PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
Nuclear LLC

MAY 1 4 2007

10CFR50.59(d)(2)

LR-N07-0105

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Salem Nuclear Generating Station Units 1 and 2
Facility Operating License Nos. DPR-70 and 75
NRC Docket Nos. 50-272 and 50-311

Subject: Report of Changes, Tests and Experiments

Pursuant to the requirements of 10CFR50.59(d)(2), PSEG Nuclear LLC (PSEG) forwards a summary of changes, tests and experiments implemented at Salem Units 1 and 2 during the period May 13, 2005 through May 1, 2007.

Should you have any questions regarding this transmittal, please contact E. H. Villar at (856) 339-5456.

Sincerely,

Thomas P. Joyce
Salem Site - Vice President

IE47

MAY 1 4 2007

C Mr. Samuel Collins, Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

USNRC Resident Inspector Office – Salem (X24)

Mr. R. Ennis, Project Manager – Hope Creek and Salem
U. S. Nuclear Regulatory Commission
Mail Stop 08B3
11555 Rockville Pike
Rockville, MD 20852

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
P. O. Box 415
Trenton, NJ 08625

Attachment
LR-N07-0105

SUMMARY OF CHANGES TESTS AND EXPERIMENTS
HOPE CREEK STATION

1) Reconfiguration of the Reactor Core for Salem Unit 2 Cycle 15 – Safety Evaluation for Operating in all Modes (S2005-005)

The proposed activity was the re-configuration of the reactor core for the fifteenth cycle (Cycle 15) of operation at Salem Unit 2. In addition, several minor nuclear/mechanical fuel design changes were implemented on the fresh (new) fuel assemblies inserted into the new core. A 10CFR50.59 evaluation was performed which focused on the effects of the proposed changes on the safety analyses contained in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). An approved Design Change Package relocated four Shutdown Bank A Control Rod Drive Mechanisms (CRDMs) and four Rod Cluster Control Assemblies (RCCAs). The new reactor configuration for Salem Unit 2 Cycle 15, in conjunction with the new RCCA pattern, was accounted for the in the 10CFR50.59 evaluation.

The NRC approved "Westinghouse Reload Safety Evaluation Methodology" was the process used to determine the effect of the proposed changes on key parameters which comprise the nuclear design dependent input to the Salem UFSAR Chapter 15 accident analysis. All of the nuclear design dependent key input assumptions contained in the licensing basis accident analyses remain bounding for Cycle 15 operation except for the input assumptions associated with the Zero Power Hypothetical Steamline Break (SLB) event. The Zero Power Hypothetical SLB event was the only event to be re-analyzed as a result of the proposed changes. The re-analysis demonstrated that the UFSAR described design functions of the reactor core are maintained. For all events other than the Zero Power Hypothetical SLB event, the conclusions of the Salem UFSAR remain valid.

Based upon the analyses performed, it was concluded that the proposed changes to the Salem Unit 2 Cycle 15 reactor core did not result in the acceptable safety limits for any accident being exceeded. The evaluation concluded that the criteria of 10CFR59.59(c)(2) was met and the activity was implemented without obtaining NRC approval.

2) Bases Change for Back-up Power Supply to Boration Flow Source (S2005-004)

Amendments 264 (U1) and 246 (U2), which were approved by the NRC, modified the definition of "OPERABLE" and revised requirements for back-up emergency electrical power sources in Modes 5 and 6. Similar technical specification changes had been implemented for Modes 1 through 4.

SUMMARY OF CHANGES TESTS AND EXPERIMENTS
HOPE CREEK STATION

Prior to Amendments 264 and 246, both normal and emergency power sources (i.e. offsite and onsite power) were required in Modes 5 and 6 in order to meet the definition of operable. If one of these required power sources became inoperable, Technical Specifications (TS) required suspension of core alterations, positive reactivity changes, or movement of irradiated fuel.

Following the issuance of Amendments 264 and 246, either normal or emergency power sources are required for operability, and TS provide an option to suspending fuel movement when a required power source becomes inoperable.

With the issuance of Amendments 264 and 246, a change to TS Bases 3/4.1.2 (Boration Systems) was needed to provide consistency with the newly issued Amendment. Specifically, the TS Bases were modified to require either offsite or onsite power to the boration source, rather than the requirement for a backup power supply from an OPERABLE diesel generator in all cases.

3) Exclusion Area Boundary, Low Population Zone, Control Room Doses Due to Non-LOCA Releases (S2006-075)

The 50.59 evaluation addressed the revised radiological consequence analysis that incorporated a number of parameter changes as a result of the replacement of the Unit 2 Steam Generators, including the modeling of the Control Room Emergency Air Conditioning System (CREACS) response to the Rod Ejection Accident, the modeling of unit-specific control room atmospheric dispersion factors (X/Q), Reactor Coolant System volume and average temperature. Radiological consequence analyses for the following design basis accidents (DBAs) were revised:

Rod Ejection Accident (REA)
Steam Generator Tube Rupture (SGTR)
Main Steam Line Break (MSLB)
Reactor Coolant Pump Locked Rotor Accident (RCPLRA)

4) Containment Fan Coil Unit (CFCU) Margin Recovery (SCN 06-007)

This 50.59 addressed an UFSAR change to reduce the CFCU heat removal rate credited in design basis accident conditions. This affected a component parameter described in UFSAR Section 6.2.2.2, "Containment Fan Cooling System" and in Section 15.4.8, "Containment Pressure Analysis". The reduction in the assumed CFCU accident heat removal rate required a revision to UFSAR Section 15.4.8, "Containment Pressure Analysis" for both a main steam line break (MSLB) and a loss-of-coolant-accident (LOCA). A revised analysis was done by Westinghouse and is documented in

Attachment
LR-N07-0105

SUMMARY OF CHANGES TESTS AND EXPERIMENTS
HOPE CREEK STATION

WCAP 16503-NP, Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit Margin Recovery Project. The significant changes to the analysis are (1) reduced CFCU accident heat transfer rate and (2) when evaluating MSLBs, crediting the SGFP trip function to limit the energy released into the containment from the faulted SG if the feedwater control valve is assumed to fail open.

The revised analysis demonstrated that the peak containment temperature and pressure following a DBA remains below the present design values. These changes were not considered a change to the accident methodology.