U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket/Report No.: 50-443/89-09 License No.: NPF-67

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

Facility: Seabrook Station, Unit No. 1, Seabrook, New Hampshire

Dates: August 18 - October 10, 1989

Inspector: A. Cerne, Senior Resident Inspector

Project Engineer: A. Chu, NRC Office of Nuclear Reactor Regulation

Approved By: Ebe C. McCabe, Jr 10/23/89
Ebe C. McCabe, Chief, Reactor Projects Section 3B Date

Areas Inspected: Operational safety, maintenance & surveillance, radiological controls, security measures & logs, quality verification activities, reportable events, and a technical review of certain design modification activities.

Results: Maintenance and surveillance were well managed, except for one questionable operational decision relating to the emergency power supply for the operable diesel generator fuel transfer pump (section 4.2).

There was timely investigation, effective response, and comprehensive corrective action planning for a Radiological Occurrence Report concerning radiography access controls (section 5.1) and for evidence of improper handling of a radiological survey by a health physics technician (section 5.2).

An active quality verification program was evident and included trending and internal reviews, and corrective action initiatives (section 7.1).

Two licensee reports require further actions and subsequent NRC review. One concerns a self-identified Technical Specification violation regarding the Condensate Storage Tank Enclosure (section 8.1) and the other involves a 10 CFR 21 evaluation of a potential Atmospheric Steam Dump Valve deficiency (section 8.2). In addition, licensee actions regarding evaluation of missed monitoring instrumentation testing is considered unresolved (section 8.1).

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DETAILS

1. Persons Contacted - New Hampshire Yankee (NHY)

   E. Brown, President and Chief Executive Officer
   *J. DeLoach, Executive Director of Engineering and Licensing
   *B. Drawbridge, Executive Director of Nuclear Production
   T. Feigenbaum, Senior Vice President and Chief Operating Officer
   J. Grillo, Operations Manager
   R. Hanley, Operations Training Manager
   *T. Harpster, Director of Licensing Services
   *D. Moody, Station Manager
   *J. Peschel, Operational Programs Manager
   *N. Pillsbury, Director of Quality Programs
   J. Vargas, Manager of Engineering
   J. Warnock, Nuclear Quality Manager

   * Attended exit meeting conducted on October 10, 1989.

   Other licensee and contractor personnel were also contacted.

2. Summary of Activities

   2.1 Senior Resident Inspector Activities

   One senior resident inspector (SRI) was assigned to the site during the entire inspection period. The SRI conducted routine and unannounced backshift, including deep backshift, inspection activities (112 total hours, with coverage provided during 12 backshift hours and 2 deep backshift hours). Certain licensee sponsored training of its employees was monitored, including procedure compliance training and a lecture on core values and work ethic presented by the NHY President and Senior Vice President. The SRI also participated in the inspection of the licensee's partial-participation annual emergency preparedness exercise on September 27, 1989; attended the public meeting on September 6, 1989 and the enforcement conference on September 7, 1989 held in response to the Region I Augmented Inspection Team inspection findings; and participated in the Systematic Assessment of Licensee Performance Board deliberations for Seabrook on August 23, 1989.

   2.2 Visiting Inspector and Management Activities

   An NRC Systematic Assessment of Licensee Performance (SALP) Board convened at the NRC Region I in King of Prussia, Pennsylvania on August 23, 1989 to evaluate the performance of activities at Seabrook Unit 1 for the period from August 1, 1987 to June 30 1989. The SALP Report (50-443/87-99) was issued on September 26, 1989.

A management meeting between the NRC and licensee personnel was held on September 6, 1989 to discuss the results of the NRC Region I Augmented Inspection Team (AIT) inspection 50-443/89-82. This open meeting was held at the University of New Hampshire in Durham, New Hampshire and included a public comment period at the conclusion of the discussion between the NRC and the licensee.

An enforcement conference was held in the Region I Office in King of Prussia, Pennsylvania on September 7, 1989. This meeting between NRC and licensee personnel was convened to discuss the assessment of safety significance, root cause(s), and interim and final corrective actions for the potential violations identified in AIT Inspection Report 50-443/89-82. An Enforcement Conference Meeting Summary was issued on September 19, 1989.

Beginning on September 25, 1989, a Region I physical security inspector conducted an unannounced five day inspection of the licensee's security controls, equipment and plan implementation. A meeting was held with licensee management on September 29, 1989 to discuss the results of this inspection. These results will be documented in Region I Inspection Report 50-443/89-12.

On September 26, 1989, the Region I Reactor Project Section 3B section chief toured the site and held discussions with senior licensee managers and control room operators.

On September 27, 1989, the licensee conducted a partial-participation annual emergency preparedness exercise. NRC observation and inspection of this exercise was performed by a team of five Region I and headquarters personnel, beginning with an announced entrance meeting and briefing with licensee personnel on September 26, 1989. The licensee conducted a drill self-critique and were informed of preliminary NRC inspection results at an exit meeting on September 28, 1989. These results will be documented in NRC Region I Inspection Report 50-443/89-10.

On October 4, 1989, a project engineer from the NRC Office of Nuclear Reactor Regulation commenced a three day visit at the site to review the status of certain issues documented in supplements to the Seabrook Safety Evaluation Report (SER). The visit results are discussed in section 9 of this report.

2.3 Plant Activities

The plant remained in operational mode 5, cold shutdown, with primary coolant temperature between 120F and 140F and the reactor coolant system vented at the top of the pressurizer. Maintenance and modification activities shifted from train 'A' to train 'B' equipment as the train 'A' residual heat removal system was returned to service. Major work was conducted on the control building air, containment building spray, service water and component cooling water systems.

2.4 Reorganization of Licensing Services

The inspector was briefed on a planned reorganization of the NHY Licensing Services group. The draft charter and expanded functions of the licensing group were reviewed and discussed with the Director of Licensing Services. The licensee indicated that revision to the Final Safety Analysis Report is required as a result of this reorganization. A licensee review is being conducted to submit the necessary FSAR changes to the NRC Office of Nuclear Reactor Regulation.

3. Operational Safety

3.1 Plant Operations

The inspector observed plant operations during regular and backshift inspections of the control room and during routine tours of the plant. In the control room, plant logs, night orders, technical specification action statement status, and alarm conditions were reviewed and operators were interviewed regarding control board indications and system lineups. The critical safety function (CSF) status and its applicable status trees were examined during control room visits to check operator understanding of plant instrument availability and system status. Even though the use of the displayed CSF status was not applicable to the current operational mode 5 conditions of the plant, control room operators on different shifts were all cognizant of the reasons why specific CSF parameters displayed unreliable or abnormal indications.

The inspector also discussed with operations personnel a recent revision to the Operations Management Manual (OPMM, Revision 20), providing guidance for mode 5 and 6 operation with respect to equipment operability for conditions not covered by the Technical Specifications. For example, "in mode 5 with the RCS loops filled, the required operable RHR pump shall be the pump associated with the operable diesel generator." The inspector's observation of shift activities confirmed operator cognizance and compliance with the new OPMM provisions.

The inspector reviewed an operations department report of control room phone unavailability, for both the Emergency Notification System (ENS) and Nuclear Alert System (NAS) lines, to the NRC headquarters and state notification centers, respectively. The licensee's internal review of phone unavailability over the last nine months identified no root cause or pervasive long term problem, but determined that further review of unavailability was prudent. This review will be accomplished over an additional nine month period, after which an analysis of the need for corrective measures will be accomplished. The inspector discussed with the licensee their plan for further root cause identification and review of control room phone problems and had no further questions. It was noted that, on both October 2 and 3, 1989, the NAS phone was taken out of service because of train 'B' electrical bus outage work. The inspector was notified of this work in advance, as was the NRC headquarters duty officer in accordance with 10 CFR 50.72. The inspector verified that backup phone lines were available for emergency notification during the time the NAS was out of service.

In anticipation of potential severe weather conditions associated with the approach of Hurricane Hugo to New England, the inspector discussed with operations personnel the licensee's severe weather preparations. Closure and temporary sealing of the outside hatch plugs to the train 'B' equipment vault were verified. The inspector reviewed abnormal operating procedure OS 1200.03 establishing plant provisions to minimize the adverse effects of severe weather conditions and conducted an unannounced deep back shift inspection of the plant on the night of September 22-23, 1989 to ensure plant readiness and operator awareness of potential weather problems. No NRC concerns were identified.

With respect to overall plant operations during this inspection period, the inspector identified no violations or unresolved safety questions and confirmed both acceptable operator performance in routine mode 5 evolutions and knowledgeable response to NRC questions regarding system lineup and component status.

3.2 Plant Tours

The inspector conducted several general inspections throughout the plant to witness work activities in progress, check equipment status and required valve positions, and spot check housekeeping and overall administrative controls of outage work conditions. The following observations and conclusions were made:

-- Plant cleanliness was acceptable, but housekeeping control of small work items and debris, particularly within the containment building, required additional attention. The inspector noted improvement in this area as the inspection progressed.

-- Control of temporary scaffolding and staging equipment within the station were in compliance with Administrative Procedure MA 4.10 and appeared adequate to ensure system and component operability were not adversely affected by temporary equipment use.

-- Roving fire watch patrols were being implemented in accordance with Technical Requirement No. 11 for hourly checks of fire seals and other fire rated assemblies rendered nonfunctional by authorized work activities.

-- Valve positions (e.g., accumulator isolation valves closed), valve system lineups (e.g., service water and fuel supply valves open for required diesel generator operability), and locked valve status (e.g., pressurizer vent for the reactor coolant system locked open) -- all were spot checked and found to be properly positioned, controlled and the status correctly documented in accordance with Technical Specification and system lineup requirements.

-- Certain sticker/tags located on equipment installed within the plant (e.g., penetration seals serving as fire barriers) incorrectly referred to controls and removal authorization specified in old construction procedures which have been superseded by station procedures. Discussion with the licensee indicated their intent to remove the outdated procedures from the Unit 1 applicable procedure control program and revise the affected sticker/tags to reflect the correct references.

With respect to the areas inspected during the plant tours and to the questions raised by the inspector on specific issues, the licensee was responsive to both the questions and any items requiring additional management attention. No violations were identified and no unresolved safety concerns remain open.

4. Maintenance/Surveillance

4.1 Maintenance & Design Change Implementation Activities

The inspector witnessed specific maintenance and design change work in progress in the plant, reviewed the associated work request and design coordination report implementation packages, and discussed the work activities with the cognizant craftsmen and technical support engineering personnel. In selected cases, records of installation, maintenance and testing were reviewed to verify compliance with the applicable code and operability requirements. The licensee's nuclear quality organization's involvement in the observed work activities was verified, either by documented evidence of hold points and inspection or by observing quality control personnel in the performance of their duties. The inspector also reviewed licensee work procedures and related design documents, as necessary, to confirm the acceptability of the work. The following list represents the different activities, equipment and records examined by the inspector.

Document*	Work
MS0514.05	MOVATS testing on residual heat removal (RHR) valve RH-FCV-610.
89W003902	Damaged seal tight fitting on limit switch for service water valve SW-V-4.
DCR 86-081	Installation of steam generator secondary manway access platforms.
ECA 99/114572A	Valve testing and setpoint data for the pressurizer safety valves.
DCR 86-032	Rerouting tubing for a steam generator level transmitter.
89W001371	Service water valve SW-V-18 flange bolting rework.
DCR 87 311	Work associated with installation of new containment building spray (CBS) and RHR check valves.
88W001765	Inspection of service water valve SW-V-5 worm gear.

*(Note: MS denotes a maintenance procedure, W indicates a work request, and DCR and ECA stand for a design coordination request and engineering change authorization respectively.)

The inspector evaluated the above work activities against the applicable design specification, code and standard requirements. Pressurizer safety valve testing was verified to be in compliance with NUREG-0737 commitments. Craftsmen appeared knowledgeable of work controls and standard practices. The cognizant engineers were able to answer all the inspector's questions on design torquing, valve testing, and visual examination criteria. Coordination of the field work with control room activities was evident from the standpoint of tagging controls, planning for specific train and subsystem outage times, and operator cognizance of field activities.

The inspector also witnessed a demonstration of strain gauge testing of Limitorque valve operator stroke settings. This technique is being developed by the licensee to replace MOVATS testing of motor-operated valves and appeared to be in compliance with the guidance and requirements of NRC Bulletin 85-03, NRC Generic Letter 89-10, and ASME Code Section XI. The engineering principles serving as bases for the use of strain gauge testing were explained by technical support personnel and the status of allowable ranges of thrust calculations was discussed. The inspector concluded that the licensee's design of this program exemplified initiative in developing a test uniquely fitted to the problems and applications of motor-operated valves at Seabrook. Also, this development work was well timed to comply with the Generic Letter 89-10 mandated reviews.

With regard to all of the above maintenance and design change activities discussed above, the inspector identified no violations and had no unresolved safety concerns. The licensee's routine program of controls appeared effective and properly implemented.

4.2 Surveillance Activities

The inspector witnessed a portion of train 'B' diesel generator monthly operability surveillance (OX1426.05) in the control room and inspected the final positions of certain diesel generator jacket water and fuel oil and service water valves to ensure operability after test and required rework completion. One problem was noted during this test: the train 'B' fuel oil transfer pump, DG-P-38B, failed to operate in the automatic mode and transfer fuel oil from the train 'B' storage tank to its respective day tank. Manual operation of the pump was possible. However, in order to repair the automatic control problem, DG-P-38B was tagged out of service and electrical supply breaker opened.

Technical Specification 3.8.1.2.b requires one operable diesel generator with an operable fuel transfer pump in Mode 5. The train 'A' diesel generator was operable at this time. The operators then lined up the train 'A' fuel transfer pump (DG-P-38A) to supply the train 'B' day tank through cross-connected piping, as allowed and discussed in FSAR section 9.5.4.3. Approximately one-half hour after setting up the cross connection, the train 'A' diesel generator was taken out of service for maintenance on its electrical bus. However, the safety evaluation in FSAR section 9.5.4.3 states: "all the motor-driven pumps are powered from the bus on which the diesel generator it serves is connected." The system lineup directed by the operators was acceptable only to the extent that offsite power was available since, with the train 'A' electrical bus 5 out of service, no emergency power supply was available to DG-P-38A to keep the train 'B' diesel generator day tank supplied with fuel. Thus, even with the cross-connected fuel oil lineup, the train 'B' diesel generator was inoperable in accordance with the provisions of Technical Specification 3.8.1.2.b and the Action Statement should have been entered.

All the Action Statement requirements required by the subject limiting condition for operation were in fact being met by existing plant conditions. Additionally, a quality assurance surveillance of this operational activity was in progress and concurrently documented the

NRC inspector's concern regarding diesel generator operability in surveillance report QASR 89-00733. A subsequent QA engineering evaluation of this situation confirmed that the subject Action Statement should have been entered.

The inspector discussed this evolution with Operations management who corroborated the position that the operators' action had not been prudent. No adverse safety impact was evident since electrical bus 5, and thus the train 'A' diesel generator, could have been restored to operable status in a short period of time, certainly before the train 'B' day tank would have approached low level conditions upon loss of offsite power. The licensee is currently developing a Technical Clarification to be issued as guidance to the operators to preclude recurrence of similar situations. The inspector raised some additional questions regarding assumptions made about fuel oil storage tank level and fuel filling/settling times in the safety evaluation of FSAR section 9.5.4.3. The licensee responded with appropriate revisions to the train 'A' and 'B' diesel generator fuel oil system operating procedures, OS1026.05 and OS1026.13.

The inspector had no further questions and considered both the operations department actions appropriate to the concerns and the QA department involvement as evidence of an active quality program.

Other surveillance activities checked by the inspector include a review of the weekly diesel generator train 'A' & 'B' surveillance report (procedure OS1426.12), and discussion with operators on shift regarding the conduct and independent inspection of the affected valves; a review of the most recent monthly surveillance (procedure OX1401.11) of the pressurizer vent valve, RC-V-468, and independent inspections of valve position and locking device by the inspector during both August and September, 1989; and examination of the pressurizer safety valve surveillance procedure (EX1804.044) with respect to both Technical Specification and ASME Section XI in-service testing requirements.

With respect to the latter inspection item, the inspector reviewed a QA surveillance report (QASR 89-00216) documenting review of procedure EX1804.044 to the guidance and criteria provided by NRC Information Notice (IN) 88-68 on setpoint testing of the pressurizer safety valves. No problems were identified by this QA surveillance. The inspector noted that EX1804.044 allows the pressurizer safety valves to be setpoint tested in place with pneumatic assist equipment. However, discussion with licensee technical support personnel revealed their plans not to use this option, but rather to send the valves off-site to have setpoint pressure tests performed. The inspector confirmed that, in addition to the three installed pressurizer safety valves, five spare valves are available for use, indicating that replacement of the valves is the preferred option over attempting to adjust setpoints in place using the pneumatic assist equipment. Thus, the concerns raised by IN 88-68 appear to have been satisfactorily addressed by the licensee's setpoint testing program. The inspector has no concern regarding the current operability of the three pressurizer safety valves, since records indicated that they were properly steam set at the factory by the Crosby Valve & Gage Company.

The inspector has no questions regarding the scope, scheduling or conduct of the licensee's surveillance activities. No violations were identified and the overall conduct of the surveillances that were examined appear to be well controlled.

5. Radiological Controls

5.1 General

The inspector observed routine access controls into the radiologically controlled area (RCA) of the plant and noted proper temporary radiological posting and controls at areas where the RCA had been opened for maintenance and outage work. The inspector spot-checked the monitoring controls for tools and major equipment egressing the RCA by health physics personnel. Personal item scanning techniques were discussed with health physics personnel at the HP checkpoint.

The inspector reviewed a radiological occurrence report (ROR) 89-17 documenting the inadvertent presence and possible exposure of one licensee employee during radiography within the plant. Analysis of the individual's dosimetry and calculation of the maximum dose to which the individual could have been exposed (i.e., approximately 2 mR) revealed no overexposure conditions. This was verified by duplicating the conditions in question with a dosimeter placed where the individual was located.

The root cause of this incident was identified to be a failure on the part of the radiographers to conduct a three-dimensional search of the area where the radiography was to take place. The individual was positioned on a platform above the shot prior to the radiographers' establishment and posting of the high radiation area. The contract radiographers were counseled by the Health Physics Department Supervisor and three dimensional searches of radiography zones, to include verbal attempts to identify personnel in the area, were emphasized. The inspector discussed with HP personnel the posting of radiography areas in general, and specifically checked how the RHR vault would be controlled, given the unshielded height, and access ladders and doors into that area. No problems were identified. The corrective action to this incident appeared appropriate to both root cause and radiological concerns.

The inspector had no further questions on this issue.

5.2 Investigation of a Health Physics Technician

On September 13, 1989 the inspector was informed by licensee management of the investigation and subsequent resignation of a health physics technician, based upon evidence that the individual submitted a documented report of a radiological survey that actually had not been performed. The licensee's investigation included a review of all work accomplished by the subject employee since he began work with New Hampshire Yankee at Seabrook Station. No Technical Specification surveillance errors, unexplained radiation monitor setpoint anomalies, or high radiation door control problems were identified. The licensee's review concluded that this individual's questionable performance raised no Technical Specification compliance questions and had no adverse impact on public health or worker safety.

Since this represents the second recent incident (reference: inspection report 50-443/89-08, section 6.2) of this nature, the licensee conducted a root cause analysis of both cases. Other investigative and corrective action undertaken by the licensee included interviews with the station health physics technicians and evaluation of other health physics work activities, briefings of the station managers, conduct of an independent cause and effect analysis, and development of a performance monitoring program to assure the detection of similar occurrences.

The inspector reviewed the licensee's action plan for implementing the above items. The overall plan appeared well directed. While no adverse safety impact or violations of license conditions or licensee management inadequacy were identified in these cases, both incidents emphasize the need for ongoing independent verification and management monitoring activities commensurate with the safety-related importance of the work being performed.

The inspector had no further questions on this issue.

6. Security

The inspector conducted a review of the licensee's Security Log for the second and third quarters of calendar year 1989. Entries were evaluated against the criteria specified in NUREG-1304 and Regulatory Guide 5.62. Evidence of an active chemical screening program in accordance with the NHY Fitness for Duty policy was noted. In the latter regard, the inspector also observed drug detection dog search activities within the protected area.

The inspector noted adequate guard posting for compensation of inoperable or open vital area security doors. In one case of guard response, where personnel exited a specific security zone and the door failed to close because of differential pressures, the inspector observed a guard respond quickly. Signs were posted at that door to alert personnel to door closure problems, a work request was issued for evaluation and repair, and a guard was subsequently assigned to attend to the door as a compensatory measure. In another example, the inspector witnessed licensee personnel action in regard to an inoperable card reader. While two personnel ignored the failed light, a third licensee individual brought the problem to the attention of health physics and security personnel and a sign was posted until the card reader was repaired. In this case, the subject door represented neither a HP nor a security door problem, but the card reader had been installed for personnel accountability and was intended to be relocated in accordance with a design change.

Additionally, the handling and physical security of safeguards information was discussed with security department personnel. One questionable incident of control was investigated, logged and properly handled. The inspector raised another question of a safeguards information nature regarding a portion of the protected area fencing. This issue is currently being researched by the licensee and NRC headquarters personnel and does not currently represent a deviation from Security Plant requirements.

The inspector found the licensee security staff responsive to any questions or areas of concern. Overall, the security program was effectively implemented.

7. Quality Verification Activities

7.1 Audits, Trending & Corrective Action Followup (RI-89-A-112)

The inspector reviewed the Joint Utility Management Audit (JUMA) report documenting the results of the August 1989 review and evaluation of QA/QC activities at Seabrook Station by an audit team comprised of members from other utilities. The Quality Trend Analysis Report, covering the first half of 1989, was also reviewed and evaluated in light of the recent NRC assessment of Seabrook activities (reference: SALP Report, 50-443/87-99). Additionally, internal licensee audit reports and compliance reviews (e.g., GE HFA and Agastat relay investigations) were spot-checked for the scope and completeness of QA inspection coverage. Overall, an active QA program with involvement in several technical issues and all plant disciplines was in evidence. QA trending of problem areas correlated well with NRC findings, and attempts to implement corrective measures had been initiated by licensee management through procedural compliance training, procedure consistency reviews, and organizational goals for personnel commitments to excellence. The effectiveness of QA corrective actions and management initiatives for self-identified problems will remain an area for future routine NRC inspection in accordance with inspection program provisions.

In regard to a specific Corrective Action Request, CAR 89-0008 (RI-89-A-112), the inspector conducted a follow-up inspection of the identified deficiency. In this particular case, a personal computer (PC) program was being utilized to sum the individual reactor containment penetration leakage results, as measured and calculated by the Type 'B' and 'C' leak rate testing performed in accordance with engineering surveillance procedure, EX1803.003. The inspector reviewed and evaluated the procedure to the criteria of 10 CFR 50, Appendix J. Technical support personnel were interviewed regarding the conduct of individual penetration test measurements. The resulting calculations and review against acceptance criteria were done by hand and verified. Only the summation of the individual results and comparison to total allowable leakage criteria was performed utilizing the PC program, which had not been independently verified in accordance with 10 CFR 50, Appendix B. At the request of the inspector, the licensee performed hand calculations to validate the accuracy of the subject PC program. The results of these calculations, to include consideration of standard deviations associated with the individual measurement errors, revealed an accuracy of the software in question to two decimal points. The inspector also checked the adequacy of licensee access and control of the software package and identified no technical problems. In the case of specific example cited by CAR 89-0008, no technical deficiencies, errors, or safety-related concerns were identified.

However, the question of generic software controls for PC and/or station computer usage in the performance of safety-related activities is a valid QA issue. A memorandum from the Nuclear Quality Manager to the Information Resource Group and Computer Systems Managers documents the need for additional computer software quality configuration management and controls. The inspector has been informed of the licensee's intention to address this self-identified concern from an overall NHY programmatic approach. The inspector had no questions regarding licensee actions on this issue to date and, as prescribed by the NRC inspection program, will follow future activities during routine inspections.

7.2 Field Assembly Inspection & Records Followup

The inspector examined structural supports located on the refueling deck inside containment. These structural saddles serve as the support for the containment equipment hatch when it is removed from position in the containment wall. The support assemblies are currently anchored to the containment liner wall by high-strength (ASTM A325) bolts. The inspector noted, however, that although long slotted holes had been utilized in the bolted assembly, no plate washers had been installed in accordance with the AISC Specification for structural joints using ASTM A325 bolts. While the subject support assembly does not have a safety-related function, USNRC Regulatory Guide 1.29 calls for it to be seismically designed and for the appropriate 10 CFR 50, Appendix B criteria to be applied to its fabrication and installation.

However, upon the request of the inspector, the licensee could produce no quality assurance records documenting inspection of this support. Discussion with licensee engineers indicated that the lack of the required plate washers appeared to be an isolated oversight. No other assemblies were found with a similar deficiency and, in fact, the inspector noted ongoing field work which required slotted holes in structural bolted connections and for which plate washers had been supplied. A work request (89X004667) was issued to replace the bolts and install the missing washers. An additional work request (89W004850) requires that an inspection of the welds and dimensional checks of the equipment hatch saddles be performed per engineering recommendations. This latter inspection is intended to confirm acceptable fabrication and provide quality documentation of the construction work in accordance with 10 CFR 50, Appendix B.

The inspector reviewed the noted work requests and discussed licensee actions on this issue with both QA and engineering personnel. He had no further questions and considered the licensee corrective measures appropriate to the safety significance of the item.

8. Review of Licensee Reports

8.1 Licensee Event Reports (LER)

(Closed) LER No. 89-010: ESF Actuation - Diesel Generator Start. This spurious event was discussed in section 6.4 of Region I Inspection Report 50-443/89-08. The inspector reviewed the LER which confirmed the previous determination of the root cause as personnel error. The undervoltage signal generated for the emergency bus, resulting from the compartment door opening, caused the spurious diesel generator starts. Emergency diesel generator 1B started as required and all safety systems functioned as designed. Corrective action included additional labeling on the subject fuse compartments, warning that opening of the potential transformer fuse cubicle doors results in deenergization of the associated electrical bus. Additional training of all auxiliary operators regarding proper potential transformer fuse operations is also planned.

The inspector had no questions regarding the cause or significance of this event and no concern regarding the adequacy of licensee corrective actions. This issue is closed.

(Closed) LER No. 89-011: Unsealed Penetrations in the Condensate Storage Tank Enclosure. On September 5, 1989, a four hour notification was made in accordance with 10 CFR 50.72(b)(2)(i) to inform the NRC that evidence was not available to confirm the fact that all penetrations through the condensate storage tank (CST) enclosure were watertight. This enclosure is a concrete structure around the CST which, as discussed in FSAR sections 3.8.4.1.g and 9.2.6.3, is intended to be "capable of retaining the contents of the tank should the tank rupture."

On October 5, 1989, the licensee submitted an LER discussing this deficiency and confirming three unsealed piping penetrations in the CST enclosure. Thus, during two previous periods of time, when the plant was at or above mode 3 for testing, the limiting condition for operation of Technical Specification 3.7.1.3 requiring "a concrete CST enclosure that is capable of retaining 212,000 gallons of water" had been violated. The root cause of this condition is being investigated and a supplemental LER is to be submitted by the licensee. Corrective actions include sealing the penetrations prior to future entry into mode 3.

Although a supplemental report is due on this event, this LER is considered administratively closed. Licensee root cause analysis and corrective actions are considered an unresolved item (89-09-01). Licensing issues which relate to this event will also be tracked with this item. Pending completion of corrective actions by the licensee, further assessment by the NRC, and understanding of the complete design and operability considerations, this issue remains unresolved.

(Open) Potential LER: Failure to perform Technical Specification Surveillances. On October 3, 1989, the licensee identified the fact that certain radioactive liquid effluent and gaseous effluent monitoring instrumentation surveillances had not been performed as required by Technical Specification 3/4.3.3.9 and 3/4.3.3.10 in the time intervals prescribed. As documented in Station Information Report (SIR) 89-061, as of September 27, 1989, the Technical Specification operability requirements for the subject instruments to be in service "at all times" were not satisfied. The licensee is reviewing whether the Action Statement requirements were being complied with coincidentally when the limiting conditions for operation were violated.

The licensee conducted and completed satisfactorily the required surveillance tests on October 3, 1989. Reporting in accordance 10 CFR 50.73 is planned. This item remains unresolved until the licensee's evaluation is complete and is evaluated by the NRC (89-09-03).

8.2 Part 21 Reports

(Closed) 10 CFR 21 Report 88-00-01: Limotorque Motor Operator Worm Gear Defects. On March 18, 1988, Limitorque Corporation filed a 10 CFR 21 notification concerning potential defects in the worm gear castings of their Type H3BC valve actuators. The licensee identified two valves in the service water system, SW-V-4 & 5, as having H3BC actuators. Work requests were issued to disassemble the two affected valves and inspect the subject worm gears.

The inspector reviewed the completed work request (88W001764) for valve SW-V-4 and inspected the valve after reinstallation in the plant. The disassembly of SW-V-5 was inspected in progress and the worm gear was visually examined by the inspector. The work request (88W001765) for SW-V-5 was reviewed as was the referenced maintenance procedure MS0519.66. The inspector discussed work controls with the maintenance technicians, observed QC inspector availability and hold point assignments, and was provided engineering response to questions regarding the correct acceptance criteria to be used in the worm gear examination. In the case of both valves, no defects were found.

The inspector has no further questions regarding licensee response and action to the 10 CFR 21 report. This item is closed.

(Open) Engineering Evaluation No. 89-026: Atmospheric Steam Dump Valve (ASDV) Failure Evaluation for Reportability in accordance with 10 CFR 21. On April 5, 1989, USNRC Information Notice No. 89-38 was issued to describe the failure of ASDVs at the Palo Verde units. Subsequently, on June 21, 1989, Control Components Inc. (CCI), the ASDV manufacturer for both the Palo Verde and Seabrook ASDVs, issued a letter to Seabrook indicating a potential problem with the Seabrook ASDVs based upon the Palo Verde incidents. Although the specific cause of the Palo Verde failures was not determined, the connection to the Seabrook valves was the fact that significant piston ring leakage would cause the ASDVs to fail to open upon demand.

The inspector reviewed NHY Engineering Evaluation No. 89-026, which concluded that the identified ASDV problems were not a substantial safety hazard reportable under 10 CFR 21. Although it was agreed that significant differences existed between the Palo Verde site, where the problems occurred, and the conditions and performance history of the Seabrook ASDVs, the inspector questioned the licensee analysis in four specific areas:

(1) assumption that the failures are random and therefore will not occur to more than one valve at the same time.

(2) reference to a probabilistic analysis in evaluating a deterministic problem.

(3) reliance upon operator action to manually open a failed ASDV.

(4) the decision not to implement the CCI recommendations for ASDV modifications.

The licensee indicated that a monitoring program would be implemented to evaluate the ASDV performance over time and that additional testing would be conducted to verify operability under mode 3 conditions. While no evidence exists to suggest that the Seabrook ASDVs have a design defect that requires repair, the notification from the ASDV vendor suggests enough doubt that the proposed licensee actions on this issue should be formally submitted to the NRC. Pending submittal by the licensee of their plan to address the issues raised by Information Notice No. 89-38 and the notification of potential problems by CCI, this issue is unresolved (89-09-02). The correctness of licensee nonreportability under 10 CFR 21 is not in question here, but the acceptability of the proposal to address this potential problem is an issue that merits formal licensee documentation of its position and review by the NRC.

8.3 Reportable Events at Other Plants

The inspector examined licensee engineering reviews of two reportable events at other nuclear sites. The first involved electrical faults propagating back through the supply system and causing a spurious trip of the electrical feeder breaker. At Seabrook Station, electrical breaker calculations have been performed to ensure proper coordination between the breakers, fuses and loads in the electrical supply system. Thus, the load protection is designed to prevent one fault from improperly taking down an entire supply circuit.

In the second case, another plant reported that its pressurizer auxiliary spray line had not been analyzed for temperatures it may be subjected to under conditions of maximum letdown, minimum charging and no reactor coolant pump operation. At Seabrook Station, the piping downstream of the regenerative heat exchanger, to include the pressurizer auxiliary spray line, is designed to withstand temperatures in excess of the most adverse operating conditions, including those associated with maximum letdown and minimum cooling.

The inspector reviewed the licensee follow-up on the design questions raised by the two noted events at other plants. The Seabrook-specific component design criteria were spot-checked. No unresolved safety issues were identified and the inspector had no further questions.

9. Technical Review of Modifications

9.1 Control Building Air (CBA) System

As a result of a deviation from FSAR commitments in the design of the CBA system identified in 1986, and a subsequent violation with respect to maintaining the control room at a positive differential pressure, the licensee committed to provide for NRC staff review the details of modifications to the CBA system. In a letter dated January 22, 1988, the licensee described the proposed CBA modifications and in another letter, dated March 30, 1989, stated that the work would be completed by September 30, 1989.

During this inspection, a project engineer from the NRC Office of Nuclear Reactor Regulation (NRR) examined the modified CBA system. The NRR engineer found that the proposed additional HEPA filter F-8038 and the proposed bypass piping with two back draft dampers were installed. The original two purge lines were capped off and their associated purge valves removed. In the report of the "Control Room Area Ventilatior System 18-Month Surveillance" dated September 29, 1989, NRC review found that the flow balance of the modified CBA system is within the acceptance criteria of the Seabrook Technical Specifications, 1100CFM +/- 10%. Based on the September 29, 1989 surveillance report, emergency filter train 'A', CBA-F-38 had a total flow of 1193 CFM which consisted of 573 CFM make up air and 620 CFM recirculation air. Emergency filter train 'B', CBA-F-8038 had a total flow of 1173 CFM which consisted of 579 CFM makeup air and 594 CFM recirculation air. These test results indicated that the makeup air was within the design value of less than or equal to 600 CFM.

The control room maintains a positive differential pressure (dp) to its adjacent areas during normal and emergency operation modes. The surveillance test indicated that the Control Room Area to outside dp was 0.15 inches water gage (WG) and the Control Room Area to Cable Spreading Room DP was 0.16 inches WG. These are greater than the Technical Specification requirement of 0.125 inches WG.

On October 3, 1989, the control room was in the emergency mode of operation due to an electrical bus E6 outage for maintenance. The NRR engineer observed that the emergency filtration train operation maintained positive control room dp.

Based on the above findings, this inspection concluded that the implemented modifications on the control room resulted in an acceptable surveillance test and, therefore, these modifications are acceptable with respect to licensee commitments for the CBA design modification. Additionally, because of the ongoing CBA modifications, the Action Statement requirements of Technical Specification 3.7.6 were imposed upon plant operations. The senior resident inspector (SRI) reviewed licensee adherence to the action requirements, to include the licensee's controls to ensure no positive reactivity changes. Technical Clarification TS-098 was issued on August 1, 1989 to provide interpretation of boron concentration limits and temperature controls in line with the intent of the Action Statement to maintain the plant in a suitable reactivity condition. The inspector had no questions regarding the implementation of reactivity controls.

Based upon both SRI inspection and NRR project engineer review, the modification to the CBA system appears to have been controlled and implemented in accordance with design commitments. Operations and licensing staff cognizance of the impact upon routine operational activities and the requisite work controls was evident and no concerns or unresolved safety questions were identified.

9.2 Safety Parameter Display System (SPDS)

Supplement No. 7 to NUREG-0896, the Seabrook Safety Evaluation Report (SSER 7), was issued in October of 1987. Paragraph 18.2 of SSER 7 describes the March 25, 1987, Partial Initial Decision of the Atomic Safety and Licensing Board (ASLB) as it related to the Seabrook safety parameter display system (SPDS). Several of the ASLB issues involve staff verification prior to Seabrook operation above five percent power or before startup following the first refueling outage. NRC Region I Inspection Reports 50-443/87-16 & 88-15 describe previous NRC inspection of certain SPDS issues. Below are listed additional SPDS issues reviewed by an NRR Project Engineer during this inspection.

(1) Board Order 2(b). For this SPDS item, the licensee created an additional Emergency Coolant Recirculation Status Tree Logic in its SPDS program. The NRR engineer observed this Status Tree Logic on a plant computer screen display which showed the designed and actual residual heat removal flow in GPM.

Hydrogen concentration can be called out onto the computer screen display on the containment status tree when the hydrogen monitor is functional. When the monitor is not functional, the display will show a black indication of the terminus exiting the hydrogen concentration logic box.

(2) Board Finding 35. Summary status of containment isolation can be called out on the computer screen display for the Critical Safety Function (CSF) Summary. Isolation valves are grouped into phase A and phase B. Individual valve position of each containment isolation valve is indicated on the screen in red for the open position, and in green for the closed position. Operators can easily determine by pattern recognition from the assigned SPDS location, the overall status of containment isolation.

(3) Board Findings 40/41/42. The data validation algorithm was improved by altering the present computer software. An additional range checking computer program was loaded onto the software. This new range checking is more representative of the operating range of the instrument. Data not within the expected range will not be included in the final average. Also, a bandwidth check is added to the software to prevent the SPDS from making a determination unless at least two reliable redundant measurements agree within a preselected tolerance.

For hydrogen concentration instrument readings, licensee controls provide a programmatic check of whether: (a) the hydrogen monitor is "on"; (b) the monitor is not failed; and (c) the concentration is reliable. When the above three conditions are being met, then the reading is an acceptable concentration value.

The NRR project engineer had no further questions regarding the design and implementation of the above SPDS items.

10. Management Meetings

At periodic intervals during the course of this inspection, meetings were held with licensee personnel to discuss the scope and findings of this inspection. An exit meeting was conducted on October 10, 1989, to discuss the inspection findings during the period. During this inspection, the NRC inspector received no comments from the licensee that any of their inspection items or issues contained proprietary information. No written material, except for publicly available NRC Headquarters Daily Reports of the types discussed in section 8.3 of this report, was provided to the licensee during this inspection.