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10CFR50.59(d)(2)

U. S. Nuclear Regulatory Commission
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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: Revised Report of Changes Tests and Experiments

Reference: Report of Changes Tests and Experiments, Dated May 14, 2007

Pursuant to the requirements of 10CFR50.59(d)(2), PSEG Nuclear LLC (PSEG) forwards a revised summary of changes, tests and experiments implemented at Salem Units 1 and 2.

This revised letter incorporates a 10CFR50.59 associated with the elimination of Steam Generator snubbers and corrects a typographical error in Attachment 1. Revision marks identified the new information.

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

Sincerely,

for RCB

Robert C. Braun
Salem Site - Vice President

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NRR

95-2168 REV. 7/99

AUG 1 0 2007

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SUMMARY OF CHANGES TESTS AND EXPERIMENTS
SALEM GENERATING 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) Technical Specification Amendments 264 (U1) and Amendment 246 (S2) (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.

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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 current 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 Generator, 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

**SUMMARY OF CHANGES TESTS AND EXPERIMENTS
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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.

5) Steam Generator Snubber Elimination – Salem Unit 2

The 50.59 evaluation addressed the removal of two snubbers on the reactor side of each steam generator. The two backside snubbers will remain in place but will be converted into rigid compression struts. The control valves and most of the hydraulic fluid will be removed. A compression collar clamp to the snubber body will convert the remaining snubbers into rigid single direction bumpers. Once the collar is installed and the hydraulic fluid from the snubber is drained, the snubber body is used with the collar to transmit compression loads from the steam generator to the building.

As part of this activity, the reactor coolant loop is reanalyzed. A non linear time history seismic analysis is performed. Time history analysis is also performed for pipe break loading. The revised configuration of the Steam Generators upper supports is incorporated into the loop analysis. Revised loads and stresses are evaluated for the loop piping, primary equipment supports, primary equipment nozzles, main steam and feedwater line piping, piping attached to the loop, and building structure embedment. Leak before break is verified to remain applicable to the primary loop piping.