U. S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NOS: 50-272/94-27
50-311/94-27
50-354/94-25

LICENSEE: Public Service Electric and Gas Company

FACILITY: Salem 1 & 2 and Hope Creek Generating Stations
Hancocks Bridge, New Jersey 08038

INSPECTION AT: Hancocks Bridge, New Jersey

DATES: September 26 - October 7, 1994, and telephone exit on December 12, 1994

INSPECTORS: J. A. Calvert, Reactor Engineer, DRS
D. T. Moy, Reactor Engineer, DRS

INSPECTOR: D. T. Moy for John Calvert 12/21/94
John Calvert, Reactor Engineer
Electrical Section
Division of Reactor Safety
Date

APPROVED BY: William H. Ruland 12/21/94
William H. Ruland, Chief
Electrical Section
Division of Reactor Safety
Date

<u>Areas Inspected</u>: This was an announced inspection of the Salem and Hope Creek station engineering program by regional personnel to determine the effectiveness of the licensee's engineering staff in providing technical support to the safe operation of the Salem and Hope Creek Nuclear Generating stations.

No significant concerns were identified.

DETAILS

1.0 PURPOSE

The objective of this safety inspection was to determine the effectiveness of the Public Service Electric & Gas Company's (PSEG) engineering organizations in providing design changes, modifications, and technical support to the safe operation of the Salem and Hope Creek Nuclear Generating Stations. This objective was accomplished by performing a review of selected plant major and minor modification packages. The inspectors reviewed the packages for the proper preparation, installation, and closeout and verified the following:

- Design changes and modifications were controlled by approved plant procedures;
- Modification packages were reviewed and approved by onsite and offsite review organizations;
- Installation procedures were adequately reviewed;
- Changes to the design as described in the FSAR and plant operating procedures were adequately controlled.

2.0 DESIGN CHANGE PROCESS

The inspectors reviewed the licensee's design modification process, which is described in procedure "Control of Design and Configuration Changes, Tests and Experiments," NC.NA-AP.ZZ-0008(Q), Revision 6, approved on March 30, 1994. This procedure established a uniform method or road map for design and configuration changes to determine the appropriate change package workbook. This procedure directs the user to the proper workbook from a series of six (6) books that discuss the following types of changes: standard design changes; engineering change authorization; equivalent and obsolete piece part replacement; document change only; as-built documentation. This procedure was previously reviewed and found to be adequate by NRC inspectors in Combined Inspection Report 50-272/94-07, 50-311/94-07, and 50-354/94-05.

3.0 DESIGN CHANGES AND MODIFICATIONS

The licensee implements design changes and modifications using the "Standard Design Changes Workbook One", Procedure NC.DE-WB.ZZ-0001(Q), Revision 4, which was last reviewed by the NRC staff in Inspection Report No. 94-07. Major modifications are prepared, documented and implemented using the above procedure.

The inspectors reviewed the modification packages to verify the effectiveness of the licensee's design change process described below.

3.1 DCR IEC-3316, Throttling Service Water Flow to Number 11 Component Cooling Heat Exchangers (CCHX)

This design change was intended to limit the maximum potential flow for service water pump in order to mitigate net positive suction head (NPSH) concerns. These concerns were originally identified in an engineering discrepancy (DEF DES-90-01434). The actual determination that available NPSH may be insufficient under certain design conditions was confirmed upon performing various hydraulic evaluations using the service water system hydraulic flow model (Calculation No. S-C-SW-MDC-131710). The licensee used this flow model which indicated that available service water pump NPSH with a single active component failure of CCHX control valves failing to the full open position, may not be maintained within the pump manufacturer's requirements. Therefore, this DCP introduced an additional flow restriction into the service water flow path for the No. 11 CCHX. The additional flow restriction was primarily accomplished by the throttling. The number 11 CCHX service water inlet isolation valve 11SW121 from the existing full open locked position to a partially open locked position.

The inspectors reviewed the modification package documents and interviewed design engineering, Station QA and operations staff to determine the quality of the engineering process for this modification. During the course of the audit, the inspectors had the following observations:

1. The purpose of the design verification procedure (NC.DE-AP.ZZ-0010(Q)) is to establish a controlled method and standard guideline for performing an independent design verification. As required by this procedure, the design verification should include a summary statement sufficiently clear to identify the method, extent and depth of verifications. Furthermore, any engineering judgement used during the verification process should also be included in the summary statement. The inspectors noted that the summary verification statement for this DCP was not detailed in compliance with the above procedure. The inspectors brought the concern to the licensee and they acknowledged the inspectors' observations.

2. The "Certification for Design Verification" section of NC.DE-AP.ZZ-0010(Q) procedure stated that the design verification signature indicates that the questions on the generic checklist have been reviewed for applicability. However, when the inspectors reviewed the checklist, they found the following statements were incorrectly checked "not applicable," but would apply to this modification:

    (a) Are assumptions necessary to perform the design activity adequately described and reasonable; and

    (b) Is the output reasonable compared to inputs?

The inspectors brought the above two observations to the licensee and PSE&G acknowledged the observations.

3.2 Small Design Change Packages

The inspectors reviewed the small change or equivalent replacement design process by document examination and personal interviews. This workbook (DE-WB.ZZ-0003(Q) is used to replace present plant components when the component is obsolete, unavailable, or unreliable. This procedure applies to "one-for-one" component replacement only. The replacement component must be evaluated to be equal to the original component or to the design application requirements. If this criteria is clearly documented as meeting the equivalent replacement 10 CFR 50.59 review and safety evaluation, then no unreviewed safety question is involved or revision to the safety analysis report (SAR) is required.

The inspectors selected the following replacement modification packages for review:

1. Change Package IEO-2353, package number 001, which replaced 12MS131 Mark No. FA109 with Equivalent Valve Mark No. F-9. This valve is the root valve for PT 526 and serves as part of the containment isolation pressure boundary.

2. Change Package IEO-2309, package number 001, which replaced 11SJ17 with an equivalent valve of a non-seal welded canopy style, W.O-930709175.

3. Change Package 2EO-2334, package number 001, which replaced Valve Internal (stem/plug/cage) 2PR1 and 2PR2 17-4 pH material with material more suitable to the actual application.

4. Change Package IEO-2362, package number 001, which was concerned with an air operator that did not cycle solenoid. The replacement was obsolete and the new model required is K831654E.

The inspectors reviewed the equivalent replacement evaluation form for each of the above packages. Some of the system parameters of the present design/specification and replacement component were different within each modification. For example, component overall size and material, flow characteristic and the flow coefficient C_v were different. Each modification detailed the justification for those changes and a safety analysis was not required to be completed by Workbook ZZ-0003. The inspectors did not identify any technical concerns with the small charge process as applied to the above change packages.

4.0 STATION QUALITY ASSURANCE (SQA) AUDITS

The inspectors reviewed station quality assurance procedure, "QA Interface for E&PE Projects," ND.QA-AP.ZZ-0024(Q), Revision 1. This nuclear quality assurance procedure provides direction for review and reports to configuration change or design change packages. However, SQA is not responsible for complete review of the design package, only spot check the modification package and changes. The inspectors reviewed several SQA station QA audit and surveillance reports relating to plant operations.

4.1 Station Quality Assurance Audit on PORV Internal Part Replacement - DCP-2EC-3190

During an unusual event Salem Unit 1 on April 7, 1994, the pressurizer power-operated relief valves (PORV) had cycled over 200 times in 35 minutes. The licensee decided to remove and inspect these valves to determine the material condition of the valve after this event. The licensee determined that the stem exhibited cracking around the roll pin area where it penetrates the pin collars. The purpose of this audit was to review the development and implementation of DCP-2EC-3190, which permitted installation of parts different than that originally intended for the Unit 2 PORVs (see Salem Inspection Reports 94-80, 94-11, and 94-13 for additional information). The licensee identified following observations relating to this event:

(a) Shared responsibility between E&PB and Salem maintenance department for installation of DCP was not well-defined nor understood by participants;

(b) When receiving material from storeroom for field installation, there was no existing process or procedure in place to verify that the "MMIS Issue Ticket" material is correct or accounted for;

(c) The modification instructions in the DCP did not provide sufficient detail to identify the upgrade material for the new valve internal to be installed; nor did these installations identify the proper procedure to be used to perform these activities. Because of the shared responsibility to implement this DCP and verifying its proper installation was deficient;

(d) Station maintenance personnel for DCP 2EC-3190 had not attended nor have they been scheduled for DCP training; and

(e) At the time of 100% DCP submittal to SQA, code job package (CJP) for valve internal parts replacement for 2PR1 and 2PR2 were not identified in Section 13.0 of DCP 2EC-3190.

The inspectors reviewed the audit report and concluded that the licensee's findings were appropriate.

5.0 INTEGRATION OF REGULATORY DOCUMENTS IN THE OPERATIONAL PROCESS

Regulatory documents (generic letters) are processed in accordance with PSE&G Nuclear Procedure, "Nuclear Licensing and Reporting," NC.NA-AP.ZZ-0035(Q), Revision 4. The licensee's commitment control involves a three-step process: (1) the identification of items requiring formal responses; (2) the preparation and approval of those responses; and (3) the tracking and closeout of commitment associated with the responses. The identification of items requiring formal responses is achieved through various administration programs.

For generic letters, the licensing representative reviews the documents and notifies the applicable department managers/designer of the required actions. Also, the representative enters the commitment into a database called "Action Tracking Task System," which is part of the MMIS.

The inspectors requested status information with regards to NRC Generic Letter 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWR pursuant to 10 CFR 50.54(F)." The information provided by the database indicated that the licensee had responded to GL 92-04 and a subsequent request for additional information. Because of the safety significance of the issue, additional information was required by the licensee, and this generic letter was re-issued as NRC Bulletin 93-03. The inspectors verified the licensee's response (NLR-N93126) to this bulletin and the acceptance by NRC staff on February 23, 1994.

The inspectors also reviewed the data for Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendation for Thermal-Hydraulic Instabilities in Boiling Water Reactors." The inspector verified Generic Letter 94-02 and responded to the staff on September 9, 1994. The inspector did not identify any discrepancy between the licensee's response and their implementation, and no unsafe conditions were noted.

6.0 USE OF PROBABILISTIC RISK ASSESSMENT (PRA)

The licensee has started a program for the integration of PRA into the engineering organization. The inspections verified that there is a PRA department engineering specialist familiar with the concepts of PRA. The licensee had submitted both Salem and Hope Creek's individual plant examination (IPE) to the NRC staff for approval, and the external event analyses (IPEEE) will be submitted for Salem and Hope Creek by May 1995 and February 1996, respectively.

The licensee stated that present training consists of the following: (a) PRA overview course, which is held quarterly; (b) PRA applications course, which was a 1-day course for nuclear engineering and reliability/assessment engineering for both Salem and Hope Creek Stations; (c) IPE descriptions and findings 2-hour seminar; and (d) discussion of IPE results in the licensed operator requalification training. The inspectors considered the training as adequate at this stage of the IPE submittal.

7.0 EXIT MEETING

Following the conclusion of the inspection on October 7, 1994, the inspectors had a telephone conference with licensee representatives denoted in Attachment 1 on December 12, 1994. The inspectors summarized the scope and results of the inspection at that time. The licensee acknowledged the inspection findings. The inspectors neither received or reviewed any proprietary material during the inspection.

ATTACHMENT 1

PERSONS CONTACTED

Public Service Electric and Gas Company

	C. Atkinson	Instrumentation & Controls Supervisor, Hope Creek
	J. Bailey	Nuclear Engineering Science Manager
	R. Chranowski	Electrical Technical Engineer, Salem
	J. Clancy	Technical Manager, Hope Creek
	A. Culliton	Standard and Assurance Supervisor
	R. Donges	Senior Staff Engineer, Standards
*	G. Englert	Nuclear Engineering Standards Manager
	E. Karpe	Principal Engineer, QA Programs
*	C. Lambert	Nuclear Engineering Service Manager
	A. Pasricha	Supervisor, Chemistry Process
	D. Patel	Civil Mechanical Engineer
*	J. Ranalli	Manager, Nuclear Mechanical Engineering
*	R. Ritzman	Lead Engineer, Nuclear Licensing
*	W. Scholl	Technical Staff Engineer, Hope Creek
	D. Smith	Principal Engineer, Nuclear Licensing
	F. Thompson	Licensing and Regulations Manager

State of New Jersey

R. Pinney	Nuclear Engineer

U.S. Nuclear Regulatory Commission

D. Moy	Reactor Engineer, Region I

* Denotes those present at the telephone exit meeting on December 12, 1994.