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United States Nuclear Regulatory Commission
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SALEM GENERATING STATION – UNIT 1
FACILITY OPERATING LICENSE NO. DPR-70
NRC DOCKET NO. 50-272

Subject: **CHANGES TO LCR S07-01**
ONE TIME LICENSE CHANGE REQUEST
STEAM GENERATOR ALTERNATE REPAIR CRITERIA (17 INCH INSPECTION DISTANCE)

References: (1) Letter from PSEG to NRC: LCR S07-01, "One Time License Change Request, LCR S07-01, Steam Generator Alternate Repair Criteria (17 Inch Inspection Distance), Salem Nuclear Generating Station - Unit 1, Docket No. 50-272, Facility Operating License DPR-70," dated January 18, 2007

(2) Letter from PSEG to NRC: "Response to RAIs on LCR S07-01, One Time License Change Request, LCR S07-01, Steam Generator Alternate Repair Criteria (17 Inch Inspection Distance)," Salem Nuclear Generating Station - Unit 1, Docket No. 50-272, Facility Operating License DPR-70," dated February 23, 2007

(3) Letter from PSEG to NRC: "Supplemental Information on LCR S07-01, One Time License Change Request, LCR S07-01, Steam Generator Alternate Repair Criteria (17 Inch Inspection Distance)," Salem Nuclear Generating Station - Unit 1, Docket No. 50-272, Facility Operating License DPR-70," dated March 9, 2007

In References 1, 2 and 3 PSEG Nuclear LLC (PSEG) submitted information to support License Change Request (LCR) S07-01; to amend the Technical Specifications (TS) for a proposed one time license change for a 17 inch inspection length alternate repair criteria for Salem Unit 1 Steam Generators.

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Following additional discussion with the NRC Staff, PSEG is submitting changes to the proposed TS wording that make the language in the TS more precise and clear. These changes are shown in Attachment 1.

The information provided in this letter supplements the analysis for the original license change request provided in Reference 1. The conclusions of the No Significant Hazards Consideration and the Environmental Consideration provided in Reference 1 are not affected by this supplemental information.

In accordance with the requirements of 10 CFR 50.91(b)(1), a copy of this letter has been sent to the State of New Jersey.

PSEG requests approval of this license change request by March 27, 2007 to support the Salem Unit 1 Spring outage.

If you have any questions or require additional information, please do not hesitate to contact Mr. Jamie Mallon at (610) 765-5507.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 3/22/07
(Date)

Sincerely,

Thomas P. Joyce
Site Vice President
Salem Generating Station

Attachment (1)

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

Mr. R. Ennis, Licensing Project Manager - Salem
U. S. Nuclear Regulatory Commission
Mail Stop 08B1
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USNRC Senior Resident Inspector – (Salem X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
P. O. Box 415
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Additional Changes to the Technical Specifications

outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gallon per minute per SG.

3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

INSERT A HERE

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria.

INSERT B HERE

The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and,

INSERT A

The following repair criteria are applicable only for Refueling Outage 18 and the subsequent operating cycle:
In lieu of the 40% of the nominal wall thickness repair criteria, the portion of the tube within the tubesheet of the inspected SGs shall be plugged in accordance with the following alternate repair criteria: Tubes with flaws located below 17 inches from the top of the tubesheet may remain in service regardless of size. Tubes with flaws identified in the portion of the tube from the top of the tubesheet to 17 inches below the top of the tubesheet shall be plugged on detection.

INSERT B

In lieu of the above, the following inspection criteria are applicable only for Refueling Outage 18 and the subsequent operating cycle:
The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube beginning 17 inches below~~from~~ the top of the tubesheet on the tube hot leg side to 17 inches below the top of the tubesheet on the tube cold leg side.

2. WCAP-8385, Power Distribution Control and Load Following Procedures - Topical Report, September 1974 (W Proprietary) Methodology for Specification 3/4.2.1 Axial Flux Difference. Approved by Safety Evaluation dated January 31, 1978.

3. WCAP-10054-P-A, Rev. 1, Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code, August 1985 (W Proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved for Salem by NRC letter dated August 25, 1993.

4. WCAP-10266-P-A, Rev. 2, The 1981 Version of Westinghouse Evaluation Model Using BASH Code, Rev. 2. March 1987 (W Proprietary) Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved by Safety Evaluation dated November 13, 1986.

5. WCAP-12472-P-A, BEACON - Core Monitoring and Operations Support System, Revision 0, (W Proprietary). Approved February 1994.

6. CENPD-397-P-A, Rev. 1, Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology, May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

6.9.1.10 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with the Specification 6.8.4.i, "Steam Generator (SG) Program." The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date, and

INSERT C HERE

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

INSERT C

h. The following reporting requirements are applicable only for Refueling Outage 18 and the subsequent operating cycle:
The number of indications detected in the upper 17 inches of the tubesheet thickness along with their location, measured size, orientation, and whether the indication initiated on the primary or secondary side.

i. The following reporting requirement is applicable only for Refueling Outage 18 and the subsequent operating cycle:
The operational primary to secondary leakage rate observed in each steam generator during the cycle preceding the inspection and the calculated accident leakage rate for each steam generator from the lowermost 4 inches of tubing (the tubesheet is nominally 21.03 inches thick) for the most limiting accident. If the calculated leak rate is less than 2 times the total observed operational leakage rate, the 180 day report should describe how the calculated leak rate is determined.