut by the top edge of the dust shield. Damage to class 1E wiring in the auxiliary switch could result in failure of the circuit breaker to perform its intended safety function. The corrective action to prevent any wiring damage was specified by the vendor. It involved encasing the damaged wires in heat shrink tubing, installation of sponge rubber strips to the top back of all dust shields, and dressing the wires close to auxiliary switch to minimize the wire contact with the dust shield. This corrective action is acceptable. The licensee has completed the above repairs and modifications. This item is closed.

(Closed) Construction Deficiency Report, 85-00-14: The licensee reported the deficiency regarding Valcor containment isolation valves which unsealed during leak testing. These valves provide containment isolation for non-ASME piping systems, and are manufactured by Valcor, Inc. of Springfield, New Jersey. During an accident, these valves could create a path for fission products to migrate from the inside of the containment to the outside. The valves were provided to isolate the differential pressure associated with normal flow into containment; but were not suitable for isolation of high reverse differential pressure resulting from an accident. The vendor recommended that for proper functioning of the valves for containment isolation, the internals would have to be replaced by balanced trim internals. The licensee has implemented the vendors recommendations, and the valves have been acceptably reworked to provide containment isolation. This item is closed.

(Closed) IE Circular 80-15: The circular contains information on the loss of reactor coolant pump (RCP) cooling and natural circulation cooldown. The circular recommended that: 1) the information should be given to licensed operators; 2) review and revise natural circulation cooldown procedures to include appropriate recovery procedure; 3) establish a natural circulation cooldown and depressurization rate envelope to preclude void formation, and provide adequate cooling; 4) evaluate design of component cooling water system to determine vulnerability to single failures that could cause loss of RCP cooling; and, 5) consider installation of a reactor vessel head temperature monitoring system.

In response to the first three recommendations, the Westinghouse Owners Group developed generic emergency response procedures. The licensee has implemented these procedures with some changes to make them site specific. Procedures E-0.2, E-0.3, and E-0.4 cover the emergency response developed by Westinghouse Owners Group. The NRC evaluation of the adequacy of emergency operating procedures is the subject of NRC Inspection Report 50-443/86-33. For recommendation (5) the licensee has installed the Reactor Vessel Level Instrument system (RVLIS) which, in part, monitors incore themocouples during an accident scenario. See paragraph 3.3.4.2 of this report for further details on RVLIS.

Regarding the design of the component cooling water system, (Recommendation (4)) it was determined that:

-- The primary component cooling water system is comprised of two independent cooling loops supplying cooling water to RCPs (motor bearing and air coolers) and miscellaneous components. Each cooling loop continuously supplies coolant to two RCPs. Since the two loops are independent (no cross connection), a single failure of either loop would not cause loss to all RCPs.

-- The RCP shaft seals are lubricated with seal injection water from Chemical and Volume Control System and water from Reactor Water Makeup System with no connection to the Primary Component Cooling Water System (PCCW)

-- The RCP shaft seal system consists of three water lubricated face seals with an external system to control and monitor the upward flow of reactor coolant in the event of a total seal failure or loss of seal water. There is also a Thermal Barrier Heat Exchanger to limit heat transfer from hot reactor coolant to the area of the pump radial bearing and heat exchanger. A single failure of either loop of the PCCW will only eliminate the water supply from the lost loop. The remaining functional loop will still supply coolant to the Thermal Barrier Heat Exchangers for all of the RCPs.

The licensee's action and analyses in response to this circular are adequate and acceptable. This item is closed.

(Open) Deviation, 85-25-03: Certain tubing associated with the Reactor Vessel Level Instrument System (RVLIS) level transmitter was not installed to class IE quality assurance requirements as specified in Regulatory Guide 1.97. Review during this inspection indicated that additional information is needed concerning the licensee's resolution to this item in order to ascertain the adequacy and effectiveness of the corrective action. This item will be reviewed again in a subsequent inspection, and remains open pending further review by the NRC.

3.0 Operations Programs and Operating Procedures

3.1 References/Requirements

-- 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants

-- Regulatory Guide 1.33-1978, Quality Assurance Program Requirements (Operation)

-- ANSI N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants

-- Seabrook Station, Unit 1, Technical Specifications, DRAFT March, 1986

3.2 Program Review

This inspection included a review of various operations programs and procedures. Previous NRC inspections of operations and inspection findings are documented in NRC Inspection Reports 50-443/86-02, 86-05, 86-06, 86-09, and 86-27. Inspection reports 86-09 and 86-27 specifically documented reviews of general operating procedures, system operating procedures, surveillance test procedures, and inservice test procedures for pumps and valves. The following areas were reviewed during this inspection:

-- Video Alarm System (VAS) including Alarm response procedures, computer alarm validation, safety reviews and approvals, and document control

-- Daily and shift logs implementing technical specifications surveillance

-- Inservice test program for pumps and valves

-- Technical specification surveillance test program

-- Operations Department Management Manual

-- Station Operations Review Committee (SORC) Meetings

-- Independment verification

-- Plant computer point validation

-- Reactor Vessel Level Instrumentation system (RVLIS) computer input validation and procedure control

The above programs and procedures were reviewed to assure compliance with the requirements listed in paragraph 3.1 above. During this inspection the following licensee program procedures and documents were reviewed:

-- Operations Management Manual, (OPMM) Revision 2, DRAFT May 20, 1986

  ° Chapter 3, Shift Operations

    1.12 Instrument and Control Adjustments
    1.13 Returning Equipment to Service
    2.0 Operating Procedures
    2.5.3 Protected Procedure Steps

  ° Chapter 8, Control of Operator Aids, Revision 0, DRAFT

  ° OP 9.5, Alarm Response Guidelines, Revision 0, DRAFT

  ° OP 10.2, Independent Verifications, Revisions 0, DRAFT

--Station Test Control Manual, (STCM) Revision 8, April 9, 1986

  ° T.C.2.2 Technical Specification Surveillances Scheduling and Performance, Revision 2, March 28, 1986

  ° T.C.3.1 Inservice Testing of Pumps, Revision 1, March 12, 1986

  ° T.C.3.2 Inservice Testing of Valves, Revision 1, February 28, 1986

--Computer Control Program Manual, Revision 0, December 24, 1985

--Yankee Atomic-Framingham Memorandum, "Seabrook Alarm Review - Video Alarm System (VAS), Interim Report", dated February 28, 1986

--Yankee Atomic-Framingham Memorandum, "VAS Changes Required as a Result of NRC Review of CRDR Submittal", dated March 27, 1986

--PSNH Memorandum, "VAS Procedure Changes", Dated January 14, 1986

3.3 Findings

3.3.1 Alarm Response Procedure

3.3.1.1 The following Alarm Response procedures were inspected by walking through each procedure while accompanied by a licensed operator:

- D405, "Safety Injection Accumulator A Pressure Low", (Priority 1 alarm)
- D4596, "Safety Injection Actuation", (Priority 1 Alarm)
- D4702, "EFW Pump A Suction Header Pressure Low", (Priority 1 Alarm)
- D4832, "RCP D No. 1 Diff Press Low", (Priority 2 Alarm)
- D5778, "RCP D Shaft Vibration High", (Priority 2 Alarm)
- D6078, "Battery Charger ID DC Volt Low, (Priority 2 Alarm)
- D6558, "DG A Lube Oil Pressure Low", (Priority 1 Alarm)
- D6716, "DG A Generator Voltage Low", (Priority 1 Alarm)
- D7486, "Main Steam Isolation Train B" (Priority 1 Alarm)

A random sampling of the procedures listed was inspected to verify that the set point, initiating devices, and associated automatic actions were correct as stated in the procedure. Each of the procedures listed was verified to comply with the requirements stated in OP 9.5 "Alarm Response Guidelines". Although the SORC approval has not been completed for OP 9.5, the procedure will establish the minimum required information to be contained in each alarm response procedure. The "Recommended Actions" section for each of the procedures was reviewed for technical accuracy by walking through each procedural step while accompanied by a licensed operator. No deficiencies were observed by the inspector.

3.3.1.2 Since alarms and alarm response procedures will be displayed on video screens in the control room, the inspector verified that systems were in place to validate the accuracy of alarm response procedures on the video display. Currently a memorandum established the mechanism by which VAS procedures are changed and placed on the computer. This memorandum also describes the interface between the Operations Department and the Computer Engineering Department. The cross checks and reviews appear to be sufficiently adequate to ensure the accuracy of video alarm procedure displays.

In addition to the video displays, computer printout hard copies of all alarm response procedures are maintained in the control room as a backup to the video display of the procedures. The inspectors verified that a mechanism was in place to assure that the hard copies of the procedures were maintained up to date. A sampling of procedures were reviewed to determine that both the video display and hard copy procedures were up to date.

A previous finding in inspection report 50-443/86-09 (86-09-01) noted that certain alarm response procedures require SORC review and Station Manager approval. Although not fully implemented, the licensee has developed a mechanism for doing this. The licensee has decided that alarms identified as priority 1, i.e. an alarm which indicates entry into emergency procedures, deviations from expected automatic actions, or immediate action required by Technical Specifications (less that 1 hour) or deviations from automatic actions required by reactor trip and ECCS deviation signal, and all alarms for emergency core cooling systems (ECCS), regardless of priority, will be reviewed by the SORC. As of the dates of this inspection, this mechanism is not fully in place and these alarm response procedures have not yet been fully reviewed and approved. This item will receive further review during a future inspection.

3.3.2 Technical Specifications Surveillance Logs

Control room logs which document daily and shift surveillances required by the Technical Specifications were reviewed. Log sheets have been prepared for Modes 1-6. A previous finding in Inspection Report 50-443/86-09 (86-09-02) identified that these logs require SORC review and station manager approval. Although a mechanism is now in place to do this, the logs have either not been approved or they are still in the process of modification.

The logs currently in use were reviewed against the current draft Technical Specifications. Based on this review the inspector determined that the logs adequately reflect T.S. requirements and parameters. However, the logs still require further modifications and the licensee review and change process for these logs is on-going.

3.3.3 Station Operations Review Committee (SORC)

The inspector attended a SORC meeting on May 22, 1986 and obser.ed that the quorum required by Section 6 of the Technical Specification, was present. Procedures discussed at this meeting pertained to Train B primary component cooling water (PCCW) operability tests, cathodic protection, chemistry and maintenance; diesel generators, and the seismic monitoring system. Many procedure changes and several new procedures were reviewed and discussed by the SORC and then recommended for approval by the station manager. The SORC meeting was conducted in a thorough and acceptable manner.

3.3.4 Plant Computer Validation

Since the plant computer provides numerous displays of plant parameters and plant alarms, the inspector reviewed the licensee's validation of plant computer information which may directly or indirectly impact plant safety. Findings concerning the video alarm system were previously discussed in paragraph 3.3.1. Additional computer related areas reviewed included the use of computer readouts for T.S. surveillance tests or inservice tests, and the reactor vessel level instrumentation system (RVLIS). Concurrently with this inspection, representatives of the NRC's office of Nuclear Reactor Regulation were on site to examine the validation and adequacy of the licensee's Safety Parameter Display System (SPDS). For this reason validation of SPDS was not a subject of this inspection.

3.3.4.1 Use of Computer Data Points In Surveillance Tests

During this inspection the following random sampling of operations surveillance tests were reviewed:

-- OX 1405.07, Quarterly Safety Injection Pump Flow Test and Quarterly Valve Test, Revision 00, DRAFT

-- OX 1406.02, Containment Spray Pump and Valve Test, Revision 00, May 21, 1986.

-- OX 1413.01, RHR Pump Flow and Valve Strobe Test, Revision 00, DRAFT

-- OX 1412.01, PCCW Train A Operability Test, Revision 00, DRAFT

In general, data for these tests was taken from hard wired instruments in the control room or from local instruments. However, the following computer points were observed in procedures for the verification of certain parameters:

Procedure	Computer Point
OX 1405.07 A	A0511, Safety Injection Pump "A' Discharge Pressure
OX 1406.02 A	A0922 Containment Spray Pump "A' Suction Pressure
	A0923, Containment Spray Pump "A" Discharge Pressure
OX 1412.01 B	A0266, PCCW Pump 11C Discharge Pressure
OX 1413.01 B	A0961 RHR Pump B Radial Bearing Temperature
	A0960 RHR Pump B Thrust Bearing Temperature

A licensee representative stated that all computer points were included in the calibration process. The inspector reviewed the following loop calibration procedures:

-- IS 1622.315, P-2313 Containment Building Spray Pump 9A Section Pressure Loop Calibration, Revision 01, May 17, 1986

-- IS 1668.361, P-919 Safety Injection Pump 6A Discharge Pressure Calibration, Revision 01, September 27, 1984

-- IS 1622.310, P-2313 Containment Spray Pump Discharge Pressure Calibration, Revision 01, May 20, 1986

It was observed that computer points AO 511, AO 922 and AO 923 were included directly in the loop calibration. In addition, the calibration source was directly from the sensor element. Computer point AO 266 had been calibrated as a preventive maintenance item. However, the licensee stated that this point would soon be incorporated into a formal calibration procedure; and further stated that computer points AO 961 and AO 960 were not calibrated because they were direct readings from the temperature element to the computer. The inspector informed the licensee that all computer points used to verify T.S. surveillance results should be included calibration procedures. The licensee stated that a review would be taken of all T.S. surveillance tests to assure that all computer points used to verify test adequacy have been incorporated into the licensee T.S. related instrument calibration program.

In addition to the calibration program noted above, the inspector observed that an extensive pre-operational test had been done to verify, calibrate, and validate all plant computer points. The inspector reviewed data from the following completed tests:

-- GT-I-101, Main Plant Computer System, May 12, 1986

-- GT-I-42, Station Computer

GT-I-101, on a sampling basis, tested computer points in the VAS, the analog input system, digital inputs, and exercised data acquisition, displays, and logging.

GT-I-42 was performed by the startup test group and tested every computer point. Based on this review the inspector determined that computer points were appropriately tested.

3.3.4.2 Reactor Vessel Level Instrumentation System (RVLIS)

RVLIS provides significant post accident information such as reactor vessel level and incore temperatures. The RVLIS computer plasma display is for backup purposes only. All RVLIS parameters are hardwired to the Control Room and are used for safety decision. Information provided by the RVLIS is used for evaluative purposes in the emergency operating procedures. The inspector observed RVLIS parameters used for accident evaluation in the following procedures:

-- E-0, Reactor Trip or Safety Inspection Subcooling

-- E-0, Loss of Reactor Coolant or Secondary Coolant

-- E-3, Steam Generator Tube Rupture

Further inspection of licensee's emergency operating procedures will be performed during Inspection 50-443/86-33.

3.3.5 Independent Verification

During inspection 86-09, a draft version of OP 10.2, "Independent Verification" was reviewed. An updated draft version of OP 10.2 was provided to the inspector during this inspection. Attached to the procedure is an extensive list of components which will require independent verification. However, the body of the procedure does not refer to this list nor define what the list is for. In addition, component nomenclature has not been included on the list. A licensee representative stated that component nomenclature was not on the list because the proper names for all components have not yet been resolved by the licensee.

The inspector informed the licensee that OP 10.2 should be revised to define the use of the attached component list and to include appropriate nomenclature for each component. This item is to be completed by plant startup and is unresolved pending completion of licensee action (50-443/86-28-04).

4.0 Licensee Control of NRC Action Items

4.1 NRC Findings and Commitments

The inspector discussed with the licensee's compliance engineer the mechanism by which NRC items such as inspection findings, commitments to the NRC, IE bulletins and information notices, FSAR commitments, etc., are incorporated into procedures when necessary.

Based on these discussions and a review of administrative procedures, the inspector noted that there was no formal station-wide mechanism for incorporating NRC commitments into procedures.

Only the Operations Department had a procedure for formalizing NRC commitments. OPMM Chapter 3, has a "protected procedure steps" paragraph. This procedure ensures that operating procedures or portions of procedures which are the results of commitments to the NRC, are clearly identified in the procedure. If a procedure should be changed or issued as result of a commitment to the NRC, a mechanism is in place which identifies the specific NRC commitment. This ensures that future changes will not inadvertently delete information because its regulatory origin is unknown.

The inspector reviewed implementation of the above in procedure OS 1026.01, "Operation of D/G 1A". Reference sections 2.7, 2.8, and 2.9 of this procedure refer to a section of the FSAR and to NRC information notices and their applicability to specific paragraphs within OS 1026.01. This should preclude inadvertent changing of these paragraphs without first considering their continued applicability to the referenced NRC items.

No other station organization had a similar mechanism; however, the inspector was informed some were considering developing similar systems. The inspector had no further questions at this time.

4.2 Incorporation of Technical Specification (T.S.) Changes into Plant Procedures

In addition to the licensee's NRC commitment system, the inspector reviewed the licensee's mechanism for ensuring that T.S. changes are incorporated into plant procedures where applicable. STCM Procedure T.C.2.2 requires that T.S. changes be routed to the test coordinator, who in turn coordinates with department or group supervisors to determine the effect on surveillance procedures and test scheduling. Currently, T.C.2.2 contains a T.S. and plant procedure cross reference. However, at the time of this inspection, a computer data base Specification Appraisal Program had been put in place and was being tested. This program is designed to perform the following tasks:

-- Schedule T.S. surveillances

-- Recognize needed T.S. surveillance completions for entry into specific reactor modes of operation

-- Provide T.S. and plant procedure cross reference, and

-- Recognize and track limiting conditions for operations (LCO's)

This program should ensure that T.S. changes will be incorporated into appropriate station surveillance procedures.

Operations Department representatives stated that they were developing their own personal computer (P.C.) program. All Operations Department procedures and their reference documents will be on the program. Changes to the T.S., FSAR, and drawings will be cross referenced to specific procedures and operating procedures can be changed when required. This program is still being developed and is not in use. The inspector observed that an informal manual system exists but appears to be cumbersome to implement.

Other departments, which have lesser numbers of procedures do not have nor plan to have a similar program. However, the Specification Appraisal Program discussed above will cover T.S. surveillance tests for all departments. The inspector had no further questions at this time.

5.0 Reactor Engineering Procedures

5.1 The status of Reactor Engineering Department procedures was reviewed in context of plant startup and low power operations (less than 5% power). Procedures were reviewed for conformance to the regulatory requirements referenced in paragraph 3.1. In addition, the status of readiness for programs for the plant computer was also reviewed. Reactor engineering procedures were reviewed for the following attributes:

-- Proper review and approval

-- Correct format

-- Conformance to Technical Specifications

-- Technical accuracy

-- Adequacy of prerequisites and precautions

-- Stepwise instructions and provision of necessary detail to ensure adequate procedure implementation.

The following Reactor Engineering procedures were reviewed:

-- RX-1700, Rod Drop Time Surveillances, Revision 00, March 13, 1985

-- RX-1701, Axial Flux Difference Surveillance, Revision 00, March 13, 1985

- -- RX-1702, Heat Flux Hot Channel Factor Surveillance, Revision 00, March 13, 1985
- -- RX-1703, Quadrant power Tilt Ratio Surveillance, Revision 01, December 5, 1985
- -- RX-1704, Moderator Temperature Coefficient Surveillance, Revision 00, March 13, 1985
- -- RX-1705, RCS flow Rate and Nuclear Enthalpy Rise Hot Channel Factor Surveillance, Revision 00, March 13, 1985
- -- RX-1707, Shutdown Margin Surveillance, Revision 00, March 13, 1985
- -- RX-1709, Reactivities Anomalies Surveillance, Revision 00, March 13, 1985
- -- RD-0715, Control of Reactor Engineering Technical Data, Revision 01, January 28, 1986
- -- RN-0717, RE Computer Program Documentation, Revision 01, May 10, 1985
- -- RN-1735, Reactivity Calculations, Revision 00, November 27, 1985
- -- RN-1740, Axial Flux Difference Control, Revision 00, November 27, 1985

5.2 Findings

5.2.1 T.S. 4.1.1.1.2 (Boration Control) states in part, "The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ±1% ΔK/K at least once per 31 Effective Full Power Days (EFPD)... The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each refueling". Procedure RX 1709, which performs this surveillance, does not incorporate the 60 EFPD normalization period. The licensee stated that the procedure would be changed to reflect this. As this is a post refueling requirement, corrective action is not required to be completed by fuel load or initial plant startup. This item is unresolved pending completion of licensee action (50-443/86-28-01).

5.2.2 Procedure 1704, paragraph 4.2 states in part, "for end of cycle life, the MTC shall be less negative than -5.6 X 10-4 ΔK/K/°F for all rods withdrawn, rated thermal power conditions"; and paragraph 8.1.1 states in part, "For EOL measurement, if the measured MTC is more negative than -4.7

X 10-4 ΔK/K/°F, the MTC shall be remeasured and compared to the EOL MTC limit...". The current T.S. values (T.S. 4.1.1.3.b) for the above MTC measurements are -4.2 X 10-4 and -3.3 X 10-4 ΔK/K/°F respectively. The licensee stated that this procedure would be changed to reflect current T.S. valves. As these values apply to the end of core life MTC, corrective action is not required to be completed by fuel load or plant startup. This item is unresolved pending completion of licensee action (50-443/86-28-02).

5.2.3 The licensee had identified approximately 50 computer programs to be developed by reactor engineering for measuring and calculating various core parameters and other primary or secondary plant parameters. A licensee representative stated that, as a minimum, those programs required for plant startup and low power operation would be ready by June 15, 1986. This item is open pending completion of licensee action to finalize required reactor engineering computer programs (50-443/86-28-03).

6.0 Independent Inspection Verifications

On a sampling basis, the inspector performed an independent verification of set points, initiating devices, and automatic actions for nine alarm response procedures by a Control Room walk-through of each procedure. See paragraph 3.3.1.1 for details.

7.0 Management Meetings

Licensee management was informed of the scope and purpose of the inspection at two entrance interviews conducted on May 12 and May 15, 1986 respectively. The findings of the inspection were periodically discussed with licensee representatives during the course of the inspection. An exit interview was conducted on May 22, 1986 (see paragraph 1 for attendees) at which time the findings of the inspection were presented.

At no time during this inspection was written material provided to the licensee by the inspectors.