U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-443/86-12

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: February 3 - March 31,1986

Inspectors: A. C. Cerne, Sr. Resident Inspector
D. G. Ruscitto, Resident Inspector
R. S. Barkley, Resident Inspector
John G. Hunter, III, Reactor Engineer

Approved by: T. C. Elsasser 4/24/86
Thomas C. Elsasser, Chief, Reactor PRojects Section 3C Date

Summary: Inspection on February 3 - March 31, 1986 (Report No. 50-443/86-12)

Areas Inspected: Routine inspection by the resident inspectors and one region-based inspector of work activities, procedures, and records relative to the containment enclosure ventilation system and solid radwaste system design and construction; the review and witness of preoperational testing activities and test results evaluation; new fuel receipt inspection; 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 10CFR50.55(e) reports, and performed plant inspection-tours. The inspection involved 496 inspection-hours by four NRC inspectors.

Results: One violation was identified in the area of design control as a failure to provide sufficient details for inclusion of a specific area of the containment enclosure as part of the air space ventilated by the containment enclosure emergency air handling system (Paragraph 6a). NRC witnessing of selected preoperational tests, to include the monitoring of in-process results review, has resulted in no major problems or safety concerns, although certain design questions, raised as a result of testing, remain unresolved (Paragraph 8c). NRC inspection of the licensee response to TMI Action Plan guidance resulted in the closure of specific items and the identification of no safety concerns, to date, and progress by the licensee on the remaining items.

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DETAILS

1. Persons Contacted

J. DeVincentis, Project Engineering Manager (NHY)
R. E. Guillette, Assistant Construction QA Manager (YAEC)
G. A. Kann, Phase 2-6 Test Group Manager (NHY)
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 (NHY)
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 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 inspector checked the installation of the pipe whip restraints on the steam generator feedwater lines (PW-4606-4, PW-4607-4, PW-4608-4, PW-4609-4). These restraints were not available for installation in time to be included in PT(I)-4.2, the Power Conversion and ECCS Thermal Expansion Test conducted during hot functional testing. Therefore, installation and testing of the subject restraints for thermal expansion was documented as a preoperational test (PT) exception which will be addressed during the startup test program (reference: Startup Test Procedure, ST-52). The inspector questioned the Piping and Pipe Support Close-Out Task Team (PAPSCOTT) supervisor on how these supports would be held in position since their support system restraints appeared to bear on the narrow range steam generator water level taps that they straddle. Review of UE&C drawing F-104065 illustrated the fact that clamps will be installed on each restraint, providing for maintenance of a position relative to the level taps. In addition, two (2) support pieces connected to the steam generator lifting trunnions will also be installed to further support the restraints. The inspector rechecked the restraints at a later time in the report period and confirmed that the support pieces and clamps had been installed. He identified no problems with the design or installation of the restraint or their supports.

The inspector traced the following systems, as noted below, for as-built configuration and agreement with the P & IDs and FSAR:

-- Control Building Air Handling System (CBA), within the control building at the 75'-0" elevation and the emergency switchgear and cable spreading rooms at the 21'-6" and 50'-0" elevations, respectively.

-- Containment Air Purge System (CAP)

-- Containment On-Line Purge System (COP)

-- Primary Building Air Handling System (PAH) at elevations 53'-0" and above.

-- Containment Enclosure Air Handling System (EAH)

-- Fuel Storage Building Air Hanlding System (FAH)

While no discrepancies between the as-built configuration and the drawing and FSAR details were identified, the inspector did question one CBA system design feature, relative to its emergency air cleaning filter subsystem activation and isolation signals. This question is being tracked as part of an unresolved item further discussed in Paragraph 8c of this report.

The resident inspectors witnessed the arrival of the first shipment of nuclear fuel (14 assemblies) to the site. The licensee's initial site inspection, transportation to the fuel building, unloading from the casks, and transfer from the fuel building track bay to the new fuel storage racks with subsequent transfer to predesignated rack locations in the spent fuel pool were all observed. The inspector also observed Health Physics technicians make a radiation survey of the area around the shipping cask prior to the removal of the cask lid. Alpha contamination swipes were made of the plastic cover on the elements after the cask lid was removed and before any work was done to remove the rigging that supported the element during shipment. The inspector confirmed that the accelerometers, which indicate whether the cask was severely jarred or dropped during shipment, were not tripped. He observed that the element was properly rigged and transferred. Inspection of the fuel assembly for obvious defects, rod misalignment, cleanliness, cladding defects and agreement between the documentation for the element and the number stamped on the fuel assembly top nozzle, was noted.

The licensee procedure for "New Fuel Receipt & Inspection" (RS0722) was reviewed and licensee activities evaluated against the applicable criteria. Additionally, the inspector witnessed operation of the spent fuel bridge assembly in the movement of new fuel from the elevator to the spent fuel storage racks. Operations procedure, OS1015.07, was reviewed and discussed with the bridge assembly operator, particularly with regard to the interlock controls which prevent movement of new fuel when the bridge is improperly positioned.

Subsequent shipments of nuclear fuel, which arrived on site during this inspection period, were spot-checked for receipt, inspection and movement operations, as well as the implementation of security, radiological and Health

Physics controls. In accordance with the Special Nuclear Materials license (SNM-1963) issued for Docket 70-3027 on December 19, 1985, specific license limitations on the storage of new fuel were reviewed and discussed with the responsible engineering and supervisory personnel. The inspector also questioned the Reactor Engineering Department Supervisor as to when the fuel assembly inserts (control rod assemblies, thimble plugs, burnable poisons) would be inspected. He was informed that they would be checked when all of the fuel assemblies received during a shipment have been transferred to the spent fuel storage pool. The inspector verified that this inspection is procedurally controlled by a licensee procedure, RS0724.

During one of the subsequent inspection-tours of the Fuel Storage Building (FSB), the inspector noted that the east door leading outside was not posted to prevent exit without Health Physics (HP) approval. The access to the FSB had been shifted from its normal location at the west door to the east door on the previous day to allow fire proofing sealant application in the area of the west door. The posting was inadvertently left off the east door when it was returned to its normally closed condition following completion of work at the west door access.

The inspector notified shift Health Physics personnel and the door was promptly reposted. The licensee also issued a project notice signed by the Senior Vice President reminding all personnel of the importance of these postings. Several follow-up inspections have verified all warning signs properly posted. Based on the fact that the door is alarmed, thereby providing assurance that the radiation boundary was not violated, as well as the prompt licensee corrective action, the inspector has no additional concerns in this area.

3. Licensee Action on Previously Identified Items

a. (Closed) Deviation (443/84-17-01): Questionable usage of ASME Code Cases with respect to FSAR commitments and regulatory guidance. The licensee issued Blue Sheet No. 077 as a request for follow-up action on this issue in October, 1984. Additionally, the licensee response to Region I (SBN-760 letter, dated February 7, 1985) committed to both a revision to the existing project controls on the handling of ASME Code Case adoption for Seabrook use and a review of previous practices to investigate potential misuse.

The inspector discussed the status of corrective action with the responsible YAEC engineering and QA personnel and reviewed correspondence directing further review by both UE&C and Westinghouse. Guidelines have been established for invoking the use of ASME Section III (Division 1 & 2) Code Cases at Seabrook (reference: UE&C memo, SM 12441). UE&C reviewed this issue to determine whether specific Code Case revisions had been annulled or superceded by ASME at the time of adoption for Seabrook use and whether specific revisions had been endorsed by the latest edition of the applicable USNRC Regulatory Guide at that time. Also, Westinghouse set forth its Code Case Policy in a letter (NAH-2698) to the YAEC Project Manager and subsequently addressed specific Code Case usage in another letter (NAH-2777).

The results of the UE&C technical review into Code Case usage at Seabrook revealed no general technical problems, although one question on minimum material tensile strength requirements is still being investigated. While specific examples of annulled or superceded Code Cases, adopted for use at Seabrook, were identified, the technical impact of such misuse was minimal, generically representing more of an administrative, rather than a technical, concern.

The inspector verified that licensee corrective actions on this issue have provided programmatic guidance for future Code Case usage at Seabrook and have addressed the impact on existing hardware. The remaining technical item, under investigation by UE&C, is being tracked even though the initial review appears to confirm the generic acceptability of the subject item.

Licensee investigation of this issue has been extensive and corrective action appears commensurate with the nature of the deviation. The inspector has no further questions. This item is considered closed.

b. (Closed) Unresolved item (443/85-25-01): Adequacy of the diesel generator brush holder assembly support design. A problem with the subject component, identified at Millstone Unit 3, was investigated at Seabrook and determined to have the potential for similar adverse impact because of a similar design by the same manufacturer. As documented in Region I Inspection Report 443/85-25, the most critical portion of the brush holder support was determined to be in the area between the base of the drilled hole in the hexagonal nut component and the threaded extension bar. The generator manufacturer, Louis-Allis, calculated the minimum required metal thickness in this critical area of the hexagonal nut to be 1/8". While radiographic examination by the licensee of the hexagonal nut revealed a minimum metal thickness of 9/64", the licensee elected to replace the brush holder supports on the Unit 1 diesel generators. The new brush holder supports have a 1/4" minimum metal thickness in the area of concern.

The inspector witnessed replacement of the brush holder supports on both diesel generators, A & B. He reviewed Engineering Change Authorization (ECA) 08/11860A which directed the replacement of the supports. The quality shipment release (Q-18475), which documented the inspection of the brush holder supports, including a dimensional check on the hexagonal nut components, was also reviewed. The inspector noted all relevant documents to be in order and the replacement work properly controlled and QC verified. The inspector considers this item closed for Unit 1.

Brush holder replacement for the Unit 2 diesel generators has not yet been scheduled and remains an NRC open item (444/85-02-01).

c. (Closed) Construciton Deficiency Report (CDR 83-00-02): General Atomic Radiation Monitor. The licensee stated in its Final Report to the NRC, Region 1 (reference: SBN-825 dated June 12, 1985) that the intermittent lock-up of the RM-23 display was due to a design problem that had been corrected on all monitors that are safety-related. This CDR was previously reviewed by an NRC inspector (reference: Region I Inspection Report, 443/85-20, dated September 18, 1985) at which time the General Atomic traveler sheet and job record cards, which documented the required hardware replacements, were requested. The licensee was unable to provide the documents before the end of that inspection. They have since supplied the traveler sheet and job record cards to the resident inspector for review. No problems with the documents were identified, as they provide evidence of the actions taken by the manufacturer to correct the subject deficiency.

This CDR is considered closed.

d. (Closed) Construction Deficiency Report (CDR 84-00-09): Limitorque motor operators. IE Information Notice 84-36 and its Supplement 1 described various generic problems associated with Limitorque Valve Operators involving loosened lock nuts on the worn gear shaft of the valve operator. Additionally, IE Information Notice 85-22 addressed a potentially significant problem pertaining to the incorrect installation of pinion gears in Limitorque Motor-Operated Valves (MOVs).

Seabrook's General Electrical Test Procedure (GT-E-33) provides general instructions and uniform test methods for Motor Operated MOVs. In July 1985, the licensee developed another General Electrical Test Procedure (GT-E-113) to conduct a one time inspection of Class IE MOVs. When deficiencies were noted from this inspection, rework and retesting were conducted in accordance with Test Program Instruction (TPI) 11 entitled "Work Requests" and GT-E-33. The GT-E-113 inspection program included checks for the deficiencies described in the above IE Information Notices as well as the five items listed below which are the subject of CDR 84-00-09.

(1) Loose Contact Screws on Contact Bridge of Torque Switch

(2) Incorrect Spring Pack

(3) Incorrect Limiter Plate on Torque Switch

(4) Improper Lug Connections

(5) Sheared Tab on Phenolic Cam of Torque Switch

Overall Limitorque valve deficiencies were also identified in Unresolved Item 85-15-11 which was closed by IR 443/86-05.

Items 2 and 3 above were determined by the licensee not to be reportable per 10CFR50.55(e), however corrective action was still required. Licensee corrective action on each of the five items is shown below.

(1) Loose Contact Screws on Contact Bridge of Torque Switch: The discrepant fiber shims under the contact bridge were replaced with the specified metal shims. GT-E-113 and GT-E-33 verified that no fiber shims remain. NCR 82-192 documented this noncomformance and the subsequent corrective action dispositions.

(2) Incorrect Spring Pack: Spring packs on the sixteen identified valves were replaced with the correct packs under Work Request (WR) CO-0480.

(3) Incorrect Limiter Plate on Torque Switch: GT-E-33 and GT-E-113 verified correct torque switch limiter plate installation.

(4) Improper Lug Connections: Deficiencies including poor crimps, exposed wire strands and damaged wire insulation were inspected under GT-E-113. Ninety valves are complete. Eighteen remain to be completed under a work request. Unacceptable conditions are either repaired or the defective parts replaced.

(5) Sheared Tab on Phenolic Cam of Torque Switch: Limitorque has implemented a design change to replace the phenolic cam with a type using a metal key to prevent shearing. This change was verified by GT-E-113. This deficiency was documented by NCR 82-257 with corrective action as dispositioned.

The inspector reviewed GT-E-113 and GT-E-33 verifying that the procedures comprehensively covered all known potential Limitorque deficiencies. This included deficiencies identified by similar 10CFR50.55(e) reports from other facilities. In addition, the test package (No. CC-4.11) for motor-operated valve CC-V1095 was selected for complete review to verify full compliance with licensee stated corrective actions. CC-4.11 contained the following documents:

-- Wiring Verification and Functional Check Record (Form GT-E-21-F01)

-- Station Computer Data Sheet (Form GT-I-42-F02)

-- IRTU 1 Cable Schematic and Table

-- Work Request CC-1364

-- MOV Inspection Check List (Form GT-E-113-F01)

-- V-1095 Schematic Diagram

-- Relevant Material/Storehouse Records

-- QC Inspection Report 85IR2966 (filed separately)

In all cases, licensee inspection and work was properly documented and reviewed. Identified areas of nonconformance with the inspection criteria were repaired. Retesting was accomplished in accordance with the appropriate sections of GT-E-21 (Wiring Verification and Functional Checks).

The inspector also conducted a field inspection of a Limitorque operator (MS-V-204) with the responsible Startup Quality Assurance Inspector.

The inspector identified no problems with either the field inspection or any of the repair documentation, which were reviewed to confirm corrective action as reported in the licensee's Final Report to the NRC Region I (SBN-890, dated November 8, 1985). This CDR is considered closed.

e. (Closed) Construction Deficiency Report (CDR 84-00-18): Control Rod Drive Mechanism (CRDM), heavy drive screw, breech guide screw. IE Information Notice 85-74, "Failure of a Heavy Control Rod (B4C) Drive Assembly to Insert on a Trip Signal," identified a potential problem with Westinghouse-designed CRDM, such as those used at Seabrook. The breech guide screw could become disengaged from the external breech of a drive rod assembly and fall on top of the CRDM latch assembly where, if it became lodged, could prevent drive line motion.

Corrective action was undertaken with ECA 08/109106C, which implemented Westinghouse Field Change Notice (FCN) NAHM-10560(B). This ECA was composed of three separate Westinghouse procedures as detailed below:

(1) Inspection Procedure for CRDM Heavy Drive Rod Assembly Breech Guide Screws: In this procedure, a reverse torque check was made to identify any screws which were loose.

(2) Repair Procedure for Heavy Drive Rod Assemblies Without Breech Guide Screws: This procedure outlines the actions to be taken on those assemblies which failed the reverse torque check and subsequently had the old screws removed. Additionally, it provides a replacement screw which is then drilled and lock-pinned in place.

(3) Repair Procedure for CRDM Heavy Drive Rod Assemblies With Installed Breech Guide Screws That Require an Additional Lock Pin: This procedure details actions to be taken to install an additional locking pin in those assemblies whose screw had been verified tight during the inspection.

This ECA was completed under the supervision of Westinghouse personnel and under surveillance of a licensee Quality Assurance inspector.

Thirty-five assemblies passed the initial torque test and had an additional pin installed. Eighteen assemblies required set screw replacement and pin installation.

The inspector checked six assemblies in the warehouse in conjunction with a Westinghouse representative, specifically verifying serial numbers, accuracy of records and type of repair for each assembly. Of the six, two required set screw replacement and the other four received a second pin. The inspector discussed all three procedures with the vendor representative and reviewed the repair records for each procedure. Additionally, the quality assurance inspector who covered this job was interviewed and his quality assurance surveillance reports were reviewed.

The inspector confirmed that the subject CRDM repairs have been conducted in accordance with the licensee's final report to NRC Region I (SBN-897, dated November 19,1985). This CDR is therefore closed.

f. (Closed) Construction Deficiency Report (CDR 85-00-15): Service Water Spill in the Primary Auxiliary Building (PAB). As reported in Region I Inspection Report 50-443/85-25, approximately 30,000 gallons of salt water (seawater from the service water system) spilled out of a disconnected piping connection located on the top floor of the PAB. Immediate corrective action was taken by the licensee to inspect and clean all affected components and evaluate chloride contamination levels for any long-term adverse impact on safety-related components. The resident inspectors witnessed clean-up activities in progress in the days immediately following the spill and discussed the use of specific cleaning materials, drainage design, and flooding protection with the responsible engineering personnel.

Nonconformance Report (NCR) 82/762B was reviewed to not only check that corrective action to address the cause of the spill was enacted, but also that adequate chemical sampling (ie: swipe tests) of affected component surfaces and corrosion engineering evaluations (post-cleanup) had been performed. Subsequent resident inspector examinations of the spill areas, which included inside closed electrical equipment, revealed no residual effects or deterioration resulting from this event.

Licensee corrective action was immediate, complete and comprehensive and in line with the Final 10CFR50.55(e) Report to Region I (SBN-883, dated October 2?,1985). The inspector has no further questions on this issue and consicers this CDR closed.

4. <u>TMI Action Plan Requirements (NUREG 0737)</u>

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 these commitments were inspected as noted below.

a. Item II.B.1, Reactor Coolant System Vents:

The licensee shall install a reactor vessel head high point vent that can be remotely operated from the control room.

The licensee has installed a Safety Class 2, Seismic Category I reactor vessel high point vent which consists of two (2) valves, in series, that are remotely operated from the control room and are powered from an emergency bus. One of the valves (RC-FV-2881) is a solenoid-operated valve powered from train B, 125 VDC power panel 112B. The other valve (RC-V323) is a motor-operated valve powered from train B, 460 V motor control center E612.

The head vent line is orificed to 3/8" so that a failure of the vent line will not cause a loss of reactor coolant greater than the capacity of the normal charging system. The vent line discharges to the pressurizer relief tank.

The inspector verified that the system was designed in accordance with the requirements of 10CFR50.44(c)(3)(iii) and installed in accordance with the design drawings.

Specifically, the following items were checked: control room indications, power supplies to each of the two vent valves, safety classification of the power supply circuits and installation and tagging of the two vent valves. The inspector noted that the analog computer point for the temperature element on the discharge of the vent was improperly labeled. A Startup Operational Problem Report (SOPR) was initiated to correct the error.

The inspector reviewed the procedure which addresses the use of the vent. He was informed by the Assistant Operations Manager that the vent would not be used during normal operation. Thus, the only reference to the head vent operation was in the station emergency procedures. The inspector reviewed Functional Restoration Procedure (FRP) I.3, "Response to Voids in the Reactor Vessel," which addressed when the vents should be used, how they are operated and when the operator should terminate vent usage.

b. Task II.D.3, Direct Indication of Relief and Safety-Valve Position:

Reactor Coolant System (RCS) relief and safety valves shall be provided with a positive indication in the Control Room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

The licensee has installed limit switches on each of the Power Operated Relief Valves (PORVs) that operate indicating lights on the Main Control Board (MCB). Backup indication for valve position is provided by a temperature element on the common discharge of the PORVs. The position of the three RCS safety valves are monitored by an acoustic monitor that senses the acoustic emissions associated with flow in the discharge line that is common to the PORVs and the three safety valves. Backup indication is provided by temperature elements on the discharge line of each safety valve.

The inspector checked the layout of the PORV status lights on the MCB. He verified that the opening of a PORV would not only turn on the status lights but also would actuate a computer alarm. The MCB meter which displays the temperature at the discharge of the PORVs was also checked with particular attention paid to its temperature range (0-400 degrees F). The inspector confirmed the installation of display meters on the MCB for each of the temperature elements on the discharge of the safety valves. He confirmed that a high temperature reading recorded by any of these temperature elements also actuated a computer alarm. The installation of an acoustic monitor on the discharge line was also confirmed as well as its ability to actuate a computer alarm in the event that steam flow is detected in the line.

The inspector noted that the acoustic monitor and the temperature element on the common discharge were used during Hot Functional Testing to identify the fact that two of the safety valves were leaking at system operating pressure and temperature. This problem with the safety valves and inspector follow-up of its resolution is addressed in Region I Inspection Report 50-443/85-31.

c. Task II.E.1.2, Auxiliary Feedwater System Automatic Initiation and Flow Indication:

The Auxiliary Feedwater System motor-driven pumps and valves shall be powered from the emergency buses. The system shall automatically initiate when needed and those initiating signals and circuits shall also be powered from the emergency buses. Safety-grade flow indication, powered from the emergency buses, shall be provided on each of the auxiliary feedwater lines.

Seabrook Station refers to its Auxiliary Feedwater System as the Emergency Feedwater (EFW) System. The EFW system consists of one 100% capacity motor-driven pump (which is powered from emergency bus E6), one 100% capacity turbine-driven pump and flow paths which permit the transfer of water from the Condensate Storage Tank (CST) to each of the four (4) steam generators. The EFW system is designed in accordance with ASME Section III, Class 3 and seismic Category I requirements.

The EFW system can be manually controlled from the MCB or from the Essential Switchgear Room (location of the Remote Safe Shutdown (RSS) panels). It is automatically initiated by any one of three signals: Lo-Lo level on any one steam generator, loss of offsite power and a Safety Injection (SI) signal. The system is provided with safety-grade, emergency-powered, train-separated flow indicators on each of the four EFW discharge lines to the steam generators. These flow indicators are displayed on the MCB and the RSS panels.

The inspector verified that the initiating signals and circuits, as well as the manual control circuit, are nuclear safety-related (Class 1E), powered from emergency buses and train-separated so no single failure will result in the loss of EFW function. He also confirmed that the initiating signals and circuits are testable at power. Examination of the EFW pump control logic showed that the failure of the automatic initiating signals would not result in the loss of the manual initiation capability of the systems. The inspector reviewed the logic diagram for the EFW discharge isolation valves and verified that the flow indicators on each of the discharge lines will initiate closure of the isolation valves on any one of the discharge lines that is feeding a faulted steam generator (as indicated by a high flow reading).

d. Item II.E.4.1, Dedicated Hydrogen Penetrations:

Plants using external recombiners or purge systems for post-accident Combustible Gas Control (CGC) of the containment atmosphere are required to provide dedicated penetrations for this purpose.

The hydrogen recombiners at Seabrook are located inside containment and therefore this requirement is not applicable.

e. Item II.F.1, Attachment 4, Containment Pressure Monitor:

Continuous containment pressure indication must be available in the control room. For concrete containments, the scale must range from (-)5 PSIG to three times containment design pressure (52 PSIG for Seabrook).

The licensee has installed two independent channels of wide range (-5 to 160 PSIG) containment pressure monitoring, which includes meter and recorder indication on the front of the MCB. This capability supplements the four narrow range (0-60 PSIG) pressure channels used for ESF actuation.

The inspector reviewed the design drawings for as-built accuracy and noted in-place design features which distinguish this configuration as being qualified as Accident Monitoring Instrumentation in accordance with Regulatory Guide 1.97. (The licensee has taken no exceptions to Regulatory Guide 1.97 with respect to containment pressure instrumentation.)

f. Item II.F.1, Attachment 5, Containment Water Level Monitor:

Containment water level indication must be provided in the control room. Narrow Range (NR) level, used for non-accident leakage detection shall cover the range from the bottom to the top of the containment sump. Wide Range (WR) level, used under accident conditions, shall cover the range from the bottom of the containment to an elevation equivalent to a 600,000 gallon capacity.

The licensee has installed instruments to measure and display both ranges. WR sump level is provided for each of the two containment recirculation sumps. The instrument range is from zero to six feet above floor level which corresponds to approximately 665,000 gallons. The maximum liquid volume in the containment is approximately 601,000 gallons. Both meter and recorder indications are installed for each train on the MCB front panel. Two channels of narrow range sump level instrumentation is provided for the equipment and floor drains sump in containment. Level transmitter output is indicated and recorded on the rear of the MCB. The licensee, in FSAR Table 7.5.1 and Appendix 7A, has taken exception to the Regulatory Guide 1.97 guidance that these instruments be considered Accident Monitoring Instruments (AMI) because the Seabrook Emergency Response Guidelines do not require that the operating crew monitor this variable during a design basis event. The licensee's position on this exception to RG 1.97 has been communicated to NRR and will be addressed in a future SER Supplement.

The inspector reviewed the Instrument Loop Drawings (ILD) for both ranges, verifying that the as-built condition in the control room is in accordance with the design drawings. In addition, the existing instrumentation was reviewed against Regulatory Guide 1.45 and 1.97.

g. Item II.F.1, Attachment 6, Containment Hydrogen Monitor:

Indication of containment hydrogen concentration must be available in the control room.

The licensee has installed two independent hydrogen monitoring systems consisting of the two analyzers themselves, the piping to and from the containment and hydrogen concentration indicators and recorders on the main control board.

The inspector traced the entire Combustible Gas Control (CGC) system within the Containment Building and the East Main Steam and Feedwater Pipe Chase. It was verified that the system was installed in accordance with design drawings. Specifically, the following items were checked: local and control room indications, train independence and separation, containment isolation valves and test connections, the capability for grab samples, and design features to prevent the entry of water into the sample lines during accident conditions. Control room indications located on both the front and rear panels of the Main Control Board (MCB) were verified to be in accordance with the ILD. Each train has an analyzer control panel set into the rear of the MCB. These panels (1-CGC-CP-173A and 174A) have dual range (0-10% H_2/0-20% H_2) meters and also feed a common two pen recorder. A single 0-10% H_2 meter is available for each train on the front section of the MCB adjacent to the containment pressure instruments (referred to in Item I.F.1, Attachment 4).

During the review, minor discrepancies were noted in the P&ID notes. These minor errors were corrected by the issuance of ECA 08/802692A.

The licensee has deviated from Regulatory Guide 1.97 criteria in that the hydrogen monitoring capability is not continuous, but available for startup within 30 minutes. This allows the monitors to be normally isolated from containment. This exception to RG 1.97 criteria, currently being reviewed by NRR, is documented in the FSAR and has previously been approved by NRR for other licensed plants.

h. Item II.G.1, Emergency Power For Pressurizer Equipment:

The motive and control power connections to the Power Operated Relief Valves (PORVs), PORV block valves and the pressurizer level indication instrument channels shall be capable of being supplied from the emergency power sources. Also, the power connections to the emergency buses for the PORVs and their associated block valves shall be via devices that have been qualified in accordance with safety-grade requirements (Class 1E components).

The PORVs and their control circuits are Class 1E and are powered from independent 125V DC vital instrument power supplies. The motive and control circuits for the block valves are also Class 1E and are powered from independent motor control centers which are supplied by the emergency power sources. In addition, since the pressurizer level indicators provide an input to the Reactor Protection System, they are Class 1E instruments powered by the vital instrument buses.

The inspector reviewed the electrical wiring schematics for the PORVs and their block valves. He verified the connection of the PORVs and the block valves to the emergency buses by confirming the installation and proper tagging of the circuit breaker on each valve's power supply circuit. Verification of train separation of the pressurizer level indicator wiring was also done. The inspector confirmed that all of the above circuits were properly denoted as safety-related on the Conduit and Cable Schedule Program (CASP).

Findings: As a result of the above review, the inspectors verified that equipment, as installed, is in accordance with NUREG-0737 and licensee commitments. All the above items are closed.

5. Allegation Follow-up

NRC inspection follow-up of four (4) separate allegations of improper construction activities at Seabrook Station was conducted during this inspection period. In the case of two of the allegations, the licensee Employee Allegation Resolution (EAR) program was informed of the concerns and NRC follow-up consisted of a review of the EAR investigation along with independent NRC inspection of the hardware. The NRC inspection, in all four allegations, focused on the quality aspects of the field work and installed components. Thus, if certain of the alleged facts could be determined neither valid nor invalid, they were assumed true and the impacted hardware was then inspected. In all cases, no adverse impact on installed equipment or safety-related work was identified.

The general subject of each allegation and ultimate disposition of the safety concerns are annotated below:

a. Unauthorized welding repairs in the west main steam and feedwater pipe tunnel: licensee inspection of welds in this area, along with EAR program interviews with craft who had worked in the area and independent

NRC inspection of both welds and the status of pipe supports/whip restraints, revealed no unacceptable conditions. (reference: Region I letters to PSNH, dated December 19, 1984 and October 25, 1985; and PSNH responses, SBN-752, dated January 18,1985; SBN-774, dated March 4, 1985; SBN-850, dated July 25, 1985; and SBN-898, dated November 21, 1985).

b. Improper implementation of sealant installation controls in the equipment vault: immediate NRC inspection of the subject seals and interviews with QC and supervisory personnel revealed no unacceptable seals. Subsequent investigation and interviews by EAR personnel, along with YAEC QA audits, provided clarification of the required procedural controls. Hardware had not been adversely affected.

c. Questionable traceability of pipe material: NRC inspection of the subject piping spool piece (E3098-479) revealed markings that provided traceability back to the manufacturer (Dravo) records, which were available for review in the licensee records storage area. ASME Code criteria, regarding traceability, had been implemented.

d. Concrete lining cracks in a circulating water system: previous NRC inspections (reference: Inspection Reports 50-443/84-12 and 50-444/84-06) had documented investigation of such cracks and review of the licensee specification for acceptance of such conditions based upon design-based, dimensional criteria. No specific allegation of cracking beyond acceptable limits was received.

All four of the above allegations were reviewed technically by the NRC and closed based upon independent evaluation of both the NRC inspection and the EAR investigation findings. In no case were the allegations substantiated to the extent that safety-related work and processes were or could be assumed to have been adversely affected.

6. Containment Enclosure Building

a. The containment building at Seabrook is enclosed by a containment enclosure building. A five (5) foot annular space separates the outside of the containment building and the inside of the enclosure building.

The function of the enclosure building is to collect any fission products which could leak from the primary containment structure following a loss-of-coolant accident. The atmosphere of the enclosure building would then be drawn down to a negative pressure of 0.25 inches w.g. (water gauge) and processed through the Containment Enclosure Emergency Air Handling System to remove radioactive iodine and particulate material before release to the environment. This feature lowers the radiation releases after an accident to within the guidelines of 10CFR100.

Since not all of the lines that penetrate containment terminate in areas treated by the Containment Enclosure Emergency Air Handling System, the potential exists for a certain amount of leakage to bypass the containment enclosure. These potential bypass leakage paths are listed in Table 6.2-83 of the FSAR.

The inspector reviewed FSAR Table 6.2-83, draft Technical Specifications' Table 3.6.1, and Preoperational Test (PT) 37.2 (Reactor Containment Type B and Type C Leakage Rate Tests), all of which address sources of containment enclosure bypass leakage. He questioned a Startup Group Supervisor as to why containment penetration 39, which services the Refueling Cavity Cleanup System, is included in the draft Technical Specifications and in FSAR Table 6.2-83 as a containment bypass leakage path, but not in PT-37.2. Also, the personnel access hatch and the equipment hatch (as well as the auxiliary personnel access hatch built into the equipment hatch) are not listed in any of these documents as bypass leakage paths even though they meet the criteria of bypass leakage paths as defined in FSAR Section 6.2.3.2.d. The inspector was informed that the licensee had already submitted an FSAR change to NRR (reference: SBN-942, dated February 12,1986) that changed the definition of containment enclosure bypass leakage paths to include all containment penetrations and isolation valves requiring Type B and Type C testing per 10CFR50, Appendix J. This change would thus include the equipment and personnel access hatches and penetration 39, as well as numerous other penetrations, as containment enclosure bypass leakage paths. If the subject FSAR change is not approved by NRR, the licensee indicated that the questioned containment penetrations would be included as bypass leakage paths and a change would subsequently be made to the Technical Specifications, the FSAR and PT-37.2.

The inspector has no further questions on this issue at this time. Routine inspection of the test results evaluation of PT-37.2 will readdress the question of the Seabrook bypass leakage definition, particularly after NRR reviews the noted FSAR change.

b. The inspector witnessed the installation of pressure seal plates in the East and West Main Steam and Feedwater (MS/FW) Pipe Chases. He discussed with the responsible UE&C craft foreman the construction of the pressure seal plate in the West Chase. Lester/Cives Foreign Print 18032, detailing the seal plate design, was also reviewed. No problems were identified with the construction of the seal plates with respect to their approved design.

   The inspector later reviewed UE&C drawing F-805052 and FSAR Section 6.2.3 and determined that these pressure seal plates were boundary extensions of the containment enclosure building. These seal plates permit the main steam, feedwater and steam generator blowdown containment penetrations to be included in the containment enclosure. However, the inspector noted an apparent lack of a vent path between the air space behind the pressure plate and the atmosphere of the rest of the containment enclosure building. Without a connection between the air spaces, the Containment Enclosure Emergency Air Handling System would not be able to draw a partial vacuum on the air space behind the pressure plate and process that air, as designed. The inspector questioned the responsible YAEC Licensing Engineer as to when and if vents would be installed to connect the air spaces. The licensee indicated in its response that the requirement for vents between the air spaces had not been translated into the design drawings. Anticipated Modification List (AML) item 492 was subsequently issued on March 24,1986 to bore two holes through the containment enclosure wall to connect the air space behind each of the pressure seal plates with the containment enclosure atmosphere.

During the exit meeting on April 1,1986, the inspector informed the licensee Field QA Manager and representaives of both UE&C and YAEC that the design drawings for the containment enclosure building were deficient in that they failed to translate a required design feature, discussed in the FSAR, into drawing details to vent the air space behind the seal plate to the containment enclosure, so that its atmosphere would be processed by the Containment Enclosure Emergency Air Handling System. This failure represents a violation of 10CFR50, Appendix B, Criterion III (443/86-12-01).

7. Solid Radioactive Waste Management System

The solid radioactive waste management system at Seabrook is comprised of several subsystems which process evaporator bottoms, spent resins and dry active wastes for disposal offsite. To process evaporator bottoms and spent resins, the licensee has installed a system which reduces the volume of the waste products and then solidifies the end product with asphalt. This produces a solid dewatered waste form suitable for disposal offsite.

The inspector conducted a walkdown of the volume reduction/asphalt solidification system to verify its as-built configuration against design details. This system consists of three distinct subsystems: one to process spent resins, one to process waste evaporator bottoms and one to solidify the effluent of the preceding two subsystems in asphalt.

The spent resin processing subsystem takes resins from the spent resin sluice tanks, adjusts their pH and dewaters the resins by mechanical agitation and centrifuging. The end product is then sent to the asphalt extruder for solidification.

The waste evaporator bottoms processing subsystem takes evaporator bottoms from the waste concentrates tank, adjusts their pH and dewaters the material by an evaporation system (ie: a crystallizer). The end product is pH adjusted further and held up in preparation for transfer to the asphalt extruder.

In the asphalt extruder, radwaste and asphalt are homogeneously mixed on a nominal one-to-one ratio, by weight. The ratio will vary depending on the waste feed radioactivity concentration and the water content of the radwaste. The radwaste is also heated to drive off any remaining moisture. The asphalt/waste effluent is then discharged into 55 gallon drums which are capped and transferred by remote control to an adjoining shielded storage area. The licensee estimates that the storage area is adequate to store all solidified waste generated by Unit 1 for over three years of commercial operation.

During the course of this inspection, the inspector made the following observations:

-- The tag on level glass WS-LG-531 did not match the corresponding number on the Piping and Instrumentation Diagram (P&ID's). This concern was relayed to the responsible Startup Test Engineer (STE) for corrective action before final system walkdown and turnover.

-- A drainline was being used on the Waste Solidification system auxiliary boiler condensate return tank for level indication. The STE informed the inspector that the tank was installed with a site glass for level indication located on the upper portion of the tank. Since the actual operating level would be in the lower portion of the tank, the STE further stated that a bottom portion gauge glass will be permanently installed at a later date.

-- A Nonconformance Report tag (NCR) 74/2990A was still hanging on waste feed recirculation pump 1-WS-P-332B even though the NCR had been properly dispositioned on May 6, 1985. The inspector notified the STE of this fact and the tag was promptly removed to preclude further confusion.

-- The relief valve for the steam side of the crystallizer heater vents directly to the atmosphere without any radiation monitoring on the discharge line. The inspector questioned the possibility of a tube leak allowing an unmonitored radiation release to the environment. He later determined from the design data listed in Table 11.4-10 of the FSAR that any leakage would be from the shell side (50 psig) to the tube side (15 psig), thereby precluding the possibility of an unmonitored release.

-- Various relief valves located in the evaporator bottoms waste subsystem discharge directly to the floor. This design creates the potential for contamination of any equipment and material located in the release areas. One relief valve in particular, WS-V235 relieves next to a pipe chase opening in the floor, creating the possibility of showering the lower elevations with potentially contaminated water. The inspector questioned the responsible UE&C design engineer as to whether the slope of the floors in the WPB is sufficient to ensure flow toward the drains and whether all of the drains in the WPB run to the Waste Liquid Drain (WLD) system. He was informed that the discharge of the relief valves would consist of only small amounts of uncontaminated water since the reliefs are only for thermal protection against overpressurization during system shut-down. Also, all floors in the WPB slope toward the drains which in turn discharge to the Waste Liquid Drain (WLD) system.

-- There is an opening in the wall at elevation 55' of the WPB, next to the waste drum storage area. There also exists a catwalk over this wall, which allows personnel access above the storage area. The inspector questioned the licensee's ALARA engineer about the anticipated radiation levels at these points and how access to these areas would be controlled, if required. He was informed that the two locations could be interpreted as high radiation areas pursuant to 10CFR20.202 and that the licensee is investigating means to either prevent or control access over the waste storage area through the noted potential access points.

With respect to all of the aforementioned observations and inspection points, no violations were identified.

The inspector also reviewed the design and installation of an alternte waste solidification system, planned for use if the volume reduction/asphalt solidification system could not be utilized. This additional system provides for the capability to pump the waste to an alternate solidification station, located in the WPB truck bay area, for mobile dewatering and solidification, if necessary. No problems or deviations were identified with this system.

8. Preoperational Testing

The inspector reviewed the following preoperational test (PT) procedures for conformance with FSAR Section 14, other pertinent technical sections of the FSAR, and the applicable regulatory guidance:

-- PT-27.1 Fuel Storage Building Ventilation System
-- PT-27.2 Fuel Storage Building Exhaust Filter System
-- PT-33.1 Emergency Diesel Generators
-- PT-34.1 Fuel Handling Equipment
-- PT-36 Primary Containment Structural Integrity Test
-- PT-37.1 Reactor Containment Integrated Leak Rate Test
-- PT-37.2 Reactor Containment Type B & Type C Leakage Rate Tests
-- PT-38 ESF Integrated Actuation Test
-- PT-39.1 Loss of Offsite Power Test
-- PT-39.2 Loss of Offsite Power with Engineered Safeguard Features Actuation
-- PT-44 Plant Ventilation Systems Filter Testing

Test results for PTs 27.1, 27.2, 34.1 and 44, as applicable to the Fuel Storage Building (FSB), were reviewed in order to verify functionality of the tested systems prior to the storage of fuel in the FSB, as committed to in FSAR Section 6.5. The inspector examined the test procedure field changes, test exceptions and the chronological logs of test conduct. The inspector also spot-checked the data sheets for the acceptability of results per the specified criteria. With respect to PT-44, the inspector noted a licensee-identified question (reference: Request for Information, RFI 99/806163B) regarding the homogeneity of the carbon filter batches used in the test. Since Regulatory Guide (RG) 1.52 provides guidance in this area, the inspector discussed this issue with the responsible mechanical engineer and reviewed certified test reports for the carbon adsorbent material batches in question. Based upon the documented quality of each batch and evidence provided by the licensee of batch "uniformity" within reasonable tolerance, the inspector determined the intent of RG 1.52 had been met. He had no further questions on that item or any other test results evaluation with regard to the subject FSB PT tests. The inspector also conducted an as-built walkdown inspection of the FSB ventilation and exhaust systems in conjunction with the applicable PT procedure reviews.

The resident inspectors also witnessed selected sections of some of the above PTs, with inspection areas and observations noted below:

a. PT-33.1 - Field changes were first reviewed to determine whether they affected the intent of the procedure. The inspector questioned the responsible Startup Test Engineer (STE) concerning an apparent increase in the fuel consumption of the diesel generator and its impact on the adequacy of the existing 7-day fuel supply for the engine. He was informed that the FSAR had been amended to incorporate the increased fuel consumption rate and that the fuel supply was still adequate for 7-days plus a 7-hour reserve for testing. The inspector independently calculated the adequacy of the fuel supply based on the new fuel consumption rate and found his calculation to support the STE's analysis.

   The inspector witnessed preparations for initial engine starting on the B diesel generator and Section 6.13, DG-1B Overspeed & Load Rejection Tests. He had no questions concerning the conduct of the test. However, following an increase in load to 6 MW, the diesel generator load suddenly dropped 3 MW. The inspector was informed by the STE that this phenomenon had occurred before and that the licensee was in the process of trouble shooting the problem.

   The licensee later issued Work Request (WR) DGG-357 to replace the ECA box of the Woodward governor (believed to be the source of the problem) with the equivalent unit from one of the Unit 2 diesel generators. Retesting of the applicable steps of GT-E-111 and subsequent testing of the diesel indicated that the problem had been resolved. The licensee did not observe this problem with the A diesel generator.

b. PTs-36 and 37.1 - The Containment Structural Integrity Test (SIT) and Integrated Leak Rate Test (ILRT) were accomplished during a concurrent time frame and were the subject of a Region I inspection (IR 50-443/86-15) by technical specialists. During this current inspection, the resident inspectors spot-checked testing activities at the specified test pressure plateaus and during various unannounced, off-shift times, and witnessed conduct of the following test procedure steps:

   -- crack mapping and inspection of the containment exterior.

   -- integrity (leak) testing of specified containment penetration welds.

   -- pressure measurement at both the official SIT gauges and the ILRT manometer recorders.

   -- recurrent checks of containment air recirculation fan motor current measurements.

   -- hydrogen recombiner room inspections.

-- control room support activities to include computer recording of one of the Mensor quartz manometer's pressure data.

-- chronological logging of events and procedural sign-off of test criteria as it occurred.

The inspectors questioned the responsible test personnel, engineers, Authorized Nuclear Inspectors (ANI), QA/QC personnel, and contractors, as necessary, during the test conduct to clarify questions regarding both the observed and recorded data. They also spot-checked relevant UE&C specification requirements, ECA/NCR design change information, and FSAR commitments for the acceptability of conditions observed during test conduct. It was noted that a recent letter (SBN-968, dated March 14, 1986) to NRR discussed the licensee position with respect to the recording of strain gauge measurements at the containment spring line during the SIT. Also, UE&C letter SBU-9817, dated February 28, 1986, revised the peak design pressure for maximum containment accident conditions from 46.1 psig to 48.7 psig. The inspectors verified that this current design information was used in the conduct of the ILRT and that the FSAR will be revised to reflect the new analysis.

c. PT-38 - The inspector conducted an as-built walkdown inspection of the Containment Enclosure Air Handling (EAH) system in conjunction with PT-38 test conduct. He observed control room activities in progress during the test to include verification of proper component (e.g., valves and pumps) position and status with respect to the required equipment line-up. The Shift Test Director and system test engineers were interviewed regarding test progress, field change justification and the documentation of test exceptions, as necessary.

In reviewing the results being compiled as the test progressed, the inspector noted two issues which appeared to merit further review, as follows:

(1) All the interlocks associated with the RHR suction valves (RC-V22, V23, V87 and V88) from reactor coolant loops 1 & 4, as discussed in the Westinghouse system description (SD-NAH/NCH-284), had not been tested.

(2) Upon reset of the safety injection "S" signal, Control Building Air (CBA) fans (FN-16A & 16B) shut off and were isolated, despite FSAR commitment that all systems serving safety-related functions remain in the emergency mode upon removal of the "S" signal.

Further discussion with startup, engineering and licensing personnel regarding these two issues revealed that PT-38 had indeed tested the applicable systems as designed (reference: UE&C drawing M503748 and NCR 82/1260A). The question then remained whether these items had been designed in accordance with the appropriate regulatory requirements and licensee commitments.

The licensee has presented evidence of the proper design of the questioned items. Included in this evidence are excerpts from the Seabrook SER (Section 6.5.1.2) and a letter from Westinghouse discussing its position on the RHR valve interlocks. Pending further review of this information and discussion with NRR, if necessary, these two design questions remain unresolved (443/86-12-02).

d. PT-39.1 - The inspector witnessed the conduct of portions of this test through Step 6.3. This included verification of prerequisites and initial conditions by the Shift Test Director (STD), as well as the pretest briefing. Section 6.1 (event #1) de-energized all offsite power, forcing both emergency diesels to start and power their respective emergency buses. Shortly after the opening of the offsite power 345KV supply breakers, all remaining 345KV breakers opened up on low SF6 pressure and would not reclose. Offsite power was restored after temporary power was rigged to the SF6 compressor for one of the 345KV breakers, allowing it to close once the low pressure trip had cleared. This failure was attributed to SF6 system leakage and is being addressed by a design change. Licensee action on this matter is not yet complete and will be the subject of continued inspection. The licensee's operators, maintenance, and startup personnel demonstrated good plant knowledge and understanding in devising a safe and workable temporary modification to restore power. Plant tagging and operating procedures were properly followed and the short term corrective action taken by the plant staff was noted to be effective.

With respect to all of the preoperational testing items inspected, as discussed above, no violations were identified.

9. Operator Training

The inspector observed two simulator sessions presented as part of the licensed operator requalification training program. Each operator receives six weeks of instruction during the annual cycle. The different phases presented include classroom and simulator training to augment actual control room manipulations. The licensee has modified the current program to take advantage of tests which are only conducted during the Startup Test Program. Specifically, Phases 3 and 4 will be conducted on shifts in the plant during low power physics testing and power ascension testing. This training is designed to allow licensed operators and instructors the opportunity to enhance their understanding of plant systems and operating characteristics.

The inspector discussed the program and schedule with the Training Center Manager and Simulator Supervisor. Based on the above discussions and the simulator observations, the inspector has no concerns.

10. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. An unresolved item disclosed during this inspection is discussed in Paragraph 8c.

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 April 1, 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.