U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No.: 50-443/88-10

License No.: NPF-56

Licensee: Public Service Company of New Hampshire

1000 Elm Street

Manchester, New Hampshire 03105

Facility Name: Seabrook Station, Unit No.1

Inspection At: Seabrook, New Hampshire

Inspection Conducted: July 6 - September 6, 1988 and September 21, 1988

Inspectors: A. C. Cerne, Senior Resident Inspector, Seabrook Station
D. G. Ruscitto, Senior Resident Inspector, Seabrook Station
E. Yachimiak, Operations Engineer (Examiner), PWR Section, Division of Reactor Safety
C. J. Conklin, Senior Emergency Preparedness Specialist, Emergency Preparedness Section, Division of Radiation Safety and Safeguards

Approved By: Donald R Haverkamp — 9/28/88
Donald R. Haverkamp, Chief, Reactor Projects Section No.3C — Date

Inspection Summary:

Areas Inspected: Routine inspection on day and backshifts by two resident inspectors and two regional specialist inspectors of actions on previous inspection findings, NRC Bulletins and Information Notices, operational safety, licensee potentially reportable occurrences and operational events, maintenance and surveillance activities, design changes, allegations, training, and electrical configuration control.

Results:

1. General Conclusions.

   A repetitive weakness was identified in the implementation of the tagging program involving physical removal of a section of non-safety related piping containing a valve which was caution tagged. While the non-safety nature of the equipment indicates that regulatory requirements were not violated, the recurrent nature of the incident indicates that further management attention in this area is warranted (Refer to paragraph 8.b).

   A weakness was identified in the licensee's reporting system with respect to diesel generator failures (Refer to paragraph 4.k)

   A weakness was identified in the calculations associated with non-class 1E loads powered from class 1E power sources. Licensee evaluation of this problem is continuing and is being tracked under existing unresolved item 88-06-01 (Refer to paragraph 4.j).

   A licensee strength was demonstrated in the handling of testing and inspection of flanges and fittings in accordance with NRC Bulletin 88-05. Strong participation by quality assurance and engineering personnel contributed to the licensee's ability to respond to this industry wide problem in a timely fashion (Refer to paragraph 6.b).

2. Violations.

   A violation was identified regarding the failure to report diesel generator failures in accordance with the technical specifications (Refer to paragraph 4.k.).

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Attachments:

* The NRC Inspection Manual inspection procedure that was used as inspection guidance is listed for each applicable report section.

DETAILS

1. Persons Contacted - New Hampshire Yankee (NHY)

   E. A. Brown, President and Chief Executive Officer
   \# W. A. DiProfio, Assistant Station Manager
   * T. C. Feigenbaum, Vice President, Engineering, Licensing and Quality Programs
   W. J. Hall, Regulatory Services Manager
   * D. E. Moody, Station Manager
   G. S. Thomas, Vice President, Nuclear Production
   * J. M. Vargas, Manager of Engineering
   * J. J. Warnock, Nuclear Quality Manager

   * Attended exit meeting conducted on September 9, 1988
   \# Attended exit meeting conducted on September 22, 1988

   Interviews and discussions with other members of licensee and contractor management, and with their staffs, were also conducted relative to the inspection of items documented in this report.

2. Summary of Facility and NRC Activities

   a. Resident Inspector Activities

      On August 8-11, 1988, the Resident Inspector attended a Resident Inspector Seminar in King of Prussia, Pennsylvania.

      On August 8-19, 1988, the Senior Resident Inspector travelled to Rockville, Maryland for a temporary assignment with the NRC Office of Nuclear Reactor Regulation.

      On August 17, 1988, the resident inspectors attended a management meeting between the NRC and NHY in King of Prussia, Pennsylvania. (Refer to paragraph 13 of this report)

      On September 1, 1988, the Senior Resident Inspector was reassigned to another duty station. The Resident Inspector was assigned as Senior Resident Inspector.

   b. Visiting Inspector and NRC Management Activities

      On July 18-22, 1988, an NRC Region I operations engineer (examiner) conducted a routine inspection of plant operations and previously identified items. His inspection findings are included in this report.

On August 16, 1988, the Director, Office of Nuclear Reactor Regulation visited the site. He held discussions with the Resident Inspector and toured the plant. The NHY inventory department staff was requested to provide information concerning the Seabrook program for material receipt inspection and identification of fraudulent or substandard parts.

On September 21, 1988, an NRC Region I senior emergency preparedness specialist conducted a routine inspection of previously identified items. His inspection findings are included in this report.

c. Plant Status

During this reporting period, the plant remained in operational Mode 5, cold shutdown, with primary temperature between 105 and 140 degrees F and depressurized. Major maintenance was conducted on service water cooling tower pump SW-P-110A, the reactor trip breakers, the chemical and volume control system, the control building air handling system, the waste gas system, the diesel generators and switchyard circuit breakers and bus ducts.

Major 18-month surveillance was conducted on the emergency diesel generators, emergency core cooling systems, engineered safety features actuation systems and ventilation filters.

On July 19, 1988, while performing surveillance testing on the train "A" containment building spray system, an improper valve lineup caused approximately 5,000 gallons of water from the refueling water storage tank to flow to the suction of the operating train "A" residual heat removal pump suction and into the reactor coolant system. Details of this event may be found in paragraph 7.f of this report.

Significant design changes were initiated on the secondary component cooling water and post accident sampling systems. Further discussion of these changes may be found in paragraph 8 of this report.

A major licensee activity involved identification and testing of flanges and fittings in accordance with NRC Bulletin 88-05. Further inspection of this bulletin may be found in paragraph 6.b of this report.

3. Operational Safety

a. Plant Inspection Tours

The inspectors observed station activities and plant status during general inspections of the plant. The inspectors examined work for any apparent defects or noncompliance with regulatory requirements or license conditions. The inspectors interviewed station staff and contractor personnel in their work areas.

During control room observation periods, during both normal working hours and on backshifts, the inspector reviewed control room logs and records including night orders, shift journals, shift turnover sheets, the temporary modifications log, and control board indications. Specific note was taken of equipment in "pull-to-lock" conditions, equipment tagged, alarm status and adherence to technical specification (T.S.) limiting conditions for operation and action statements. Also, boron samples, taken from the reactor coolant system and connected water supplies, were spot-checked for concentration, sample frequency and documentation in accordance with specified zero power license conditions.

The inspector verified the proper position, in accordance with operational procedure or work controls of various valves, switches and breakers during system walk-downs and checked the valve and switch status in the control room. Similarly, temporary modifications and component tagging, maintenance work, and design change implementation activities, as observed during plant inspection tours, were evaluated for evidence of both proper field controls and coordination of the subject work activity with the control room and operations personnel on shift. In certain cases, the operability of specific components and the applicability of the observed work to the T.S. requirements were discussed with the operators.

The inspector identified several minor discrepancies in material conditions. A list of items was provided to the licensee. Action taken on each issue is described below.

(1) Design coordination report (DCR) 87-0185 changed out certain switches on the main control board (MCB). The inspector questioned when the new identification labeling will be completed. The licensee provided work request (WR) 87W007159 initiated on September 30, 1987 to have the labeling finished.

(2) The startup rate meter for nuclear instrument channel N31D on the MCB frequently sticks downscale and requires manual agitation to free the pointer. The inspector questioned the status of resolving this issue since it has been a recurring problem. Request for engineering services 87-452 was initiated on January 6, 1987. Meter operation under normal neutron flux will be observed during the upcoming test program to verify that the present condition is being caused by low core activity levels.

(3) The lens on the indicating light on the MCB for safety injection accumulator SI-TK-9C nitrogen vent valve (SI-FV-2477) requires engraving. The licensee initiated WR 88-2514 to accomplish this task.

(4) The inspector identified a disassembled conduit clamp on instrument rack MM-IR-73 in the service water pumphouse. The licensee took corrective action to reclamp the conduit.

(5) The 345kV schematic drawing posted on the wall of the relay room was not being controlled as an approved operator aid. The licensee provided a new controlled copy of the drawing and posted it in accordance with NHY guidelines delineated in the Operations Management Manual, Chapter 8, "Operator Aids".

On July 13, 1988 while touring the tank farm, elevation 20'-0", the inspector noted valves CBS-V39 and CBS-V44 unlocked and closed. These valves are normally locked open. The inspector verified that the locked valve log in the control room reflected the current status of the valves and determined that adequate controls were in place to ensure that the valves would be returned to their proper positions when required.

While touring the control room on July 20, 1988, the inspector noted that suction pressure for train "B" emergency feedwater (EFW) pump FW-P-37B indicated 6 psig, while the suction pressure for the train "A" pump (FW-P-37A) indicated zero psig. The inspector verified by inspecting the EFW pumphouse that the suction valves to each pump were danger tagged closed and that plastic isolation "pancakes" had been installed downstream of the suction valves to keep the pump casings dry. Since the tap for the FW-P-37B suction pressure instrument is between the "pancake" and the closed suction valve, any leakage past the suction valve or trapped pressure would be sensed by the suction pressure instrument. Based on this information, the inspector had no further questions.

While touring the essential switchgear rooms the inspector noted that the indicators for containment building spray (CBS) system sump level were not identified. These level indicating tranmitters CBS-LIT-2384 and CBS-LIT-2385 were installed by engineering change authorization 03/109038H in 1985. The inspector reviewed the above ECA along with the applicable design change notice (DCN 65/0259A) and budget expense revision (BER 742A). The licensee stated that the indicators will be labeled.

b. Operational Events

(1) Paragraph 4.g of this report details a reporting deficiency concerning diesel generator failures. As described in that paragraph the licensee instituted a new reporting procedure utilizing the station information report (SIR) process. Subsequent to this procedural modification, two additional failures occurred. The inspector reviewed the preliminary SIRs on the failures which occurred on August 11 and 12, 1988 on the train "B" engine. These failures will be the subject of 30-day reports to the Commission in accordance with Seabrook Technical Specification 6.8.

(2) On August 10, 1988 the electrical load dispatcher offsite opened up 345 kV circuit breaker No.163 in the switchyard. At the time, 345 kV circuit breaker No.11 was open and out of service for maintenance. Train "B" emergency diesel generator (EDG) was also out of service for maintenance as permitted by Technical Specifications. The result of this breaker opening was an undervoltage condition to buses E5 and E6 and the resulting automatic start of the train "A" EDG. As expected, no transfer of power from the unit auxiliary transformer to the reserve auxiliary transformer occurred, and power was restored by manual operator action without incident. The licensee made a non-emergency report to the NRC operations center in accordance with 10 CFR 50.72. The inspector reviewed the preliminary station information report and will followup licensee activities under the licensee event report when issued.

(3) On July 8, 11, 13, 15, 20, 26, August 3, 1988, the licensee made 48-hour, non-emergency calls to the NRC Operations Center via the emergency notification system pursuant to NRC Bulletin 88-05. Additional information on this issue may be found in paragraph 6.b of this report.

4. Licensee Action on Previous Findings

a. (Closed) Unresolved Item 86-54-02: Containment Building Spray (CBS) Pump Suction Piping Design Questions. The primary issue raised with this unresolved item involved questions of code compliance and adequacy of the overpressure protection of a portion of the CBS system piping. Since the residual heat removal (RHR) system piping is designed to higher system pressure requirements than that of the CBS system, the adequacy of a single check valve in each of four lines interconnecting the RHR and CBS systems was evaluated with respect to design commitments, American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code interpretations, and current ASME Code guidance.

The inspector held several meetings, including telephone conferences, with licensee engineering and licensing personnel during the first half of 1987 to discuss the subject design questions. The original temperature/pressure design data for the CBS piping was reviewed and an ASME Code subcommittee member was interviewed in regard to precise interpretation and requirements of Section NB-3612.4 of the ASME Code, Section III (1971 Edition, Winter 1972 Addenda). Furthermore, the NRC Office of Nuclear Reactor Regulation (NRR) became involved in the question of original design adequacy and FSAR commitments. As stated in Supplement No. 7 to NUREG-0896, the Seabrook Safety Evaluation Report (SER) issued in October, 1987, the NRC staff concluded that:

"Although the current guidelines in Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME Code) stipulate the use of two series connected check valves for such system interface applications, the applicant is in compliance with the ASME Code requirements under which the Seabrook RHR and CBS system piping was designed and constructed."

Therefore, the question of the code compliance of the original CBS system design was reviewed and determined to be adequate by NRR. However, based upon concern over the potential for RHR system leakage to the CBS system pump suction piping, as had been noted to occur in late 1986, the licensee committed to implement both short and long-term corrective actions. The major element of the licensee's short-term actions involved the installation of a piping thermal monitoring system (PTMS) which generates an alarm in the control room when the CBS system piping temperature profile indicates that leakage from the RHR system is occurring. Operators could then evaluate and estimate the RHR-to-CBS system leak rate and respond with the appropriate valve and system realignments.

The inspector witnessed field activities associated with the installation of the PTMS, examined the final thermocouple locations and reviewed the operator alarm response actions. As documented in Supplement No. 7 to the SER, the NRC staff concluded that the licensee's short-term actions were sufficient to resolve concerns of CBS system overpressurization due to RHR check valve leakage and to allow operation with the present CBS/RHR pressure isolation configuration until the first refueling outage.

The performance of longer-term corrective measures, such as the installation of redundant motor operated gate valves in series with the existing check valves, is currently being scoped and analyzed by the licensee. The need for such action is a full-power licensing issue/condition, as noted in SER Supplement No. 7, which resides under the purview of NRR for future evaluation.

With respect to the acceptability of existing field conditions and to the adequacy of licensee contingency actions in response to the subject RHR check valve leakage, no concerns remain and no additional safety questions have been identified. While NRR has further licensing action on this matter, as an inspection issue all the relevant parts of this item have been resolved. This issue is considered closed.

b. (Open) Unresolved Item 87-10-02: NRC Information Notice 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRS": This Information Notice (IN) addressed the degradation of the FSAR four-loop emergency core cooling systems (ECCS) injection flow rate if RHR crossover line valves were closed. As documented in NRC:RI Inspection Report 50-443/87-10, the licensee's Independent Safety Engineering Group (ISEG) recommended that NHY Engineering perform an analysis, based upon Westinghouse Owners Group (WOG) data, which would address the problems associated with the normal RHR shutdown cooling configuration during Mode 4 operation with a closed crossover valve.

Due to a delay in the WOG response to IN 87-01, the licensee's analysis has yet to be performed. In a licensee memo dated July 14, 1988, a commitment to initiate the WOG solution to the IN 87-01 generic problem was made. If the current RHR shutdown cooling procedures are not in accordance with the new solution, then they will be revised with appropriate corrections. As a result, this item remains open.

Additional inspection effort was devoted to the follow-up of operator training in this area. Discussions with on-shift operators revealed that they were familiar with the problems associated with degraded emergency core cooling systems ECCS operability and the closure of the RHR crossover line valves. Procedures which address the valve alignments for shutdown cooling (OS1000.01, OS1013.03, and OS1013.04) and RHR technical pecification surveillance testing (OX1413.01) were reviewed and found to have incorporated the appropriate cautions/statements regarding this problem.

c. (Closed) Unresolved Item 87-16-03: Operation of the Startup Feedwater Pump (SUFP) on an Emergency Bus. Based on the results of preoperational test PT-39.2, "Loss of Offsite Power with SI," reviews of procedure OX1426.02, "D/G 1A 18 Month Operability Surveillance," and subsequent discussions with both the licensee and NRR, two concerns regarding the operation and testing of the SUFP on emergency bus E5 were identified.

The licensee's corrective action for the operations concerns was to revise the applicable emergency operating procedures to ensure that operators would verify that emergency diesel generator (EDG) 1A would have adequate load carrying capability before loading the SUFP on to bus E5. This was verified by a review of the following procedures: E-0, ES-0.1, E-3, FR-H.1, ECA-0.1, and ECA-0.2. In each of these procedures, the maximum allowable EDG 1A load of 3600 kW is addressed as either a caution on the summary page or has been incorporated into the procedure as a required step/action.

The testing concern for EDG 1A and the SUFP loading will be addressed by interpreting Technical Specification 4.8.1.1.2 in accordance with a proposed NRC Generic Letter, which clarifies the description of auto-connected loads. The inspector had no further questions in this area and considers this item to be closed.

d. (Closed) Open Item 87-22-01: Siren Modifications. This item indicated that the sirens located in Rye, New Hampshire required modified antenna ground planes and that several additional sirens required application of the anti-icing coating. The inspector reviewed the repetitive task sheets for the antenna change outs and application of anti-icing coatings for seven Rye sirens.

Based upon the above, this item is closed.

e. (Closed) Open Item 88-09-01: TSC/EOF Technical Support. The inspector participated in the NRC evaluation team which observed the 1988 Annual Graded EP Exercise on June 27-28, 1988, as documented in NRC:RI Inspection Report 50-443/88-09. Several open items were generated concerning exercise weaknesses. The following presents amplification and clarification of certain technical concerns identified in paragraph 3.1 of the above report. Inspection Report 50-443/88-09 stated,

"The Technical Support Center (TSC) and Emergency Operations Facility (EOF) staff displayed questionable engineering judgement and/or did not recognize or address technical concerns (50-443/88-08[9]-01)."

Several issues addressed below were cited as examples. Overall engineering judgement displayed in both the TSC and EOF was adequate, however, the following activities were noted to be isolated areas of weakness which were intended to be addressed by the licensee. In follow-up subsequent to the exercise with licensee technical support, operations and emergency preparedness staff, the following additional information was provided. The resolution of each sub-item of inspector follow-up item 88-09-01 is described individually below.

(1) "Efforts continued to restore the emergency feedwater pump (EFW) after a large break LOCA"

The licensee correctly stated that the EFW pump would be required to operate to support steam generator cooldown in the recovery phase and continued repair efforts were prudent. The inspector agrees and determined that the stated activity did not detract from the overall recovery effort, nor did it diminish other high priority recovery action in progress or planned, and that TSC judgments were made with long-term recovery in mind.

(2) "A questionable fix for the containment building spray (CBS) system"

The inspector met with the Technical Support Manager and a Technical Support Engineer and discussed the rationale behind the corrective action taken to rig an alternative water source for the CBS system. Although the capability of the proposed modification to the system to reduce containment pressure was never proven due to the eventual repair of a CBS pump, the inspector determined, based on this additional information, that the engineering judgment and methodology involved in the proposed system and operating procedure changes were acceptable. The licensee actions were appropriate since this fix was considered to be a "last resort" measure after all prudent and subsequent extraordinary measures had failed to provide containment spray by other means due to additional scenario controller intervention.

Additionally, the licensee had previously determined that the composition of the present TSC engineering staff, while adequate, could be enhanced by providing an augmented staff roster. NHY has committed to implement this initiative.

(3) "A lack of effort to locate and isolate the release path"

This apparent lack of effort was the result of licensee decisions not to pursue entry into the containment enclosure due to high radiation levels. Discussion with the licensee confirmed that indirect measures, such as remote temperature, pressure and sump level indications, were taken in a timely fashion to provide an alternate assessment of potential leakage paths. The inspector was unaware of these activities during the drill. The licensee decision to postpone entry into the containment enclosure was intentional, based upon other recovery efforts associated with depressuring the containment. Restoration of a CBS pump was imminent and activation of this system would have stopped the release. CBS restoration was subsequently, and repeatedly, delayed by controller intervention so that the operators were prevented from affecting repairs. The licensee decisions in this regard were appropriate.

(4) "No effort was noted to blowdown steam generators (S/G) to lessen the heat load in containment"

This comment implied that S/G blowdown was appropriate. The actual concern was that a step in the emergency procedure required the S/G to be depressurized. This step was not performed because the TSC staff was unsure of the integrity of the S/G tubes because no sample was available due to blowdown system isolation. This TSC staff concern was expressed to the inspector when he questioned them during the exercise. The NRC position in this area is that improved guidance to the operator may be warranted and should be evaluated, however the decision not to vent or blowdown the S/Gs without sampling appears to have been reasonable and appropriate.

(5) "Neither the EOF or TSC staff questioned a release of greater than 7C30 curies per second with only clad damage and no core uncovery"

The inspector reviewed the player and controller logs for selected TSC, EOF and engineering support center (ESC) staff. These logs revealed that several staff members did question and/or comment on the mismatch between the reactor coolant activity and the release rate. Subsequent discussions with the TSC and EOF controllers and players also indicated that they were aware of this mismatch. In actuality, the ESC staff made very accurate core damage assessments based upon the data supplied by the TSC. The EOF dose assessment staff made accurate dose projections based upon the release rate, as well as correlation of field data to the release rate. A review of previous drill comments, as well as the player instruction for this exercise, indicated that this level of activity is recognized to be an unrealistic number, which is required to provide the offsite dose rates necessary to exercise the entire emergency planning zone. The technical staffs had repeatedly identified and questioned these mismatches in previous drills and were told by the controllers that this high release rate was necessary to test the off-site plans, and that they should not challenge the data.

Although NRC review of the specific scenario used for the exercise was acceptable, the above described problem indicates that the licensee should place more effort in developing exercise scenarios where core damage and release rates are consistent.

With respect to the above identified weaknesses, the exercise inspection confirmed that the TSC/EOF staff possesses adequate capabilities to protect public health and safety. This open item is considered closed.

f. (Closed) Open Item 88-09-02: TSC/OSC Multiple Access Points. This item indicated that the TSC and Operational Support Center (OSC) have multiple entrances and exits that are not controlled. As a result, contamination controls were ineffective at times as personnel entered without frisking and it couldn't be determined if continuous accountability was, or could be, maintained.

The TSC has a main entrance where contamination controls and initial and continuous accountability is established and maintained. The TSC also has a back entrance which is not locked. Although this entrance is not normally used, the licensee agrees that it could be used, in effect bypassing the controls established at the main entrance. The licensee has agreed to change ER 3.1, "Technical Support Center Operations", to control access through this entrance as well as move the main entrance controls.

The OSC also has multiple entrances. However, this was a condition that was artificial to the exercise. At the time of the exercise, the radiological control area (RCA) had not been implemented at the station. The licensee procedures clearly show that when the RCA is implemented there will be only one entrance into the OSC from the RCA.

The inspector noted that the licensee established and maintained habitability throughout the exercise. Although some minor contamination could have occurred in the TSC, it is clear it would have been promptly recognized and would not have adversely impacted TSC operations.

g. (Closed) Open Item 88-09-02: Departing Shift Dosimetry. This item indicated that no apparent consideration was given to the departing first shift to account for possible dose when leaving the plant during the release, as they were not given dosimetry.

A subsequent review of the TSC logs, as well as discussions with TSC and OSC staff, indicated that consideration was given to the departing shift. Contamination and radiation surveys were ordered and taken. Results indicated all areas were below background. Because of this and the current wind direction, the TSC staff elected to allow the departing shift to exit the site without dosimetry.

Based upon the above review, this item is closed.

h. (Closed) Open Item 88-09-04: Media Center Responses to the Press Inquiries. This item concerned the licensee representative's responses to some questions in the Media Center which were not considered adequate. The licensee has agreed that these questions were not fully answered. Although the answers given were current, they did not have enough substance. The licensee has agreed to upgrade the training for the Media Center spokesperson, including more information on the NRC Incident Response Team capabilities and roles. Additionally, during a real emergency, federal spokespersons would have been available to provide clarification as the need arose. This item is closed.

i. (Closed) Unresolved Item 88-02-01: Accumulator Isolation Valve Actuation Logic Questions. In meetings with licensee operations and engineering representatives in June and August, 1988, the resident inspectors discussed questions regarding the "maintain CLOSED" switch, its function and design features. Licensee personnel adequately addressed the compliance of the current design with Institute of Electrical and Electronic Engineers (IEEE) Standard 279 and IE Bulletin No. 80-06 guidance. Additionally, the inspector reviewed system test packages for the wiring verification and functional checks (reference: general test procedure, GT-E-21) of the subject valve circuitry to confirm the opening of the accumulator isolation valves upon receipt of a safety injection signal with the switch in the "maintain CLOSE" position.

The licensee stated that the FSAR described a valve capability for future operational testing which, while currently available, was prohibited from use by technical specification requirements. The inspector evaluated this position and determined that the governing administrative and LCO controls were adequate to prevent safety problems during routine operation and shutdown activities. Only specific plant transitional situations and mode changes (particularly entry into Mode 3) represent potential problem areas. It was noted that the Westinghouse Owners Group is evaluating accident scenarios in Mode 3 below 1000 psig reactor coolant system (RCS) pressure and in Mode 4 on a generic design basis.

In order to address the inspector's specific concerns regarding the adequacy of current procedures/drawings and of future operational controls if technical specification requirements are revised to allow accumulator isolation valve closure in higher modes for testing in accordance with FSAR provisions, the licensee implemented the following actions:

(1) Issued Revision 10 to the "SI-Accumulator Isolation Valves Logic Diagram", 1-NHY-503907, to delineate the pressure setpoint above which an alarm is actuated if the valve is not fully open.

(2) Initiated revisions to the affected alarm response procedures to correct the recommended action references relative to the proper RCS pressure setting at the safety injection (SI) unblock pressure.

(3) Recommended revision to the SI system description, SC-NAH/NCH-284, Foreign Print No. 52005, for the accumulator tank isolation valves discussing valve closure after resetting an SI signal with the valve controls in a "maintain CLOSED" position.

The inspector reviewed licensee engineering memoranda, including one issued by the Yankee Atomic Electric Company, Nuclear Services Division, on the accumulator isolation valve actuation logic and considered the adequacy of the current Emergency Response Procedures to the SI valve response design, including SI signal reset. No problems with existing controls were identified.

The inspector determined that the questions on the subject system design and controls have been adequately addressed and that the licensee has taken steps to ensure the continued adequacy of design control if the technical specifications are amended to incorporate the full accumulator design features discussed in the FSAR. This unresolved item is considered closed.

j. (Open) Open Item 88-06-01: Non-Class 1E Loads Powered from Class 1E Sources. This item was originally opened to resolve the issue surrounding the tachometer on the emergency feedwater pump (EFW) turbine. Subsequently the NRC concern has been expanded to include the entire program for design, identification and testing of non-class 1E loads powered off of class 1E sources.

(1) Background

NRC:RI Inspection Report 50-443/88-06 described a non-class 1E circuit (EFW tachometer) which was not included in the NHY Technical Requirements Manual (NYTR) list of devices to be tested per technical specifications (T.S.).

The T.S. involved in this issue consists of two parts which deal with containment penetration conductor overcurrent protective devices and protective devices for class 1E power sources connected to non-class 1E circuits. This discussion concerns only the class 1E power sources connected to non-class 1E circuits. This specification states that each protective device for class 1E power sources connected to non-class 1E circuits shall be operable in Modes 1-6.

With one or more of the protective devices inoperable, the circuit must be de-energized by tripping the circuit breaker or racking out or removing the inoperable device within 72 hours. In addition, the above status must be verified every seven days thereafter. The surveillance requirements necessary to declare operability include periodic testing, inspection and preventive maintenance of the device. The list of protective devices to be tested per T.S. Surveillance Requirement 4.8.4.2 were incorporated into NYTR Table 16.3-10 (Technical Requirement 15) under the T.S. Improvement Program.

The NHY Systems Support Department Manager reported on May 2, 1988 that his review of the circuit indicated that the tachometer for the turbine-driven emergency feedwater pump was a non-class 1E load connected to safety-related bus E5 via 120 vac motor control center E515 distribution panel F3E, circuit 4. Request for engineering services (RES) 88-226 was written on May 6, 1988 to determine whether this circuit should be included in Table 16.3-10 of the NYTR. A station information report (SIR) was initiated on July 26, 1988 to document this situation and further clarify the reporting requirements. Licensee event report (LER) 88-002 and its supplement document previous instances where other non-class 1E circuits were omitted from Table 16.3-10 of the NYTR. Additional NRC inspection of this previous LER may be found in NRC:RI Inspection Reports 50-443/88-06, paragraph 5c and 50-443/88-07, paragraph 5.

Licensee evaluation of this issue was conducted as an SIR follow-up. Engineering review of calculation 9763-3-ED-00-46-F, "Failure of non-class 1E Loads on class 1E Buses" revealed several additional loads requiring immediate resolution to ensure compliance with the T.S. As of the end of this reporting period temporary modifications had been made to nearly all of those circuits and a permanent design change is in progress.

(2) Chronology

January 1988	Licensee review indicates that the supply breaker to inverter 2B off of unit substation E51 is not on the list in the NYTR.
February 1988	Following evaluation of preoperational testing previously conducted on the breaker, it is determined that the breaker must be tested. It fails the test, is repaired and the system is restored to operable status.
March 1988	LER 88-002 is submitted indicating that a review of all unit substations reveals that the above finding is an isolated case.
April 1988	The inspector provides a copy of a January, 1988 daily report from another nuclear facility about the power supply to the auxiliary feedwater pump tachometer which is similar to the above finding.

May 1988 Request for engineering review of Seabrook EFW pump turbine tachometer is issued by NHY (RES 88-226). The licensee determines that the EFW pump tachometer is not class 1E. The tachometer circuit is not disconnected electrically from its 1E power source as required by the T.S. action statement.

Licensee discovers the breakers between 2 pairs of unit substations are also not on NYTR list. Substation tie breakers are added to list. Supplement 1 to LER 88-002 issued.

July 1988 Licensee review of the relevant engineering calculation determines that two separate problems exist:

(1) Coordination of the tie breakers in the unit substations

(2) EFW tachometer circuit

Circuit breaker for EFW pump is opened per T.S. after discussion with the inspector.

August 1988 Continued review of calculations indicate that trains "A" and "B" have additional circuits which are not analyzed and are required to be disconnected per T.S. Temporary modifications are initiated so as to be completed prior to expiration of the 72-hour LCO. A permanent design change is in progress.

(3) Inspection

The inspector held frequent discussions with the Technical Support Manager and Lead Technical Support Electrical Engineer concerning progress of the analysis and installation of the temporary modifications. A licensee event report will be submitted. Preliminary NRC review of the train "B" temporary modifications revealed no concerns.

(4) Findings

Based on the above, the following issues remain unresolved:

(a) Adequacy of the original determination of which components were to be incorporated into the NYTR list.

(b) Licensee actions taken upon discovery of the non-class 1E EFW tachometer powered from a class 1E bus.

(c) Reportability of the above findings in accordance with 10 CFR 50.73.

An additional question that must be resolved concerning the NYTR is whether non-class 1E loads which meet seismic design criteria may be omitted from the NYTR listing. Licensee and NRC activities are ongoing and will be the subject of continuing evaluation. This item under expanded scope remains open.

k. (Closed) Violation 88-06-02: Emergency Diesel Generator (EDG) Failure Reporting

(1) Background. NRC:RI Inspection Report 50-443/88-06 described a trip of the train "B" emergency diesel generator which occurred on February 24, 1988. Open Item 88-06-02 was written to document NRC questions related to the reportability of this failure. Based upon the NRC questions, NHY conducted a comprehensive review of the diesel generator logs and determined that seven failures had occurred since issuance of the zero power license in October 1986. The failures were analyzed and summarized in a letter to the NRC (NYN-88102) dated July 22, 1988. The informational requirements of T.S. 4.8.1.1.3 were addressed for the most recent failure on February 24, 1988. Additionally, the six previous failures were reported to bring the record up to date.

(2) Requirement. The above T.S. is applicable in Modes 5 and 6. Surveillance Requirement 4.8.1.2 states that the required ac electrical power sources shall be demonstrated operable by performance of Specification 4.8.1.1.3. This surveillance specification states that all diesel generator failures shall be reported to the Commission in a Special Report within 30 days.

(3) Findings. None of the above failures were reported within the 30-day time frame required by T.S. 4.8.1.1.3 and this failure to report constitutes a violation of the Seabrook Technical Specifications (88-06-02).

(4) Licensee Corrective Actions. Licensee corrective actions as a result of this violation and actions to prevent recurrence were provided to the NRC in letter NYN-88102. NHY reporting procedures have been revised to address EDG failures. The station information reporting system will be utilized to ensure that appropriate post failure actions are taken.

Based upon the above and appropriate licensee actions initiated on two recent diesel failures, the inspector considers this issue closed and no additional response is required.

5. Licensee Reports

a. (Closed) Construction Deficiency Report (CDR) 86-00-09: Veritrak/Tobar Transmitters. NRC:RI inspection reports 50-443/87-24 and 88-06 both document the progress made in the installation of Rosemount transmitters to correct this deficiency. Design coordination report (DCR) 86-349 was implemented to control the rework and complete the corrective action documented in the final 10 CFR 50.55(e) report to the NRC.

During this inspection, the inspector examined the completed field installation of all 23 Rosemount transmitters in the Unit 1 containment building. The rework associated with change authorization No. 7 to DCR 86-349 was checked and specific installation details (e.g., compression fittings) were examined. The inspector also noted that the installed components were Rosemount Model 1154 transmitters, different from the Model 1153 transmitters that have exhibited manufacturing deficiencies at other nuclear power plants.

The inspector reviewed the DCR for calculations affecting instrument setpoints and determined that certain technical specification tabular data and limiting conditions for operation setpoints require revision. The licensee submitted letters to the NRC dated May 27, July 8 and August 3, 1988 (NYN-88075, NYN-88091, and NYN-88109 respectively), which discuss the methodology used in the Rosemount setpoint analysis and transmit the proposed technical specification changes and a supplemental analysis of the relevant safety considerations. The inspector reviewed these documents, noting consistency with the Westinghouse setpoint methodology (also discussed in NRC:RI inspection report 50-443/87-24) and with the values calculated in DCR 86-349. The inspector's review of the proposed technical specification revisions were discussed with NRR project and technical reviewer personnel.

The inspector confirmed that system operability considerations will be adequately controlled by the proposed technical specification changes, that a license amendment has been requested and is being processed, and that the licensee has completed all corrective actions relevant to its final 10 CFR 50.55(e) report. Adequate consideration of the level measurement error due to reference leg heatup for the steam generator level reactor trip and emergency feedwater actuation setpoints was also verified to have been included in the Rosemount data calculations. A licensee request (NYN-88082) dated June 9, 1988, regarding the need for operator action in response to level measurement errors also has been transmitted to NRR for review.

All corrective measures commitments have been completed and no further action is required of the licensee at this time. This CDR is considered closed.

b. (Closed) 10 CFR 21 Report (87-88-04): Gould Relay Failures. The failure of Seabrook-specific modified Telemecanique J-10 relays in April and August, 1987 resulted in a licensee investigation into the number and use of relays installed at Seabrook Station. NHY engineering evaluation 88-001, "J-10 Relay System Evaluation", concluded that plant operation with the defective relays in service was acceptable during Modes 5-6, but was unacceptable during Modes 1-4.

Of the 112 J-10 relays which were found to be in service in the plant, 57 were installed in safety-related applications. These were replaced in accordance with DCR-87-390.

Because of the unique voltage requirements specified for the original relays, Telemecanique was unable to ensure a qualified 40 year operational design life for the replacement relays. Analysis showed that a design life of only 4.3 years could be guaranteed. This reduction in design life resulted in the generation of maintenance procedure MS0514.17, "Telemecanique J-10 Relay Magnet Block Replacement". This procedure provides the instructions necessary to change out all safety-related J-10 relays prior to the end of their design life.

To verify that these changes were made, the inspector conducted a field walkdown of selected replaced relays with the cognizant technical support engineer. This sampling included the following relays:

System	Relay	Work Package
CBA	E42/9a-3-3	87W008095
CBA	E42/9a-3-4	87W008096
PCCW	RYY-2192-1L, 2L, 3L	87W008132, 8133, 8134
PCCW	RYY-2292-1L, 2L, 3L	87W008135, 8136, 8137
EAH	E3E/8-R1	87W008112
EAH	E3F/8a-R2	87W008113
EPA	RBC7a	87W008114

All of the above listed relays were verified to have been replaced. A document review of the above listed work packages was performed. No discrepancies were identified. The inspector has no further questions in this area and considers this item to be closed.

c. (Closed) 10 CFR 21 Report (87-88-03): Service Water System Valve Liners and Seats. A generic problem was identified with the discovery in May, 1987 of the premature deterioration of the liner/seats of certain butterfly valves supplied by Fischer Controls. The subject valves, installed in the service water system, had been modified previously as corrective action in accordance with a 10 CFR 50.55(e) report (85-00-13) in which liner detachment problems were noted. The root cause of the most recent deterioration problem was attributed to inadequacies in the modified seat design and in the elastomer liner bonding process applied to correct the original detachment problem.

This issue was first opened in NRC:RI inspection report 50-443/87-13 and was reviewed by an NRC:RI specialist inspector, as discussed in report 50-443/87-18. The licensee submitted a 10 CFR 21 report (NYN-87091) to Region I on July 28, 1987. The inspector reviewed the licensee's "Summary Report on Service Water System Valves", dated July 29, 1987, noting discussion of both short term and long term corrective action programs. With respect to the short term, NRC inspectors, over the past year, have witnessed licensee implementation of a repair and test program for the subject valves. Twenty-eight valves were modified with an improved valve liner/seat design which has increased the liner thickness to preclude deterioration (reference: DCR 87-249). Also, the installation of design modifications (DCR's 87-315 and 87-401) to the piping downstream of certain of the valves was inspected. These changes allowed for the subject valves, previously utilized in throttling applications, to be positioned either fully opened or closed, thus reducing the potential for future deterioration. By July, 1988, all the design changes associated with the service water valve rework and system redesign had been completed.

Longer term corrective action consists primarily of a monitoring program to ensure that short term corrective action has been effective. The licensee plans to conduct an inspection of four of the modified valves, including two that were changed from a throttling application, during the first refueling outage. The inspector verified that this activity has been formally noted in the licensee's integrated commitment tracking system (action no. RE02082). The inspector also reviewed scheduled maintenance data sheets which prescribe the inspection of two additional modified valves for seal/liner damage. Such checks will occur each time the service water strainers in proximity to the valves are removed for cleaning, at a frequency of about every two months or whenever differential pressure indications dictate. Also the licensee has fabricated test coupons of the modified elastomer liner material bonded to valve-like metal. These test coupons have been immersed in the circulating water pump house basin to monitor the effect of seawater on both the elastomer and the bonding process. The inspector examined two work requests describing the removal of the test coupons to be conducted in the latter part of 1988 for transmittal to the elastomer manufacturer, Belzona Molecular Laboratory, for pull testing.

The insepctor noted that both the short and long term corrective actions taken or planned by the licensee in response to this design deficiency were consistent with the 10 CFR 21 report submitted to the NRC and with the discussion of the deficiency documented in NRC:RI inspection report 50-443/87-18. Short term corrective actions have been completed and long term corrective actions are scheduled and being tracked. The inspector has no further questions at this time with respect to the licensee evaluation of the problem, the testing conducted to effect a workable design solution, the actual repairs or the plans for future monitoring of the valves to check for liner deterioration. The licensee's overall approach to this problem from a technical standpoint has been methodical and comprehensive. The NRC has been kept informed of new developments and licensee plans. This 10 CFR 21 Report is considered closed.

d. Station Information Reports. Licensee station information reports (SIR) are used to internally report and evaluate operational events that may require further investigation, notification to a regulatory agency or require root cause analysis. Licensee Event Reports and 10 CFR 21 reports normally originate with an SIR. The reports discussed below were reviewed for compliance with the implementing instruction. Supervisory, regulatory services, management and SORC reviews were verified. Also examined were the technical evaluation of each event, root cause analysis and recommendation.

(1) SIR 88-010: On January 15, 1988 the train "A" emergency diesel generator (EDG) was unloaded and shutdown during a post maintenance test because of a lifting relief valve in the auxiliary cooling water system. As a result of this SIR several minor design changes were instituted to improve engine reliability and performance. The inspectors discussed these modifications with the Systems Support Manager and the cognizant Lead Systems Engineer.

(2) SIR 88-054: This SIR was initiated to investigate the root cause of a mispositioned circuit breaker in the service water system. The licensee evaluation revealed minor administrative work control deficiencies and some human factors improvements which should be made in the labeling of the affected motor control centers.

6. NRC Bulletins and Information Notices

a. (Closed) NRC Bulletin 87-02, Supplements 1 and 2: Fastener Testing to Determine Conformance with Applicable Material Specifications. As documented in NRC:RI inspection report 50-443/87-26, Bulletin 87-02 was closed based upon the conduct of testing and submittal of test results by the licensee to the NRC. The inspector assessed all the actions taken by the licensee in response to this bulletin and determined that they were both complete and adequate.

Subsequently, the NRC issued Supplements 1 and 2 to NRC Bulletin 87-02, requesting, and then clarifying the request for, additional information on the suppliers and manufacturers from which the subject fasteners may have been purchased. On July 21, 1988, the licensee responded to the supplemental requests by letter (NYN-88099) to the NRC. Enclosed with the letter were a list of approved vendors who supplied or may have supplied ferrous fasteners suitable for safety-related applications and a list of vendors who supplied commercial grade fasteners. The licensee response also discussed the basis for compilation of the lists and a commitment to notify the NRC of any additional suppliers or manufacturers identified by on-going procurement record reviews.

The inspector reviewed the information submitted in response to Supplements 1 and 2 to NRC Bulletin 87-02. No questions or concerns regarding this submittal were identified. This bulletin remains closed for inspection purposes.

b. (Closed) NRC Bulletin 88-05, with Supplements 1 and 2: Nonconforming Materials Supplied by Piping Supplies, Inc. at Folsom, New Jersey and West Jersey Manufacturing Company at Williamstown, New Jersey. NHY responded to NRC Bulletin 88-05 by letter (NYN-88114) on August 25, 1988. This letter included the detailed results of the licensee effort to determine the impact of suspect materials at Seabrook. The NHY program consisted of the following:

- Identification of affected materials in safety related systems
- Verifying acceptability of installed materials
- Reporting to the NRC in accordance with the requirements of the bulletin

A total of 369 flanges and fittings were identified in safety related systems. A test program was developed to measure the hardness of carbon steel items and ferite content in stainless steel items. Licensee representatives participated in an Electric Power Research Institute workshop on the use of the Equotip test equipment. NHY q.lity control (QC) inspectors performed the field testing of each flange and fitting. The data sheets were evaluated by the cognizant quality assurance (QA) engineer. On July 15, 1988 in the service water cooling tower, the inspector observed field hardness testing of the service water system flanges. The testing was conducted in accordance with procedure NHY-EHT-1, "Equotip Hardness Testing" (Revision 01, Change 01). The inspector reviewed the procedure and work request 88W3339 and verified that licensee QC personnel were knowledgeable concerning both the procedure and test equipment.

Independent measurements were also performed on separate pieces of suspect material by J. Dirats and Co. and Bechtel Corporation to confirm the Equotip test results. Additionally, test results were sent to the Nuclear Management and Resources Council (NUMARC) for generic industry data compilation and analysis. Of the 369 flanges and fitting tested at Seabrook, 30 were found to be below the minimum Brinell hardness value of 137. This is the minimum value specified in the American Society of Mechanical Engineers (ASME) material specification SA-105. The 30 fittings were individually evaluated and found to exceed existing tensile strength requirements in accordance with the ASME code. The evaluation demonstrated the inherent conservatism of the code as well as the correlation between hardness and tensile strength. NHY made seven calls to the NRC Operations Center over the course of the testing as required by the bulletin. These non-emergency notifications were part of the 48-hour reporting requirements that were subsequently discontinued by the issuance of Supplement 2 to the bulletin.

Throughout the course of the test process, the inspector maintained close liaison with licensee QA/QC inspectors, engineers and managers. The methodology employed in identifying, testing and analyzing the suspect fittings was labor intensive. The licensee devoted adequate resources to ensure timely completion. The two shift testing schedule was particularly rigorous and the total support of NHY engineering and quality assurance departments were in evidence. Additional NRC Headquarters review of this bulletin may occur as a result of generic evaluation of the PSI/WJM concern. For inspection purposes, this bulletin is closed.

c. NRC Information Notice 88-46 and Supplement 1: Licensee Report of Defective Refurbished Circuit Breakers. This Information Notice (IN) describes discovery by another utility that certain non-safety related circuit breakers manufactured by the Square D Company were actually refurbished equipment rather than new stock. It has been determined that certain suppliers were refurbishing components and re-labeling them as new equipment. The licensee is conducting its own inspection to determine what effect, if any, this IN may have on Seabrook. During a visit to the facility on August 16, 1988, the Director of the NRC Office of Nuclear Reactor Regulation discussed this issue with members of the licensee inventory and material requirements departments.

The inspector will continue to follow this issue and its relationship to receipt inspection of commercial grade items as well as any future additional NRC correspondence such as NRC Bulletins or additional IN Supplements. For inspection purposes, this is an open item.

d. NRC Information Notice 88-25: Minimum Edge Distance for Expansion Anchor Bolts. An analysis of site specific data affecting the capacity factors of Hilti Kwik-Bolts installed at the minimum specified distance from an unsupported concrete edge revealed safety factors greater than twice the allowable design loads. This analysis, accomplished by the Yankee Atomic Electric Company (YAEC) for the Seabrook Project, utilized conservative assumptions based upon Seabrook design criteria, Kwik-Bolt installation specifications and concrete compressive strength test data. Since no safety concern was identified, the YAEC recommendation to support a Nuclear Management and Resources Council (NUMARC) initiative for generic industry-wide action on this issue was adopted.

The inspector noted that a previous NRC unresolved item, 443/82-03-07, had addressed consideration of the Kwik-Bolt shear cone interaction, including the influence of the spacing of anchors at concrete corners. As documented in NRC:RI inspection report 50-443/85-25, testing was conducted at the Hilti Test Facility in Tulsa, Oklahoma to check the reduction in Kwik-Bolt capacities, in part, at outside corners. The results of such testing, while indicating a reduction in ultimate capacity, were acceptable when considered with respect to the overall expansion anchor design. The unresolved item was therefore closed.

The inspector noted that the past testing of the Hilti Kwik-Bolts, while not accomplished specifically to address the 10 CFR 21 concerns raised in IN 88-25, has confirmed the conservatism of the design, the acceptability of Seabrook site-specific applications and the assumptions made by licensee engineering personnel in calculating design loading data. Thus the licensee positions that Kwik-Bolt installations at Seabrook represent no immediate safety concern and that future reviews can be adequately handled through NUMARC appear to be well founded.

No violations were identified. This item is closed for inspection purposes.

e. IE Information Notice 86-50: Inadequate Testing to Detect Failures of Safety Related Pneumatic Components or Systems. The inspector reviewed internal licensee memoranda providing evidence of engineering review and regulatory cognizance of the subject information notice. The licensee continues to evaluate their methods of air system and component testing and instrument air quality sampling in accordance with FSAR commitments.

The inspector confirmed that although no specific action is required by this information notice, the licensee appears to be investigating the applicability of the relevant safety issues and tracking regulatory commitments and criteria accordingly. No violations were identified. This item is closed for inspection purposes.

7. Maintenance/Surveillance

a. OX 1456.81: Operability Test of ISI Valves. On July 22, 1988 a re-test of the motor operated suction isolation valve to the train "B" safety injection (SI) pump, CBS-V-53, was performed in accordance with surveillance procedure OX145681, "Operability Test of ISI Valves". The test was completed under work request 88W2735 and consisted of the stroking of the valve to gather the required inservice testing (IST) valve stroke time data. The inspector observed the test locally at the valve in the residual heat removal vault. The results of this test were an opening time of 10.69 seconds and a closing time of 10.22 seconds. The maximum allowable stroke time was 15 seconds for each direction. No violations were identified.

b. EX 1804.044: Safety and Relief Valve Setpoint Pressure Test. On June 17, 1988 another nuclear facility reported problems associated with setting main steam safety valve (MSSV) lift setpoints using nitrogen. When these valves were subsequently lift tested with steam, setpoint drift was noted. The inspector reviewed surveillance procedure EX1804.044, "Safety and Relief Valve Setpoint Pressure Test" and verified that Seabrook MSSV's are presently tested in place with system pressure 15-25% below valve set pressure. An assist motor is used to provide the additional test pressure. Therefore the above described problems can not occur at Seabrook.

c. EX 1804.016: Diesel Generator Auxiliary Coolant System Quarterly Test. On May 13, 1988 the train "B" emergency diesel generator (EDG) was returned to service following maintenance. Operability of the EDG is normally verified by four separate surveillance tests; engine start, fuel oil transfer pump performance, cooling water and air start valve performance and auxiliary coolant performance. An administrative error resulted in declaring the EDG operable on May 16, 1988 prior to completion of the test on the auxiliary cooling system (EX 1804.016). Station information report (SIR) 88-048 was initiated because of this occurrence. The SIR indicated that the root cause of the problem was inadequate scheduling because of an error in the Specification Appraisal computer program. The inspector reviewed licensee corrective actions which included adjustment of the program model and had no further questions.

d. IX 1680.921: SSPS Train "A" Actuation Logic Test. On August 19, 1988 the inspector witnessed portions of I&C Department Surveillance Procedure IX 1680.921, SSPS Train "A" Actuation Logic Test. The purpose of the test is to functionally test the train "A" solid state protection system (SSPS) in accordance with technical specification 4.3.1.1 and 4.3.2.1. The inspector witnessed selected steps concerning reactor trip breaker operation locally in the essential switchgear room. The inspector noted effective communications established with the control room, the presence of a knowledgeable electrical quality control inspector and proper control exercised over the procedure by the control room personnel. No violations were identified.

e. EX 1804.015: Diesel Generator 1B 18-Month Operability and Engineered Safeguards Pump and Valve Response Time Testing Mode 5 Surveillance. This is a seven-event surveillance test which satisfies several train "B" Mode 5 technical specification surveillance requirements. The inspector observed portions of event three and event six. Event three involved an emergency diesel generator (EDG) start initiated by resetting the train "B" low steamline pressure safety injection ("S") actuation signal from the main control board. The inspector witnessed the diesel start to a standby idling condition and the starting of the train "B" emergency core cooling system (ECCS) pumps as well as feedwater isolation and main steam line isolation. The test was run twice because of high speed recorder problems which were eventually corrected. In all cases the plant responded as designed. Event six followed the 24-hour run of the train "B" EDG and tested the ability of EDG 1B to start and load upon concurrent loss of offsite power and an "S" signal and to verify that bus E6 sheds its load. ECCS pump and valve response times were obtained and the EDG's ability to accept a cooling tower actuation ("TA") signal while loaded with auto connected loads was also verified. Following successful service water system transfer to the cooling tower, the EDG's ability to accept a large load rejection was tested by simultaneously tripping the cooling tower pump and charging pump. The inspector noted that the control room operators and test director were intimately familiar with the procedure and expeditiously performed the critical post safety injection steps required by procedure. The equipment also was verified to properly perform its intended function. No violations were identified.

f. OX 1406.02: CBS Pump and Valve Quarterly Test and 18 Month Remote Position Indication. On July 19, 1988 while performing surveillance procedure OX 1406.02, "CBS Pump and Valve Quarterly Test and 18 Month Remote Position Indication", about 5000 gallons of water was inadvertently transferred from the refueling water storage tank (RWST) to the reactor coolant system (RCS) via the residual heat removal (RHR) system. The event occurred because valve CBS-V-2, the train "A" RWST to RHR isolation valve was opened with RH-V-22 and RH-V-23, the train "A" RCS to RHR suction valves still opened. The operator immediately realized that the lineup was incorrect and re-closed CBS-V-2.

NRC:RI Inspection Report 86-54 (paragraph 4.a) described a previous similar event which occurred on September 5, 1986 and describes the design bases for the system. Also addressed was the standard Westinghouse design for interlocks in these valves and the NHY position on how certain design features (alarms) would be added to prevent recurrence of the September 5, 1986 event.

The inspector met with the Assistant Operations Manager and discussed several issues related to this event. The licensee's ongoing corrective actions will be observed during a subsequent NRC inspection.

g. Residual Heat Removal (RHR) System

NRC Region I Inspection Report 50-443/87-24 described a discrepancy in the dimensional gap between the train "B" RHR pump casing and impeller. The licensee subsequently disassembled the train "A" RHR pump and found a similar problem. The dimensional gaps were found to be 0.0235 inches and 0.025 inches for the train "B" and "A" pumps respectively. The manufacturer (Ingersoll-Rand) specifies a diametrical clearance between 0.030 to 0.036 inches. Both pumps wearing rings were machined within specification and the pumps restored to service.

On March 18, 1988 the inspector observed the clearance measurements made on the Unit 2 RHR pumps. These pumps were never installed in Unit 2 and were transported from storage to the Unit 1 turbine building for disassembly. The inspector noted appropriate quality control hold points in the procedure. Both quality control and maintenance personnel were considered to be knowledgeable in their tasks. The Unit 2 clearances as measured were found to be within specification.

The licensee conducted an evaluation of this technical issue pursuant to 10 CFR 21. Engineering evaluation 88-016 concluded that given the "as found" dimensions under design thermal and seismic conditions, pump damage would not have occurred and therefore, a substantial safety hazard did not exist. This condition was therefore not reportable under 10 CFR 21.

The licensee conducted a detailed review of all relevant documents to determine whether the wearing rings were modified in some way during the construction or startup phases. The NHY effort consisted of a review of installation and work records and a review of spare part receipt and inventory records. Ingersoll-Rand documents indicated that the clearances were within specification when shipped from their facility. Construction and maintenance records revealed no modifications or replacements were ever performed on the wearing rings. The cause of the out of tolerance condition could not be identified even though the records check was extremely detailed and the quality of the records was found to be acceptable. The licensee concluded that all available prudent action had been taken and therefore considers the issue closed. The inspector discussed the results of the engineering evaluation with the Manager of Engineering and the Lead Mechanical Engineer and had no further questions.

8. Design Changes and Modifications

a. Post Accident Sampling System (PASS). In order to meet the requirements of NUREG-0737, "TMI Action Plan Requirements", (Item II.B.3), a PASS was installed at Seabrook. During hot functional testing, difficulty was experienced in obtaining consistent sample results because of inadequate sample temperature control. As a result, design coordination report (DCR) 88-081 was generated to add an additional sample cooler to the system. The inspector reviewed DCR 88-081, as well as its DCR implementation plan, and made frequent field inspections of work in progress with special emphasis in the piping supports in the primary auxiliary building (PAB). Although the primary component cooling water lines which cool the new heat exchanger are not safety related, they are constructed to seismic criteria due to the design requirements of the PAB. The inspector had discussions with the Systems Engineering Supervisor concerning the identification of seismic/non-seismic class breaks in relation to licensee commitments documented in NRC:RI Inspection Report 50-443/86-14. Field inspection of piping and pipe supports revealed no violations of NRC requirements. Completion of pre-operational testing on the PASS requires the plant to be hot and is scheduled for accomplishment in the heatup prior to initial criticality. Actual testing of the PASS will be the subject of future NRC inspection to close out TMI Item II.B.3.

b. Secondary Component Cooling Water System

(1) Background. The secondary component cooling water (SCCW) system provides cooling water to non-safety related secondary loads in the turbine building. Typical cooling loads are the air compressors and condensate pump air and oil coolers. The system includes three 50% capacity each contrifugal pumps and two 100% capacity each large horizontal heat exchangers. The heat exchanger shells and tube sheets are clad with 90-10 copper nickel. All other carbon steel inner substances are lined with neoprene. The tubes are 90-10 copper nickel. These heat exhangers are cooled by a non-safety related leg of the service water (SW) system.

System inspections in 1986 and 1987 revealed significant tube corrosion due to low fluid velocities at low flow.

(2) Licensee Evaluation and Corrective Action. The NHY engineering department prepared engineering evaluation 88-04 in February, 1988 which proposed several solutions including installation of low flow heat exchangers for use during low heat load condition. This would allow the main heat exchangers to be placed in layup when not in use. Design coordination report (DCR) 88-088 was

initiated to add two additional low flow heat exchangers to the SW/SCCW systems. The heat exchangers were procured from existing stock as they are the original Unit 2 air removal heat exhangers. Once the new auxiliary heat exchangers (SCC-E-185A, B) are installed, the main heat exchangers (SCC-E-29A,B) may be removed and reworked or replaced with the Unit 2 coolers.

(3) Inspection. Despite the fact that this system is not safety related, this design change is of general NRC interest because of its relationship to heat exchanger degradation in primary systems as well as general workmanship and work control throughout the plant. The inspector reviewed engineering evaluation 88-04 and DCR 88-088 and made frequent inspections of the worksite.

On July 22, 1988, the inspector identified a section of drain piping which had been cut off the main SCCW line in preparation for weldolet installation. The line contained valve SCC-V-344 and a tubing connection for chemistry corrosion monitoring. The above valve was still caution tagged and the tubing fittings were identified as "Temporary Modification #10-Other". The inspector discussed this activity with the shift operators and Assistant Operations Manager. The inspector stated that removal of a caution tagged valve and temporarily modified assembly appeared to violate station procedures concerning equipment tagging and temporary modifications. Maintenance Procedure MA 4.2, Revision 7, "Equipment Tagging and Isolation" states, "No person shall physically remove any equipment that is tagged "DANGER/CAUTION". Maintenance Procedure MA 4.3, Revision 7, "Temporary Modifications" indicates that changes to temporary modifications be re-routed with appropriate notations, initialled and dated by all reviewers or a new temporary modification be prepared. In light of the non-safety related nature of this modification activity, no violation of NRC regulations existed, however, it is noted that corrective action for violation 87-20-01 that occurred in July, 1987, did not prevent recurrence of a similar although significantly less serious situation. It is also noted that another related occurrence was reported in station information report 87-108 in November, 1987.

(4) Conclusions. It appears that additional attention is warranted in this area especially with respect to temporary modification control. These modifications are clearly identified and removal or modification requires similar procedural controls as installation. This area will be the subject of continuing NRC inspection with respect to routine plant operations as well as readiness for initial criticality.

9. Allegation Review

As documented in NRC:RI inspection report 50-443/88-07, a written response on the licensee's investigation by its Employee Allegation Resolution (EAR) program personnel of five separate allegations was requested. By letter (NYN-88116) dated August 29, 1988, the licensee responded with the determination that the subject allegations are either inaccurate or relate to issues which were identified and dispositioned through internal quality programs. An enclosure to the licensee letter summarized each concern, its review and the licensee conclusions.

The inspector reviewed the above letter, its enclosure and additional EAR files and documonts relating to the investigation of each allegation. As was documented in the 88-07 inspection report, the inspector had previously conducted preliminary reviews of each allegation and performed both field inspection and records research where appropriate. During this inspection, the results of the licensee investigation were evaluated not only with regard to completeness and substantiating evidence, but also with respect to the inspection data independently collected and checked by the NRC. The following represent the conclusions reached for each of the five open allegations.

(a) Uncertified piping material supplied by Boston Pipe.

The inspector reviewed UE&C audit and nonconformance reports (NCR) covering the Boston Pipe & Fittings Co. of Cambridge, Massachusetts and the material supplied by this company for Seabrook Station. At least one of the NCR's documented the receipt of fittings on site without certification. Additionally, a Pullman Power Products NCR was found to have identified certain refrigeration system and support material which lacked the appropriate documentation.

Each case of a nonconforming condition resulting from incomplete certification appeared to be properly dispositioned with evidence of completed corrective action and reinspection by quality assurance (QA) personnel. The inspector also noted that contractor receiving inspection reports required and recorded document verification and traceability of the subject material as a requisite part of the inspection criteria. Thus, while the existence of the noted NCR's indicates that this allegation may have some basis in fact, the identification and disposition of these problems by the licensee also indicates that the receipt inspection process was working effectively. The inspector found no evidence to suggest uncertified material supplied by Boston Pipe had been installed in the plant.

(b) Uncertified electrical equipment supplied by Massachusetts Gas and Electric.

The inspector checked a sample of purchase orders from the Massachusetts Gas & Electric Light Supply Company, noting that most wire and circuit breakers were procured for general jobsite temporary power and lighting. Despite the nonsafety-related use of such material, at least one NCR was issued to document the lack of proper material certification. The inspector also noted that both UE&C and Fischbach, the electrical installation contractor, conducted receiving inspections which required document checks for certificates of compliance of the inspected material in accordance with specification requirements.

As similarly discussed with allegation (a) above, the fact that the licensee quality programs require receipt inspection checks for proper material certification and that NCR's have been issued when complete documentation was not available provides one measure of confirmation that the material installed meets fabrication specifications. Even in the case of a nonsafety supplier like Massachusetts Gas and Electric, evidence of such QA checks are available in licensee records. The regulatory requirements governing certificates of compliance, versus material certifications like mill test reports, are not in conflict with the licensee position that the manufacturer provides the requisite certifying documentation.

The inspector identified no information or facts that indicated that the Massachusetts Gas and Electric Light Supply Company had improperly certified material or that electrical components had been installed in the plant in applications for which they were unqualified.

(c) Acceptable level installation of the reactor coolant pumps.

The inspector reviewed Westinghouse and contractor records which substantiated the licensee conclusion documented in the NYN-88116 letter to the NRC. The Westinghouse Nuclear Service Division "Procedure for Setting of Major NSSS Components", Revision 2, issued in February, 1979, delineates the level criteria for the reactor coolant pumps. The inspector checked the Pullman-Higgins installation records for two reactor coolant pumps (RCP), including RCP-1C which represented the component originally questioned in the technical concern addressed in NRC:RI inspection report 50-443/87-07 (reference: UE&C engineering change authorization 08/1557A). For each pump, the inspector examined the "RCP-Volute Level Data Sheet - After Adjustment" and independently calculated the maximum level deviation. Although RCP-1C was slightly more off-level than RCP-1D, both pumps were measured to be level within the Westinghouse acceptance criteria.

Furthermore, the inspector noted that a Westinghouse memorandum issued in March, 1982 acknowledged the adjustment that was made to the RCP support and the resulting change in the RCP volute main flange differential elevation. Westinghouse engineers approved the change at that time. The inspector reviewed additional evaluation of the RCP level concerns by the licensee corporate engineering staff to include recent Westinghouse studies on RCP "tilt" conditions. These newer studies appear to indicate that the original Westinghouse level criteria, which the Seabrook RCP's meet, are conservative.

Therefore, with regard the question raised by this allegation, the inspector confirmed that the reactor coolant pumps have been installed and inspected to the Westinghouse design criteria and that acceptable level conditions for each RCP were verified after implementation of the engineering change which resulted in the repositioning of the base of one support.

(d) Weldolet in the emergency feedwater (EFW) pump room with wrong taper and counterfeit identification number.

Visual inspection of weldolets in the EFW pump room by an NRC inspector revealed no deficient or nonconforming conditions. The inspector also reviewed licensee nuclear quality group evaluations of elbolets and weldolets in the EFW pump room to ensure American Society of Mechanical Engineers (ASME) code compliance, acceptable markings and traceability and weld quality and taper. The licensee evaluation included documentation reviews, visual and ultrasonic thickness examinations, and inspection tracing of the scribed field marks to vendor documents which verify the quality and further traceability of the installed components. The licensee evaluation concluded that ASME code compliance had been confirmed.

The inspector checked the licensee's Thickness Data Sheet resulting from the ultrasonic testing field examinations and reviewed a sample of Dravo pipe fabrication sketches, establishing traceability of weldolet/elbolet field scribe marks to the heat number codes documented in the manufacturers' mill test reports.

The acceptability of field conditions for a number of components, which might represent the subject of the stated allegation, was verified by independent NRC and licensee inspections. The inspector concluded that this allegation could not be substantiated.

(e) Qualification of an Authorized Nuclear Inspector (ANI) trainee.

The inspector reviewed EAR records documenting licensee investigation of an allegation regarding the qualification of an ANI trainee and his authority to conduct independent inspections. As discussed in NRC:RI Inspection report 50-443/88-07, NRC inspection of a similar concern resulted in substantiation of certain of the facts, but in a conclusion that neither a noncompliance with the ASME Code, nor evidence of wrongdoing was identified.

The EAR records confirmed that the allegation previously reviewed by the licensee involved the same ANI trainee that was the subject of the allegation raised to the NRC. The licensee investigation concluded that during the period of time from May to December, 1985 when the subject ANI trainee was assigned to Seabrook, he performed assignments in accordance with his assigned training program. NRC inspector review of documents dating back to the 1985 time frame verified that qualified ANI's had evaluated and monitored the ANI trainee's training, progress and inspection work.

While the facts surrounding this allegation may be true, both NRC and licensee reviews of the stated concerns have identified no impropriety with respect to the certification or conduct of work on the subject ANI trainee while at Seabrook Station.

The five allegations listed as open in NRC:RI inspection report 50-443/88-07 were addressed by the licensee in the response letter, NYN-88116. Independent NRC inspection of these issues prior to raising the questions with the licensee had identified no hardware problems or quality concerns. Subsequent licensee EAR investigation of the allegations concluded that the allegations had no substantive merit. This inspection has included a review of those EAR investigation results and the process by which they were achieved. The inspector verified that licensee actions were comprehensive relative to the information provided in the allegations. The allegations generally either could not be substantiated, or represented issues with some factual basis, but with no adverse safety impact.

These five allegation issues are considered closed.

10. Training

a. General Employee Training

NRC:RI Inspection Report 50-443/87-16 discussed the topic of cheating on general employee training (GET) exams and the lack of written policy on cheating. During this inspection period this issue was re-visited. The inspector reviewed the GET examination cover sheet which listed instructions to be read aloud by the instructor prior to the examination. These instructions specifically addressed the steps to be taken should suspected cheating occur. Additionally, the inspector reviewed the draft of training procedure NT-7010, "Examination Administration and Integrity" which also formalized the station policy on cheating. The inspector determined that licensee follow-up actions this issue have been appropriate and had no further questions.

b. Operator Training

On July 20, 1988, the inspector discussed the recent Nuclear Management and Resources Council meeting on operator requalification testing, and the status of Institute of Nuclear Power Operations (INPO) accreditation with the Training Manager. In the area of INPO accreditation, the licensee stated that an INPO programmatic inspection is due to be performed in November of this year.

11. Electrical Configuration Control

As documented in NRC:RI inspection report 50-443/88-06, several engineering discrepancies and configuration control problems identified in the electrical area were resolved with the issuance of licensee engineering evaluation 88-011. NRC open item 87-24-01 was therefore closed.

During this inspection, the inspector identified certain field conditions for which questions of electrical detail and adequacy were raised. Specifically, electrical fire wrap requirements in accordance with engineering change authorization 03/11295G, the protection of spared cable terminations, the conformance of SF6 switching station breaker alignment to the plant technical specifications, and the status of missing condolet covers were all checked and found to be either acceptable or under work request control. Additionally, the inspector reviewed a quality assurance (QA) assessment (reference: QAIR 88-0597) of electrical design changes where the potential for interface problems from engineering to construction to startup/operational control appeared to be high. Only minor discrepancies were identified as a result of this assessment.

Another QA surveillance report 87-00983 was reviewed with regard to the implementation of work request activities in the cannibalization of Unit 2 equipment and spare part components, including electrical items. The Station Procurement and Materials Manual (Chapter 5.5) delineates criteria for the control and document tracking of the cannibalization process. The subject surveillance activity resulted in no adverse findings.

With respect to the licensee's programs of control for electrical work activities and its efforts to ensure electrical field configurations meet design requirements, the inspector noted comprehensive QA/QC department involvement. Based upon internal licensee assessments and NRC inspector spot-check and review, no generic problems or violations were identified.

12. Management Meetings

On August 17, 1988 a meeting was held in King of Prussia, Pennsylvania with NHY senior managers at the request of the NRC. The purpose of the meeting was to discuss licensee plans for heatup, initial criticality and low power testing. In addition, the current status of NRC Bulletin 88-05 was presented. Both parties agreed to meet again prior to initial criticality. A copy of the meeting handouts and attendance sheet is appended to this report as Attachments A and B, respectively.

At periodic intervals during the course of this inspection, meetings were held with plant managment to discuss the scope and findings of this inspection. An exit meeting was conducted on September 9, 1988 to discuss the inspection findings during the period. An additional meeting was held on September 22, 1988 between the Assistant Station Manager and the Senior Resident Inspector to discuss item status not covered in the previous exit meeting. During this inspection, the NRC inspector received no comments from the licensee that any of their inspection items or issues contained proprietary information. No written material was provided to the licensee during this inspection other than a listing of minor inspection deficiencies summarized in paragraph 3.a of this report.

ATTACHMENT A

NHY/NRC MEETING ON AUGUST 17, 1988
NRC REGION I, KING OF PRUSSIA, PENNSYLVANIA

Name	Title	Organization
W. Russell	Regional Administrator	NRC/RI
W. Kane	Director, Division of Reactor Projects	NRC/RI
W. Johnston	Director (Acting), Division Reactor Safety	NRC/RI
J. Wiggins	Chief, Projects Branch 3	NRC/RI
R. Gallo	Chief, Operations Branch	NRC/RI
D. Haverkamp	Chief, Reactor Projects Section 3C	NRC/RI
M. Shanbaky	Chief, Radiation Safety Section	NRC/RI
A. Cerne	Senior Resident Inspector	NRC/RI
D. Ruscitto	Resident Inspector	NRC/RI
D. Brinkman	Project Manager	NRC/NRR
R. Wessman	Director, Project Directorate I-3	NRC/NRR
F. Brown	President	NHY
G. Thomas	Vice President, Nuclear Production	NHY
T. Feigenbaum	Vice President, Engineering, Licensing and Quality Programs	NHY
D. Moody	Station Manager	NHY
J. Vargas	Manager of Engineering	NHY
J. Warnock	Nuclear Quality Manager	NHY
R. Sweeney	Washington Licensing Representative	NHY

ATTACHMENT B

New Hampshire Yankee

Presentation

to

USNRC Region 1

08-17-88

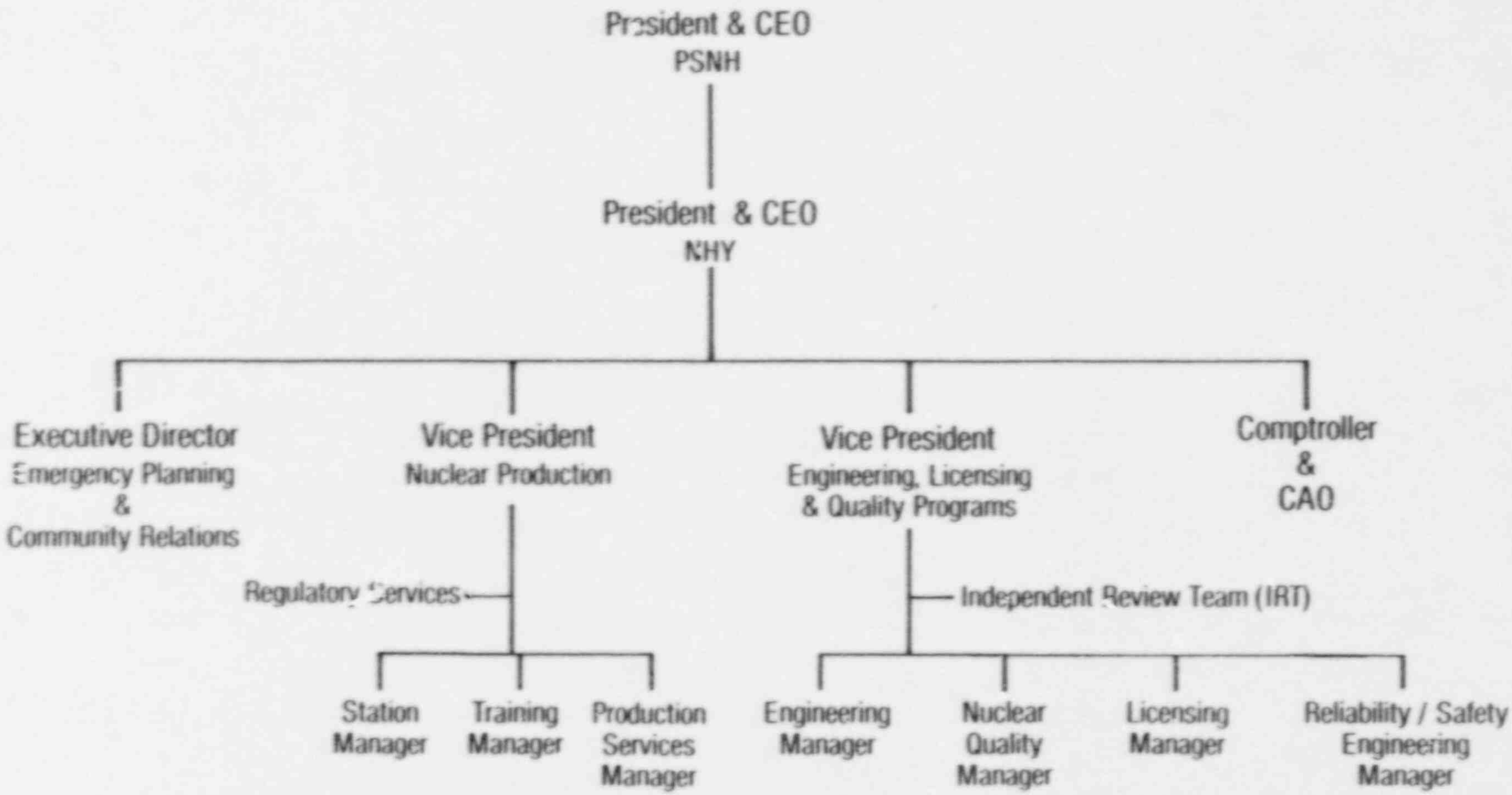
AGENDA

Introduction	G. S. Thomas
NHY Organization	E. A. Brown
Low Power Test Program	G. S. Thomas
Self-Assessment	T. C. Feigenbaum
Status of Bulletin 88-05	J. J. Warnock
Conclusion	G. S. Thomas

NEW HAMPSHIRE YANKEE
ORGANIZATION

E.A. Brown

NEW HAMPSHIRE YANKEE ORGANIZATION

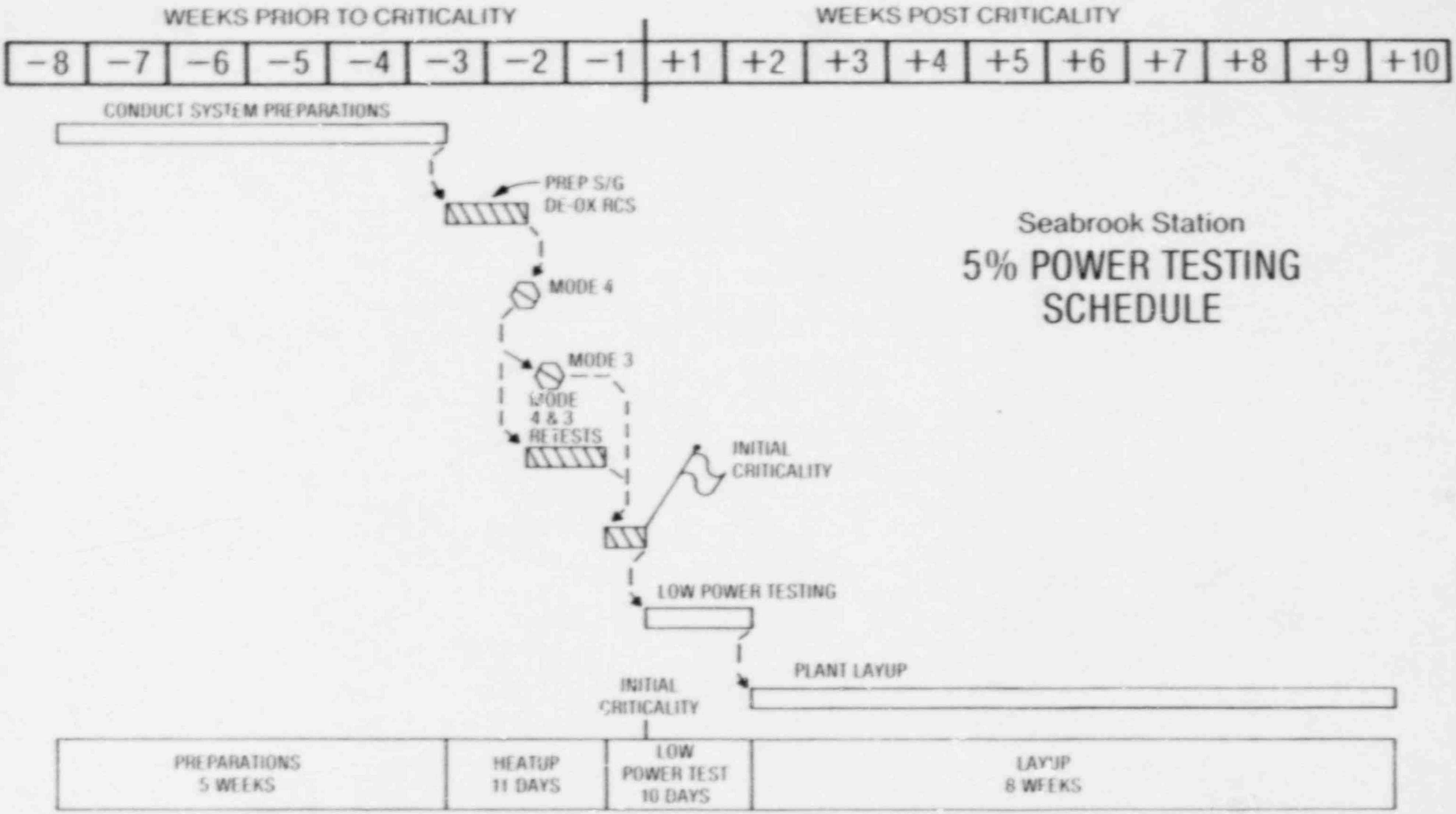


LOW POWER TEST PROGRAM

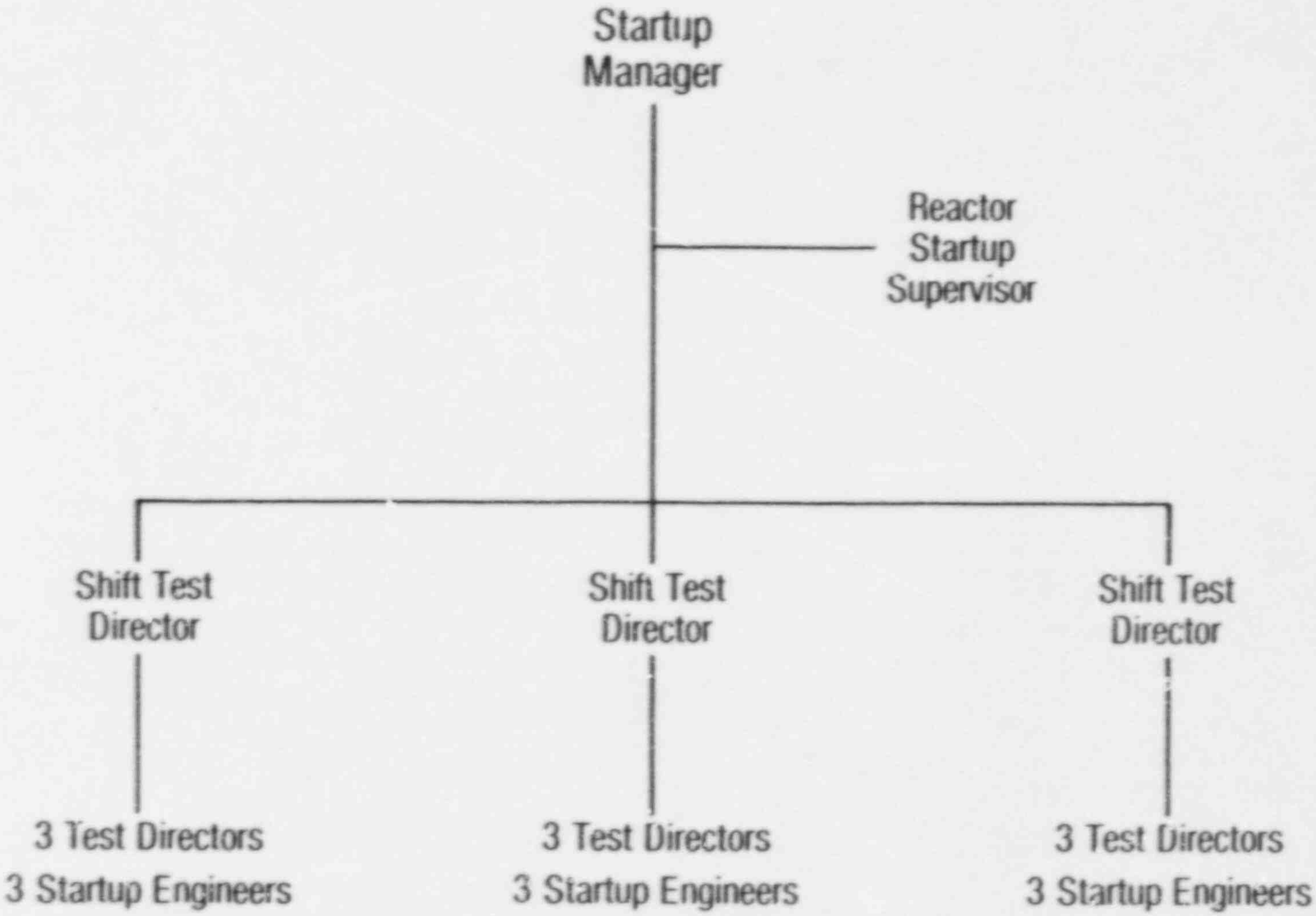
G.S. Thomas

Seabrook Station

5% POWER TESTING SCHEDULE



LOW POWER TEST PROGRAM
STARTUP ORGANIZATION



LOW POWER TEST PROGRAM
STARTUP ORGANIZATION

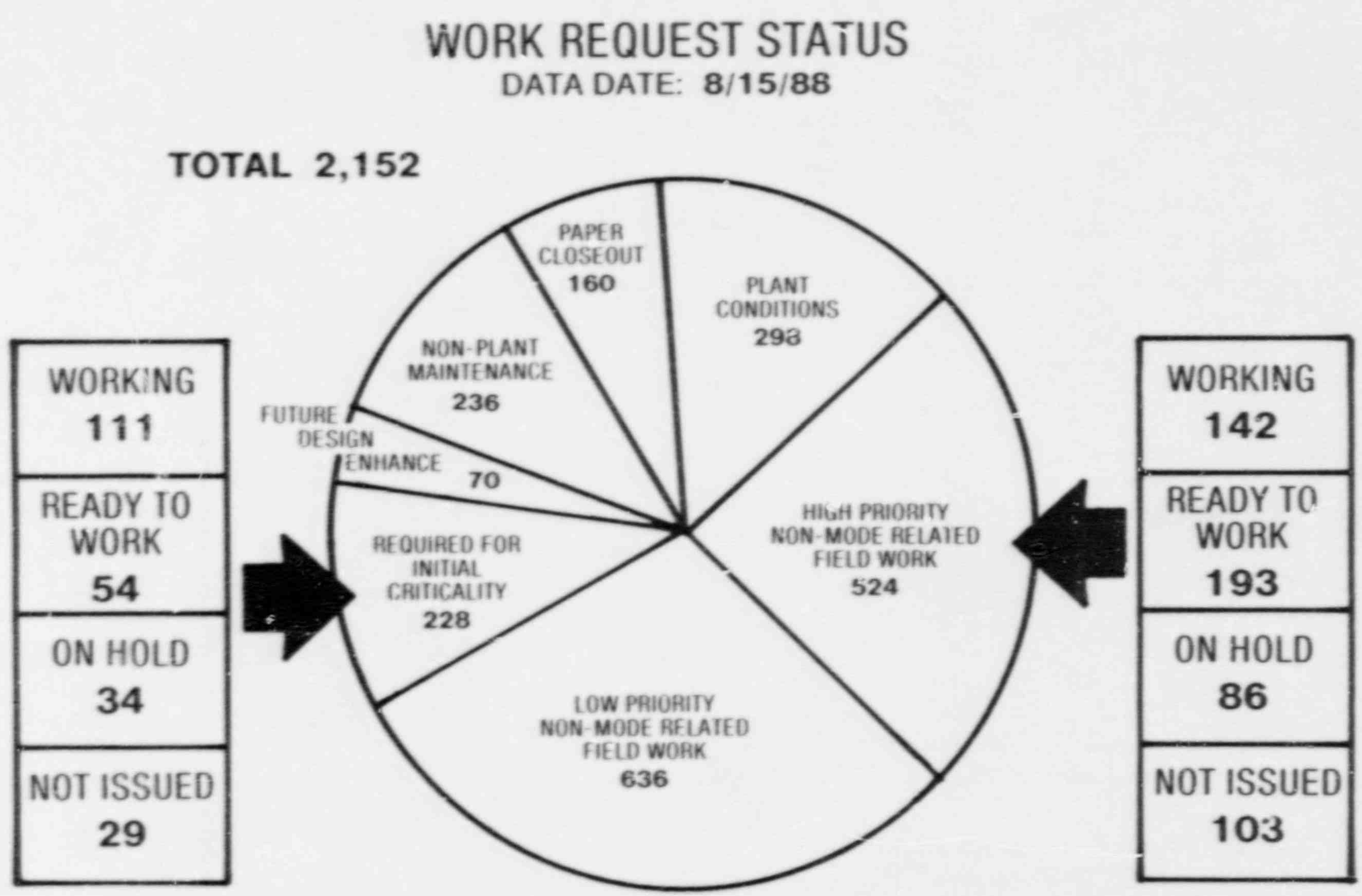
- Shift test directors and test directors (directors of test activities) will be qualified in accordance with the requirements of Reg Guide 1.8 as specified in the FSAR.

- All shift test directors and test directors have previously worked in the Seabrook preoperational and startup test programs.

- Test personnel will be formed from the following organizations:
  - Technical Support
  - Engineering
  - Operator Training
  - Regulatory Services
  - Yankee Atomic Electric Company
  - Westinghouse Electric Company

TECHNICAL SPECIFICATION SURVEILLANCE TESTS

- All local leak rate tests (Type B & C) have been reperformed
- Emergency diesel generator and engineered safety features actuation testing scheduled for the last two weeks in August
- Other surveillance testing has been incorporated into the schedule


WORK REQUEST STATUS
DATA DATE: 8/15/88
TOTAL 2,152
WORKING 111
READY TO WORK 54
ON HOLD 34
NOT ISSUED 29
FUTURE DESIGN ENHANCE 70
NON-PLANT MAINTENANCE 236
PAPER CLOSEOUT 160
PLANT CONDITIONS 298
HIGH PRIORITY NON-MODE RELATED FIELD WORK 524
LOW PRIORITY NON-MODE RELATED FIELD WORK 636
REQUIRED FOR INITIAL CRITICALITY 228
WORKING 142
READY TO WORK 193
ON HOLD 86
NOT ISSUED 103

PREVENTIVE MAINTENANCE
Data Date 8/2/88

ACTIVITIES PERFORMED

DEPARTMENT	1987	1988
Mechanical	1868	1144
Electrical	2834	1558
I&C	2634	1435
Utilities	536	207
TOTAL	7872	4344

MAN-HOURS CONSUMED

	1987	1988
TOTAL	47232	20267

PREVENTIVE MAINTENANCE
Data Date 8/2/88

RATIO OF PREVENTIVE TO TOTAL MAINTENANCE

	1987	1988
Activity	56%	51%
Man-hours	32%	22%

RADIATION PROTECTION

TIME PRIOR TO CRITICALITY	ACTIVITY
4 weeks	Start reissue of dosimetry to qualified rad workers
1 week	Establish Radiological Control Area for training
Just prior	Establish full Radiological Controlled Area (RCA)
Just prior	Implement full radiation protection program

OPERATIONS

Licensed Operators	23	SRO*
	9	RO
	32	Total
Staff Licenses	5	SRO–Operations
	9	SRO–Training
	14	Total

* Includes 16 STA - Qualified Operators

EMERGENCY PREPAREDNESS

ON-SITE EMERGENCY RESPONSE ORGANIZATION (ERO)

- Fully staffed and trained
- Demonstrated during 1986, 1987 and 1988 Graded Exercises
- Fully implemented since receipt of Zero Power License
- Meets requirements of proposed change to 10CFR 50.47(d)

SELF-ASSESSMENT
of the
LOW POWER TESTING EVOLUTION

T.C. Feigenbaum

PURPOSE:

To perform a self-assessment of the preparation for and the conduct of activities associated with the Seabrook Station low power testing evolution in order to assess the readiness and effectiveness of personnel, programs and equipment and to identify areas requiring immediate or long term management attention.

SCOPE:

The scope of the self-assessment effort will include, as a minimum, the following topical areas:

1. Plant operations
2. Radiological controls
3. Maintenance
4. Surveillance and testing
5. Safety assessment / Quality verification
6. Control room operations
7. Effectiveness of internal problem identification and resolution
8. Plant chemistry and health physics

FOCUS:

Within the above topical areas, the self-assessment effort will focus on the following organizational conduct and activities:

1. Organizational interfaces and management effectiveness
2. Plant configuration control
3. Program/procedural adequacy and compliance
4. Communications and teamwork
5. Operational Quality Assurance effectiveness
6. Timeliness and adequacy of support of Station activities
7. Training program adequacy and effectiveness
8. Timeliness and adequacy of corrective action reporting and follow-through
9. Adequacy of design based on Low Power Test Program elements

SELF-ASSESSMENT TEAM ORGANIZATION:

MANAGEMENT OVERSIGHT COMMITTEE

- E.A. Brown — President and CEO
- G.S. Thomas — V.P. Nuclear Production
- T.C. Feigenbaum — V.P. Engineering, Licensing and Quality Programs
- D.E. Moody — Station Manager

SELF-ASSESSMENT TEAM MANAGER

- N.A. Pillsbury — Independent Review Team Manager

SELF-ASSESSMENT TEAM MEMBERS*

AREAS of EXPERIENCE:

- Operations
- Maintenance
- Chemistry/Health Physics
- Training
- Engineering/Technical Support
- QA/QC
- Independent Safety Engineering Group

* Approximately 30% of each work week to be dedicated to evaluation activities

NRC INTERFACE:

- Periodic updates by Team Manager and members of the Management Oversight Committee (bi-weekly suggested)

- Final report available to NRC Resident and Region 1 office (approximately 6 weeks after completion of Low Power Testing)

- Normal daily contact with NRC Resident as required

- NHY/NRC critique of performance following completion of major activities

SCHEDULE of ACTIVITIES:

WEEK	LOW POWER TESTING	SELF-ASSESSMENT	MGMT. OVERSIGHT COMMITTEE
1	Preparation		
2	Preparation	Team Preparation	Team & Mgmt. Briefing
3	Preparation	Self-Assessment Team Start	
4	Preparation	Self-Assessment	Team Status Report
5	Preparation	Self-Assessment	
6	Heatup	Self-Assessment	Team Status Report
7	Heatup	Self-Assessment	Team Status Report – Precritical Concurrences Status –
8	Low Power Tests	Self-Assessment	Team Status Report
9	Low Power Tests/Cooldown	Self-Assessment	
10	Layup	Self-Assessment	Team Status Report
11	Layup	Self-Assessment	
12	Layup	Self-Assessment	Team Status Report
13	Layup	Self-Assessment End	
14	Layup	Draft Report	D/R Internal Distribution
15	Layup		
16	Layup	Issue Final Report	Team & Mgmt. Debriefing

STATUS of BULLETIN 88-05

J.J. Warnock

BULLETIN 88-05 SUMMARY

- Falsified CMTRs — WJM/PSI/Chews Landing

- Identify Installed Fittings and Flanges (F/F) and other material and test

- Engineering evaluation for F/F as required

- Written report to NRC

SEABROOK APPROACH

- Documentation review
- Field walkdowns
- Procedures developed
- Testing of installed F/Fs
- Laboratory testing of selected F/Fs
- NUMARC/EPRI support
- Engineering Evaluation
- Additional confirmations
  - — DRAVO
  - — Radnor Alloys
  - — Other suppliers
  - — Continued NUMARC support

BULLETIN 88-05 RESULTS

DOCUMENTATION REVIEW

- Complete
- 358 WJM flanges/fittings installed; 12 F/F vendor markings not available (B31.1 only)
- 13 S/R ASME systems affected; 1 S/R B31.1 systems affected
- Predominently carbon steel (5 stainless steel flanges)

TEST RESULTS

- 368 tested
- 30 requiring engineering evaluation
- No replacement anticipated

OTHER

- DRAVO review consistent
- Supplier responses