U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No.: 50-443/88-06

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 Hampshire

Inspection Conducted: March 29 - May 23, 1988

Inspectors: A. C. Cerne, Senior Resident Inspector, Seabrook Station
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Approved By: Donald R. Haverkamp — 6/15/88
Donald R. Haverkamp, Chief, Reactor Projects Section No. 3C — Date

Inspection Summary: Inspection on March 29 - May 23, 1988 (Report No. 50-443/88-06)

Areas Inspected: Routine inspection on day and backshifts by two resident inspectors and two specialist inspectors of actions on previous inspection findings; routine plant operations; physical security; licensee potentially reportable occurrences and operational events; maintenance and surveillance activities; RHR system mid-loop testing, and service water system piping remote video inspection.

Results:

1. General Conclusions on Adequacy, Strength or Weakness in the Licensee's Program

   The licensee demonstrated comprehensive inspection/modification planning and implementation during the April 26-28, 1988 service water piping inspection (Section 4.b). Additionally, NHY successfully conducted the RHR/RCS mid loop test, STP-105 on April 5-9, 1988. Close observation by NRC inspectors of all shifts revealed strong test/operations planning, test administration and plant control (Section 6.c).

2. New Open Items Identified

   An open item was identified requiring licensee evaluation of the existence of non-safety circuits powered off a 1E bus. This issue will be followed up under Open Item 88-06-01 (Section 7.c)

   The low lube oil pressure trip of the train "B" emergency diesel generator which occurred on February 24, 1988 raised a concern with the reportability of the diesel generator failure in accordance with Technical Specifications. Review of this issue will be tracked under Open Item 88-06-02 (Section 7.e).

3. Clarification of Previous Unresolved Item

   NHY actions taken in repairing the failed RHR system welds were evaluated by the inspector and determined not to be in violation of NRC regulations (Section 4.c). However, NRC review of this evolution raised several concerns related to potentially non-conservative plant management determinations of equipment operability in accordance with the Seabrook Technical Specifications. Specifically, the pressure boundary of the only operating RHR train was opened during weld repairs. Recurrence of a similar situation after initial criticality could require that the affected RHR train be declared inoperable. Additionally, the RHR system was operating without a safety-related heat sink. System heat loads were being dissipated via non-safety-related systems connected to the PCCW system.

   The questions raised in Section 4.c of this report relate to a delineation between system functionality and system operability. In the specific case involving the RHR system, the removal of decay heat is evidence of a functioning system under the conditions existing at that time, but not necessarily for all design basis conditions. For example, the noted in-process repairs or reliance on non-safety support systems could be interpreted to render the RHR system inoperable because of the system dependence upon components or activities not analyzed in the system design.

Such questions, particularly involving Mode 5 operations, require further regulatory review. Therefore, portions of the unresolved item discussed in Section 4.c remain open until an NRC position is developed and licensee response is solicited.

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*The NRC Inspection Manual inspection procedure that was used as inspection guidance is listed for each applicable report section.

DETAILS

1. Persons Contacted

E. A. Brown, President and Chief Executive Officer
* W. A. DiProfio, Assistant Station Manager
T. C. Feigenbaum, Vice President, Engineering, Licensing and Quality Programs
G. R. Gram, Executive Director of Emergency Preparedness and Community Relations
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 May 20, 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

a. Resident Inspector Activities

The Senior Resident Inspector attended meetings with regional management and conducted Vermont Yankee Inspection preparations on March 30 - April 1, 1988 in King of Prussia, Pennsylvania.

The Senior Resident Inspector participated in an inspection of the Vermont Yankee Nuclear Power Station on April 4-8, 1988.

The Senior Resident Inspector and the Resident Inspector participated in inspections of the nuclear facility at the University of Lowell on April 14, April 29 and May 12, 1988.

The Resident Inspector attended a one week simulator training course in advanced boiling water reactor technology at the NRC Technical Training Center on April 11-16, 1988.

The Resident Inspector and Senior Resident Inspector attended the NRC Resident Inspector Couterpart Meeting on April 19-21, 1988 in King of Prussia, Pennsylvania.

The Senior Resident Inspector attended a training course on May 2-6, 1988 in King of Prussia, Pennsylvania.

b. Visiting Inspector Activities

On April 6-8, 1988, a regional test specialist conducted a routine inspection of the NHY special RHR/Mid Loop test, STP-105. The inspection findings are incorporated into this report, paragraph 6.

On April 27, 1988, the Resident Inspector and Regional State Liaison Officer accompanied the Regional Administrator from NRC Region I on an inspection of Unit 1. For additional discussion of a meeting held with NHY senior executives following that inspection, refer to paragraph 12.

On May 20, 1988, a Senior Reactor Engineer from the NRC Office of NRR conducted a site inspection related to Generic Letter 87-12, RHR/RCS Mid-Loop Operation (Refer to paragraph 6). In preparation for this announced visit, a list of discussion topics, all related to the technical issues of Generic Letter 87-12, was provided to the licensee at the request of the NRR Senior Reactor Engineer in order to facilitate the collection of information requested by NRR.

c. Plant Status

During this reporting period, the plant remained in operational Mode 5, cold shutdown, with primary temperature between 105 and 140 degrees F and depressurized. Major maintenance was conducted on the circulating water and service water systems. Passivation procedures for layup of secondary systems continued to be implemented.

On April 5-9, 1988, NHY conducted a special test to determine how low the reactor coolant system could be drained before cavitation occurred on the residual heat removal pumps. Further discussion of this test may be found in paragraph 6 of this report.

On April 26-28, 1988, the licensee conducted a remote video camera inspection of selected, safety-related portions of the service water system. Further discussion of this inspection may be found in paragraph 4.b of this report.

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 in their 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 (TS) limiting conditions for operation 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 conditions.

The inspector verified the proper position, in accordance with operational procedure or tag-out controls, of various 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.

The inspector identified several minor discrepancies in material conditions. A list of items was provided to the licensee. Action taken on each issue is described below.

(1) Two instrumentation cables had insulation damage in the cable spreading room. Work request 88W1636 was initiated to remove the conduit fire seals for access to the damaged cables and work requests 88W1637 and 88W1638 were issued for each individual cable repair.

(2) The air pressure regulator to ventilation damper 2-SWA-DP-39A was leaking. Work request 88W1447 was issued on April 18, 1988 for regulator replacement.

(3) Although door P1100 is labeled as a fire door, it appeared that the label was not consistent with the design for the wall the door penetrates. Licensee review indicated that the wall was not a fire barrier and the label will be removed per work request 88W1976.

(4) In the primary auxiliary building boric acid crystallization was noted on several instrument connections. Work requests 88W2271 and 88W2273 were initiated to correct the conditions.

(5) Two air pressure regulators in the chemical and volume control system were leaking air. Work requests 88W2269 and 88W2270 were written to replace the diaphragms.

The inspector performed a walkdown of the rocker arm lube oil systems in both emergency diesel generators and verified system lineup and configuration. The inspector noted one minor discrepancy with two broken tubing clamps in the train "A" system. The licensee was aware of this as evidenced by work request 87W005775 originated in July, 1987. Repairs are scheduled for the next diesel outage.

The inspector conducted a walkdown of the five fire protection system sprinkler stations for the cable spreading room. He verified proper alignment and readiness for actuation. Within the cable spreading room, the inspector noted several penetration seals which were not intact due to mainte nce activities. He verified that each seal was identified on the Secondary Roving Fire Watch List for periodic monitoring.

The inspector discussed the operability status of the train "A" residual heat removal (RHR) system with the Unit Shift Supervisor and Shift Superintendent during the maintenance outage of 460 volt motor control center (MCC) 521. Both operators were aware of the impact that the loss of MCC 521 had on the RHR system. They both demonstrated a sensitivity to the differences between this outage and an outage in which decay heat existed. Actions to be taken to initiate train "A" RHR upon loss of train "B" RHR were discussed. The inspector reviewed the electrical schematics relevant to the outage and verified that RHR and safety injection system configurations were consistent with the loss of power. He also verified that the appropriate action statements of Technical Specifications 3.7.6 concerning the control building air handling system were entered prior to de-energizing MCC 521. No violations were identified.

b. Operational and Security Events

On April 24, 1987, a one-hour report to the NRC Operations Center was made pursuant to 10 CFR 73.71b when a security procedure was found missing. Subsequent review of the contents of the missing procedures revealed that it contained no information that would meet the requirements of 10 CFR 73.71b and the event was downgraded to a security log entry rather than a one-hour report (See also section 9).

4. Licensee Action on Previous Findings

a. (Open) Deviation 86-54-01: CBA System Design. This issue was most recently updated in NRC:RI Inspection Report (IR) 50-443/87-02. Other related issues concerning the control building air handling system (CBA) may be found in NRC:RI IRs 50-443/88-02 and 50-443/87-16.

On April 9, 1987, NHY submitted additional information to NRC:HQ concerning the CBA system evaluation for operation up to 5% power (NYN-87051). Supplement 7 to the Seabrook Safety Evaluation Report (NUREG-0896) approved operation up to 5% power with the present design. On January 22, 1988, NHY submitted a summary of proposed modifications to the CBA system (NYN-88007), and in a licensee/NRC:HQ conference call on March 1, 1988 this submittal was discussed. Additional questions were raised which will be addressed in a future submittal.

The CBA design changes were formalized in design coordination report (DCR) 86-709. The inspector conducted a preliminary review of the DCR noting that it is particularly comprehensive. However, final approval of the design changes rests with the NRC Office of Nuclear Reactor Regulation (NRR). Following design review, NRR must also approve the as yet unfinalized proposed Technical Specification (TS) for the CBA system. With respect to the current TS, NHY is considering submittal of a request for enforcement discretion or other appropriate temporary license amendment for the time frame during which design change implementation is actually in progress, rendering the CBA system inoperable. The inspectors will continue to follow licensee progress on this deviation which remains open.

b. (Open) Open Item (88-02-02): Service Water Piping Corrosion. As part of the licensee's continuing program to evaluate the extent of corrosion in the service water (SW) system, a remote video camera inspection of a portion of the train "B" underground piping was conducted on April 26-28, 1988. The inspection involved removal of a piping section located in the piping pit of the service water pump house. The modification of this pipe spool including installation of flanged end connections to facilitate future inspection and was controlled by design coordination report (DCR) 88-022. The design implementation was provided by work request (WR) 88W1425. The inspector reviewed work activities in progress including fit-up and surface preparation. A review of WR 88W1425, the ASME weld traveller sheets, the non-destructive examination data sheets and process control procedures revealed no deficiencies. While the piping spool was removed, a remote video camera inspection of approximately 692 feet of twenty-four inch diameter, cement-lined piping was conducted.

Preliminary review of inspection results yielded no significant corrosion problems. Each indivivual field weld was examined radially and mapped. The inspector witnessed the preparation, camera insertion and video inspection both at the contractor's video studio van and locally in the SW pumphouse. The entire inspection was supervised by the cognizant NHY system engineer. Final video tape reviews have not yet been completed.

The technical support organization continues to demonstrate the ability to execute complex multi-disciplinary activities with a minimum of difficulty. The inspectors will continue to monitor licensee progress with respect to further inspection of the safety related portions of the SW system.

c. (Open) Unresolved Item 87-16-02: RHR Line Weld Failures. Following inspection described in report 50-443/88-02, five sub-items remained open. Each issue is dealt with separately below, as items (1) and (5) are closed and questions remain with items (2), (3) and (4).

(1) Evaluation of whether Interpretation XI-1-83-85 applies to ASME repairs as well as replacements.

The inspector reviewed ASME Interpretation XI-1-83-26R issued on November 21, 1985 which indicated that the exemption of piping and fittings applied to replacements also applied to repairs. This position was also concurred with by the site Authorized Nuclear Inspector. Based upon NRC questions, a change was made to procedure MA3.1 to clarify the policy already in effect that ASME components 1 inch nominal pipe size and less that are specifically exempted from the requirements of MT8.1 shall be repaired or replaced in a manner at least equivalent to that specified in the original design basis and requirements, materials specifications and inspection requirements. The repairs in question were conducted in accordance with the above procedure change, however the commitment had not been proceduralized. It is concluded that NHY, while recognizing the exemptions, has instituted its own requirements which exceed those of the code. The inspector had no further questions

(2) Evaluations of whether operations of the primary component cooling water system with cross connects through the thermal barrier heat exchangers and containment air handling fan coolers is an acceptable method of heat removal with respect to single failure criteria.

The licensee completed Engineering Evaluation 87-026 on December 23, 1987. This evaluation determined that operation of the thermal barrier heat exchangers as identified in NRC:RI inspection report 50-443/87-16 was consistent with the Seabrook Station Operating Procedures and Technical Specifications. While the technical basis supports such a conclusion, the inspector noted that the determination of acceptability was made after the fact rather than in advance. Additionally, while the operators were aware of the systems configuration as well as the heat transfer mechanism, an evaluation should have been performed to determine the technical basis for acceptability.

(3) Operability of systems with inoperable subsystems or support systems when operational criteria are met, but design bases have not been addressed.

In this case with a new core and no decay heat, the design bases for the RHR system are coolant circulation and removal of the heat generated by the RHR pumps. As with sub-item (2) above, operation of the RHR system without its associated PCCW/SW train available would not be acceptable after initial criticality. Furthermore reliance on non-safety class systems such as the containment air handling coolers is not considered an acceptable practice under any condition with irradiated fuel in the vessel or spent fuel pool.

With respect to sub-items (2) and (3) above, although no violations of NRC requirements were identified, it is expected that in the future a more rigorous interpretation of the technical specification operability requirements will be made.

Two NRC Information Notices (IN) address issues related to this item. IN 84-42 entitled "Equipment Availability for Conditions During Outages Not Covered by Technical Specifications", indicated that certain corrective actions as a result of an event at another nuclear facility included:

- A review of the management control of equipment for plant conditions not covered by the requirements of the Technical Specifications. The review would specifically address electrical system requirements during cold shutdown to ensure sufficient equipment remains available to maintain the plant in a safe condition and to meet the commitments of the Site Emergency, Security, and Fire Protection Plans.

- Establishing minimum equipment availability for specific conditions not covered by the Technical Specifications.

IN 83-56 entitled "Operability of Required Auxiliary Equipment" specified that auxiliary equipment needed to support safety related equipment should be identified in advance and maintained in a similar high state of operability. NHY should review the above IN's to assure that more concrete guidance is available to control room operators when similar conditions are experienced in the future.

(4) Acceptability of the repair of the train "B" residual heat removal (RHR) recirculation line weld failure on August 7, 1987 while still declaring the system operable.

At the time of the above failure, the RHR system demonstrated functionality in fulfilling zero-power design requirements. However, the determination made by the licensee that the system remained operable was questionable and merits further licensee attention. In this particular case, the weld repair opened the pressure boundary of the only operating train of the RHR system. Automatic closure of the recirculation flow control valve, RH-FCV-611, with the pipe cut open by the repair process, would have resulted in a pressurized leak of the reactor coolant system (RCS). While such a leak would have been isolable, this could only be accomplished by removing the RHR system train from service, thus not meeting the minimum Technical Specifications requirements for operability.

In the observed situation, the actual repair was conducted with the plant in an operating condition where the reactor coolant system contained neither radiological activity nor decay heat. Therefore, the inspector determined that no violation of NRC requirements occurred, because no actual inoperability was identified, as the repair was completed without the need to isolate the RHR system. However, future operating conditions with an irradiated core and core decay heat will dictate the need for a more comprehensive approach to and analysis of any similar problems. Under those circumstances, any questions of system operability, of the requirement to enter the appropriate Technical Specification action statement and of the advisability of opening a hole, thus creating a potentially larger leakage path into the RCS, would all require careful evaluation by the licensee. Additional planning and preparation for contingency actions would be considered appropriate.

The inspector discussed with station management personnel the above questions on the choice of repair options and the RHR system operability interpretations made during the conduct of the subject weld repair activity. It was noted that similar situations encountered in the future will require the analysis of similar questions with the answers dependent upon both existing plant conditions and the safety impact of the selected approach.

(5) Investigation into the common mode failure of these welds.

NRC:RI Inspection Report 50-443/88-02 detailed the review of Special Test Procedure (STP) No.109 and repairs to RH-FCV-611, the train "B" mini-flow recirculation valve. During this reporting period, the inspector reviewed the final test results of STP-109 which were forwarded to Engineering for use in Engineering Evaluation 87-037. The inspector had no further questions.

With respect to all the above items, no violations were identified. This item is closed.

d. (Closed) Open item (87-24-01): Miscellaneous Electrical Issues. This inspection item was opened to track the licensee's progress in resolving several engineering discrepancies and configuration control problems in the electrical area. During this inspection, the licensee published Engineering Evaluation No.88-011 which provided a root cause analysis and safety significance assessment for seven separate electrical configuration control problems, as follows:

1. Distribution panel discrepancies
2. Solenoid valve coils and lights on different control circuits
3. Improperly spared cables
4. Motor control center discrepancies
5. Nylon pull ropes and temporary cables
6. CASP pull slip discrepancies
7. Circuit breaker rating disc epancy

The inspector reviewed Enginee·ing Evaluation No.88-011 to determine the completeness of the overall design assessment. QA involvement was noted, not only in the original identification of the problem areas (e.g., surveillance activities), but also in the verification activities planned to ensure that a breakdown of the change document control process is not evident. The inspector evaluated the adequacy of the specified licensee corrective action, where appropriate, and confirmed the conduct of licensee field walkdown activities to fully define the scope of the problem areas. Engineering calculations were available for review where the impact upon design bases considerations were in question. Also, the operations department involvement in problem resolution was evident where component status or electrical switch lineups were of concern.

Overall, the inspector considered the licensee's handling and assessment of the identified miscellaneous electrical issues to be comprehensive and focused on the potential for a larger electrical configuration control problem. The results of the licensee analysis indicated that no generic problem was in evidence.

The inspector has no further questions in this area and considers this item to be closed.

5. Licensee Reports

a. (Open) Construction deficiency report (CDR) 86-00-09: Veritrak/ Tobar Transmitters. As documented in NRC:RI inspection report 50-443/87-24, replacement of all 23 of the subject transmitters with new Rosemount transmitters has progressed in accordance with the details of design coordination report (DCR) 86-349. The inspector reviewed the DCR implementation plan (DIP) for component installation and work controls and examined the Rosemount transmitters, as installed in their field locations within containment.

Based upon engineering evaluation, a change authorization No. 7 to DCR 86-349 is being processed to take into account the thermal stresses on the tubing connecting the newly installed Rosemount transmitters to the process fluid manifolds. Complete rework of this interconnecting tubing and the associated swagelok fittings is anticipated. The final corrective action attendant to CDR 86-00-09 will not be considered complete until all the rework associated with DCR 86-349, including all change authorizations, is concluded.

Additionally, the inspector reviewed the calculations included in DCR 86-349, as they affect the Technical Specification setpoints, allowances and other allowable values. Discrepancies were noted with respect to not only the comparable values listed in the current Technical Specifications, but also the instrumentation setpoint data listed in the draft Technical Specifications changes intended for issuance with the low-power license, when authorized. The inspector indicated to licensee operations and regulatory services personnel that operability of the instrumentation affected by the installation of the Rosemount transmitters was equally dependent upon the existence of correct Technical Specification data to determine and control operations. Licensee management personnel agreed that either the noted discrepancies must be clarified before the affected components could be considered operable or further justification was required to support a position that operations under the current Technical Specifications would be conservative relative to the actual instrument setpoints.

Pending NRC inspection of the completed rework associated with change authorization No.7 to DCR 86-349 and review of the process that verifies setpoint and allowable value consistency between the design data for the Rosemount transmitters and the Technical Specifications in effect at the time operability is required, this item remains open.

b. (Closed) 10 CFR 21 Report (87-88-06): Inadequate Capacity of the Thermal Compensating Accumulator for Type A Actuators on Rockwell Main Steam Isolation Valves. As documented in NRC:RI inspection report 50-443/87-10, inspection followup of the corrective action taken in response to LER 87-009 involved a system and procedural control review of the main steam isolation valve (MSIV) design. Subsequent to this review, Rockwell International corporation notified the licensee of a design deficiency in the thermal compensating accumulator installed on the A-260 actuator of each of the four MSIVs installed in Unit 1 at Seabrook Station (reference: P21-87-011 notification, Rockwell letter dated May 8, 1987). The accumulator deficiency involved an insufficient capacity to compensate for thermal expansion of the hydraulic fluid, resulting in high fluid pressures and potential damage to the hydraulic solenoid dump valves. If this were to occur, the MSIV might not be capable of performing its safety-related function, when required.

In accordance with Rockwell recommendations and with the assistance of a Rockwell represenative on site, the licensee initiated design coordination report (DCR) 87-232 and installed a relief valve and tubing on each thermal compensating accumulator. This modification provides for the hydraulic fluid to discharge directly to the MSIV hydraulic fluid reservoir when the operating temperatures create pressure increases which approach the allowable qualified limits. The relief valve setpoint has been calculated to conservatively protect the actuator below its maximum operating pressure.

The inspector reviewed the DCR, spot-checked the DCR implementation plan, interviewed the cognizant implementation engineer and examined the completed rework on each of the four MSIVs. The seismic calculations in the DCR package were checked and the orientation of the relief valve installation was questioned in relation to the documented calculations. Acceptability of the as-built condition of the modification work was verified in subsequent documented liaison between licensee and Rockwell engineering personnel. The inspector also questioned some of the relief valve specification data which was included in the DCR. A request for engineering services was initiated by the system engineer to update the supporting design change information with the current available vendor data.

The inspector verified that all testing relative to the modification and its impact on MSIV functionality was conducted with acceptable results. Operability determinations will be made in accordance with the appropriate MSIV Technical Specification surveillance requirements prior to the operational mode in which operable MSIVs are required. However, in the case of Technical Specification 3.7.1.5, entry into MODE 3 is authorized prior to verifying and timing the

MSIV full closure operability provisions. The inspector noted that the subject design change had provided for an orifice in the thermal accumulator adapter. This feature limits hydraulic fluid flow through the relief valve to a value which precludes the MSIV from drifting closed with a failed-open relief valve. Thus, the design change implemented to correct the deficiency reported under 10 CFR 21 has not affected the operability criteria or the functioning and failure modes analysis of the MSIVs. Additionally, this modification does not alter the findings documented in NRC:RI inspection report 50-443/87-16 regarding the acceptable fail-safe application of the MSIV design.

The inspector had no further questions on the implementation of corrective action in response to 10 CFR 21 Report 87-88-06 and considers this item to be closed.

c. (Closed) Licensee Event Report 88-002: Technical Specification Surveillance Not Performed. This Licensee Event Report (LER) involved the discovery that certain circuit breakers which connect non-Class 1E devices to Class 1E power sources were omitted from the NHY Technical Requirements Manual (NYTR). Table 16.3-10 of the NYTR lists those circuit breakers that must be tested to satisfy the requirements of Technical Specification (TS) Surveillance 4.8.4.2. In all, three circuit breakers were omitted. The licensee review following the discovery of the first omission in February, 1988 was not comprehensive and in April, 1988 the additional two omissions were found by the licensee staff. The preliminary versions of the initiating station information reports (SIR 88-003 and 88-037) were reviewed by the inspector as well as the initial LER itself which was submitted to NRC by letter NYN-88031 on March 14, 1988. NHY updated the initial LER with Revision 1 on May 16, 1988 (NYN-88067).

As verification of the licensee review to ensure that no other breakers were omitted from the table, the inspector conducted an independent review of the 480 Vac unit substation drawings and the 460 Vac motor control center drawings and verified that all non-safety related loads were listed in the NYTR. No errors were found. This LER is closed.

6. NRC Bulletins, Generic Letters and Information Notices

a. (Open) IE Bulletin 85-03: Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings. The licensee responded to this bulletin with letters to the NRC dated May 15, 1986 (SBN-1052) and November 30, 1987 (NYN-87137). Review of these responses by the NRC Division of Operational Events Assessment has revealed the need for additional information to adequately assess the licensee program for valve operability.

Accordingly, a licensee on-site Regulatory Services Department representative was provided a copy of the "Request for Additional Information (RAI) RE: Review of Responses to Action Item e of IE Bulletin 85-03", which has been attached to this inspection report. The licensee has been requested to provide written reponse to this RAI within 30 days of the date of the cover letter transmitting both the inspection report and the subject attachment.

Pending licensee response, further review by the NRC and inspection followup of the MOVATS test program implementation, results evaluation and status of the remaining open action items, IE Bulletin 85-03 will continue to be tracked as an open item.

b. (Closed) NRC Bulletin 88-01: Defects in Westinghouse Circuit Breakers. This bulletin was initially discussed in NRC:RI Inspection Report 50-443/88-02. On March 30, 1988, NHY responded to the bulletin in letter NYN-88038. Welds were identified not to meet acceptance criteria in all of the Type DS-416 reactor trip breakers. New pole shafts were ordered from Westinghouse and installed. The inspector conducted a visual examination of a sample pole shaft replacement noting a significantly higher standard of welding than that found on the failed shafts.

Breaker reassembly is completed for the four main breakers and testing is in progress and well directed. The licensee also intends to repair the spare breaker. The inspector confirmed that the fifth new pole shaft is on order to accomplish the repair. This bulletin is therefore closed.

c. (Closed) Generic Letter 87-12: Loss of Residual Heat Removal While the Reactor Coolant System is Partially Filled. NRC Region I Inspection Report 50-443/88-02 described the NHY response to Generic Letter (GL) 87-12 "Loss of Residual Heat Removal While the Reactor Coolant System is Partially Filled". During this inspection period, Special Test Procedure (STP) No. 105, "RHR/Reactor Coolant Mid Loop Operation" was conducted. The purpose of the test, performed on April 5-9, 1988, was to determine the maximum residual heat removal (RHR) system flowrate while operating at the reactor vessel (RV) mid-plane without RHR pump cavitation. The information will be used to respond to GL 87-12 as well as provide setpoint data for the planned design change concerning shutdown RV level instrument improvements.

Two NRC inspectors provided around the clock coverage of the test. While witnessing the test, the inspectors verified the following:

- The test procedure and its changes were reviewed and approved as required
- A designated test director conducted the test
- Qualified personnel performed the test
- Test precautions and prerequisites were met
- QA personnel witnessed the test performance
- Special measuring and test equipment was installed as required
- Shift turnovers and pre-test briefings were thorough
- The procedure was technically adequate
- Testing was performed in accordance with the procedure
- Test personnel actions were correct and timely during the test
- Communications were established and maintained to support the test

The inspectors also verified tubing lineups and connections for the special level indicators. They observed the test procedure both in the control room and locally at the pumps. Licensee personnel took the conservative option when decisions regarding pump shutdown were made. For example, they did not wait for control room indication of erratic RHR pump motor current or RV level swings before tripping the RHR pumps. Their actions in this regard ensured that the pumps were not damaged without compromising the validity of the test results.

While witnessing the performance of step 6.2.17 of STP-105, the inspector noted that the "A" pressurizer spray valve did not indicate full open when the valve was open. The open and closed indicating lights remained illuminated when the operator went to the open position on the switch. The inspector asked the operator about the condition and was informed that work request 88W928 had already been generated to address the problem. He had no further questions on this matter.

The inspector observed that the test data sheets and data instructions did not appear to be as complete as they might have been. The instructions for data compilation were somewhat vague. It was determined, however, that the data actually recorded was appropriate in scope and frequency. The difficulty in standardizing level measurements due to different design references emphasized the necessity for implementation of design coordination report (DCR) 87-136 at the earliest possible opportunity. This DCR provides improvements in shutdown RV level instrumentation. The test results are to be shared with the Westinghouse Owner's Group to ensure lessons learned are translated into useful industry feedback.

The successful performance of this complex test is a credit to the Seabrook Station organization and is a notable initiative typical of NHY's efforts to expand the knowledge base of its operations staff. Inspection coverage of this test also closes out NRC Region I Temporary Instruction 88-02. The need for further inspection effort in this area will be determined following the issuance of further NRC information and guidance (e.g., an NRC Bulletin) on this subject.

d. NRC Information Notice 87-21: Shutdown Order Issued Because Licensed Operators Asleep While on Duty. This NRC Information Notice described events surrounding an incident at another nuclear facility where licensed operators were inattentive to duty. The inspector disclosed and discussed with several key licensee managers a list of lessons learned from this incident in order that those lessons could receive wide dissemination throughout the operations staff. He also discussed with licensee operators another related incident report regarding the discovery of suspected alcohol in the Control Room of another facility. Discussions with several shift crews at Seabrook demonstrated a responsible and professional attitude towards their watchstanding duties as well as a familiarity with the conditions of their operator's licenses. During normal and backshift inspection tours, all operators were noted to be alert and carrying out their duties in a professional manner. The inspectors had no further questions.

e. NRC Information Notice 88-13: Water Hammer and Possible Piping Damage Caused by Misapplication of Kerotest Packless Metal Diaphragm Globe Valves. This Information Notice was issued on April 18, 1988 to alert licensees of potential problems resulting from the improper application of Y-pattern packless metal diaphragm valves supplied by the Kerotest Manufacturing Corporation.

Kerotest globe valves are used in a variety of applications in nuclear power plant reactor systems and are designed to meet each owner's equipment specifications. However, incidents have occurred involving flow throttling and reverse flow as a result of the misapplication of these valves.

NHY systems support department established a matrix of these valves installed at Seabrook, and identified the propbability that each valve would experience flow throttling or reverse flow in its specific design application. In each case, the design feature of the valve is isolation for maintenance or calibration and therefore the Seabrook valves exhibit a very low probability that reverse flow usage could occur. The inspector had no further questions.

7. Follow-up Issues

a. Cold Overpessure Protection Actuation Circuitry

In January, 1987, Millstone Unit 3, a Westinghouse pressurized water reactor similar in design to Seabrook experienced an overpressure event in which certain systems designed to mitigate the effects of such an incident were determined to be inoperable. NHY was provided with a copy of the relevant NRC:RI Special Inspection Report, 50-423/88-03 and evaluated whether such an event could occur with the Seabrook design. The inspector reviewed licensee instrumentation drawings and discussed the event with cognizant licensee engineers. The actuating signals for the Seabrook low temperature/overpressure protection system are developed in the process instrumentation cabinets whereas the Millstone 3 design utilizes the solid state protection system. Therefore, it is not possible for a similar event to occur at Seabrook. The inspector had no further questions.

b. Environmental Qualification of Torque Switches in Limitorque Valve Operators. Following notification by the inspector, NHY contacted the Limitorque Corporation on the applicability of a problem identified at another nuclear facility. The deficiency related to the environmental qualification of torque switches installed in type SMB, size 00 motor operators. The Limitorque representative indicated that these switches are an earlier design than those used at Seabrook. The inspector had no further questions.

c. Emergency Feedwater Pump Turbine Tachometer Power Supply Isolation. The inspector notified the licensee of a report made by another pressurized water reactor under 10 CFR Part 50.55(e). This construction deficiency involved discovery that certain non-safety instruments powered off a 1E bus were not provided with electrical separation. Preliminary review of the Seabrook design indicates that this problem may also exist on the turbine driven emergency feedwater pump. The cognizant NHY system support engineer initiated a request for engineering services on April 28, 1988. The inspector will follow up licensee action on this issue (Open Item 88-06-01).

d. Design Coordination Report 87-401: Orifice Installation for Cooling Tower Direct Recirculation Lines. The inspector reviewed the subject design coordination report (DCR) which installs orifices downstream of service water valves SW-V-27 and 56 in the cooling tower. By orificing the flow in each train of cooling tower recirculation piping from the pumps to the spray headers, the need for throttling SW-V-27 & 56 is eliminated. Thus, the possibility of pipe or valve lining damage due to the flow turbulence associated with throttled valve operation is prevented.

Prior to implementation of this DCR, the design requirements prescribed that the subject valves be throttled to be 38% open at the "full-open" position of the valve stroke. By increasing the valve opening, the stroke time is affected, as - the relative valve position interlocks with other discharge va. which are required to prevent deadheading of the operating pumps. erefore, the inspector examined and evaluated the logic diagrams for the cooling tower pumps, discharge valves and recirculation valves to ensure that the detailed DCR analysis had considered all factors related to changing the "full-open" stroke position of valves SW-V-27 and 56. The inspector verified that automatic valve sequencing of both timing and opening had been considered for a range of design conditions and that engineering basis for the determination of the optimal valve open setting to coincide with the "full-open" position was well founded.

The inspector discussed orifice sizing with the cognizant system engineer, witnessed flow balancing and testing activities in progress, examined the completed orifice installation in the field and questioned component and system operability criteria during the period of time between orifice installation and the modifications to the valve opening settings. All design, engineering change, testing and modification work appeared to address the relevant safety issues and comply with existing programmatic controls. The inspector concurred with the licensee position that no unresolved safety question was raised as a result of this design change and noted that FSAR revisions had been drafted, where applicable.

Followup of both the processing and implementation of DCR 87-401 revealed no safety concerns or system redesign questions. No violations were identified.

e. Station Information Reports. The inspector conducted a routine review of several station information reports (SIR's). The reports were reviewed for compliance with the implementing instruction, supervisory review, regulatory services review and management review including SORC review. Also examined were the technical evaluation of each event, root cause analysis and recommendation. The inspector concluded that SIR's issued recently are of significantly higher quality than those issued previously. As such, they are a more useful tool in managing station operations and maintenance as evidenced by SIR 88-011 which has generated a maintenance procedure for J-10 relay replacements. Additionally, the turnaround time on these reports has improved. No violations were identified.

The inspector conducted a detailed review of SIR 88-025 concerning a trip of the train "B" diesel generator on low lube oil pressure on February 24, 1988. NHY investigation of the problem included a testing program with special instrumentation installed on the engine. The conclusion drawn in the SIR is that the cause of the trip was the unusually long time taken for the engine to reach rated speed. The extended starting time was too lengthy for oil pressure to build up prior to the completion of the time delay which re-institutes the low lube oil pressure trip. Licensee contact with the vendor revealed that the reason for similar failures is not fully understood, but is caused by fuel evaporation during a lengthy shutdown period (greater than one month).

The inspector met with the cognizant technical support department engineers and discussed the diesel failure. The NHY engineers believe that the problem is more likely caused by fuel draining from between the injector and injector pump causing a delayed start due to the additional time to get fuel to the cylinders. The inspector expressed concern that objective evidence did not exist to guarantee that the 31-day surveillance interval would preclude future slow starts. The licensee indicated that the diesel generator surveillances conducted in accordance with the Technical Specifications, to date, since issuance of the zero-power operating license, have demonstrated reliable operability. In fact, no similar failures have been reported during performance of the monthly surveillances. NRC routine inspection and SIR analysis will continue to verify reliability in the starting of the station's emergency diesel generators.

Additionally, TS surveillance requirement 4.8.1.1.3 requires that all diesel generator failures, valid or non-valid, shall be reported to the Commission in a Special Report pursuant to TS 6.8.2 within 30 days. Review of the licensee's interpretation of the reporting requirements of the subject diesel generator trip shall be tracked under Open Item 88-06-02.

8. Relocation of Seismic Monitoring Instruments. On April 28, 1988, the licensee submitted a letter (NYN-88061) to NRC:RI describing the relocation of three seismic monitors which have been rendered inoperable due to their location change. Triaxial Peak Accelerograph (1-SM-XR-6703), Triaxial Response Spectrum Recorder (1-SM-XR-6706) and Triaxial Peak Accelerograph (1-SM-XR-6702) have all been relocated with the new positions approved by NRC. The inspector confirmed that the above letter was sent pursuant to Technical Specification 3.3.3.3 and 6.8.2 because of operability considerations. Even though the instruments are fully functional, they are inoperable because they are located in an area different from that specified in the Technical Specifications. It is intended that the affected specification will be revised at its next issuance. The inspector noted that the instrument relocation has no effect on the ability of the monitors to perform their design functions, and had no further questions regarding this matter.

9. Security Log Review

The inspector reviewed the NHY Security Event Log for the current quarter, commencing on April 1, 1988. A log entry on April 24, 1988 was noted with regard to the missing security department procedure discussed in paragraph 3.b of this report. The inspector evaluated the licensee information in accordance with the reportability requirements of 10 CFR 73.71 and the guidance of USNRC Regulatory Guide 5.62 and NUREG-1304. In addition, the Security Department Supervisor and the contract guard force Chief of Security were interviewed regarding the security department implementation of a random chemical screening and testing program and routine follow-up actions taken by the licensee in response to incidents involving personnel fitness for duty.

As was documented in NRC inspection report 50-443/88-05, the random chemical screening program of the security force has resulted in five personnel terminations since the latter part of 1987. The inspector reviewed individual records and determined that three of these personnel were terminated as a result of prehire testing, while the other two individuals were assigned to the non-nuclear staff of the contractor guard force. None of the five subject personnel had been authorized clearance for protected area access. The inspector noted no entries in the Security Event Log for the current CY-88 second quarter relating to fitness for duty issues.

The inspector also verified that a copy of the Security Event Log for the first quarter, CY-88 had been submitted to the NRC in accordance with 10 CFR 73.71(c). An internal licensee summary report for the first quarter logable events was reviewed to confirm trending and analysis of each safeguards item. With respect to a more recent "safeguards" event involving the missing security procedure, the inspector examined the Station Information Report (SIR 88-39), noting root cause evaluation and corrective action and checking that the document was correctly classified and handled as "safeguards" information.

No violations were identified.

10. 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 May 20, 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, except for a list of discussion topics related to Generic Letter 87-12, as discussed in Section 2.b, relating to the visit by the Senior Reactor Engineer from NRR; the inspector follow-up questions generally discussed in Section 3.a of this report; and the RAI with respect to Bulletin 85-03, included as Attachment I to this report. In addition, lessons learned from an attentiveness to duty incident at another plant were disclosed and discussed with several key licensee managers.

On April 27, 1988, following a site inspection, the Regional Administrator and State Liaison Officer from NRC Region I held discussions with the NHY President and the Vice-President Nuclear Production. Discussion topics included the impact of PSNH bankruptcy on the Seabrook project, licensee emergency preparedness initiatives, the progress of hearing issues related to license issuance, observations from the site inspection and the licensee/NRC activities that will be required to support startup testing. Other items of mutual interest were also discussed. It was agreed that a management meeting with Region I should be scheduled sufficiently in advance of startup to discuss licensee startup plans with an emphasis on determining readiness to enter higher operating modes.

REQUEST FOR ADDITIONAL INFORMATION (RAI) RE:

Review of Responses to Action Item e of IE Bulletin 85-03

Licensee:
Public Service Company of
New Hampshire
P. O. Box 330
Manchester, New Hampshire 03105

Unit(s): Seabrook 1,2
Date of Response: 05-15-86
*11-30-87

*Initial Response
to Action Item f.

Respondent:
Robert J. Harrison,
President and Chief
Executive Officer

The information provided in your response to Action Item e of IE Bulletin 85-03 was found to be deficient in some areas. Provide the additional information necessary to resolve the following comments and questions:

1. Has water hammer due to valve closure been considered in the determination of pressure differentials? If not, explain.

2. If MOVATS is planned for application to some MOVs which are not included in its data base, commit to and describe an alternate method for determining the extra thrust necessary to overcome pressure differentials for these valves.

3. The AFW MOVs listed for testing in the response of 05-15-86 are shown on Drawing PID-1-FW-B20685 Revision 0, in discharge lines to the steam generators. Verify that these valves are truly paired in series to each steam generator. Page 27 of the WOG Report of March 1986 shows them paired in parallel.

4. On Page 2 of Table 1 of the response of 05-15-86, the design closing differential pressure of 850 psig is less than the maximum operating value of 2530 psig. Justify use of the lower pressure.

5. The proposed program for action items b, c and d of the bulletin is incomplete. Provide the following details as a minimum:

   (a) commitment to a training program for setting switches, maintaining valve operators, using signature testing equipment and interpreting signatures,

   (b) commitment to justify continued operation of a valve determined to be inoperable (for Unit 1 only),

(c) consideration of pipe break conditions as required by the bulletin, and

(d) stroke testing when necessary to meet bulletin requirements.