U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-443/86-34

Docket No. 50-443

License No. CPPR-135 Priority -- Category A/B

Licensee: Public Service Company of New Hampshire
1000 Elm Street
Manchester, New Hampshire 03105

Facility Name: Seabrook Station, Unit 1

Inspection at: Seabrook, New Hampshire

Inspection conducted: May 24 - July 7, 1986

Inspectors: A. C. Cerne, Sr. Resident Inspector
D. G. Ruscitto, Resident Inspector
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Approved by: Harold Eichenholz for 7/24/86
T. C. Elsasser, Chief, Reactor Projects Section 3C Date

Summary: Inspection on May 24 - July 7, 1986 (Report No. 50-443/86-34)

Areas Inspected: Routine inspection by four resident inspectors and four region-based inspectors of work activities, procedures, and records relative to building turnover preparations; design and construction of selected portions of the safety injection, chemical & volume control, and control building air handling systems; and the follow-up of licensee scheduled activities and controls for TMI Action Plan items. The inspectors also reviewed licensee action on previously identified items, including 10 CFR 50.55(e) reports, and performed plant inspection-tours. The inspection involved 333 inspection-hours by eight NRC inspectors.

Results: Review of selected TMI action plan items, construction deficiency reports and licensee response to IE Bulletins, Circulars and Information Notices revealed no safety concerns. No violations were identified.

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DETAILS

1. Persons Contacted

J. DeVincentis, Director of Engineering (NHY)
G. F. McDonald, Construction QA Manager (YAEC)
D. E. Moody, Station Manager (NHY)
R. E. Guillette, Assistant Construction QA Manager (YAEC)
D. A. Maidrand, Assistant Project Manager (YAEC)
D. G. McLain, Startup Test Group Manager (NHY)

Interviews and discussions with other members of the licensee and contractors management and staff were also conducted relative to the inspection on items documented in this report.

2. Plant Inspection Tours

The inspectors observed work activities in-progress, completed work and plant status in several areas during general inspections of the plant. They examined work for any obvious defects or noncomplaince with regulatory requirements or license conditions. Particular note was taken of the presence of quality control inspectors and quality control evidence such as inspection records, material identification, nonconforming material identification, housekeeping and equipment preservation. The inspectors interviewed craft personnel, supervision, and quality inspection personnel as such personnel were available in the work areas.

The inspector reviewed a recent licensee submittal to NRR which stated their intention to change the pressure for determining containment integrity from 48.7 psig to 49.6 psig. He confirmed that this increased pressure was exceeded during the Containment Integrated Leak Rate Test (CILRT) and that the leak test data obtained was still acceptable even though the test acceptance criteria were based on the original containment pressure. Thus, no retest of the CILRT is mandated by the proposed change.

The fire door on the (-)26'-0" elevation of the primary auxiliary building (PAB) and three bullet-proof doors restricting access to the containment enclosure air handling area are containment enclosure boundary doors. The containment enclosure building is designed to operate at a (-)0.25" water vacuum and is capable of withstanding the rupture of the Chemical and Volume Control System (CS) letdown line which passes through it. However, no credit is taken for the integrity of the containment enclosure boundary (including the enclosure doors) in mitigating the radiological consequences of a rupture of the letdown line. The inspector reviewed receipt inspection report (RIR)-7913 for one of the bullet-proof doors (EM-409). No problems with the procurement of the door were identified although the purchase order had no special requirements governing the doors use as a containment enclosure boundary. However, due to the low negative pressure that the fire door and the bullet-proof doors will be subjected to, they appear to be structurally adequate to serve as part of the containment enclosure boundary. Their structural adequacy will be verified by conduct of the containment Enclosure Air Handling (EAH) preoperational test (1-PT-23) when the containment enclosure will be drawn down to a (-)0.25" water vacuum. The inspector also reviewed FSAR Section 6.2.3 and 15.6.2 relative to this issue. No problems were identified.

Preoperational test (1-PT-43), entitled "Reactor Post Hot Functional Inspection" required the removal of the steam generator (S/G) primary side manways for visual inspection of the S/G interior. When the licensee tried to remove the bolts holding the steam generator manways, 20 of the bolts were found to be seized. The stuck bolts were removed per the disposition of nonconformance report (NCR) 82/1172D which included, in some cases, drilling the bolts out. Thread damage to the bolted holes occurred as a result of the drilling.

The inspector reviewed the disposition to NCR 82/1172D. He also reviewed an engineering evaluation performed by Westinghouse (reference: Westinghouse Service Technology Division Report, 4.6.2-6216, dated March 17, 1986) on the probable causes for the large number of seized bolts. The inspector questioned the Startup Test Department Primary Side Lead Engineer as to the corrective actions that will be taken to prevent a recurrence of this problem. He stated that the bolts would be reinstalled using Felpro N-1000 (a copper-based lubricant) which was originally used. Experience at other nuclear stations has shown a significant reduction in bolt seizures using this lubricant. He further stated that the manway cover seating surfaces were machined to a finer finish to aid in preventing the bolt seizures and to remove the galling that existed when all the bolts were finally removed.

The inspector reviewed controlled memos 866 and 878 which evaluated the potential reportability of the bolt seizures per 10CFR50.55(e). The problem was determined to be non-reportable. With regard to the above inspection points, no violations were identified.

The inspector noted that many of the threaded drain caps on safety-related systems were missing. Specifically, the drain caps for lines containing drain valves CBS-V-88, CBS-V-89, CBS-V127, SI-V-235 and SI-V-236 were all missing. While the drain caps are non-safety related, they are shown on the Piping and Instrument Diagrams (P&IDs). The inspector questioned the Operations Manager as to when these caps would be installed and how their installation is being tracked. He stated that their installation is not being tracked and committed to either have the drain caps on all safety-related systems installed or investigate with engineering personnel the absolute need for the drain caps to determine if their optional use is acceptable. The inspector indicated that drain cap installation would be randomly checked on future plant inspection-tours. No violations were identified.

IE Information Notices 82-51 and 84-19 identify a generic problem of radiation overexposures at pressurized water reactors (PWR) caused by entries into locked high radiation areas containing the incore neutron monitoring detectors and thimble tubes. Inspection report 50-443/85-17 documented the inquiries made by the inspector into the measures that would be taken by the licensee to prevent such an occurrence at Seabrook.

At that time, all the health physics (HP) administrative controls governing entry into the reactor cavity (where the incore neutron monitoring thimble tubes are located) and containment (during power operation and immediately after shutdown) were not yet finalized.

The licensee plans to institute a two keylock system on both entrances to the reactor cavity area. One key is to be controlled by HP supervision. The other is to be held by the shift supervisor (SS). Entry into the reactor cavity and the containment during operation and initially following a shutdown will not be permitted except under unusual circumstances unless the incore neutron monitoring fission detectors are in their appropriate storage position or at the bottom of the core and their control panel is tagged out. A unique radiation work permit (RWP) and continuous HP coverage will also be required.

The inspector reviewed HP procedures HP0960.02 entitled "Radiological Requirements for Entry Beneath the Reactor Vessel" and HP 0960.03 entitled "Radiological Requirements for Containment Entries".

He also reviewed the instrumentation and control (I&C) procedures governing the withdrawal and insertion of the incore thimble tubes (reference: IS1690.815 and IS1690.816). For human factors reasons, the licensee chose to place special warnings and administrative controls on reactor cavity entries when the thimble tubes are removed from the reactor core in the I&C procedures versus the applicable HP procedure.

The inspector questioned the HP Department Supervisor as to when the entrances to the reactor cavity would be enclosed and locked as required. The HP supervisor stated that the entries would be enclosed in the near future and that their installation was being tracked by Incomplete Items List (IIL) number F1C1-0019. He had no further questions. No violations were identifed.

3. Licensee Action on Previously Identified Items

a. (Closed) Unresolved Item (85-31-01): Hot Function Testing (HFT) Activities. This item relates to those activities associated with HFT and contains five separate concerns. These concerns are addressed below in the same order as listed in NRC Region I Inspection Report (IR) 50-443/85-31.

(1) Snubber Leakage: As a result of HFT inspection, seventeen of twenty steam generator upper lateral support snubbers appeared to be leaking and had fluid levels less than that prescribed by technical manual requirements. Leakage was mainly from the rod extension seals and is not a unique characteristic of hydraulic snubbers. The inspector verified that the leaking snubbers were refurbished and properly performance tested prior to reinstallation. These snubbers will be retested in accordance with startup test 1-ST-52, entitled "Power Ascension Thermal Expansion Test."

(2) Maximum Excess Letdown Flow: The inspector reviewed the licensee's engineering justification for acceptance of a 53 gallon per minute (GPM) flowrate through hand control valve (HCV)123 which was obtained during HFT. Westinghouse design criteria for the maximum excess letdown flowrate is 25 GPM. This is accomplished by regulating flow through HCV-123 to maintain proper pressure and temperature indications downstream of the excess letdown heat exchanger. Failure of HCV-123 in the open position could result in combined letdown flow beyond the capacity of one charging pump. The inspector reviewed Westinghouse letter NAH-300 which provides engineering justification for a 53 GPM flowrate. The inspector determined, based on this letter, that no additional safety concerns exist based upon increased flow through HCV-123 since operator actions can reduce total letdown flow within the capacity of one charging pump using safety related systems. In addition, an engineering evaluation shows that relief valves on the seal water return line inside containment will not lift with a 53 GPM flowrate.

(3) Reactor Coolant System (RCS) Vibration with Safety Injection (SI) Hot Leg Flow. During safety injection pump (SIP) runs with flow to loops 1 & 4 hot legs, accomplished in conjunction with 1-PT-9 testing, vibration levels near an residual heat removal (RHR) suction line off hot leg 1 (RC-V-22) were detected. Reestablishment of this condition was performed to provide addition data. The inspector reviewed the evaluation of data obtained from the vibration measurements to determine if the maximum measured displacemnt of 13.9 mils was acceptable. Ebasco Services, who were contracted by the licensee to provide allowable vibration levels covered by NRC Regulatory Guide 1.68, indicated that a displacement of 35 mils is acceptable at the location of the noted vibration. In addition, ANSI/ASME OM-3-1982, "Requirements for Pre-Operational and Initial Startup Vibration Testing for Nuclear Power Plant Piping Systems" shows this vibration level to be acceptable. ANSI/ASME OM-3-1982 defines the acceptable criteria for vibration for the FSAR.

(4) Effect of High Basin Salinity on Cooling Tower Performance. A concern was raised during HFT that make-up water, supplied to the cooling tower in the case when a seismic event rendered other sources unusable, may affect thermal performance of the tower. The licensee's FSAR states that make-up water could be obtained from the Browns River which had a salinity reading of 31,130 PPM on November 13, 1985. The inspector reviewed "Cooling Tower Operating Guidelines" provided by the vendor and determined that although increasing salinity affects thermal performance, a maximum cooling tower water salinity level of 72,000 PPM allows the tower to meet thermal performance during safeguards operations. The Brown River is therefore a viable source of make-up water.

(5) Following improper control board temperature indication on TI-126 during hot functional testing, loose terminations were found on an instrument cable at an electrical penetration. The initial troubleshooting was conducted under work request (WR) CS-1579. As a result of these findings, the Quality Assurance Group issued surveillance report (SR) Y-258 to determine the cause. The Startup Test Department (STD) initiated an inspection of 100 terminals in five different electrical penetrations. This inspection (WR PEN-0001) did not reveal any discrepancies, however unrelated work in another penetration did identify some loose teminations. As a result, three more WR were written to conduct 100% inspection of all instrumentation and electrical terminations both inside and outside containment in each penetration. WRs PEN-0002, PEN-0010 and PEN-0011 revealed 370 loose terminations.

The inspector noted that UE&C Field Electrical Procedure FEP-505 and Fischbach Boulos Manzi Field Electrical Construction Procedure FECP-505, both entitled "Constructor Procedures for Installation of Cable Terminations" only require that the terminal be "tightened" and Quality Control Procedure QCP-505 of the same title only calls for verifying that terminations are tight. When questioned, the various electrical technicians stated that few if any of the loose connections could be taken up much more than 1/4 of one turn. Additionally, the total number of terminations inspected was several thousand so the percentage of those which were found slightly loose was very small.

The inspector determined that the licensee's initial action was appropriate based on the potential problem identified by the initial loose terminal pair. Subsequent inspection has proven that the CS pair to be an isolated incident where the degree of looseness effected instrument indication. IR 85-31 indicated that the potential source of the problem was rework performed on cable H44-TF9. Subsequent license investigation indicates that the leads lifted for this work were not the same ones which were originally found loose. The inspector noted startup QC involvement in each of the final inspections.

With regard to all five subitems above, the inspector had no further questions. This item is closed.

b. (Closed) Violation (443/86-12-01): Lack of Air Paths Connecting the Containment Enclosure to the Air Space Behind the Main Steam/Feedwater Pipe Chase Pressure Seal Plates.

The licensee's response to the violation (reference: SBN-1078, dated May 30,1986) stated that the cause of the violation was an oversight between interfacing engineering disciplines in not providing all of the required information in the installation details. To correct the existing hardware deficiency, engineering change authorization (ECA) 01/807485B was issued to core bore through the containment enclosure wall to connect the air space behind each of the pressure seal plates with the containment enclosure atmosphere. An engineering evaluation was also done to determine if other such unique containment enclosure volumes existed. None were found to exist. Thus the violation was considered an isolated occurrence.

To preclude recurrence, a management directive was issued to emphasize the importance of the interdisciplinary review.

The inspector physically verified that the core bore connecting the air space behind the east main steam/feedwater pressure seal plate to the containment enclosure was bored. He reviewed the above described directive along with the engineering evaluation of the containment enclosure. No problems were found. The inspector concurs with the licensee's determination that this violation was an isolated occurrence and is closed.

4. Licensee Action on Construction Deficiency Reports

a. (Closed) Construction Deficiency Report (CDR) 83-00-14: Service Air System NNS Supports. The CDR described a deficiency regarding supports for the service air system and other systems which carried a Non-Nuclear Safety (NNS) designation instead of the correct Non-Nuclear Seismic (NNS-I) designation, which should have been assigned due to the system's proximity to safety-related equipment. In addition, a portion of these supports, plus approximately 700 supports properly classified as NNS-I, did not have retrievable formal calculations. This deficiency and completion of corrective actions were described in licensee letters to NRC Region I dated August 12, 1983, September 28, 1983, November 10, 1983, December 2, 1983, June 19, 1984 and June 5, 1986. The licensee's contractor identified 1,362 pipe support points that required an evaluation. The changes resulting from those evaluations included 212 pipe supports that required a reclassification from NNS to NNS-I, and 60 pipe supports (some installed) that required modificiations to meet NSS-I design criteria.

The inspector reviewed various licensee/architect-engineer correspondence, licensee internal memoranda and design review documents related to the licensee's evaluations and modifications of NNS supports. The licensee's contractor has certified. and the licensee's quality assurance department has verified completion of required corrective actions for Seabrook Unit 1. The pipe support points identified for evaluation were listed in a UE&C memorandum, serial CM#00899, dated May 22,1986. The inspector had no further questions regarding CDR 83-00-14. This item is closed.

b. (Closed) Construction Deficiency Report 84-00-05: Fuel Transfer Tube Seam Closure Welds. The CDR described a condition reportable under 10 CFR 50.55(e) regarding the potential for overstressing welds that join the expansion joint assemblies to the fuel transfer tube. Based on the licensee's analysis of the fillet closure welds for the fuel transfer tube at postulated design basis conditions, the fillet welds on both the containment and fuel building sides had calculated stresses in excess of applicable code limits. The licensee's corrective action was to rework the fuel transfer tube welds to satisfy ASME III, Code Class 2, criteria, as described in their letters to NRC Region I dated April 27, 1984, June 19, 1984 and May 20, 1986. Specifically, leak test channels and stiffener plates were welded over the seam closure welds. The inspector reviewed engineering design change and modification installation documents related to the licensee's corrective actions. These documents included ECA 08/2250B and 08/2259D, fuel transfer tube drawings, process sheets nonconformance reports and various licensee internal memoranda. The inspector had no further questions regarding CDR 84-00-05. This item is closed.

c. (Closed) Construction Deficiency Report 84-00-08: Service Water Valve Bracket Welds. Thirty-six valve operator mounting brackets supplied by Fisher Controls Company on butterfly valves installed in the Unit 1 service water system (SWS) were found to have welds not conforming to AWS standards for size, voids, undercut, fusion and corner fill. Failure of these welds may have reduced the operability of the valves during accident conditions. The licensee submitted his final 10 CFR 50.55(e) report on May 20,1986 (letter SBN-1060). The inspector reviewed documentation of the licensee's corrective actions and inspected a sample of the brackets that were repaired as part of those corrective actions. Documents reviewed included NCR 82-194, 82-190, 2320, 2545, 73-5309, 74-2544 and 94-959B; nonconformance review board response forms (NRBRF) associated with these NCRs and material request orders 24873 and 24551.

Fabrication welds on brackets supporting valve operators on the following service water valves were visually inspected for size, overlap, profile, undercut, length and location: 1-SW-V02, V-22, V-29, V-31, V-34, V-20, V-69 and V-68. No discrepancies were noted.

The inspector also reviewed a compilation from the Automated Valve List of all other Fisher Controls (Specification 248-05) valves installed in Unit 1. There were 60 valves listed, components of chlorination, circulating water, floor drains, condenser air evacuation and screen wash systems, none of which are safety related.

The inspector determined that all substandardly welded brackets on safety related valves supplied by Fisher Controls in Unit 1 have been adequately repaired. This item is closed.

d. (Closed) Construction Deficiency Report 84-00-13: Control Circuit Cable Lengths. This CDR described a deficiency that occcurred during the design process of control circuits for various devices such as relays, solenoids, etc. Specifically, portions of the circuit were not considered in all cases when evaluating the total circuit length for voltage drop calculations, and the inadequate voltage at the terminals of such devices due to increased voltage drop could potentially prevent the device from performing its intended safety function. This deficiency and completion of corrective actions were described in licensee letters to NRC Region I dated August 30, 1984, December 6, 1984, June 17, 1985 and May 28, 1986. The licensee contractor's preliminary review of all control circuits identified 358 potential problem circuits, and a detailed review and voltage drop calculation for each potential problem circuit identified 142 circuits, both safety-related and non-safety-related that required modification to insure adequate terminal voltage of the various devices. These modifications involved rewiring with larger conducters, the development and use of a new relay coil with a lower minimum operating voltage and reconfiguration of some circuits.

The inspector reviewed various licensee/architect-engineer correspondence, licensee internal memoranda and design study documents and calculations and engineering change authorizations that were related to the licensee's study of potential control circuit cable problems and implementation of modifications. The Startup Test Department has certified and the Quality Assurance Department has verified completion of required corrective actions for Seabrook Unit No.1. The inspector had no further questions concerning CDR 84-00-13. This item is closed.

e. (Closed) Construction Deficiency Report 85-00-04: Defective Rubber Lining in Airflex Instrument Hose. Improper site fabrication of Parker 204 Ethylene Propylene air hoses used to connect solenoid-operated air valves to safety-related air-operated valves resulted in partial blockage of the hoses by damaged inner linings. Such damage may have retarded or blocked safety-related valve motion during accident conditions. The applicant made his final report on this deficiency on May 20, 1986. The inspector reviewed documentation of the development and implementation of corrective actions leading to the elimination of the deficiency. The review included the disposition to NCR 82-397; ECAs 05/114674A and 74/106974A; WR RH-0248, CBS-0433, RH-0270, RC-1095, NG-0028 and CS-1072; In Process Inspection Reports 86-IR-5197, I538, I01118 and I0903; Memoranda STD86-205 and Q1.1.Y/YFQA-1095; and letters SBU-99036 and SBN-1058. The inspector noted that I&C personnel were trained to inspect and repair Parker flex hoses in accordance with vendor guidelines and the disposition to NCR 82-397.

Based on the above documentation, The inspector determined that the applicant has removed defective safety related valve air hoses and replaced them either with hoses that were properly fabricated or with instrument tubing if the application did not require a hose. This item is closed.

f. (Closed) Construction Deficiency Report 85-00-10: Emergency Feedwater System High Flow Isolation Logic Seal-In. The CDR described a design deficiency in the isolation logic for emergency feedwater (EFW) flow to a steam generator in the event of a feedwater or main steam line break. In addition to causing isolation of EFW to the affected S/G, the isolation logic includes an interlock to block a concurrent high flow isolation of the nonfaulted steam generators. However, the final design did not seal-in the high flow logic and could have resulted in the sequential isolation of EFW to all steam generators. The licensee's corrective action was to provide a seal-in for the high flow signal so that the interlock will remain in place until manually reset from the control room. In addition, the high flow isolation/logic seal-in was alarmed. This CDR and the completion of corrective actions were described in licensee letters to NRC Region I dated August 1,1985 and Febraury 18,1986. The inspector reviewed ECAs 99/107679A and 99/108705B and licensee startup test department and quality assurance department memoranda that certified and verified satisfactory completion of modications to EFW controls. The inspector had no further questions concerning CDR 85-00-10. This item is closed.

g. (Closed) Construction Deficiency Report 85-00-17: Gould-Supplied Motor Control Starters. The CDR describes a deficiency regarding motor control starters supplied by Gould, Inc. (now Telemecanique). Specifically, the deficiency involved the potential for motor control starters, unitized type NEMA Size 1 and 2, to experience binding or seizure of the contact carrier assembly if the space between the contact carrier post and support housing is less than 0.010 inch. A similar deficiency was identified in 1982 and determined to be reportable per 10 CFR 50.55(e). In addition, Gould Inc. had previously issued a 10CFR21 report to the NRC regarding this potential deficiency. Corrective action then taken by the licensee involved inspection and modification of starters identified by Gould, Inc. as having the subject deficiency. However, recent licensee testing identified starters with less than the specified clearance. The manufacturer's explanation for the recurrence of this problem was that previous corrective action did not emphasize the importance of washers that should be installed under the contact carrier cover. Without these washers the cover screws can be overtightened.

The licensee's corrective actions involved a 100% reinspection of the subject starters which are utilized in safety-related Class 1E applications to assure that washers are installed under the contact carrier cover and that adequate clearance exists between the contact carrier post and support housing. This CDR was described in licensee letters to the NRC dated November 18, 1985, January 24, 1986, March 31, 1986 and May 28, 1986. The inspector reviewed licensee/vendor/architect-engineer correspondence, licensee internal memoranda, NCRs 74-1404 and 82-766D and various work authorization documents and quality control inspection reports. The licensee's startup test department has certified and the quality assurance department has verified that corrective actions have been completed regarding the Gould starters binding problem. The inspector had no further questions concerning CDR 85-00-17. This item is closed.

h. (Closed) Construction Deficiency Report 86-00-04: Improper Cable Lug Connections in 25kVA Inverters. The CDR described a deficiency regarding potential malfunction of 25kVA inverters manufactured by Elgar Corporation due to improper stacking of cable lugs on the back of the battery input breaker assembly and also on the AC input power supply connection. The electrical connections to the fuse blocks were made via nuts and studs instead of direct connection of the cable lug to the bus which connects to the fuse block. The licensee identified this deficiency during their inspection of the inverters, based on the results of an NRC inspection of Elgar Corporation inverters at River Bend. The licensee corrective actions were to determinate and properly reterminate the affected cable lugs in a manner to preclude abnormal heating due to high resistance.

This deficiency and completion of corrective actions were described in licensee letters to NRC Region I dated May 9, 1986 and June 3, 1986. The inspector reviewed licensee NCRs 86/0004 and 95/0012A, WRs 86WM00330, 86WM00626, 86WM00627, and 86W01756 and other internal licensee memoranda/documents regarding correction of this deficiency. The licensee quality assurance (QA) department has verified completion of work for Seabrook Unit 1 inverters. The inspector had no further questions regarding CDR 86-00-04. This item is closed.

5. Licensee Action on IE Bulletins and Circulars

a. (Closed) IE Circular (IEC 78-05): Inadvertent Safety Injection During Cooldown. This circular described an SI transient that occurred at an operating four-loop Westinghouse PWR during a normal plant cooldown. The SI occurred due to low pressure in the steam line from one of the steam generators. The contributing factors included: (1) operation of a single reactor coolant pump instead of all reactor coolant pumps, (2) lack of pressure recording instruments for the steam lines, and (3) use of atmospheric relief valves instead of steam dump valves to cooldown. The inspector reviewed the licensee's assessment of actions which would minimize the frequency of inadvertent SI during cooldown, as described in an internal memorandum, dated June 5, 1986. The specific cause of the inadvertent SI described in IEC 78-05 is not applicable to Seabrook, which does not have a safety injection signal from steam line differential pressure. The three SI initiation signals applicable to Seabrook, were assessed as not being anticipated causes of an inadvertent SI during normal plant cooldown. The licensee also noted in their memorandum that the Seabrook design includes a recorder for pressure in each steam line. The inspector also reviewed draft changes to procedure OS1000.04, entitled "Plant Cooldown from Hot Standby to Cold Shutdown", initiated on May 29, 1986, regarding the use of condenser steam dumps whenever possible for cooldown. The inspector had no further questions concerning IE Circular 78-05. This item is closed.

b. (Closed) IE Circular (IEC 78-19): Manual Override (Bypass) of Safety System Actuation Signals. This circular described two events that occurred at operating power reactors which raised questions about safety system circuit designs which incorporate manual override/bypass features. The events directly related to the practice of containment purging during normal plant operation by manually overriding containment isolation signals. In those instances the automatic isolation function of the purge system containment isolation valves was unintentionally made inoperable and this condition was neither continuously indicated in the control room nor known to the plant operators. The circular recommended that holders of power reactor construction permits review the design of all safety actuation signal circuits which incorporate a manual override feature to ensure that overriding of one safety actuation signal, does not also cause the bypass of any other safety actuation signal, that sufficient physical features are provided to facilitate adequate administrative controls, and that the use of each such manual override is annunciated at the system level for every system impacted.

The inspector reviewed the results of the licensee's design review of safety system actuation signal circuits for Seabrook Unit 1, as described in letters from the Westinghouse Electric Corporation, to YNSD, serial NAH-4705, dated July 5, 1985 and from UE&C to YSND, serial SBU-95433, dated September 11, 1985, and in a licensee memorandum from YNSD to NHY, serial SBP-85-69, dated October 1, 1985.

Based on their design review, the licensee determined that Seabrook Unit 1 safety system instrumentation complies with the recommendations of IEC 78-19 with the exception that manually overriding a containment ventilation isolation (CVI) due to a high radiation signal from both trains of the protection system will prevent CVI due to an SI signal. YNSD Engineering recommended that administrative controls be put in place which prohibit overriding a containment ventilation isolation due to a high radiation signal on both trains. The inspector verified that appropriate controls were incorporated in draft procedures OS 1252.02 entitled "Airborne High Radiation", and OS 1252.03 entitled "Area High Radiation", and had no further questions concerning IEC 78-19. This item is closed.

c. (Closed) IE Bulletin (IEB 79-06, 79-06A, 79-06A (Revision No.1), 79-06B, 79-06C): Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident. This bulletin and its follow-on supplements were issued to all PWR licensees with the exception of B&W reactors. Construction permit holders were not required to respond to this IEB; it was for information only. All of the required actions were individually addressed in NUREG-0660 entitled "NRC Action Plan Developed as a Result of the TMI-2 Accident". A cross reference of the IEB requirements is found in NUREG-0660, Appendix C, Table C.1. These items are being closed individually as part of the TMI Action Plan. This bulletin is therefore closed.

d. (Closed) IE Bulletin (IEB 79-15): Deep Draft Pump Deficiencies. The inspector reviewed the initial applicant response to IEB 79-15 in letter SB-8261, to NRC RI dated September 5, 1979 and found it to be accurate and timely. At the time of the response the pumps had not been installed, hence operating histories were unavailable. The inspector therefore reviewed the following information, provided subsequent to preoperational testing of the six deep draft pumps at Seabrook, to determine performance and operability status: design specifications, drawings, sectional assemblies, parts lists, pump curves, preoperational test results and machinery histories. The preliminary conclusion was that the pumps were adequately installed and tested so as to provide for reliable operations. The inspector also noted that the bulletin addressed manufacturing deficiencies in components supplied by three vendors other than the one supplying deep draft pumps to Seabrook.

A November 8, 1981 letter from the NRC Office of NRR to the licensee set forth recommended operability guidelines for deep draft pumps. The applicant responded in letter SBN-217 dated March 1, 1982 by providing his own program to demonstrate the long-term operability of deep draft pumps. That program included using the vendor installation procedure with a vendor representative present for at least one pump installation, performance testing to verify that design criteria are met, 168 hour pump runs with vibration analysis every 24 hours, monthly surveillance testing with vibration measurement following preoperational testing until issuance of OL, surveillance and inservice pumps testing in accordance with ASME Section XI, IWP, and plant Technical Specifications after issuance of operating license (OL), performance of maintenance using approved procedures and performance of surveillance testing after significant maintenance to determine pump operability. The post-OL program commitments are generally accepted methods of ensuring pump operability. It appears that this program will meet the intent of the NRC-recommended guidelines and will demonstrate long-term operability of these pumps. The Pump and Valve Operability Review Team, during their November 5-8, 1985 inspection, also found the applicant's program was adequate.

The inspector noted that the control of pump, driver and shaft alignment during installation would have a significant impact on the deep draft pump long term operability. He therefore reviewed the instruction/checklist package for installation and alignment of SW pumps 1-SW-P-41A,B,C and D and service water cooling tower pumps 1-SW-P-110A&B. The manufacturer's instruction manual, FP 53041, was referenced for P-41A,B,C,D installations. A review of that manual revealed a simple, clear, step-by-step installation and alignment procedure. Details on coupling separation and shaft total indicated runout were provided in the instruction/checklist for the six pumps applicable to Seabrook 1. These details were verified against the vendor's technical manual with no discrepancies noted.

An alignment procedure was provided by the instruction/checklist and data recording was mandated by it. The inspector reviewed those data sheets and determined that values for final runouts were better than the manufacturer's preferred valves in all cases. Bulletin 79-15 is closed.

e. (Closed) IE Bulletin (IEB 79-28): Possible Malfunction of NAMCO Model EA180 Limit Switches at Elevated Temperatures. NRC Inspection Report 50-443/85-20, paragraph 3.d, stated that this item would remain open pending NRC review of the licensee's examination of safety-related valves installed at the site. The inspector reviewed licensee memorandum (CEM-86-021) from the I&C Engineering Supervisor to the NHY Licensing Engineer, dated May 23, 1986 which described the results of the licensee's followup to this item. During their harsh environment equipment walkdown effort the licensee identified five additional NAMCO EA180 limit switches stamped with date codes specified in IE Bulletin 79-28. However, none of the limit switches are located in environmental areas where the temperature can potentially exceed 175 F. Therefore, gasket replacement is not required per IEB 79-28. The inspector had no further questions concerning this IEB. This item is closed.

f. (Closed) IE Circular (IEC 80-05): Emergency Diesel-Generator Lubricating Oil Addition and Onsite Supply. This circular described potential problems that were identified at an operating nuclear power plant regarding the addition of lubricating (lube) oil while an emergency diesel-generator was operating and regarding the onsite availability of lube oil. The circular recommended that certain specific actions be taken by holders of operating licenses and stated that holders of construction permits should be aware of the potential problems and initiate appropriate procedures prior to initial fuel loading. The inspector reviewed the licensee's action correspondence report No.85-017, which certified that action has been completed for IEC 80-05, and the inspector reviewed draft station operating procedure MX0539.06 entitled "Emergency Addition of Lube Oil to the Diesel Generator". In addition, the inspector reviewed records of diesel-generator operation and maintenance training completed by licensee employees and a description of training to be provided, as discussed in the Seabrook FSAR, Request for Additional Information (RAI) 430.63. The inspector determined that licensee action in response to IEC 80-05 was completed satisfactorily and had no further questions regarding this matter. This item is closed.

g. (Closed) IE Circular (IEC 80-09): Problems with Plant Internal Communications Systems. This circular described communications systems problems that had occurred at operating power reactor facilities during loss of offsite power events. The circular recommended that several actions be considered including: (1) determine the source of power for plant internal communications systems; (2) upgrade the internal communications to assure operability during the loss of offsite power or other foreseeable events; (3) determine whether any plant electronic equipment may be adversely affected by portable radio transmission; and (4) instruct employees on the use of radios in areas susceptible to electromagnetic interference. The inspector reviewed the licensee's assessment of the use of portable radio transmitters, as described in a memorandum, dated May 30, 1986 and procedures/restrictions on the use of radios within the protected area, as described in a memorandum to Distribution, dated June 6, 1986. The inspector also reviewed the sources of power for plant internal communications systems as described in the Seabrook FSAR, RAI 430.67(9.5.2) and RAI 430.67(9.5.2). There is sufficient diversity in design to assure reliable internal communications systems operation during the loss of offsite power or other foreseeable events, and, therefore, these systems do not require upgrading. The inspector had no further questions regarding IEC 80-09. This item is closed.

h. (Closed) IE Circular (IEC 80-13): Grid Strap Damage in Westinghouse Fuel Assemblies. Fuel for Seabrook 1 has anti-snag straps in accordance with Westinghouse Specification F-5. Further, new fuel will be loaded using the Westinghouse approved sequence that minimizes risk of fuel assembly interaction. This item is closed.

i. (Closed) IE Circular (IEC 81-12): Inadequate Periodic Test Procedure For PWR Protection System. This circular addressed problems associated with testing reactor protection systems (RPS). The test procedures at other facilities described in this circular did not independently verify the shunt trip and undervoltage (UV) trip functions.

   At Seabrook, the shunt trip and UV trip functions are tested independently prior to reactor startup after refueling in accordance with operations procedure OS1410.04 entitled "Post Refueling Pre-Startup RX Trip Breaker Surveillance". The inspector reviewed this procedure during closeout of IEB 83-04 below. This circular is closed.

j. (Closed) IE Bulletin (IEB 83-04): Failure of Undervoltage Trip Function of Reactor Trip Breakers. IEB 83-04 discussed failures of General Electric AK-2 type reactor trip circuit breakers during testing of the UV trip function. It required operating reactors to assure proper operation of reactor trip breakers (RTB) in the future.

   This item is closely related to IE Circular 81-12 above and Generic Letter 83-28 which will be the subject of a future inspection.

   This IEB was first addressed in NRC Region I IR 50-443/85-20. Additionally, CDR 83-00-07 and IEB 85-02 concern failures of RTB and were closed in NRC Region I IR 50-443/85-31.

   The Seabrook design utilizes Westinghouse (W) DS-416 type RTB with both UV and shunt trip features.

   As written, IEB 83-04 does not require an official licensee response since Seabrook does not yet have an operating license, however, it does require that action be taken. The five required actions as stated in the bulletin are addressed individually below:

   (1) This item required surveillance testing be performed upon receipt of the bulletin. Since Seabrook is not operating, these surveillances are not required until just before startup.

   (2) The RTB maintenance program must include the latest manufacturer's recommendation, including frequency and lubrication.

       The W Maintenance Manual for the DS-416 RTB divides maintenance activities into two groups, A and B. Group A is "breaker only" related maintenance. Group B is "switchgear enclosure" related maintenance. The license has addressed Group A and B requirements in maintenance procedures MS0507.17 and MS0507.18, respectively.

(3) All operators were to be informed of the Salem ATWS event and the testing failures at San Onofre Units 2 and 3. They were also to review the appropriate emergency procedures for ATWS.

The Salem ATWS event and San Onofre failures were covered in the cold license training program during module 6 entitled "Industry Events". The emergency procedures for ATWS (FR-S.1) were also covered in the training program.

The inspector reviewed the lesson objectives for the above topics and noted that coverage was comprehensive, with integrated discussion of failure history, symptoms, emergency actions, system interrelations and transient analysis.

(4) Written reply was required of operating plants. Seabrook is not yet licensed and although actions in accordance with this IEB are appropriate, no reply was required.

(5) Reports of breaker failures as a result of testing required by this bulletin need not be made since no specific testing is required until the plant is operational. Should breaker failures occur, they would be reported in accordance with existing requirements.

The inspector reviewed the above referenced maintenance procedures as well as the procedures listed below. He had no concerns. This IEB is closed.

-- OS1410.03, Rev.00, "Reactor Trip Breaker Operations Cycle Weekly Surveillance"

-- OS1410.04, Rev.00, "Post Refueling Pre-Startup RX Trip Breaker Surveillance"

-- MS0507.19, Rev.00, "Corrective Maintenance of Westinghouse DS-416 Reactor Trip Breakers"

k. (Closed) IE Bulletin (IEB) 85-01: Steam Binding of Auxiliary Feedwater Pumps. This problem was first identified in IE Information Notice (IN) 84-06 issued in January, 1984. Subsequently, IEB 85-01 was issued in October, 1985 based on increasing reports concerning the steam binding problem. Preliminary inspection was conducted prior to issuance of IEB 85-01 and documented in Region I IR 50-443/85-20. During this report period, the inspector reviewed operating procedure OS1236.02, Rev.00 entitled "Response to EFW Header Check Valve Backleakage" which provides instructions on appropriate actions to be taken when the EFW discharge header temperature is 50 degrees F higher than ambient or trending upward.

Additionally, he verified that the Roving Auxiliary Operator (AO) logs contain spaces for hourly recording of EFW piping temperatures. Review of the AO training material entitled " Mechanical Components-Pumps" revealed that the AOs are being trained to identify steam binding and the reasons why it is a problem.

The licensee will submit a final report to the NRC after core load within the time frame required by the bulletin. Based on these actions and those identified in the previously referenced report, this IEB is considered closed.

6. TMI Action Plan Requirements (NUREG-0737)

Licensee commitments in response to the requirements of the TMI Action Plan have been reviewed by the NRC staff as documented in the Safety Evaluation Report (SER). During this inspection, the licensee's actions in implementing several commitments were inspected and are considered closed, for NRC inspection verification tracking purposes, as noted below.

a. I.A.1.1 Shift Technical Advisor (STA) - (Closed)

Each licensee shall provide a technical advisor on each shift. The individual must possess specific qualifications in accordance with Generic Letter (GL) 86-04 entitled "Policy Statement on Engineering Expertise on Shift".

New Hampshire Yankee has chosen Option 1 of this letter which allows elimination of the separate STA position by combining one of the senior reactor operator (SRO) positions with the STA into a dual role SRO/STA position. This NHY plan was communicated to the NRC office of NRR by SBN-1027 dated May 2, 1986. This letter referenced an earlier submittal on the subject, SBN-943 dated February 14, 1986. The NHY program, although appearing to meet the intent of GL 86-04 with respect to the technical qualifications of the STA candidates, differs sufficiently in academic degree titles to require a detailed staff review. This review is currently underway at NRR. This item is considered closed and will be re-opened only if staff review indicates that the NHY program is not in accordance with the Policy Statement.

b. I.A.2.1 Upgrading of RO/SRO Training and Qualifications - (Closed)

Applicants for a senior reactor operator license must have been a licensed reactor operator for one year or be a degreed staff engineer. The requirements for licensed operator eligibility at a new facility are more definitively spelled out in NUREG-1021 "Operator Licensing Examiner Standards".

All of Seabrook's operators were licensed in accordance with NUREG-1021 in 1984 and 1985. Additionally, the licensee has incorporated the requirements of this document in the Training Center Management Manual

(TCMM). This manual also specifies that candidates meet the requirements of ANSI/ANS 3.1-1981 entitled, "American National Standard for Selection and Training of Nuclear Power Plant Personnel."

c. II.B.4 Training on Mitigating Core Damage - (Closed)

Licensees are required to develop a training program to teach the use of equipment and systems to control and mitigate the effects of accidents with the potential for core damage.

New Hampshire Yankee, with Westinghouse assistance, developed a detailed training program to cover the required topics. The program was based on the INPO guidelines. As required by Item II.B.4, participation in particular segments of the curriculum were spelled out on Chapter 13 of the FSAR. This training not only involved licensed operators but also various managers and technicians.

The inspector reviewed the participation requirements and the training text. He found the participation requirements to be logical and commensurate with the responsibilities of the participants. His review of the text noted a particularly comprehensive coverage of relevant topics. The emergency procedures are incorporated into the text which is a strength. This course was integrated into the cold license training program for licensed operators. Initial instruction was provided by W trainers and has been nearly completed for all others by the station training staff.

d. I.C.1 Short Term Accident Analysis and Procedure Review - (Closed)

I.C.8 Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants - (Closed)

Both of the above items deal with the development of effective emergency operating procedures (EOP). The Seabrook procedures were developed after the accident at TMI-2 and in accordance with the Westinghouse Owner's Group (WOG) guidelines. The NHY procedures generation package (PGP) is the subject of review at the NRC Office of NRR. This review has been documented in the Safety Evaluation Report (SER) and the third supplement to the SER (SSER 3).

Additionally, those procedures have been observed in practice by the operator licensing examiners in the four cold license exams given to date at Seabrook. They have also been the subject of requalification inspection by the resident inspectors. The Seabrook EOPs are integrated into the licensed operator training program in both the classroom and simulator phases and to date the operators have demonstrated an above average knowledge in their use and bases.

NRC Region I IR 50-443/86-20 discussed TMI Item I.C.7 NSSS Vendor Review of Procedures and stated that the vendor review of the EOPs would be the subject of further inspection. Reference is made to paragraph 7.b of this report.

e. Item III.D.1.1 Integrity of Systems Outside Containment Likely to Contain Radioactive Material for Pressurized-Water Reactors and Boiling-Water Reactors - (Closed)

The licensee has developed a leakage reduction surveillance program for systems outside containment which could contain highly radioactive material during a severe transient or accident. The program involves visual inspections of potential leakage locations (i.e. joints, valve packings, heat exchanger flanges, pump shafts). Any leakage that is noted is measured and recorded on data sheets. The data sheets also record work request numbers that are initiated when leakage beyond acceptable limits is identified.

The inspector reviewed station operating procedure ES1801.02, entitled "Leakage Reduction Program Surveillance", which addressed the leak reduction program conducted during pre-operational testing on the RHR, CS, SI, and the Containment Building Spray System (CBS).

He reviewed the data sheets for the RHR, CS (Train B), and SI (Train A) systems. The results of the surveillance conducted during the preoperational and startup testing programs will be included in the Startup Test Report. The surveillance is required to be repeated at intervals not to exceed each refueling per Technical Specification (TS) 6.7.4 (Final Draft Version).

The inspector noted that the scope of the surveillance program is still under discussion between the licensee and NRR and that issue remains an SER Confirmatory Item. However, no problems with the conduct of the surveillance program exist. Based on the status and conduct of the leakage reduction surveillance to date, the program meets the intent of the Task III.D.1.1 contingent upon the satisfactory resolution of the final scope of the program between the licensee and NRR.

f. Item II.E.4.2, Containment Isolation Dependability - (Closed)

The applicant's conformance to the seven subparts of this item has been evaluated by NRR and found acceptable. The inspector further verified component and system operation were as designed by reviewing the results of various preoperational tests. No deficiencies were noted, as discussed in the sections that follow.

(1) This requirement was to have diversity in the parameters sensed for the initiation of containment isolation. The Seabrook T signal (containment isolation phase A) automatically isolates non-essential systems on a high containment pressure of 4.3 psig, low compensated steam line pressure or low pressurizer pressure. The P signal

(containment isolation/phase B) isolates reactor plant component cooling water lines and control rod drive mechanism cooling fans while starting the hydrogen mixing system on a containment pressure of 18 psig. Diversity was found to be adequate in SER Section 6.2.4. The inspector reviewed the results of general instrumentation test procedure GT-I-120 for loops 1-SI-P-934 through 1-SI-P-937 and was satisfied that the four containment pressure transmitters and loops were installed and did function as intended. He also reviewed the results of test 1-PT-19.1 on the reactor protection system and was satisfied that the logic and coincidence for containment isolation on high containment pressure functioned as intended. Reference is also made to TMI item II.F.1, Attachment 4 of Region I IR 50-443/86-12.

(2,3,4) These requirements necessitate definition of essential and non-essential systems, isolation of non-essential systems during containment pressurization and deliberate operator action to reopen non-essential penetrations after the "close" signal is removed.

The licensee's design on non-essential vs. essential systems, was provided to NRR in letter SBN-470 of February 17, 1983. The inspector reviewed the results of pre-operational test 1-PT(I)-38 entitled "ESF Integrated Actuation Test" and was satisfied that all valves in non-essential systems functioned as required during T and P signal manual initiations and that no valve automatically repositioned when the initiating signals were reset.

(5) This subpart requires containment isolation setpoint pressure to be reduced to the minimum compatible with normal operating conditions. The applicant reduced the original 5 psig setpoint to 4.3 psig, based on a maximum normal containment pressure of 1.5 psig plus the 1.0 psig margin allowed by NUREG 0737, II.E.4.2, clarification 6 plus a 1.8 psig protection system setpoint statistical allowance. The SER Review accepted this 4.3 psig setpoint. The inspector reviewed technical specifications for normal containment pressure limits and found limiting condition for operation (LCO) 3.6.1.4 has a maximum allowance of 16.2 psia (1.5 psig) while in modes 1-4. The inspector also reviewed pressure transmitter data to validate the 1.8 psig statistical allowance. Barton 752 transmitters are used for PT-934 through PT-937 having a range of 0-60 psig. The Protection System Setpoint Study provided a statistical allowance of 3% for these instruments; the product of these 2 values provides the 1.8 psig allowance.

(6) This subpart requires that containment purge valves which do not satisfy BTP CSB6-4 operability criteria must be sealed shut and verified so every 31 days while in Mode 4 or above. The

applicant provided for sealing shut the 36 inch valves CAP-V-1, 2, 3, 4 as required. Also, draft technical specifications were submitted to allow 1000 hours per year of operation with the 8 inch valves COP-V-1,2,3,4 open. However, the NRC staff did not concur that these smaller valves would meet the above operability criteria. Based on discussions with the applicant's licensing staff and NRR reviewers, the inspector determined that TS LCO 3.6.1.7.b has been changed to ensure that the 8 inch containment purge supply and exhaust valves will remain sealed shut unless required for specific containment operations.

(7) This requires that containment purge and isolation valves must close on a high radiation signal. The inspector determined that process and effluent monitors 6527A and B (G-M detectors with a range of $10\text{-}10^6$ CPM) will initiate an isolation of 8 inch valves COP-V-1,2,3,4 on a high radiation signal and that area monitors 6535A and B (G-M detectors with a range of $10^{-1}\text{-}10^4$ mR/hr) will initiate an isolation of both COP-V-1, 2, 3, 4 and 36 inch valves CAP-V-1,2,3,4 on a high radiation signal. The latter circuitry is for dropped fuel accident mitigation.

The inspector reviewed the results of acceptance test 1-AT-30 on these four radiation monitor circuits. The test was initiated by lowering the alarm setpoint below background; it was a full system test except for calibration of the detectors. The test showed that all valves in an alarming train (A or B) shut and remained shut until the alarm cleared.

TS LCO 3.3.3.1 covers operability of these four radiation monitors and Table 3.6-2, Section C specifies closure times for the 8 inch and 36 inch ventilation valves.

g. I.G.1 Training Requirements - (Closed)

This item requires licensees to establish and conduct a training program in preparation for low power testing. NHY incorporated this training requirement into their 1986 requalification program schedule. The course consisted of both classroom and simulator sessions and was completed for all licensed operators in April, 1986. The inspector reviewed the attendance records and while reviewing the lesson plans noted that the training was a coordinated effort between the Startup Test Department and the Training Department.

h. II.K.1 IE Bulletins on Measures to Mitigate Small Break LOCAs and Loss of Feedwater Accidents - (Closed)

Three sub-paragraph requirements of item II.K.1 are addressed below.

(5) Review all valve position and positioning requirements and positive controls and all related test and maintenance procedures to assure proper Engineered Safety Features (ESF) functioning.

The inspector confirmed that the licensee verified proper ESF functioning during the applicable portions of the pre-operational test program. Pre-operational tests on the EFS (PT-14.1), SI (PT-8), and the ESF's Integrated Actuation (1-PT-38), among others, have been reviewed and witnessed by Region I inspectors during the course of the pre-operational test program. The results of those inspections have been documented in several previous NRC inspection reports 85-23, 85-26, 85-30, 86-01, 86-12, and 86-13.

(10) Review and modify procedures for removing safety-related systems from service and restoring to service to assure operability status is known.

To enable the control room operators to easily access the status of safety systems, Seabrook has installed a safety system status inoperable panel in the main control board. The panel consists of sixteen push buttons, eight per train, each connected to a status light. The eight safety systems that are displayed are:

Containment Building Spray
Primary Component Cooling water
Chemical and Volume Control
Emergency Feedwater
Residual Heat Removal
Safety Injection
Service Water
Service Water Cooling Tower

All of the status lights are manually actuated by the control room operators. Actuation of a status panel push button causes the corresponding status light to come on and an alarm message to either be displayed on the Video Alarm System (VAS) or logged by the plant computer. It also causes lighting of the status lights for systems which are supported by an inoperable system.

The inspector reviewed the VAS response procedures for alarms governing CS pump inoperability and SI train inoperability. He confirmed that the control room operators are directed to depress the applicable status light in response to these alarms. He discussed with the Operations Administrative Supervisor the procedural methods that will be used to direct the control room operators to activate the appropriate status light when maintenance or testing render a safety system inoperable and the action is not alarmed (i.e. realigning manual valves). He stated that these procedural directives had not been finalized but committed to have a procedural mechanism in place in the near future which satisfied their commitments to NRR on this issue. The adequacy of these procedural directives will be reviewed when they are in place and remain an unresolved item (86-34-01).

The inspector reviewed the loop and logic diagrams for all the safety system status alarms and verified that actuation of a status light generates an alarm message that is either displayed on the VAS or logged by the computer. He also reviewed the shift superintendents relief checklist and confirmed that he is required to review the status of the safety system status panel during shift turnover.

In addition to the safety system status panel, Chapter 10 of the Operations Management Manual (OPMM), entitled "System Status", addresses the administrative controls to be implemented to ensure the status of safety systems is independently verified and controlled. TMI Action Plan item I.C.6, which also addresses the issue of independent verification of safety systems and components will be addressed as a separate item in a future inspection.

(17) Trip the pressurizer low level coincident signal bistables so that safety injection is initiated when the pressurizer low-pressure setpoint is reached regardless of the pressurizer level.

The inspector reviewed Westinghouse foreign print 70320 listing the ESF actuation signals. He confirmed that a reactor trip and safety injection signal is generated on low pressurizer pressure regardless of pressurizer level.

i. <u>II.K.3.5 Automatic Trip of Reactor Coolant Pump (RCP) During Loss of Coolant Accident (LOCA)</u>

The WOG answered GLs 83-10c and 83-10d with a generic response concerning RCP trip criteria. GL 85-12 was subsequently issued to inform licensees and applicants of the staff's conclusion regarding the WOG submittals and to provide guidance concerning implementation of RCP trip criteria. This GL directed each facility to select one of the acceptable options approved in the safety evaluation of GL 85-12 for implementaion. NHY responded in SBN-976, dated March 31, 1986. Subsequent correspondence and discussion with NRR has revealed a NHY position that subcooled margin (SCM) will be used for RCP trip criteria.

The inspector reviewed the Seabrook plant specific Emergency Contingency Actions (ECA), Functional Restoration Procedures (FRP) and the Emergency Response Procedures (ERP) noting the RCP trip criteria of less the 30 degrees F SCM, concurrent with having at least one centrifugal charging Pump (CCP), or safety injection pump (SIP) running.

Discussions with licensed operators revealed an adequate level of knowledge with respect to use of the trip criteria. Additional discussions were held with the Shift Superintendent responsible for development of these procedures.

j. II.K.3.9 Proportional Integral Derivative Controller Modification

Westinghouse recommended modifications to the Proportional Integral Derivative (PID) controller for the power operated relief valves (PORV).

The Seabrook design includes a PID controller for the PORVs. The derivative feature has been deleted, effectively preventing a rate compensating signal input to the controller setpoint. The inspector reviewed surveillance procedure IX1662.390 entitled "PC-455 Pressure Control Loop Calibration", noting that there was no provision for a rate adjustment. Discussion with the I&C Department Supervisor and an I&C Department Working Foreman confirmed that the derivative function jumpers had been removed from the applicable W 7300 series cards.

k. Item II.K.3.12 Confirm Existence of Anticipatory Reactory Trip Upon Turbine Trip

The Seabrook design provides an automatic reactor trip upon turbine trip at power levels above the P-9 setpoint.

7. SER Items (Confirmatory Action Items Requiring Follow-up per the Safety Evaluation Report)

a. 6.2.8 Containment Pressure Monitor (Item II.F.1.4)

The hardware installation associated with this system was field inspected in NRC Region I IR 50-443/86-12. Therefore this item is closed.

b. 13.5.2.3 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

The Seabrook SER indicates that NSSS vendor review of the EOPs is not required since they are based on the WOG guidelines. The W review of the startup test procedures (including power ascension) was inspected in IR 50-443/86-20. This item is therefore considered closed.

c. 5.4.12 Reactor Coolant System (RCS) Vents

SSER 4 indicated that the RCS vents operability requirements and inservice inspection (ISI) and inservice test (IST) programs must be established. Additionally, operations procedures for the vents must be written and approved.

NRC Region I IR 50-443/86-12 addressed the procedures for using the vents. The TS specify the operability requirements (T.S.3.4.1) and the valves were verified to be covered under the NHY IST Program, Rev.1.

The head vent lines are either ASME Class 1 or 2 from just downstream of the reactor vessel (RV) head penetration until the outlet of RC-V-232, the motor operated isolation valve. Included in this piping run is the removable piece which allows removal of the RV head. The entire line is exempt from ISI requirements, in accordance with IWC-1000 of the ASME Boiler and Pressure Vessel Code, Section XI.

The inspector reviewed the design drawing for the vent system in verifying code boundaries and compliance. As part of the assessment of the TS surveillance requirements (T.S.4.4.11.1, 4.4.11.2), the inspector reviewed the below noted operations surveillance procedures:

-- OX1401.09, "Reactor Coolant Vent Path 18 Month Surveillance" (Rev. 00)

-- OX1401.08, "Reactor Coolant Vent Path Quarterly Check" (Rev.00)

While verifying that the surveillance procedures meet the requirements of the TS, he noted that inconsistencies existed in the acceptance criteria between the body of the procedure and the T.S. and between the procedure and its attached repetitive task sheet (RTS). There was also inconsistency between the procedure and the RTS concerning the modes in which the surveillance could be conducted. Other minor discrepancies were also brought to the attention of the Operations Administrative Assistant. The licensee will correct the procedures and issue a new revision of each procedure. The inspector considered the licensee's planned procedure revisions acceptable and had no further questions concerning this matter. This item is therefore considered closed.

8. <u>As-Built Verifications</u>

A detailed system walkdown was conducted on the makeup portion of the CS system and the SI system located within the RHR vaults. The installed piping, instrumentation, and components were checked against the P&ID. In addition, discrepancies in the as-built configuration were verified to be covered by approved changes.

In the makeup system, a local flow indication on the boric acid pump (BAP) recirculation line to the boric acid tank (BAT) was not installed as shown on the P&ID and as described in the FSAR. Additional inspection revealed that ECA 99/112818A had been issued in January 1986 to remove these items (one per BAP) from the P&ID. The FSAR was not listed as a reference document. The licensee intends to amend the FSAR description. This item remains unresolved pending additional review of the NHY engineering change process both before and after it was implemented (86-34-02).

The SI piping within the RHR vaults was also traced. SI-V-249, the relief valve off the common mini-flow recirculation line from each SIP is shown on the drawing to be located in Vault No. 1. Its actual location is in Vault No. 2. This discrepancy is related to unresolved item 50-443/85-35-01 regarding the development of the NHY computer assisted drafting (CAD) drawings. This item remains open with expanded scope to include the adequacy of NHY design control and generation process.

9. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. Unresolved items disclosed during this inspection are discussed in paragraphs 6.h and 9.

11. Management Meetings

At periodic intervals during the course of this inspection, meetings were held with senior plant management to discuss the scope and findings of this inspection. An exit meeting was conducted on July 8, 1986 to discuss the inspection findings during the period. During this inspection, the NRC inspectors 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.

On June 10, 1986, a meeting was held at Seabrook Station by mutual licensee/ Region I agreement to discuss the project status, schedule, and operational readiness of Seabrook, Unit 1. This meeting was held at the conclusion of the Systematic Assessment of Licensee Performance (SALP) meeting, documented in the SALP Report No. 50-443/86-99.

During this readiness meeting, New Hampshire Yankee management personnel presented information on construction completion, startup testing progress, operational preparedness, licensing activities and the Seabrook Completion Items List (SCIL) for Unit 1. NRC questions related to the tracking of incomplete construction and testing activities, station staff readiness for operations and the status of licensing. Both the licensee and NRC management agreed that such meetings are beneficial on a periodic basis to provide for the interchange of information and for a consistent understading of both regulatory developments and the current project status.