U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-443/89-07

Docket No. 50-443

License No. NPF-56

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: June 19-23, 1989

Inspector: L. J. Prividy — 8/16/89
L. J. Prividy, Senior Reactor Engineer — date

Approved by: P. K. Eapen — 8/16/89
P. K. Eapen, Chief, Special Test Programs Section — date

Inspection Summary: Routine Unannounced Inspection June 19-23, 1989 (Report No. 50-443/89-07)

Areas Inspected: The focus of this inspection is design, design changes, and modifications. Test results associated with the modifications were also reviewed.

Results. The inspector determined that the design change process was being properly controlled by Administrative Procedures. The modification packages including the post modification testing were adequate. Two unresolved items were identified concerning the evaluation of steam leakage through the auxiliary feed pump turbine control valves (Section 2.2) and the evaluation for leakage through reactor coolant system pressure isolation valves (Section 3.3). There were no violations.

DETAILS

1.0 Persons Contacted

Public Service Company of New Hampshire

*R. Belanger, Lead Compliance Engineer
*C. Beverly, Operational Programs Engineer
*S. Burhwald, QA Supervisor
*J. Cady, ISEG Supervisor
J. Connally, Program Support Test Engineer
S. Corcoran, Technical Support Engineer
*D. Covill, Surveillance Supervisor
*R. Cyr, Maintenance Manager
*E. Desmarois, Independent Review Team
*W. DiProfio, Assistant Station Manager
*R. Faix, Westinghouse Engineer in NHY Engineering
*T. Feigenbaum, Vice-President, Engineering and Quality
*J. Grillo, Operations Manager
*P. Gurney, Reactor Engineering Supervisor
*R. Gwinn, Operational Programs Engineer
J. Hanley, Training Manager
*G. Kann, Startup Manager
*G. Kline, Technical Support Manager
*J. Marchi, Audit and Evaluations
*R. Martel, Staff Engineer
*D. McLain, Production Service Manager
*D. Moody, Station Manger
*D. Perkins, Operational Programs Engineer
*J. Peterson, A.O. Manager
*R. Sherwin, Planning and Outage Manager
*E. Sovetsky, T.P. Supervisor
*W. Temple, Licensing Coordinator
J. Tipton, Technical Support Engineer
*J. Vargas, Manager of Engineering
*C. Vincent, QC Supervisor

U.S. Nuclear Regulatory Commission

*N. Dudley, Senior Resident Inspector
*D. Haverkamp, Reactor Projects Section Chief, Region I
*J. Trapp, Reactor Engineer, Region I

*Denotes those present at exit meeting.

2.0 Design Changes and Modifications (37700 and 37828)

The primary objective of this inspection was to ascertain that design changes and modifications were in conformance with the requirements of the Technical Specifications (TS), 10 CFR, the Safety Analysis Report, and the licensee's Quality Assurance program. This objective was accomplished by reviewing the following modifications: (1) Design Coordination Report (DCR) 87-422 "Replace Miniflow Valves RH-FCV-610 and 611" and (2) Minor Modification (MMOD) 89-542 "Modify Pilot Plug Vent Holes for MS-V393, V394 and V395." The inspector reviewed these modification packages and the installation of the design changes and verified the following:

- The modifications were technically sound and they adequately addressed the root cause for the change.
- The modification packages were reviewed and approved by onsite and offsite review organizations.
- Design changes and modifications were controlled by approved procedures.
- Post modification test procedures and results were adequately reviewed.
- Station procedure modifications were made prior to the modification being declared operable.
- Operator training was conducted prior to declaring the modification operable.
- Marked up copies of as-built drawings were distributed prior to declaring the modification operable. Also administrative controls were established to maintain as-built drawings.
- Preventive maintenance and inservice inspection and test programs were properly updated.
- Changes to the FSAR were properly controlled and updated.
- Installation of modifications conformed with design change package.

The licensee has been submitting quarterly reports of 10 CFR 50.59 safety evaluations concerning facility and procedure changes which were performed or implemented without prior NRC approval. The inspector reviewed the last four quarterly reports which covered the period of April 1, 1988 to March 31, 1989. These summaries contained sufficient modification information as required by 10 CFR 50.59.

2.1 References

- 10 CFR Part 50, (50.59, Appendix B and other Sections)
- Reg. Guide 1.33, Rev. 2, Quality Assurance Program Requirements (Operation)
- ANSI N45.2 - 19[illegible], Quality Assurance Requirements for Nuclear Facilities
- ANSI N45.2.11 - 1974. Quality Assurance Requirements for the Design of Nuclear Power Plants
- Seabrook Station Final Safety Analysis Report

2.2 Inspection Findings

Design changes and modifications at the facility are controlled by Engineering Procedure 34025, "Scoping, Planning and Scheduling of DCRs and Facilities Modifications and Other Engineering Activities." This procedure provides the necessary guidance to design, budget, schedule, develop, review and approve proposed design changes and facilities modifications. The key personnel involved in this process are the cognizant design engineers who are responsible for the detailed engineering work. After a DCR is completed and approved by the Station Operation Review Committee, it is implemented in accordance with Procedure No. MT 3.1, "DCR Implementation Plan." At this point the onsite technical support group assumes field engineering responsibility. The cognizant implementation engineer in the technical support group is usually the responsible system engineer who interfaces closely with the cognizant design engineer to ensure proper DCR implementation. The inspector verified adequate implementation of the licensee's modification programs using the attributes listed in Section 2. Additional observations and concerns are discussed below. These observations and concerns were based on an overall review of these modifications with the cognizant design engineer and plant walkdowns with the responsible system engineer.

DCR 87-422 - "Replace Miniflow Valves RH-FCV-610 and 611"
This modification had been designed and approved but not yet installed. Basically the modification entails changing the miniflow valve from a gate to a globe valve in the residual heat removal (RHR) system recirculation piping and modifying the piping arrangement. The modified design is intended to improve the RHR system performance from a vibration standpoint. Weld failures at instrument connections in this piping had occurred in mid-1987 and subsequently were attributed to flow induced vibration. In addition to the weld failures, problems were encountered with damage to the disc guides and motor operator for RH-FCV-611. Although all of

these problems occurred somewhat separately, they were all related in that they occurred in the same RHR system recirculation piping and a comprehensive problem resolution was not apparent until the development of DCR 87-422.

The inspector reviewed the calculation for the preliminary sizing of the new flow measurement orifice to be installed in the RHR system recirculation piping. The calculation was in accordance with good engineering practices with clear statement of the problem, assumptions, details and conclusions. The calculation was independently reviewed per the licensee's design control procedures. The inspector independently reviewed the calculation and verified that the bases of the calculations were technically sound.

MMOD 89-542 - "Modify Pilot Plug Vent Holes for MS-V393, V394 and V395"

This modification concerned the steam supply control valves for the emergency feedwater (EFW) pump turbine and it had already been implemented. It consisted of increasing the vent hole area in the pilot plug for these valves since the valves would not open as required under operating temperature, pressure and flow conditions. The post modification testing was performed satisfactorily in accordance with Procedure No. STP-101, "Turbine Driven Emergency Feedwater Start Verification Test," to demonstrate operability. The results of this acceptance testing indicated that steam leakage through these valves while in the closed position was determined to be acceptable. The inspector discussed the basis for this determination with the licensee's program support and engineering managers who indicated that Engineering Evaluation 89-021 dated June 20, 1989 thoroughly evaluated and accepted this valve leakage. Part of this evaluation considered the impact of steam leakage on the four-inch check valves, MS-V94 and V96 which are located downstream of MS-V393 and V394, respectively. The licensee concluded in the evaluation that valve leakage should have no detrimental effects on system operation and thus, safe shutdown capability was not compromised. The licensee recognized that valve leakage should be repaired in accordance with established maintenance practices and that check valves MS-V94 and V96 should be inspected for possible seat wear and hinge pin and hanger wear. However, when the inspector conducted a plant walkdown with the system engineer, steam leakage through MS-V394 was sufficient to cause repetitive slamming of the disc (approximately 2-3 cycles per second) of check valve MS-V96. The inspector was concerned that this repetitive slamming had the potential to cause fatigue in the internals and eventual valve failure. The licensee's engineering manager inspected these in-plant conditions and acknowledged the inspector's concern. It appears that Engineering Evaluation 89-021 did not adequately consider the negative impact of increasing steam leakages on system performance. Further evaluation is required not only to determine a resolution to the steam leakage

problem through the control valves but also to ensure a comprehensive engineering solution for a reliable steam supply system for the EFW pump turbine. This item is unresolved (50-443/89-07-01) pending the completion of maintenance on these valves and a reevaluation of the acceptability of EFW pump-steam isolation valve leakage especially during extended power operation when steam leakages could increase.

3.0 Other Engineering and Technical Support Issues (37700 and 73756)

In addition to the detailed review of the modifications discussed in Section 2 above, the inspector reviewed (1) engineering involvement in procedure development and (2) engineering response to RHR system check valve leakage.

3.1 Engineering Organization and Training Observations

- The design and technical support engineering staffs are experienced with many engineers having been at the plant during construction.

- Engineering management was generally well informed and abreast of technical issues.

- Engineers and engineering management were responsive to NRC concerns. The inspector particularly noted the engineering manager's personal, in-plant inspection of the MS-V96 check valve conditions.

- Based on a discussion and review with the training manager, the inspector concluded that adequate training is available for the engineering staff.

- The total engineering work effort was adequately controlled. Such work includes response to DCRs, MMODs, engineering evaluations, requests for engineering services (RESs), and special engineering activities (e.g., task forces). However, the licensee recognized that the number of open RESs of 650-700 was high and the engineering organization was making a concerted effort to reduce this work backlog.

3.2 Engineering Involvement in Procedure Development

The inspector reviewed several pump and valve surveillance test procedures to determine if the tests adequately verified component and system requirements. The inspector concluded that the surveillance test procedures selected for review were technically adequate. For example, surveillance test procedure OX1436.13 periodically tests the backflow capability of check valves MS-V94 and V96. Also, the EFW pump discharge check valves are periodically

tested for backflow. However, upon further review of the feedwater system, the inspector noted that there is no periodic backflow testing for the six-inch, normally closed, check valves FW-V216 and V357. These valves are in the startup feed pump discharge piping which connects into the EFW pump discharge header. Both valves would be required to remain functional under a seismic event since the seismic boundary is designated immediately upstream of FW-V216. These valves are tested properly in the forward flow direction. The program support manager noted that there is no easy way to individually test these valves in the backflow direction. However, he noted that it may be possible without any hardware changes to modify existing test procedures and perform a backflow test monitoring for combined leakage through both valves. The program support manager agreed to evaluate the feasibility of such a test. The inspector had no further comment.

3.3 Engineering Response to RHR System Check Valve Leakage

On June 19, 1989 while performing the RHR A pump surveillance test during Mode 2 where the RHR pump was operating on recirculation flow, leakage past the RHR to SI cold leg injection check valves (RH-V15, RH-V29, RH-V30 and RH-V31) was observed by the licensee. This leakage was apparently due to a reduced differential pressure across the check valves, while running the RHR pump, allowing flow into the RHR system from the safety injection accumulators. During this time, the RHR system was pressurized above the RHR pump suction relief valve set pressure of 450 PSIG. This relief valve lifted and approximately 500 gallons of water was drained from the four SI accumulators. Pressure in the system was subsequently vented off through the safety injection test line header to the RWST via RH-V28, SI-V62 and SI-V70. In order to perform the RHR pump surveillance (DX1413.01), a procedure change was made to maintain the SI test header vent path to the RWST open for the duration of the test.

RES 89-381 (June 22, 1989) was issued to address concerns expressed by the inspector and the technical support staff. This RES requested the licensee's design engineering group to evaluate the ability of the present Reactor Coolant System (RCS) pressure isolation valves (PIVs) installed to effectively isolate lower pressure ECCS piping at lower differential pressures. The inspector noted that check valves RH-V15, V29, V30 and V31 had been tested on June 1, 1989 in accordance with the PIV requirements of the Technical Specifications and only RH-V30 had a measurable leakage of 94 ml/minute. The maximum allowable leakage for these valves per Technical Specifications Table 3.4-1 is 3.0 GPM. However, most of this leak testing is done with higher differential pressures. This item is unresolved (50-443/89-07-02) pending satisfactory resolution of RES 89-381 by the licensee concerning an evaluation of the ability of the RCS PIVs installed to effectively isolate lower pressure ECCS piping at lower differential pressures.

4.0 Licensee Action on Previously Identified Open Items

(Closed) Unresolved Item (50-443/88-11-01): This item was opened to address a potential problem of reporting modification work items as complete when they actually were incomplete. Specifically, certain work associated with Engineering Change Authorization (ECA) 05/112374 was determined to be incomplete when it should have been completed. This item was left unresolved pending the licensee's verification that all work associated with ECA 05/112374 was complete as this was an isolated instance. Subsequent to these developments, additional reviews and plant walkdowns concerning field verification were conducted by the licensee's quality group to determine proper implementation of randomly selected ECAs. The results of these reviews and plant walkdowns are given in licensee memo NQG No. 89244 dated May 3, 1989 and they concluded that no generic problem existed in this area. Based on NRC inspection report 50-443/88-10 which verified the licensee's action in this regard, the above item is now closed.

5.0 Unresolved Items

Unresolved items are matters about which more information is required to ascertain whether they are acceptable, deviations or violations. Two unresolved items were identified during this inspection and they are discussed in Sections 2.2 and 3.3.

6.0 Management Meetings

Licensee management was informed of the scope and purpose of the inspection at the entrance meeting conducted on June 19, 1989. The findings of the inspection were discussed with licensee representatives during the course of the inspection. An exit meeting was conducted on June 23, 1989 at the conclusion of the inspection (see Section 1 for attendees) at which time the licensee management was informed of the inspection results.

At no times during this inspection was written material provided to the licensee. The licensee did not indicate that proprietary information was involved within the scope of this inspection.