U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-443/87-26

Docket No. 50-443

License No. NPF-56

Permit No. CPPR-135

Licensee: Public Service Company of New Hampshire

1000 Elm Street

Manchester, New Hampshire 03105

Facility Name: Seabrook Station, Unit No.1

Inspection At: Seabrook, New Hampsnire

Inspection Conducted: December 8, 1987 - February 1, 1988

Inspectors: A. C. Cerne, Senior Resident Inspector
D. G. Ruscitto, Resident Inspector
C. A. Carpenter, Resident Inspector (Yankee Nuclear Power Station)

Approved By: Donald R. Haverkamp 3/2/88
Donald R. Haverkamp, Chief Date
Reactor Projects Section No. 3C

Inspection Summary: Inspection on December 8, 1987 - February 1, 1988 (Report No. 50-443/87-26)

Areas Inspected: Routine safety inspection during normal and backshift periods by three resident inspectors (98 hours). The areas reviewed included operational safety, licensee action on previous inspection findings, follow-up issues, unresolved safety issue A-26, allegation follow-up, control room ventilation, design changes/modifications and technical specifications.

Results: No violations were identified. One issue, involving the temporary loss of control room pressurization, is still under review for reportability in accordance with 10 CFR 50.73. The actions taken by the licensee to alert the control room operators of any similar situation are also being evaluated with respect to the adequacy of corrective measures to preclude reoccurrence. This issue is discussed in paragraph 8 of this report and remains unresolved.

In paragraph 5, licensee review of problems occurring at other plants and licensee action to evaluate nonsafety problems at Seabrook are discussed. The licensee responsiveness to these technical issues was effective in providing evidence and assurance of no adverse impact or relationship to safety-related equipment at Seabrook Station. Generic implications were assessed and appropriately addressed by the cognizant licensee technical personnel.

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*NOTE: The applicable NRC Inspection Manual inspection procedure or temporary instruction is listed in parentheses for the appropriate report sections.

DETAILS

1. Persons Contacted

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

   * Attended exit meeting conducted on February 8, 1988.

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

2. Summary of Facility and NRC Activities

   During this reporting period, the plant remained in operational Mode 5, cold shutdown, with primary temperature between 110 and 140 degrees F and depressurized.

   On January 28, 1988, Public Service Company of New Hampshire, one of the named licensees of Seabrook, Unit No.1, filed a voluntary petition for relief under Chapter 11 of Title 11 of the United States Code in the United States Bankruptcy Court for the District of New Hampshire. By letter (NYN-88013) dated February 2, 1988, PSNH notified the NRC Region I Office of this Chapter 11 filing in accordance with 10 CFR 50.54(cc).

   Subsequent to the filing, the inspector discussed its effects on current and future operation of the Unit No.1 facility with senior New Hampshire Yankee management. No adverse impact on scheduled plant maintenance and test activities was identified and appropriate contingency planning on the part of plant supervisory personnel was noted.

   On December 15-17, 1987, NRC Region I (NRC:RI) emergency preparedness specialist inspectors witnessed an exercise of the onsite portion of the facility emergency plan. During that inspection (Inspection Report No. 50-443/87-25), several NRC inspector follow-up items were reviewed and closed. On January 4-5, 1988, an NRC:RI operator licensing examiner, accompanied by a contract engineer, conducted a re-examination (Report No. 50-443/88-01) of one licensed operator candidate.

Also, during this inspection period, the resident inspectors attended an NRC:RI Resident Counterpart Meeting on December 15-17, 1987 and participated in a meeting between NRC:RI and the Employee's Legal Project on December 29, 1987, both meetings held in King of Prussia, Pennsylvania. The resident inspector was a member of the NRC:RI team that conducted an inspection of the Yankee Nuclear Power Station on January 11-15, 1988. The resident inspector also attended a three-week training course in boiling water reactor technology at the NRC Technical Training Center, commencing on January 25, 1988. Additionally, the senior resident inspector served as team leader for an NRC:RI team inspection at the Oyster Creek plant on January 25-29, 1988 to follow up an operational incident.

3. Operational Safety

a. 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 apparent 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 station staff, craft, quality inspection and supervisory personnel as such personnel were available in the work areas.

During control room observation periods, during both normal working hours and on backshifts, the inspector reviewed control room logs and records including night orders, shift journals, shift turnover sheets, completed repetitive task sheets, the temporary modifications log, weekly surveillance schedules and control board indications. Specific note was taken of equipment in "pull-to-lock" conditions, equipment tagged, alarm status and adherence to technical specification limiting conditions for operation (LCO) and action statements. Also, boron samples, taken from the reactor coolant system and connected water supplies, were spot-checked for concentration, sample frequency and documentation in accordance with specified zero power license (NPF-56) conditions.

The inspector verified the proper position, in accordance with operational procedure or tag-out controls, of specific valves during system walk-downs and checked the valve status in the control room. Similarly, temporary modifications and component tagging, maintenance work, and design change implementation activities, as observed during plant inspection tours, were evaluated for evidence of both proper

field controls and coordination of the subject work activity with the control room and operations personnel on shift. In certain cases, the operability of specific components and the applicability of the observed work to the TS requirements were discussed with the operators.

No violations were identified.

b. Operational and Security Events

Several events of minor safety significance and/or reportable in accordance with 10 CFR 50 or 10 CFR 73 occurred during this inspection period. These events are documented below:

(1) On January 29, 1988, a control building air handling system (CBA) isolation occurred as a result of planned, modification work to replace certain relays in the CBA system. The subject relays ensure the system functions in accordance with the single failure criterion by implementing an actuation/isolation logic which affects components in both CBA trains. Thus, the relay replacement work inherently affects both trains and caused a spurious ESF actuation, i.e., total CBA isolation. Upon actuation, the ESF equipment performed as designed.

Following notification by the control room that a CBA isolation had occurred, the original relay was returned to its initial condition, control room remote air intake ventilation restored, and work suspended on the subject relay replacement. An LER (88-001) on this event is currently under development by the licensee.

Subsequently, the inspector discussed with the shift superintendent the impact of making both trains of CBA inoperable to complete the required relay replacement work. Technical Specification (TS) 3.7.6 was reviewed and a Technical Support Group written evaluation was performed to determine the options and preferred course of action before reinitiating the field work. It was determined that although temporary jumpers could be installed to allow a CBA fan to continue to operate during the relay replacement, both trains of CBA would still be technically inoperable because the surveillance requirements for the CBA isolation capability could not be met. Also, the use of the jumpers and associated system modifications would require both trains of CBA to be inoperable for a significantly longer period of time than merely shutting down and isolating the entire system.

The license therefore decided to enter TS Action Statement 3.7.6.b, declaring both control room area ventilation systems inoperable. Control room air conditioning would not be affected, so as not to adversely impact the equipment and instrumentation temperature controls or limitations. Inspector review of the licensee documentation, discussion with both operations and technical support personnel, and assessment of the option chosen to allow the relay work to continue without additional spurious ESF actuations were conducted. All revealed proper consideration of the technical concerns and the appropriate use of both analysis and judgement.

While no violations were identified, LER 88-001 will be reviewed in its written form, after issuance, to inspect and evaluate the complete licensee analysis of this event.

With respect to the initial follow-up of the above event and other inspection issues, as noted, no violations were identified.

(2) On December 21 and 23, 1987 and January 17 and 28, 1988, four separate security incidents occurred for which one-hour notification to the NRC Operations Center was initiated in accordance with 10 CFR 73.71. Two Licensee Event Reports (LER) 87-026 and 87-027 and one Security Event Report (SER) 88-001 have subsequently been issued and a second SER is being processed.

Two incidents involved station employee access controls while another involved visitor escort controls. In the fourth incident, a security officer was discovered apparently asleep on post. The inspectors reviewed these events following their occurrence for the immediate consequences and licensee analysis/action. No adverse safety impact, directly resulting from these incidents, was identified. In the case of the inattentive security officer, the time duration in question was confined to a period of 17 minutes for which computer access logs revealed no unauthorized entries. The subject officer was suspended and subsequently discharged.

For the other incidents, the adequacy of existing programmatic controls for plant access and visitor escort were initially reviewed for any contributory causal relationship to the events. Based upon past incidents (reference: NRC Inspection Reports 50-443/87-23 & 24), the effectiveness of licensee corrective action requires further NRC review, which will be accomplished

during the course of inspection and closure of the individual LERs and SERs. For the protection of the "Safeguards" nature of certain of the event details and also for a comprehensive, expert review of the generic issues involved with both the current incidents and the related, previously reported events, the subject LERs and SERs will be reviewed by a Region I security specialist during the next scheduled NRC security inspection.

No violations were identified.

4. Licensee Action on Previous Inspection Findings

a. (Closed) Unresolved item 86-58-01: Shunt Trip Relay Spare Stock Items. During an NRC inspection (443/86-58) in December, 1986, the inspector noted that Class 1E automatic shunt trip relays had been added as a design modification to the reactor trip breaker circuitry, as recommended by Westinghouse. However, at the time of the inspection, no spare relays for the automatic shunt trip feature had yet been procured.

   During the current inspection, two Potter & Brumfield, Type MDR-5134 relays were procured, receipt inspected and placed in spare parts storage at Seabrook for issuance when needed. The inspector reviewed the Westinghouse quality documentation, including certificates of conformance and operating characteristics, for the subject relays. Licensee QA inspection had noted the need for additional certification data from Westinghouse with respect to Class 1E applicability and IEEE standard compliance. Such quality documentation was subsequently received, and all material purchase requisition requirements met by the vendor (Westinghouse).

   The availability of spare parts for the reactor trip switchgear (1-CP-CP-111) is now in conformance with the recommendation of the Westinghouse Reactor Trip Breaker Type DS-416 Maintenance Manual. This issue is therefore resolved and the item is closed.

b. (Closed) NRC Compliance Bulletin (No.87-02): Fastener Testing to Determine Conformance with Applicable Material Specifications. In accordance with the prescribed actions to be taken by the licensee, the NRC inspector participated in the selection process of the sample of twenty safety class and ten non-safety class fasteners, with typically appropriate nuts. This composite sample lot encompassed all types of the fasteners listed in Bulletin 87-02, which were available in the Seabrook storage supply.

The method of testing, both mechanical and chemical; the material purchase requisition requirements for the testing services; and QA surveillance over the sample selection and shipping activities were all discussed by the inspector with the cognizant licensee regulatory services, procurement and QA personnel. On January 21, 1988, the licensee response (NYN-88004) to Bulletin 87-02 with the required Fastener Testing Data Sheets and test results was forwarded to the NRC for review. The timeliness of this response was evaluated by the inspector with regard to the 60-day reporting requirement prescirbed in the Bulletin. Although the licensee response letter was delayed in order to procure and analyze additional testing data from the testing laboratory, this delay had been discussed with the NRC inspector. The inspector concurred with the licensee decision to submit the complete test results in one package, given that the expected delay time was not considered excessive.

The inspector reviewed the subject test results and accompanying fastener data for conformance with the bulletin requests for information. The licensee discussion of the sampling technique, procurement requirements and storage controls was evaluated for consistency with the material control programs in effect, both during construction and under the current operating license. Safety-related fasteners were appropriately referenced to the governing material and fabrication specifications (e.g., ASME Section III or ANSI with IEEE Class 1E controls). The licensee evaluation of test data for the one bolt evidencing hardness results outside the material specification limits was reviewed for both code justification and safety impact.

The inspector assessed all the actions taken by the licensee in response to Bulletin 87-02 and found them to be properly scoped and in compliance with the bulletin intent and sound engineering judgement. No violations were identified.

While further analysis and generic review of the test results will be undertaken by the NRC Office of NRR, for inspection purposes at Seabrook, NRC Compliance Bulletin 87-02 is considered closed.

c. (Closed) Licensee Event Report (LER 87-025): Incomplete Surveillance Testing Data. This deficiency involved incorrect selector switch settings on the vibration meter used during the conduct of ASME Section XI Inservice Testing. The inspector was monitoring from the control room the performance of the vibration testing of the primary component cooling water pumps at the time the problem with the readings was discovered. Subsequent licensee evaluation of this issue determined that although certain frequencies had been filtered out during previous tests, valid vibration data had still been compiled from those tests. Review of that data would have indicated the existence of a potential problem.

Licensee corrective action was reported to the NRC by letter (NYN-88001), dated January 6,1987. The inspector reviewed this report and evaluated the control room operator activities associated with this surveillance activity. No discrepancies or performance problems were identified. This LER is closed.

5. Follow-up Issues

a. Equipment Problems at Other Plants

Two specific equipment problems which had occurred at other plants were brought to the attention of the licensee and reviewed for applicability at Seabrook. One involved auxiliary feedwater pump turbine trips and oscillations caused by undersized springs in the Woodward governor and excessive condensate trapped in the steam supply lines to the Terry turbine. At Seabrook, based upon extensive redesign and post modification testing of the steam supply to the emergency feedwater pump turbine, both the governor controls and condensate build-up were frequently checked. Analysis of the problems encountered at the other plant by the licensee revealed no similarity to either the governor control or condensate problems. The inspector reviewed the licensee evaluation of the events documented in Nuclear Network Operating Event Report No. OE 2259 and concurred with the rationale for a nonapplicability determination at Seabrook.

The other problem involved failed safety-related pump starts at another plant due to Westinghouse electrical supply breaker malfunctions. Further review revealed no relationship between the problematic "DB" breakers supplied by Westinghouse to the plant where the failures occurred and the ITE Gould breakers supplied to Seabrook. Licensee technical support personnel evaluated the available details of the DB breaker component clearance problems and confirmed their nonapplicability at Seabrook.

No violations were identified.

b. U.S. Tool and Die Inspection

As documented in NRC Region I inspection report No.443/87-16, the licensee was informed of problems identified by the NRC Vendor Inspection Branch at U.S.Tool and Die, Incorporated. The specific problems were related to the fabrication and testing of spent fuel storage racks which at Seabrook were not fabricated by the subject vendor. However, since U.S.Tool and Die had supplied the new fuel storage racks, the licensee initiated an evaluation of the specific deficiencies for their applicability to Seabrook.

The licensee analysis included not only a review of past vendor surveillances and audits conducted at U.S.Tool and Die and of new fuel rack receipt weld inspections, but also the conduct of an additional visual examination of a sample of the rack welds, in place in the fuel storage building. The inspector reviewed the overall licensee analysis of this issue, specifically noting that the problems identified in the NRC vendor inspection report No. 99901082/87-01 were not in evidence at Seabrook. The QA involvement in the evaluation of this issue was noteworthy. While the licensee weld inspection did reveal certain configurations requiring additional engineering analysis, no major quality-related problems were noted.

The inspector had no further questions on the licensee actions taken to follow-up this issue. No violations were identified.

c. Industrial Crane Failure

As documented in NRC Region I inspection report No.443/87-24, a structural failure on an overhead crane in the circulating water pump house caused an industrial accident on November 20, 1987 which injured two workers. An accident investigation committee was formed to evaluate this incident from a personnel safety perspective, as well as for generic applicability to other cranes throughout the plant. New Hampshire Yankee contracted Teledyne Engineering Services to perform an evaluation of the failed crane equalizer sheave and support assembly, including inspection and metallurgical analysis.

The NRC safety interest in the follow-up of this accident relates to any potential for adverse impact to safety-related equipment which might be attributed to a common defect in the design or fabrication of other cranes manufactured by the same supplier, Shepard-Niles. The inspector reviewed the final accident report, dated December 22, 1987, issued as a result of the committee investigation, chaired by the Assistant Station Manager. The cause of the accident was attributed to a combination of an impact loading with a low safety factor, and marginal material quality in the susceptible area of the crane equalizer sheave and support assembly.

Because of the potential for generic applicability, the licensee conducted an inspection and evaluation of all Shepard-Niles hoist systems at Seabrook. The results, documented in the final accident report, indicate proper consideration of the evaluation criteria required by NUREG-0612 and the conduct of additional nondestructive examination where equalizer sheave and support design details similar to the failed crane were identified. Only one crane still remains tagged out pending the evaluation of the magnetic particle testing.

The inspector determined that the licensee investigation of this accident and the resulting report were responsive to NRC concerns regarding the generic safety impact. The inspector had no questions on the planned licensee actions to preclude recurrence of such an incident from a personnel safety standpoint. No violations were identified.

d. Caustic Fill Line Leak

The NRC Region I inspection report No.50-443/87-10 described a leak on the caustic fill line drain connection, which is part of the steam generator blowdown recovery system, in April,1987. A request for engineering services RES 87-568 was initiated as a result of this occurrence. Design coordination report (DCR) 87-192 was issued to provide seals for the subject penetrations. The inspector reviewed the DCR implementation plan for full service notification and the applicable implementing work requests. No violations were identified.

6. Unresolved Safety Issue (USI) A-26, Reactor Vessel Pressure Transient Protection

a. Background

The technical issue involved with Unresolved Safety Issue A-26 relates to the safety margin-to-failure for reactors which may be subject to severe pressure transients while at relatively low temperatures. The majority of industry events in this area have occurred during shutdown and startup while the reactor coolant system was in a solid condition. Plants licensed after March 13, 1978 were required to install fully automatic protection systems.

A review of the NHY actions taken in response to licensing commitments concerning this unresolved safety issue was conducted. Previous inspection activity conducted in this area is documented in NRC Region I inspection report No.50-443/86-12 (TMI Item II.D.3, Direct Indication of Relief and Safety Valve Position and TMI Item II.G.1, Emergency Power for Pressurizer Equipment). Also, licensing committments were made with respect to this issue and are documented in Section 5.2.2 (Overpressure Protection) of the Seabrook Final Safety Analysis Report, the NRC Seabrook Safety Evaluation Report (SER) and subsequent SER Supplements.

b. Inspection

Field verification of pressurizer power operated relief valve (PORV) position indication and power supplies was conducted during inspection 50-443/36-12. During this current inspection, the inspector performed a detailed review of the PORV and PORV block valve actuation circuitry, and discussed the design of the system with the cognizant NHY and YAEC engineers. The applicability and scope of the requirements to provide safety-grade control circuits for the PORV and block valves were questioned by the inspector with respect to the Seabrook design. It was agreed that in accordance with the intent of the TMI Action Plan, safety-grade controls were required for manual operation only. The inspector verified that the manual controls were capable of overriding any failure of the non-safety grade automatic controls.

The inspector also confirmed that the subject controls were from independent power supplies and were electrically separated in accordance with the appropriate industry standards. No violations were identified.

7. Allegation Follow-up

Two allegations involving component quality during the construction phase of Seabrook, Unit 1 were brought to the attention of the NRC. In one case, the alleger indicated that the concern had been reported to and evaluated by the licensee. For the other allegation, no specific details of the concern were initially documented, nor was there any indication that the licensee had been informed of the concern. In both cases, subsequent information was provided by the allegers to clarify the details of the individual concerns in a manner which would permit meaningful inspection of each allegation. The NRC inspection follow-up of these two issues is documented below:

a. Valve Material Traceability Problem

Original Allegation: "There was a materials traceability problem. Certain valves on the steam generator lacked the engraved manufacturer's number."

Additional Information: "Relief valves 456A and 456B, on the steam generator, noted during the first hydro test in Unit 1 (RC IT 01A)."

Inspection Follow-up: The inspector noted that the two valves specified by the alleger represent the power operated relief valves (PORVs) for the pressurizer, not a steam generator. The inspector visually examined the current condition and status, including valve identification and tagging, of these reactor coolant (RC) system valves, RC-PCV-456A & B, and identified no marking or traceability deficiencies.

Additionally, conduct of the reactor coolant system hydrostatic test (RC-IT-01) in April, 1985, was witnessed by the resident inspectors (reference: inspection report 50-443/85-01) and Region I specialist inspectors (reference: inspection report 50-443/85-08). The RC-PCV-456A & B represented hydrostatic test boundary valves which were spot-checked by the inspectors during the conduct of RC-IT-01. Subsequently, during a Region I resident inspection (reference: inspection report 50-443/85-25), the inspector witnessed PORV modification in accordance with an engineering change authorization and under the direction of Westinghouse and Crosby Valve engineers. This work was verifed to have been properly conducted in accordance with ASME Section XI requirements and additional hydrostatic test requirements for the PORV pressure boundary were delineated. The hydrostatic test of RC-PCV-456A & B was later witnessed by an NRC resident inspector, as is documented in inspection report 50-443/85-31.

At no time during the conduct of previous NRC inspections, as documented above, of the subject valves (RC-PCV-456A & B) were traceability or component identification problems identified. QA coverage of the PORV modification work, documented in inspection report 50-443/85-25, programmatically checked material identification/traceability prior to any ASME welding or part replacement.

Based upon previous NRC inspection of the valves in question and the conduct of a recent inspection to ensure that the current valve identification is both proper and traceable, the allegation was not substantiated.

b. Crack in the Core Barrel

Original Allegation: "A quality assurance man in Pullman-Higgins believed the core barrel was cracked. The QA person reported this to the company, and engineers responded to his concern, but the QA inspector was never satisfied that the core barrel was not cracked."

Additional Information: "About 2 1/2 to 3 years ago, while he was with a New Hampshire Yankee inspector and another inspector, they were there while the core barrel was being moved. He saw a crack about 18 inches long which changed direction about four or five times. Both of the other inspectors saw it as well, so it was reported at the time to NHY."

"The crack was located in an area where there is an upper and lower flange protruding out from the core barrel. Sketches to follow later."

A sketch, marked up to designate the alleged crack, was transmitted to the NRC with a letter noting the following applicable information: "He drew the zig-zag line on the enclosed drawing of the core barrel to indicate where he saw the crack and drill holes at the junction of each crack."

Inspection Follow-up: The inspector reviewed the Employee Allegation Resolution (EAR) program files documenting the concern, interviewed the QA inspector that accompanied the alleger on an inspection of the core barrel and discussed with other NRC personnel the independent NRC inspections of the core barrel while in its storage position in the refueling cavity and during insertion and removal from the reactor pressure vessel for testing. Historically, NRC inspectors have conducted examinations of the reactor pressure vessel internals, including the core barrel, both in storage and during the installation process (e.g., Region I inspection reports, 50-443/81-01, 82-08, 82-12 and 83-05). In addition to the licensee programmatic quality

assurance controls and inspections of the core barrel, Westinghouse engineering personnel were on site to observe and provide direction to the initial internals package installation process. Subsequently, during the preoperational testing phase of Seabrook, Unit No. 1, the core barrel was visually examined by an NRC inspector for evidence of physical damage or any abnormalities.

Separate from any routine NRC inspections of the core barrel, the EAR file review revealed that the alleger, accompanied by a certified QA inspector, conducted a reexamination of the core barrel in an attempt to find the "crack" he had seen. An NRC interview with this QA inspector indicated that no evidence of a "crack" was discovered and that the alleger had expressed apparent satisfaction that no problem existed at that time. In July, 1986 upon completion of his employment at Seabrook, the alleger again raised the same concern to EAR program personnel. An EAR investigator conducted an additional examination of the core barrel and discovered no crack. The EAR investigation also reviewed the relevant quality records associated with both the Unit No.1 and No.2 core barrels and identified no evidence of cracking indications.

Given the extent that the core barrel was examined, both prior to and after this allegation was raised, and given that a zig-zag crack with drill holes at the junction of each crack would be expected to be clearly visible upon examination, the inspector determined that this allegation was not substantiated. The inspector also noted that the EAR investigation had included checks for evidence of unauthorized repair welding on the core barrel, to include a review for any design modifications on structural attachment which could mask an unauthorized weld repair. None were identified.

With respect to the follow-up of both of the noted allegations, no violations were identified. The inspector considers both allegations closed. No further follow-up is planned, pending Region I management review of the satisfactory closure of these allegations.

8. Control Room Ventilation

On September 20, 1987, licensee personnel discovered that the control room differential pressure was zero inches (water gauge, WG) with the control room exhaust fan, CBA-FN-15, running and its exhaust damper full open. Seabrook Technical Specification 3.7.6 requires that a minimum of (+)1/8" WG differential pressure be maintained between the control room and adjacent areas to preclude the entry of contaminants.

Subsequent licensee investigation determined that the cable spreading room supply fan CBA-FN-17, had tripped due to a battery test on the local fire protection panel on the previous day. This fan is not designed to automatically restart following a trip. Therefore, the pressure in the cable spreading room became slightly negative with respect to atmosphere since the exhaust fan CBA-FN-18 continued to run. The control room pressure controller senses differential pressures between the control room and either atmosphere or the cable spreading room below. It was noted that this is an auctioneered high signal and therefore provides control functions based upon the highest differential signal. In this particular case, the differential pressure between the control room and cable spreading room was greater than that between the control room and atmosphere.

Therefore, the pressure controller was attempting to reduce the control room/cable spreading room differential pressure to (+)1/8" WG by maximizing the exhaust from the control room. However, CBA-FN-15 is much smaller than CBA-FN-18. Even with the control room exhaust damper fully open and the control room depressurized, CBA-FN-18 operation continued to cause a differential pressure greater than (+) 1/8" WG between the control room and cable spreading room. This resulted in a continued erroneous signal to the pressure controller which maintained the control room exhaust fan operating with the supply fan off.

Subsequently, by taking manual control of the pressure controller, the control room operator was able to reestablish pressure in the control room by closing the exhaust damper. The inspector verified that no limiting condition for operation (LCO) was violated since TS 3.7.6 requires action be taken to place the control room in the recirculation mode after seven days of component inoperability. The specific condition discussed above existed only for about fourteen hours.

The inspector reviewed the sequence of events and system design which resulted in a depressurized control room for the 14-hour period. Further discussion with licensee personnel resulted in two issues which require additional evaluation or action by the licensee, as follows:

(1) The potential reportability of this event based upon a loss of an engineered safety feature.

(2) Short-term actions requires of the licensee to ensure that a similar situation cannot exist undetected for any length of time until a final design modification to the subject pressure control functions is implemented.

Pending the presentation by the licensee of evidence that both of the above issues have been adequately addressed, NRC concerns regarding the corrective measures remain unresolved (50-443/87-26-01).

9. Design Changes/Modifications

The inspector reviewed the following design coordination reports (DCR's) and spot-checked work in progress in the field relative to work control, change authorization and engineering overview:

-- DCR 87-082, "Modify Thermal Barrier Cooling System Head Pipe"

-- DCR 87-315, "Add Restricting Orifice to DG-E-42A and B Outlet"

The observed work was discussed with the cognizant engineering personnel. In the case of DCR 87-082, corrosion in the drain lines off the thermal barrier heat exchangers was chemically analyzed. The piping upstream of valve CC-V-1090 was visually inspected by the licensee and also examined by the inspector. Subsequent analysis revealed that the corrosion was specific to the stagnant conditions in the drain piping and not representative of the conditions of the thermal barrier cooling system in general. The DCR 87-082 specifies the installation of rupture discs on the large vent openings on the thermal barrier head tank piping and the addition of a smaller expansion vent. These modifications, along with the installation of a new chemical addition connection, will minimize corrosion due to excess oxygenation of the system and will allow for the more efficient addition of corrosion inhibitors to the system.

With respect to DCR 87-315, the inspector raised specific questions regarding orifice sizing, weld relocation and modifications to the service water valves, SW-V-16 & 18, previously designated as throttling valves. Existing DCR change authorizations were found to provide the answers to certain NRC questions and justification for physical configuration changes in the design. For the valve rework, specific stroking criteria were added to in-process change authorization, CA-08, to clarify the need for full-open position adjustments. The inspector also verified that the fail-safe testing of these valves to their 100% open position would be routinely accomplished by an engineering surveillance, EX 1804.029, in accordance with in-service testing requirements. The inspector also verified the conduct of a weekly operational surveillance, OS 1426.12, for stroking the subject valves.

The inspector had no unresolved safety concerns regarding implementation of the above design coordination reports. No violations were identified.

10. Technical Specifications

a. Administrative Controls

The inspector reviewed Section 6.0, Administrative Controls of the Seabrook Technical Specifications to determine the consistency in the descriptions of the licensee off-site organization and station staff with the current organizational structure of New Hampshire Yankee.

The inspector noted that recent organizational changes have affected the official titles and both the function and responsibility of certain personnel. Also, the organizational charts included in the Technical Specifications been superseded by the NHY reorganizations which have taken place since the license was issued in October, 1986.

The inspector discussed the noted inconsistencies with licensee regulatory service personnel and was provided a draft copy of Section 6.0 revisions to the Technical Specifications which will be submitted to the NRC Office of Nuclear Reactor Regulation (NRR) for proposed future amendment to the Technical Specifications. The NRR staff is expected to either (1) approve the changes as a separate licensing action, or (2) incorporate such administrative changes, along with any required technical revisions, into the revised Technical Specifications which will serve as conditions to the low-power operating license, when issued. The inspector had no further questions on the way such organizational changes would be processed.

The inspector also examined the licensee controls over the Technical Specification Improvement Program, as discussed in Section 6.7 of the Technical Specifications and Chapter 16.3 of the FSAR. Revision 3, effective December 29,1987, to the Technical Requirements Manual was reviewed to verify the requisite Station Operation Review Committee (SORC) and 10 CFR 50.59 evaluations. The inspector checked the NHY Programs and Procedure Manual, noting a provision in procedure 11210 that prescribes safety evaluations for all procedural changes that require SORC review. The Technical Specification Improvement Program, as implemented by the Technical Requirements Manual, is administratively controlled by both the Technical Specification conditions and the SORC review process. A Nuclear Safety Audit and Review Committee review of any change is also required. Therefore, even though prior NRC approval to the Technical Specification Improvement Program revisions is not mandated by the existing facility license and Technical Specifications requirements adequate procedural control and post-implementation reporting to the NRC appears evident in the NHY program.

No violations were identified.

b. Containment Leak Rate Testing

The inspector noted an apparent inconsistency between the surveillance requirements, Technical Specification (TS) 4.6.1.2.a for primary containment leakage and a commitment in FSAR Chapter 6.2.6.4 for the scheduling of Type A tests (Overall Integrated Containment Leakage Rate). The inspector verified that the TS provisions were in conformance with 10 CFR 50, Appendix J criteria and discussed the FSAR write-up with the appropriate technical support and regulatory service personnel. The licensee subsequently provided the inspector with a FSAR change request, revising Chapter 6.2.6.4 to bring it into conformance with the Technical Specifications.

Additionally, the inspector reviewed an engineering surveillance procedure, EX 1803.004, effective August 14,1987, which discusses the containment interior steel liner and exterior concrete inspections conducted during each plant shutdown prior to the Type A containment leakage rate tests. Compliance with the requirements of TS 4.6.1.6, with reference to TS 4.6.1.2 schedular provisions, was confirmed.

No violations were identified.

11. Unresolved Items

An unresolved item is a matter about which more information is required to ascertain whether it is an acceptable item, a deviation, or a violation. An unresolved item is discussed in section 6 of this report.

12. Management Meetings

At periodic intervals during the course of this inspection, meetings were held with plant management to discuss the scope and findings of this inspection. An exit meeting was conducted on February 9, 1988 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.