

U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No. 50-443/85-20

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, Units 1 and 2

Inspection at: Seabrook, New Hampshire

Inspection conducted: July 8 - August 27, 1985

Inspectors: A.C.Cerne, Sr. Resident Inspector  
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J.M.Grant, Reactor Engineer  
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Reviewed by:

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9/6/85

date signed

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9/12/85

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Inspection Summary:

Inspection on July 8 - August 27, 1985 (Report No. 50-443/85-20)

Areas Inspected: Routine inspection by the resident inspectors and region-based inspectors of work activities, procedures, and records relative to I&C installation; piping and component supports; fuel building ventilation systems and spent fuel rack erection; and the review and witness of preoperational testing activities. A visiting PWR technology instructor from the NRC Training Center (Office of Inspection & Enforcement) conducted a review of operational readiness and training activities. The inspectors also reviewed licensee action on previously identified items, including 10CFR50.55(e) reports and I&E Bulletins and performed plant inspection-tours. The inspection involved 343 inspection-hours.

Results: Two violations were identified (Paragraphs 4 e & h), both resulting from follow-up of licensee corrective action on previously reported construction deficiencies. While the licensee has taken immediate measures to correct any hardware problems which have been identified with these violations, further investigation into their cause is required to preclude future recurrence.

DETAILS1. Persons Contacted

W.B.Derrickson, Senior Vice President (NHV)  
J. DeVincentis, Project Engineering Manager (NHV)  
R.E.Guillette, Ass't Construction QA Manager (YAEC)  
G.A.Kann, Phase 2-6 Test Group Manager (NHV)  
D.C.Lambert, Field Superintendent of QA (UE&C)  
D.A.Maidrand, Assistant Project Manager (YAEC)  
G.F.McDonald, Construction QA Manager (YAEC)  
D.G.McLain, Startup Test Group Manager (NHV)  
J.W.Singleton, Field QA Manager (YAEC)  
C.E.Walker, Systems Engineer (Westinghouse)

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

2. Plant Inspection Tours

The inspectors observed work activities in-progress, completed work and plant status in several areas of the plant during general inspections of the plant. The inspectors examined work for any obvious defects or noncompliance 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.

Specifically, the inspectors checked "J-tube" replacement on the four steam generators, visually examining fit-up & complete welds on the "C" steam generator, interviewing craft and Westinghouse engineering personnel, and verifying quality assurance activities to include visual remote examination of the inner diameter of the feedwater ring headers after welding to assure cleanliness and debris removal. Further inspection by QA of all the steam generator secondary sides was confirmed and it was noted that the generators were placed in wet-layup (demineralized water with overlying nitrogen gas blanket) on their shell sides to their respective feedwater and main steam isolation valves. An inspector also verified that the tube lane blocking devices are installed in the Model F steam generators at Seabrook and that they will be used in line with FSAR commitments.

On the main steam lines, the inspectors noted and discussed with craft personnel, the reinstallation of main steam relief valves which had been removed to adjust the guide ring settings in order to achieve full valve travel. At the same time, mechanical ball joints were being installed in the discharge piping from the relief valves to the vent stacks in order to provide a pressure boundary vent path for the steam while assuring expansion capability in line with design loads associated with the opening of the relief valves. Installation controls were evident as was continued licensee analysis to confirm proper valve guide ring settings in line with the valve disc travel required to support FSAR accident analysis assumptions.

An inspector reviewed the as-built configuration of the Spent Fuel Building Heating, Ventilation and Air Conditioning (HVAC) System to verify that it conformed with FSAR commitments. The inspector specifically checked:

- Design Changes related to the ventilation exhaust and tornado dampers.
- The nuclear grade classification (class 1E) of various exhaust and tornado damper electrical components.
- Certified Material Test Records (CMTR's) on Air Handling Unit 1-FAH-F-74 and the limit switches for exhaust damper 1-FAH-DP-14.
- Certificate of Compliance on the Dow Corning 795 sealant used by Pullman Construction Industries (PCI) in HVAC applications.
- Nonconformance Reports (NCR's) 82/645A&B governing restorations to hold down studs in Air Handling Units 1-FAH-F-74 & 41.
- Work Request FAH-0034 governing work per the disposition of NCR 82/645B.
- Qualifications of the welder performing Work Request FAH-0034.
- Administrative controls governing operation of the Air Handling Units during fuel movement.

The inspector also discussed with the licensee staff the status of the operations procedure governing control of the HVAC system during fuel movement. After such procedures receive approval by the Station Operations Review Committee (SORC), routine NRC inspection of procedural content will be conducted to confirm agreement with commitments made in the FSAR.

The inspector also reviewed the design of the cooling tower spray distribution piping. He visually examined and made inquiries into the installation of hangers on the Unit 2 portion of the distribution header piping which runs to the common cell of the cooling tower. The inspector verified that the hangers will be completed as part of the Unit 1 construction program, although they will remain designated as Unit 2 hangers. He also questioned the completion status of Grinnell Mechanical Snubbers 1-M/S-90-1-90-SV-2 and 1-M/S-90-1-89-SV-2 on the reactor vessel seal leakoff lines. The inspector verified that the supports were in process and determined that use of the Pullman Field Process Sheet governing the snubber installation will assure satisfactory completion of the partial installation.

The inspector reviewed the licensee's plans and procedures to detect and handle steam binding of auxiliary feedwater pumps due to check valve back leakage. The problem of auxiliary feedwater pump steam binding was first addressed in I&E Information Notice No. 84-06 and is potentially applicable to the Seabrook Emergency Feedwater (EFW) system. The inspector surveyed the licensee actions related to this matter and determined thru interviews that the licensee is planning to monitor EFW piping temperature once every twelve hours utilizing a pyrometer attached to the piping. The licensee also plans to test pump discharge check valves 1-FW-V64 & 70 under their inservice testing program. While no procedural changes to address steam binding of the subject pumps are presently being contemplated, the licensee's planned monitoring program appears to provide adequate assurance of the early detection of EFW steam binding events. Based upon discussions with the licensee and review of the EFW system arrangement, the inspector has no further questions on this issue at this time.

In line with a Region I Nondestructive Examination (NDE) van inspection conducted during the same time period as this resident inspection, the resident

inspectors devoted additional inspection time to the independent inspection of welds and material and initial review of the licensee inservice inspection program. The results of the NDE van inspection are documented in inspection report 50-443/85-19.

Also during this inspection, a lead NRC instructor from the PWR Technology Branch of the I&E Technical Training Center visited Seabrook and conducted an inspection review of certain operational readiness areas. Included in this review were system checks and plant walkdowns of the Emergency Core Cooling System, the Boron Thermal Regeneration System, and the Main Control Board. He also reviewed current procedures being used by the Seabrook Operations Training Center staff and evaluated future plans for the handling and control of operational drawings.

On August 15, 1985, the resident inspectors were informed of a metallurgical fracture in the support assembly holding the electrical contact brushes on the diesel generator exciter ring at the Millstone Unit 3 plant. Since the diesel generators at Seabrook are supplied by the same manufacturer, utilizing the same generator system, they were examined and found to have a similar design except that two brush holder supports per generator were present instead of the one on the failed Millstone 3 unit. The licensee had already been informed of the Millstone problem and has begun investigating its potential applicability to Seabrook. The licensee informed the inspectors that a 10CFR50.55(e) evaluation will begin when more facts as to the cause of the Millstone problem become known.

With regard to all of the above inspection items and issues, no violations or unresolved safety questions were identified.

### 3. Licensee Action on I&E Bulletins & Circulars

- a. (Open) Circular 78-16: Limitorque Valve Actuators. The inspector reviewed a letter dated August 31, 1978 (Limitorque to UE&C) describing Limitorque's evaluation of the clutch wear problem identified in the subject Circular and its recommendations for periodic manual operation of the actuator and verification of proper actuator engagement following manual operation. The licensee provided no further evidence that these recommendations had been addressed and/or incorporated into site procedures for ensuring electrical operability of the SMB-0, 1, 2, and 3 Limitorque Operators used at Seabrook. Pending review of the licensee's follow-up to Limitorque's recommendations, this item will remain open.
- b. (Open) Bulletin 79-14: Seismic Analyses for As-Built Safety-Related Piping Systems. At the time this Bulletin was issued, the licensee had not yet erected any safety-related piping at Seabrook (letter, dated August 15, 1979, PSNH to NRC, Region I). The licensee has since developed a program for verifying the validity of the seismic analyses for safety-related piping. This program called PAPSCOTT (Piping & Pipe Support Closeout Task Team) will involve reconciling the as-built condition (e.g., pipe geometry; pipe support and pipe restraint design, location, function and clearance; floor and wall penetration location and clearance; embedments; pipe attachments; valve and valve operator location; in-line instrument geometry and location) with the original seismic analyses. Westinghouse (W)



is responsible for the seismic analysis of the Reactor Coolant System and the majority of remaining safety class 1 piping. UE&C is responsible for the safety class (1,2,and 3) piping outside of W's scope. This reconciliation effort will be reviewed for implementation in a future inspection. Pending the outcome of that NRC inspection, this item will remain open.

- c. (Closed) Bulletin 79-21: Temperature Effects on Level Measurements. The licensee was not required to respond to this Bulletin. However, NRR has identified the subject of this Bulletin as an open item in Seabrook's SER, NUREG-0896, dated March 1983. Because no response was required, and this issue is being reviewed by NRR in its evaluation of the Seabrook FSAR, this Bulletin is considered closed.
- d. (Open) Bulletin 79-28: Possible Malfunction of NAMCO Model EA 180 Limit Switches at Elevated Temperatures. The inspector reviewed the licensee's response to this Bulletin. Per letter dated February 2, 1980 (PSNH to NRC, Region I), the licensee stated that at the time the Bulletin was issued only one safety-related valve had been installed, and it did not use a NAMCO EA180 limit switch. The licensee also committed to examining other safety-related valves at the site. The inspector has requested the results of that examination. Pending review of those results, this item will remain open.
- e. (Open) Bulletin 80-03: Loss of Charcoal from Standard Type II, 2-inch, Tray Adsorber Cells. Because the charcoal adsorber cells for Seabrook had not yet been purchased and/or received, the licensee was not required to respond to this Bulletin. (They have since been delivered to Seabrook.) The licensee did request information from CVI Corporation, the manufacturer from which Seabrook's carbon adsorbers were ordered. In turn, CVI Corp. requested information from its carbon tray supplier. Response from the carbon tray supplier was not evident, and has been requested by the inspector. This Bulletin will remain open until the inspector is provided the carbon tray supplier's response so it can be reviewed in relation to the subsequent receipt inspection records for the adsorber cells and other available, pertinent documentation.
- f. (Closed) Bulletin 81-01: Surveillance of Mechanical Snubbers. The licensee was not required to respond to this Bulletin. However, the licensee has addressed the issue in its program for testing snubbers. All mechanical snubbers at Seabrook are Pacific Scientific (PSA) manufactured. Each of these snubbers will be tested per PSNH Engineering Procedure, EX 1805.03, "Functional Testing of Snubbers". This will include testing each snubber over the range of stroke in both compression and tension, and when a failure occurs, evaluating that failure before work continues. This procedure sufficiently addresses the Bulletin's concerns, and therefore, this Bulletin is considered closed.
- g. (Open) Bulletin 81-03: Flow Blockage of Cooling Water to Safety System Components by Corbicula sp. (Asiatic Clam) and Mytilus sp. (Mussel). Per letter, dated July 8, 1981 (PSNH to NRC, Region I), the licensee determined that Mytilus sp. is found in the source water for Seabrook; Corbicula sp. is a fresh water bivalve and is therefore not found in Seabrook's source water. Thus far, the licensee has not encountered any infestation problems because the potentially affected systems have not yet been placed in service utilizing the source water (ie: service water from the Atlantic Ocean).

As stated in the Seabrook Final Environmental Statement, NUREG-0895, dated December 1982, the licensee plans to control Mytilus sp. through a combination of low level chlorination and intermittent shock treatment, possibly supplemented by thermal backflushing. The decision to use such treatments is based, in part, on past successes at similar nuclear and fossil plants situated on the coast, such as Pilgrim (nuclear) and Canal Station (fossil).

The EPA and State of New Hampshire recently issued the Operations Discharge Permit, which would allow continuous chlorination for controlling biofouling. This permit, which is now in a public comment period, requires the licensee to submit a chlorination minimization usage plan to monitor the biofouling control program. The licensee is developing this program and will be submitting it to EPA and the State of New Hampshire in late September or early October. Pending review of the approved discharge permit and chlorination minimization usage plan, this will remain open.

- h. (Open) Bulletin 83-04: Failure of the Undervoltage Trip Function of Reactor Trip Breakers. Although the licensee was not required to respond to this Bulletin, the licensee addressed it in an interoffice memorandum, concluding that this bulletin did not apply to Seabrook. However, following discussions with the licensee, it was agreed that this Bulletin was indeed applicable because Seabrook uses W DS-416 Reactor Trip Breakers. This applicability was previously documented in Region I Inspection Report, 443/83-05, where it was determined that W DB-50 trip breakers (ie: the source of the original concern for trip breaker failure) were not in use at Seabrook. Until the licensee adequately addresses the subject concerns for DS-416 breakers, this Bulletin will remain open.
- i. (Closed) Bulletin 83-08: Electrical Circuit Breakers with an Undervoltage Trip Feature in Use in Safety-Related Applications Other Than the Reactor Trip System. Per letter, dated March 27, 1984 (PSNH to NRC, Region I), the licensee stated that no undervoltage trip feature is utilized in safety-related breakers at Seabrook other than in the Reactor Trip system. The inspector discussed this Bulletin with licensee representatives and reviewed responses to the licensee from UE&C and Westinghouse (W) which concluded that UE&C's design philosophy for Seabrook does not include the use of the undervoltage trip feature anywhere other than in the Reactor Trip System, and W has supplied no W type DB, DS or GE type AK-2 circuit breakers, other than the Reactor Trip Breakers, in safety-related applications at Seabrook. The inspector had no further questions. This Bulletin is considered closed.
- j. (Open) Bulletin 84-02: Failures of GE Type HFA Relays in Use in Class 1E Safety Systems. UE&C identified five HFA relays per unit for Seabrook that perform safety-related functions within a class 1E safety system (used on 480V unit substation buses E52, E53, E62, E63 and E64). The licensee committed to replacing these HFA relays with Century Series HFA relays (per letter, dated June 8, 1984, PSNH to NRC, Region I). The licensee has also determined that W has not supplied any GE type HFA relays for safety systems under its scope of work (ie: the Class 1E Reactor Protection Instrumentation Systems). The licensee plans to continue using HFA relays with nylon or lexan type coil spools in nonsafety-related applications and has committed to establishing administrative controls dealing with maintenance,

storage and handling of spare relays and parts to ensure segregation of safety-related and nonsafety-related relays and parts. The inspector requested further information on the subject administrative controls. Pending review of these administrative controls and inspection of the relays that were replaced, this item will remain open.

#### 4. Licensee Action on Construction Deficiency Reports

- a. (Closed) Construction Deficiency Report (CDR 82-00-08): Potential for power surges to disable an actuation train in the Solid State Protection System (SSPS). A field modification in the SSPS output relay test circuitry was performed to provide positive indication of proper circuitry configuration after the conduct of the in-service relay testing. Per Final 10CFR50.55(e) Report (SBN-816, dated June 10, 1985), PSNH reported to the NRC, Region I that the hardware changes had been completed.

The inspector reviewed the Startup Work Request (WR SSPS-0049) directing the modification to the SSPS relay test panel per ECA 03/101585. He noted that Startup Quality Control inspection of this work had been provided and documented (84-IR-1677). Upon completion of the work, output relay testing was postponed until conduct of General Test, GT-I-102C, which was a prerequisite to the conduct of Preoperational Test (PT 19.1) for the Reactor Protection System. The inspector reviewed the GT-I-102C test records package, interviewed the startup test engineer, and examined revised wiring drawings being utilized by operations instrument technicians in the field to confirm completion of the required hardware modifications. The governing design change (ECA 03/101585) had as its basis a W Field Change (FCN NAHM-10534) which was issued to insure detection of a potentially undetectable test circuit failure.

The inspector witnessed the ongoing conduct of PT 19.1 and thru the above interviews, record reviews, and drawing change verifications confirmed that the design change had been implemented. This CDR is closed.

- b. (Closed) Construction Deficiency Report (CDR 82-00-12): Adequacy of tornado missile shield protection for the Fuel Storage Building. As previously documented in Region I inspection report (IR 443/84-13), the licensee concluded that this issue was not a reportable deficiency based upon a probabilistic analysis performed by the Applied Research Associates for Seabrook (Report C569, dated September, 1983). This report was transmitted by Region I to the NRC office of Nuclear Reactor Regulation (NRR) for analysis and review of the deterministic probabilities.

Correspondence between PSNH and NRR led to the issuance of a Final 10CFR50.55(e) Report (SBN-793, dated April 18, 1985) by the licensee, submitting the responses to certain questions raised on this subject by NRR and reconfirming the licensee position that this item was not a reportable deficiency. The Office of NRR reviewed the licensee report and responses to requests for additional information and contracted with the National Bureau of Standards for a further technical evaluation of this issue.

Upon completion of the NRR review, NRR reported to Region I (letter dated June 25, 1985) that the licensee "has satisfactorily demonstrated compliance

with the requirements of General Design Criteria 2 and 4 regarding protection of safety-related plant equipment from the effects of tornado and high wind generated missiles." Thus, no physical corrective action is required by the licensee and this CDR is considered closed.

- c. (Open) Construction Deficiency Report (CDR 83-00-02): General Atomic Radiation Monitors (RM -23's). The inspector reviewed Receiving Inspection Reports (RIR) 8228, 8682 and 9519 for radiation monitors (1-RM-CP-180 A&B and 2-RM-CP-180 A&B) received at Seabrook. Prior to shipment to Seabrook, General Atomic corrected a design deficiency which caused the monitor display to "lock-up". Per letter, dated July 30, 1984, General Atomic to UE&C, the RM-23 design was revised. The inspector requested the traveler and job record cards that were included as an attachment to that letter, but the licensee was unable to furnish them by the end of this inspection. This deficiency will remain open pending review of the traveler and job record cards associated with the repair of the RM-23's.
- d. (Closed) Construction Deficiency Report (CDR 83-00-03): Detachment of transistor heat sinks from the integrated circuit cards in the W 7300 series process cabinets. A UE&C nonconformance report (NCR 1852) was issued and dispositioned to correct this deficiency by providing a new heat sink, screwed into the circuit cards. Per PSNH Final 10CFR50.55(e) Report (SBN-828, dated June 19, 1985) to the NRC, Region I, the licensee reported that the affected heat sinks had been replaced.

The inspector reviewed Startup Work Request (WR-0018) requiring replacement of the defective heat sinks per the disposition to NCR 1852. He confirmed Startup QC visual inspection of this activity on 55 of the affected cards (reference 83IR248). It was noted that electrical isolation of the newly installed heat sink assemblies was provided by the use of nonmetallic washers and the conduct of ohmmeter checks for resistance. The inspector visually examined some of the modified cards in the 7300 series cabinets, requesting one card (PY-403B, C04-445) to be pulled for closer inspection to the design change details of the W Standard Drawing 403A947 .

Interviews with YAEQ QA personnel revealed their intent to conduct further surveillance to confirm the completeness of the modification work for the Unit 2 cabinets in storage at Newington Station. This was subsequently accomplished and documented on a YAEQ Surveillance Report, dated July 15, 1985. The inspector has no further questions on the hardware modifications used to correct this deficiency and considers this CDR to be closed.

- e. (Open) Construction Deficiency Report (CDR 84-00-12): Cracking of the Diesel Generator Exhaust Silencer Concrete Pedestals. Corrective action on this design deficiency consisted of the redesign of the support pedestals as steel structures with bolt clearances sufficient to accommodate expansion due to exhaust temperatures and which provide sufficient heat sink for limiting the concrete and grout temperatures. After several design change iterations, the final design details were issued as disposition to NCR 82/221G. Startup Work Requests (WR DGN-0521 & 0738), controlling the conduct of the rework, were signed off as complete and NCR 82/221G was closed based upon QC verification of the work and records under the UE&C QA program



(QAS-5) for seismic components important to safety, but not "safety related". By PSNH Final 10CFR50.55(e) Report (SBN-819, dated June 10, 1985), the licensee reported that the required rework per the redesign details had been completed.

However, visual examination of the exhaust pedestal rework by NRC resident inspectors after the issuance of PSNH Final 10CFR50.55(e) Report indicated that certain portions of the steel bolting work had not yet been completed, as evidenced by the existence of temporary bolts and loose connections. The licensee confirmed the lack of complete work and issued a NCR 82/616A to document the discrepancy between the closed out records and the status of the rework. UE&C Corrective Action Request (CAR167) was also issued to recommend corrective actions in the way the control of work packages and status are maintained and documented. Recognition was given to the fact that one cause of the problem lay with the inadequate QC procedures under which the subject re-work was being accomplished. The licensee sent another letter (SBN-835, dated July 10, 1985) to the NRC, Region I, clarifying the previous Final Report and the current status of the diesel generator exhaust silencer pedestal redesign and rework.

Notwithstanding the corrective actions that have been taken since the identification of the subject discrepant rework, the inspector noted that Test Program Instruction (TPI-11) on the handling of Work Requests had been procedurally violated and that the Seabrook Project Policy No.27, regarding QA verification of CDR verification had not been adequately followed. He informed the licensee Assistant Construction QA Manager in an exit meeting on August 19, 1985 that these corrective action failures constituted a violation of 10CFR50, Appendix B, Criterion XVI (85-20-01). Additionally, this CDR will remain open pending further independent verification of the rework on the steel pedestals by NRC inspection personnel.

- f. (Closed) Construction Deficiency Report (CDR 84-00-14): Seismic qualification of the control rod drive mechanism (CRDM) cooling shroud. Per PSNH Final 10CFR50.55(e) Report (SBN-847, dated July 25, 1985) to the IRC, Region I, W has performed analyses which seismically qualify the CRDM shroud as an intervening structural element providing support to reactor pressure vessel (RPV) piping attached to it. The licensee has therefore determined that this issue is not a reportable deficiency under 10CFR50.55(e) criteria.

The inspector reviewed the W calculations to include consideration of both static and seismic loadings on the shroud frame and shroud flange. He confirmed that the calculations were done for the most conservative loading point by visual examination of the shroud installed on the RPV head in the field. Other assumptions and design conditions used in calculating loads reactor vessel level indication system (RVLIS) piping were checked in the field as work progressed for the installation of affected pipe supports per ECA 25/101251A.

The inspector's review of the W calculations, in conjunction with the independent field checks, verified the adequacy of the licensee/W investigation of this issue. No physical corrective action is required and this CDR is considered closed.

- g. (Closed) Construction Deficiency Report (CDR 85-00-03): Service Water Pumps Reduced Discharge Head. The licensee stated in their Final Report to the NRC, Region I (SBN-824, dated June 17, 1985) that the deficiency in the performance of the service water and cooling tower pumps was not reportable under the conditions of 10CFR50.55(e). The determination was made on the basis of calculations which showed that the pumps were still capable of performing their safety-related function. The inspector reviewed the analysis and discussed with the Startup Mechanical Test Director the bases for specific portions of the pump tests. The inspector also reviewed the Yankee Atomic Electric Company (YAEC) nonconformance reports on the subject pumps, as well as, two (2) startup work requests (SW-1001 and SW-1003) governing changes made in the impeller lift of cooling tower pump 110B. All documents were complete and in order. The NCR's additionally stated that FSAR sections 9.2-12 and 9.2-13 would be revised to reflect the actual performance of the service water pumps. Based upon the results of the licensee's analysis, the inspector agrees that the deficiency is not reportable under the terms of 10CFR50.55(e). This CDR is considered closed.
- h. (Open) Construction Deficiency Report (CDR 85-00-05): Safety Injection Limit Switch Brackets. As a result of NCRs 74/1903 and 74/2914, which require the strengthening of limit switch brackets to three safety injection valves (1-SI-V-89, -90, and -93), the licensee reported a potential significant construction deficiency in accordance with 10CFR50.55(e); i.e., if the subject brackets were to fail and prevent the attached limit switches from performing their safety-related function, the design operation of the safety injection pumps could be impeded or prevented. After investigation, the licensee submitted a Final Report (SBN-830, dated June 18, 1985) to the NRC, Region I which indicated that the subject brackets were capable of performing their safety-related function with the as-delivered design (i.e., with a 1/16" thick plate as the supporting bracket) and therefore, this deficiency was nonreportable.

The inspector reviewed NCRs 74/1903 and 74/2814 and noted that although the licensee had concluded that the 1/16" brackets were adequate, the disposition of each NCR required that the brackets be replaced with 3/16" thick A-36 material plate. The inspector examined the actual installation of the new brackets and discussed both NCRs with the licensee, UE&C and Pullman-Higgins (P-H) representatives.

Revision B (the last revision) to NCR 74/1903 required removal of the existing 1/16" bracket on 1-SI-V-93 and replacement with a 3/16" plate. Sheet (2) of the Nonconformance Response Form (NRF) to NCR 74/1903B supplemented the written instructions with a drawing representing the approved installation. Responsibility for completing the work was assigned to P-H, who in turn issued Field Instruction FI-413 to do the work. (Note: A Field Instruction is not a design document - it is only a construction aid.) NCR 74/1903B was closed on November 26, 1984. A new NCR (74/2914) was then issued to replace the limit switch brackets on valves 1-SI-V-89 and -90. Revision B to NCR 74/2914 was issued on December 1, 1984 to also include the limit switch bracket on 1-SI-V-93. (Note: This bracket was supposedly replaced per the disposition of NCR 74/1903B.) FI-413 was also revised on December 1, 1984 to include 1-SI-V-89 and -90. According to Weld Rod Stores Requisition

No.329799, weld rod was drawn against FI-413 on December 1,1984. (Note: It was indicated on the rod slip that the installations to FI-413 were nonsafety-related.) The installations were completed and NCR 74/2914B was closed on December 5,1984.

As a result of the inspector's discussions with licensee, UE&C, and P-H representatives, review of NCR 74/1903, NCR 74/2914, FI-413, and Weld Rod Stores Requisition No.329799, and the examination of the actual installation, the inspector identified the following deficiencies:

- (1) The weld design approved and illustrated on NCR 74/1903B and FI-413 calls for a fillet weld. (Note: No weld design is given or referred to in NCR 74/2914.) In discussions with the licensee and UE&C engineers, it was agreed that a double-flare-bevel-groove weld is the appropriate design. It was also agreed, however, that no technical concern exists because of the adequacy of the actual installation; i.e., the actual welding resulted in a double-flare-bevel-groove weld with sufficient effective throat, and the welder who installed the brackets was qualified for both fillet and groove welding.
- (2) The details illustrated on FI-413 do not reflect the approved design of NCR 74/1903B. (Note: NCR 74/2914 provides no details from which FI-413 could have been derived.)
- (3) NCR 74/2914B does not provide sufficient details for the new installation, nor does it refer to any design documents that provide those details.
- (4) Both NCR 74/1903B and NCR 74/2914B indicate that ASME Section III, safety-related components were affected. It appears that this classification refers to the valves (1-SI-V-89, -90, and -93) rather than the brackets because the brackets are not part of the pressure boundary. UE&C recognized this by identifying on NCR 74/1903B that the limit switch bracket to 1-SI-V-93 should be installed under ANSI B31.1 requirements. However, UE&C did not clarify the "safety or nonsafety" classification of the limit switch brackets and made no clarification on NCR 74/2914B.
- (5) When NCR 74/1903B and 74/2914B were assigned to P-H for work completion, P-H erroneously interpreted the disposition as requiring installation of limit switch brackets that were nonsafety-related. Weld Rod Stores Requisition No. 329799, used to draw rod against FI-413, indicates this "nonsafety-related" classification.
- (6) When NCR 74/1903B and 74/2914B were reviewed by UE&C for closure, UE&C did not identify the failure of P-H to install the limit switch brackets as safety-related components.

The inspector discussed the classification of the limit switch brackets with the licensee who agreed that these brackets were indeed safety-related because of their safety-function as supports to the limit switches on safety injection valves 1-SI-V-89, -90, and -93. Following further discussion and review, the inspector determined that UE&C did not identify the subject brackets as safety-related components and further, did not provide design details appropriate to nonconforming conditions requiring rework on the limit switch brackets. In

turn, P-H erroneously considered the brackets as nonsafety-related, and therefore did not implement all quality assurance controls required for safety-related components, such as QC inspection. Furthermore, the inspector determined that UE&C QAE reviewed both NCRs for closure without identifying the misclassification, resulting lack of adequate inspection and the unclear design details and resulting incorrect installation. The inspector informed the licensee at an exit meeting on July 18, 1985 that failure of UE&C and P-H to properly control the design change rework, resulting in the installation of material and welds in conflict with design and without the required QA controls, represents a violation of 10 CFR Part 50, Appendix B, Criterion III (85-20-02).

In conjunction with the licensee's determination that the original significant deficiency was nonreportable, the inspector requested to review the calculations performed by Westinghouse (W) and used as the basis for that determination. The inspector subsequently reviewed these calculations but identified some discrepancies between the analysis and the as-built limit switch bracket hardware, as discussed above. At the end of this inspection the licensee indicated that W has performed additional calculations to justify the adequacy of the existing configurations. Pending NRC review of this revised analysis, this CDR remains open.

- i. Other Construction Deficiencies: To verify that the licensee was aware of construction deficiencies identified at sites similar in design and/or supplied by the same vendors, the inspector requested information pertaining to the following issues:
  - 1) environmental qualification of equipment for high energy line breaks outside of containment (design; W),
  - 2) solid state logic protection system power surge failure (design; W),
  - 3) valve stem nut disengagement resulting in degraded diesel generator performance (vendor: Colt),
  - 4) defect in fuel injection pump delivery valve holders on diesel generators (vendor: Colt),
  - 5) overpressurization in W component cooling water systems (designer: W),
  - 6) cracks on the inside bend of Unistrut fittings (vendor: Unistrut).

The inspector reviewed information related to the above deficiencies and concluded that the licensee had adequately addressed each issue. One item which the licensee was unable to address, is a deficiency involving a broken boss found on the lube oil pump discharge nozzle of the Shoreham diesel generators (vendor: Colt). The licensee is continuing its review of this item to determine whether or not it has been previously evaluated. Pending review of the licensee's efforts, this item will remain unresolved (85-20-03).

## 5. Licensee Action on Previously Identified Items

- a. (Closed) Violation (443/84-13-04): Discrepancies in the Steam Generator (SG) lateral support installation records. The inspector reviewed the response to YAEK Blue Sheet No. 76, requesting clarification of the subject discrepancies, interviewed one of the Pullman-Higgins (P-H) field engineers on this issue, and examined QC inspection records which were not included in the original



records package. This evidence indicates that ECA 01/2374C and the P-H Field Instruction (FI-120) were properly implemented in the erection of the SG lateral support members.

Also, W issued a new Quality Release (QR 36012) on June 11, 1985 which adds to the lateral support records a Southern Bolt letter cross referencing bolt material heat numbers and their certification reports to the heat codes, stamped on the bolt heads. A P-H NCR (73/7796A) was issued to document and correct the premature closure of an earlier NCR (73/0591) on the traceability for the lateral support bolting.

These actions in augmenting and correcting, as necessary, the records for the SG lateral supports installation are consistent with the corrective measures documented in the PSNH response to the violation (SBN-744, dated December 28, 1984). In conjunction with the evidence provided in the response to Blue Sheet No. 76, indicating correct installation and torquing of the lateral support bolts, these corrective measures were judged by the inspector to adequately address both the technical aspects of this violation and the prevention of future recurrence. This item is closed.

- b. (Closed) Unresolved Item (443/84-20-01): Need for better control of jurisdictional work interfaces. Startup Test Department (STD) internal memorandum STD INT 85-87 to all System Test Engineers (STE) stressed the importance of adhering to jurisdictional controls. Each STE signed a form indicating that they had read and understood the STD INT 85-87 memorandum. The NRC inspectors verified that the questioned level transmitters (LT) 930, 931, 932, and 933 were now properly labeled with the STD turnover stickers for STD jurisdictional control. The inspector also spot-checked the following STD activities to verify that work was being performed under the correct jurisdictional control:

- 1) the preparation effort for starting the thermal barrier isolation valves CCW-395, 428, 438
- 2) feed water hydrostatic test of ASME and ANSI B.31-1 piping in accordance with procedure FW-IT-05
- 3) rework of the screen wash system flanges to the filters SCW-185A and B.

Based on the above STD actions and additional verifications by the inspector, this item is considered closed.

- c. (Closed) Unresolved Item (443/85-09-01): Questioned classification of excess letdown line, solenoid-operated, air control valves as non-Class 1E. The inspector discussed this item with both the YAEC Systems Lead Engineer and a systems reviewer from the NRC Office of NRR. It was determined that a hand-controlled drag valve, existing in the subject excess letdown line, will be set for a maximum flow of 25 GPM and confirmed during Hot Functional Testing. Since such a loss of primary coolant, given a failure of the questioned non-Class 1E valves, could be made up by normal charging flow from the Chemical & Volume Control System, the classification of the subject valves was deemed acceptable.

The inspector also confirmed that the categorization of these valves (CS-V-175 & 176) as "active", per Table 3.9(N)-11 of the Seabrook FSAR was

correct. Valve failure mode ("Fail-closed") was proper even though closure of the valves is not considered "safety-related" under the required accident analyses. The adequacy of this design was corroborated by the NRR reviewer to be in line with the General Design Criteria. This item is therefore considered closed.

#### 6. Instrumentation & Control (I&C) Design Change Implementation

The inspector examined both the in-progress and completed work for the following two design changes relating to I&C activities on safety-related components:

- 1) ECA 05/103176: support details for the steam generator (SG) level instrument tubing.
- 2) ECA 05/107372: addition of a volume booster to the instrument air supply actuator for the main steam atmospheric relief valves.

ECA 05/103176 required sixteen drilled and tapped holes (each 5/8" diameter by 1.25" deep) in the shell of each steam generator. The inspector checked the adequacy of control exercised not only over the installation activities, but also over the engineering approval and concurrence for this design change. He interviewed Westinghouse engineers and reviewed the W Model F Steam Generator Stress Report (SG-85-03-038), which included secondary shell stress analyses at the level instrument tubing support attachment points. UE&C Installation Instructions (M-1) dated February 8, 1985 were noted to prescribe the use of "bottoming taps" for the bolt holes and to require visual examination and functional "go/no go" checks of each hole, with the inspection witnessed by a W representative. These precautions were confirmed by the NRC inspector, as were other criteria like hole placement and proximity to SG welds, by direct visual examination after completion of the work. The inspector also reviewed the fabrication drawing (Foreign Print, FP53394) for the Seabrook steam generators to check the consistency of the transition cone and upper shell thicknesses used in the W stress analysis.

No violations were identified.

With regard to ECA 05/107372, the inspector noted that the installation controls included QC inspection under the UE&C nonsafety-related, seismic QA program (QAS-5). However, the current revision to the ECA (Rev.B) listed the work under the jurisdiction of ASME, Section III criteria. This discrepancy was clarified when Revision C was issued to the ECA listing the subject work as nonASME. This was further discussed with licensee QA and licensing personnel in view of licensing commitments for a dedicated air supply for the affected main steam atmospheric relief valves to assure operability, given a failure of the nonsafety instrument air system. The inspector learned that the design details for the safety-related air supply have not yet been issued and that at this time, they bear no relation to the volume booster installation, which while seismic, is still considered a nonASME installation. The inspector additionally checked the completed tubing configurations, installed per ECA 05/107372, with the volume booster.

The inspector has no further questions in regard to the main steam atmospheric relief valve I&C work at this time. No violations were identified.

## 7. Preoperational Testing

### a. Test Procedure Review

The inspectors reviewed the following preoperational test (PT) and acceptance test (AT) procedures for conformance with FSAR Section 14 and the applicable regulatory guidance:

- 1-PT-16.1 (Revision 0) - Primary Component Cooling Water (PCCW) System
- 1-PT-16.2 (Revision 0) - Reactor Coolant Pump (RCP) Thermal Barrier Cooling System
- 1-AT-1.3 (Revision 0) - Startup Feed Pump Test

While AT-1.3 is an acceptance test on a nonsafety system, it is required by USNRC Regulatory Guide 1.68 and therefore will be subject to Startup QA/QC coverage per existing Startup Test Department (STD) notification procedures. The inspector verified that during conduct of the AT, control logic verification of the startup feed pump operation from electrical Bus 5 (Class 1E, Train "A" safety-related power) will be accomplished as well as the normal operation from electrical Bus 4 (nonsafety power). This is consistent with the discussion in the Seabrook Safety Evaluation Report (SER), section 6.8, which analyzes the availability of the startup feedwater pump and its capability to be powered by onsite emergency power, if required.

With regard to the review of the test procedures, listed above, no unresolved safety questions or violations were identified.

### b. Test Witness

The inspector witnessed portions of the following preoperational test conduct, evaluating performance against the reviewed and approved PT procedure, interviewing test engineers and technicians during the course of the test, reviewing documentation associated with the prerequisites and test instructions in particular, monitoring control room activities in support of the test, and confirming QA coverage to include the control and witness of designated mandatory STD QA witness points.

- 1-PT-8 - ECCS Performance Test  
(Note: NRC inspection/witness of this test occurred in June, 1985, prior to this current report period, but it was not documented in an earlier inspection report)
- 1-PT-19.1 - Reactor Protection System

For the conduct of PT-8, the inspector visually examined each of the twelve safety injection (SI) needlepoint valves which were adjusted to balance

SI flow to all four coolant loops during the test. He also spot-checked the Validyne digital devices used for measuring flow across the local flow element upstream of each valve, verifying calibration and measurement configuration. Following the test, he confirmed that handwheels had been removed and casings locked or welded over the valve operators to preclude any changes to valve position thereafter.

The inspector reviewed the Test Exceptions List, the available Test Procedure Field Changes, and the QA Notification Points listed within the PT-8 procedure. He discussed with STD engineers the sequence of running the weaker pump of the redundant ECCS trains (as it was determined by the Pump Head Curve Verification checks) prior to the stronger pump in order to establish a flow balance condition while preventing pump runout of the weaker pump. For operations involving each train discharging thru separate flow paths (eg: hot leg recirculation SI), the relative strength of the SI pumps was not considered relevant. This assumption was checked against the Westinghouse system description for hot leg recirculation operations and confirmed to be valid.

The inspector also subsequently reviewed the YAEK STD QA surveillance reports for the conduct of PT-8 and discussed the designation and sign-off of QA witness points with the lead startup discipline QA engineer.

With respect to the conduct of the ECCS Performance Test, no violations were identified.

In regard to PT-19.1, the inspector reviewed the prerequisite General Test Packages (GT-I-102 A thru D), examined the test console and the test loop diagrams (WR SSPS-0137), and interviewed the startup test engineers to determine the function and adequacy of the test console to electrically mimic a trip signal to the reactor protection system (RPS) channels. The RPS trip tests and coincidence logic of PT 19.1 were evaluated against the FSAR Table 7.2-1, UE&C design drawings (C509042 thru C509049), and the standard W trip design. Calculations for test trip setpoints for input to the test console were spot-checked.

The inspector also witnessed several trip tests, to include both the series of logic checks and circuit response time measurements. Documentation of the recorded information, to include calibration and test data, was reviewed. The inspector confirmed startup QA surveillance of PT 19.1 and noted the use of QA verification witness points during the test. The need for and adequacy of test procedure field changes were randomly sampled over the course of the test.

With respect to the conduct of the Reactor Protection System preoperational test, no violations were identified.

## 8. Piping & Component Supports

- a. The inspector examined a section of residual heat removal piping (RH-158), specifically checking the status and condition of supports, snubbers, and struts, as detailed on the design drawings and installed. The following three discrepancies were identified:



- 1) Support 158-SG-28, as shown on the latest revision (Rev.15) of isometric drawing, F800158 was neither installed nor listed for installation in the support tracking system because of an error in the Change Document Tracking system which incorrectly deleted the support drawing.
- 2) Strut 158-SG-32 was missing a spacer on one side of the rod end assembly despite a procedural requirement (Pullman JS-IX-6) to provide spacers to ensure the ball bushing remains centered on the load pin and also despite completion of a Sway Strut Checklist indicating spacers were installed.
- 3) Snubber 158-RM-31 was incompletely erected (note: records indicated it was still in-process) without evidence that Pullman Procedure JS-IX-6 adequately covered the installation of the double bolt pipe clamps which were part of this hydraulic snubber.

The inspector discussed the above discrepancies with licensee engineering and QA personnel and subsequently reviewed documentation of the following licensee investigation and corrective action, where required:

- 1) Investigation revealed that the UE&C Pipe Support Group had erroneously deleted the support because of misinterpretation of a design work request. Other design evidence was available to confirm the validity of the need for support 158-SG-28 and to illustrate that the Piping and Pipe Support Closeout Task Team (PAPSCOTT) effort would have identified and reconciled this error and that the support would have been installed, as required.
- 2) A nonconformance report (NCR 73/11590A) was issued against the missing spacer on strut 158-SG-32, with disposition to install a spacer as procedurally required. Additionally, YAEQ QA personnel conducted a reinspection of twelve similar strut supports and identified no further discrepant conditions.
- 3) Prior to the resident inspector's question on the adequacy of installation procedures for snubber 158-RM-31, Pullman-Higgins had already requested engineering clarification (RFI 73/7911A) as to how this type of hydraulic snubber pipe clamp should be installed. The response to this request, illustrating the design details, was issued on August 22, 1985.

Based upon the above licensee actions and results, the inspector concluded that the missing support would have been identified and corrected by the licensee PAPSCOTT program, that the missing spacer was an isolated case which has now been corrected, and that Pullman-Higgins had taken adequate action to supplement their procedure on the design details for the installation of the hydraulic snubber pipe clamps. He has no further questions on these items.

- b. During an inspection of the Letdown Heat Exchanger (CS-E-4) and the heat exchangers of the Boron Thermal Regeneration System (BTRS), the inspector noted discrepancies in the connections of the heat exchanger support beam angles to the embed plates. The majority of the connections were made with a 5/16 inch fillet weld and two high strength bolts. The remainder of the

connections lacked high strength bolts. One of the connections (the Letdown Heat Exchanger support beam connection in the northeast corner of the heat exchanger shield cubicle) lacked not only the high strength bolts but also one of the two welds connecting the beam to the embed plate.

The inspector reviewed UE&C drawing F101567, Section F, outlining the type of connections required for these beams. The drawing specified either a 3/16 inch fillet weld on three sides of the connection or two high strength bolts. The use of both bolts and welds in the same connection was not addressed. However, the Lester/Cives installation drawing (Foreign Print 13279) specified both bolts and welds in these connections. Such an installation would be non-standard per American Institute of Steel Construction (AISC) design details. While the inspector reviewed ECAs which authorized the deletion of the weld on the northeast support beam connection for CS-E-4 and substitution of a single 5/16 inch fillet weld for the three 3/16 inch fillet welds, he discussed his other concerns with representatives of YAEC Quality Assurance and the UE&Cs Civil Engineering staff. They agreed that a discrepancy does exist and are taking measures to investigate and resolve this issue.

The inspector was also informed that the subject work was incomplete and would be finished at a later date. This led the inspector to question how the licensee is ensuring that all incomplete work from the previous structural steel contractor at Seabrook (Perini Power Constructors, Inc.) would be completed by UE&C. Discussions with UE&C Civil/Structural Quality Control (QC) and YAEC QA revealed that a "punchlist" of incomplete structural steel construction items was developed by Perini before they left the project. The inspector was told that these items had to be completed prior to turnover of a building from Construction to the Operating Staff. The inspector reviewed the UE&C procedure which addresses structural steel erections (FSP-153). After review of the procedure, it is still not clear to the inspector that there are adequate controls to ensure that the Perini "punchlist" items will be completed and receive proper QA/QC attention. Specifically, the inspector requested response to the following questions:

- 1) What steps in the building turnover procedure ensure that the Perini "punchlist" items will be completed?
- 2) In relation to the specific problem of both welds and bolts in the same connection, which method of connection is proper and is actually taken credit for in carrying the vertical and axial loads?

Pending resolution of these two questions and a random check by the inspector of a sample of the completion status of the questioned "punchlist" items, this issue remains unresolved (85-20-04).

## 9. Installation of High Density Spent Fuel Storage Racks

The inspector witnessed receiving, unloading, cleaning and placement of one of the high density spent fuel storage racks in the spent fuel storage pool. The inspector also reviewed the installation procedure for the racks (Purchase Order NSSS-229) and verified that the procedure contained adequate measures to

ensure the racks would not be damaged during installation. The inspector observed proper QC coverage of the work and adherence to the installation procedure and checked thru interviews, the cognizance of supervisory personnel to the technical requirements. Spent fuel rack installations continued in-progress at the conclusion of this inspection. The inspector has no questions, to date, regarding the observed work.

No violations were identified.

#### 10. Allegation Resolution

The licensee has established an Employee Allegation Resolution (EAR) program for addressing concerns raised by employees at Seabrook. This program has also been used to address requests from the NRC for further investigation into alleged deficiencies. Per letters, dated December 19, 1984 and February 11, 1985, from the USNRC, Region I to PSNH, the NRC requested the licensee to investigate two concerns regarding unauthorized weld repairs and welder identification inconsistencies, respectively. The inspector reviewed the EAR files addressing each of these NRC concerns. The inspector concluded that the EAR investigation (interviews, documentation review, reinspection) was sufficient to address both issues. The inspector had no further questions and informed the licensee that the EAR results would be evaluated in conjunction with the other pertinent information received by the NRC for the follow-up of concerns and allegations.

#### 11. 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 the inspection are discussed in Paragraphs 4i and 8b.

#### 12. 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. Documentation marked as proprietary information was reviewed by the inspectors during the course of this inspection. However, the subject matter of this inspection report, regarding those items considered proprietary, was discussed with licensee management and engineering personnel at a formal exit meeting on August 19, 1985 and at a follow-up meeting on August 27, 1985. The NRC inspectors received no comments from the licensee that any of the discussed inspection items or issues contained proprietary information. No written material, of a subject matter or nature not documented in this report, was provided to the licensee during this inspection.