UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 30, 2015

Mr. Robert Braun
President and Chief Nuclear Officer
PSEG Nuclear LLC
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING ADOPTING TECHNICAL SPECIFICATION TASK FORCE TRAVELER (TSTF)-510, REVISION 2, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION" (TAC NOS. MF4574 AND MF4575)

Dear Mr. Braun:

The Commission has issued the enclosed Amendment Nos. 309 and 291 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. These amendments consist of changes to the technical specifications (TSs) in response to PSEG Nuclear LLC's application dated July 28, 2014, as supplemented by letter dated January 15, 2015.

The amendments revise TS requirements regarding steam generator tube inspections and reporting as described in Technical Specifications Task Force (TSTF) Standard Technical Specifications Change Traveler (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection." In addition, the amendments revise Salem, Unit No. 2 TSs to remove unnecessary information related to the original Salem, Unit No. 2 Westinghouse steam generators.

A copy of our safety evaluation of the amendment is enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Carleen J. Parker, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 309 to DPR-70
2. Amendment No. 291 to DPR-75
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PSEG NUCLEAR, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 309
Renewed License No. DPR-70

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by PSEG Nuclear LLC, acting on behalf of itself and Exelon Generation Company, LLC (the licensees) dated July 28, 2014, as supplemented by letter dated January 15, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-70 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 309, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications, and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeffrey A. Whited for

Douglas A. Broaddus, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License
and Technical Specifications

Date of Issuance: July 30, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 309

RENEWED FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following page of Renewed Facility Operating License No. DPR-70 with the attached revised page as indicated. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
Page 3	Page 3

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
Page 3/4 4-7	Page 3/4 4-7
Page 6-19b	Page 6-19b
Page 6-19c	Page 6-19c
Page 6-19d	Page 6-19d
Page 6-19e	Page 6-19e
Page 6-24a	Page 6-24a

instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at a steady state reactor core power level not in excess of 3459 megawatts (one hundred percent of rated core power).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 309, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications, and the Environmental Protection Plan.

(3) Deleted Per Amendment 22, 11-20-79

(4) Less than Four Loop Operation

PSEG Nuclear LLC shall not operate the reactor at power levels above P-7 (as defined in Table 3.3-1 of Specification 3.3.1.1 of Appendix A to this renewed license) with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensees and approval for less than four loop operation at power levels above P-7 has been granted by the Commission by Amendment of this renewed license.

(5) PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety

STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained and all SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a.* With one or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program:

1. Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection within 7 days; and

2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or SG tube inspection.

b. With SG tube integrity not maintained or the required Action of a. above not met, be in at least HOT STANDBY within 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a SG tube inspection.

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* Separate Action is allowed for each SG tube.

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

6.8.4.h Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of the census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.8.4.i Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each 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), 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 plugging 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.

The following alternate plugging criteria shall be applied as an alternative to the 40% depth based criteria:

1. Tubes with service-induced flaws located greater than 15.21 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 15.21 inches below the top of the tubesheet shall be plugged upon detection.

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 plugging criteria.

The portion of the tube below 15.21 inches from the top of the tubesheet is excluded from this requirement.

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. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

    a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;

    b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and

    c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

3. If crack indications are found in portions of the SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary leakage.

6.8.4.j Inservice Testing Program

This Program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies and to other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities,

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities, and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.

6.8.4.k Reactor Coolant Pump Flywheel Inspection Program

In addition to the requirements of the ISI Program, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

2. WCAP-8385, Power Distribution Control and Load Following Procedures - Topical Report, (W Proprietary) Methodology for Specification 3/4.2.1 Axial Flux Difference.

3. WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code (W Proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor.

4. WCAP-10266-P-A, The 1981 Version of Westinghouse Evaluation Model Using BASH Code, (W Proprietary) Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor.

5. WCAP-12472-P-A, BEACON – Core Monitoring and Operations Support System, (W Proprietary).

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

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. 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 degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,

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

UNITED STATES
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PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 291
Renewed License No. DPR-75

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by PSEG Nuclear LLC, acting on behalf of itself and Exelon Generation Company, LLC (the licensees) dated July 28, 2014, as supplemented by letters dated January 15, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

   (2) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 291, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Tiffany A Whited for

Douglas A. Broaddus, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License
and Technical Specifications

Date of Issuance: July 30, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 291

RENEWED FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following page of Renewed Facility Operating License No. DPR-75 with the attached revised page as indicated. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
Page 3	Page 3

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
Page 3/4 4-9	Page 3/4 4-9
Page 6-19b	Page 6-19b
Page 6-19c	Page 6-19c
Page 6-19d	Page 6-19d
Page 6-19e	Page 6-19e
Page 6-19f	Page 6-19f
Page 6-24a	Page 6-24a
Page 6-24b	Page 6-24b

(4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source or special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at steady state reactor core power levels not in excess of 3459 megawatts (thermal).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 291, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3/4.4.6 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.6 SG tube integrity shall be maintained and all SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a.* With one or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program:

1. Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection within 7 days; and

2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or SG tube inspection.

b. With SG tube integrity not maintained or the required Action of a. above not met, be in at least HOT STANDBY within 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.6.2 Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a SG tube inspection

* Separate Action is allowed for each SG tube.

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

6.8.4.h Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of the census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.8.4.i Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each 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), 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 3.4.7.2, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube plugging 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.

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 plugging criteria. 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. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary leakage.

6.8.4.j Inservice Testing Program

This Program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies and to other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities,

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities, and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.

6.8.4.k Reactor Coolant Pump Flywheel Inspection Program

In addition to the requirements of the ISI Program, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

2. WCAP-8385, Power Distribution Control and Load Following Procedures - Topical Report, (W Proprietary) Methodology for Specification 3/4.2.1 Axial Flux Difference

3. WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code, (W Proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor.

4. WCAP-10266-P-A, The 1981 Version of Westinghouse Evaluation Model Using BASH Code, (W Proprietary) Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor.

5. WCAP-12472-P-A, BEACON – Core Monitoring and Operations Support System, (W Proprietary).

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

7. WCAP-10054-P-A, Addendum 2, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model."

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. 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 degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 DELETED

6.9.4 When a report is required by ACTION 8 OR 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 309 AND 291 TO RENEWED FACILITY OPERATING

LICENSE NOS. DPR-70 AND DPR-75

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated July 28, 2014, as supplemented by letter dated January 15, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML14210A484 and ML15015A060, respectively), PSEG Nuclear LLC (PSEG, the licensee) submitted a license amendment request (LAR) to revise the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2 Technical Specifications (TSs). The proposed amendments would incorporate the guidance of Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," and to remove unnecessary information related to the original Salem, Unit No. 2 steam generators (SGs), which were replaced in 2008. The guidance of TSTF-510 revises STS TS 3/4.4.6, "Steam Generator (SG) Tube Integrity"; TS 6.8.4.i, "Steam Generator (SG) Program"; and TS 6.9.1.10, "Steam Generator Tube Inspection Report." Salem, Unit Nos. 1 and 2, utilize a different number system than the STS.

The licensee stated that this LAR is consistent with TSTF-510, Revision 2, which was noticed as being available for use as part of the consolidated line item improvement process in the *Federal Register* on October 27, 2011 (76 FR 66763). Because the LAR includes TS revisions for both implementation of TSTF-510 and removal of unnecessary information related to the original Salem, Unit No. 2 SGs, this LAR is not being processed under the consolidated line item improvement process.

The supplement dated January 15, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 28, 2014 (79 FR 64227).

2.0 BACKGROUND

The current SG TS requirements for Salem were based on TSTF-449, Revision 4, "Steam Generator Tube Integrity" (NRC *Federal Register* Notice of Availability (70 FR 24126)). The TSTF-449 changes to the STS incorporated a new, largely performance-based, approach for ensuring the integrity of the SG tubes is maintained. The performance-based requirements were supplemented by prescriptive requirements relating to tube inspections and tube repair limits, to ensure that conditions adverse to quality are detected and corrected on a timely basis. As of September 2007, the TSTF-449, Revision 4, changes were adopted in the plant TSs for all pressurized-water reactors (PWRs).

The changes in TSTF-510, Revision 2, reflect licensees' early implementation experience with their current TSs. The changes in TSTF-510, Revision 2, are editorial corrections, changes, and clarifications intended to improve internal consistency, consistency with implementing industry documents, and usability, without changing the intent of the requirements. The proposed changes are an improvement to the existing SG inspection requirements and continue to provide assurance that the plant licensing basis will be maintained between SG inspections. The NRC staff approved TSTF-510, Revision 2, for use with the consolidated line item process on October 19, 2011 (ADAMS Accession No. ML112101604). The staff's model safety evaluation (SE) is available under ADAMS Accession No. ML112101513. Other than the variations or deviations discussed in Section 4.0 of this SE, the licensee is not proposing any variations or deviations from the TS changes described in the TSTF-510, Revision 2.

3.0 REGULATORY EVALUATION

The SG tubes of a PWR have a number of important safety functions. They are an integral part of the reactor coolant pressure boundary (RCPB) and as such, are relied upon to maintain primary system pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon to transfer heat from the primary to the secondary system and are relied upon to isolate the radioactive fission products in the primary coolant from the secondary system. In addition, SG tube integrity is relied upon to prevent uncontrolled fission product release under conditions resulting from core damage during severe accidents.

As part of the application for a construction permit, 10 CFR 50.34(a)(3)(i) requires the applicant to provide, in a preliminary safety analysis report (PSAR), the principal design criteria for the facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The General Design Criteria (GDC) in 10 CFR Part 50 Appendix A establish minimum requirements for the principal design criteria for water-cooled nuclear power plants. When a construction permit holder applies for an operating license, they are required by 10 CFR 50.34(b) to submit a Final Safety Analysis Report (FSAR) that includes information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole.

Salem, Unit Nos. 1 and 2, were designed and licensed to the Atomic Industrial Forum's GDC, which predates the July 7, 1971, GDC contained in Appendix A to 10 CFR Part 50. A comparison of the Salem, Units 1 and 2, plant design was done with 10 CFR Part 50,

Appendix A. The comparison was documented in the Salem Updated Final Safety Analysis Report (UFSAR), Section 3.1.3, which states, in part, that "the Salem Plant design conforms with the intent of the "General Design Criteria for Nuclear Power Plants," dated July 7, 1971," which were published in Appendix A to 10 CFR Part 50. There are no exceptions noted to GDC 14, "Reactor coolant pressure boundary," GDC 15, "Reactor coolant system design," GDC 30, "Quality of reactor coolant pressure boundary," GDC 31, "Fracture prevention of reactor coolant pressure boundary," and GDC 32, "Inspection of reactor coolant pressure boundary."

Thus, the principal design of fission product barriers considers:

> *Criterion 14—Reactor coolant pressure boundary.* The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
>
> *Criterion 15—Reactor coolant system design.* The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Also, the principal design of fluid systems for the Salem reactors considers:

> *Criterion 30—Quality of reactor coolant pressure boundary.* Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.
>
> *Criterion 31—Fracture prevention of reactor coolant pressure boundary.* The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.
>
> *Criterion 32—Inspection of reactor coolant pressure boundary.* Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Paragraph 10 CFR 50.55a(c) specifies that components, which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). Section 50.55a(g) further requires that components and supports, which are classified as ASME Code Class 1, must be designed and provided with access to enable the performance of inservice examination of these components to the extent practical, and must meet the pre-service examination requirements set forth in the editions and addenda of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code incorporated by reference in 10 CFR 50.55a(b) that were applied to the construction of the particular component. Section XI requirements pertaining to inservice inspection of SG tubing are augmented by additional SG tube surveillance requirements (SRs) in the TSs.

As part of the plant licensing process, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents such as an SG tube rupture and main steamline break. On February 17, 2006, the Commission issued Amendment Nos. 271 and 252 to Salem, Units 1 and 2, respectively, to incorporate a full-scope application of an alternate source term methodology in accordance with 10 CFR 50.67. Accordingly, the evaluation of the consequences of applicable design basis accidents at Salem must demonstrate with reasonable assurance that the total effective does equivalent (TEDE) meets the criteria of 10 CFR 50.67(2)(i) through (iii).

The regulation at 10 CFR 50.36, "Technical specifications," establishes the requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls. Salem, Unit No. 1 TS 3.4.5 "Steam Generator Tube Integrity," and Salem, Unit No. 2 TS 3.4.6, "Steam Generator Tube Integrity," provides that SG tube integrity shall be maintained and all SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program. Salem, Unit No. 1 TS 3.4.6.2 "Reactor Coolant System Operational Leakage," and Salem, Unit No. 2 TS 3.4.7.2 "Reactor Coolant System Operational Leakage," limits Reactor Coolant System leakage.

Paragraph 10 CFR 50.36(c)(5) defines administrative controls as, "The provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." The TS include programs established by the licensee to operate the facility in a safe manner, including the SG Program. The SG Program is described in TS 6.8.4.i. TS 6.9.1.10 requires a SG Tube Inspection Report be submitted to the NRC for Salem, Unit Nos. 1 and 2.

TS 6.8.4.i, "Steam Generator (SG) Program," for Salem, Unit Nos. 1 and 2, requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. TS 6.8.4.i.b. requires the SG Program to include provisions addressing performance criteria for SG tube integrity. Specifically, TS 6.8.4.i.b. states that "SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage." TS 6.8.4.i.a requires a condition monitoring assessment be performed during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met. TS 6.8.4.i.d includes provisions regarding the scope, frequency, and methods of SG tube inspections.

4.0 TECHNICAL EVALUATION

Each proposed change to the TS is described individually below, followed by the NRC staff's assessment of the change.

4.1 TS Numbering

The TSs for Salem, Unit Nos. 1 and 2, use different numbers and titles than TSTF-510, Revision 2.

For Salem, Unit No. 1, the "Steam Generator (SG) Program" TS is numbered 6.8.4.i rather than 5.5.9; the "Steam Generator Tube Integrity" TS is numbered 3.4.5 rather than 3.4.20; and the "Steam Generator Tube Inspection Report" is numbered 6.9.1.10 rather than 5.5.9.

For Salem, Unit No. 2, the "Steam Generator (SG) Program" TS is numbered 6.8.4.i rather than 5.5.9; the "Steam Generator Tube Integrity" TS is numbered 3.4.6 rather than 3.4.20; and the "Steam Generator Tube Inspection Report" is numbered 6.9.1.10 rather than 5.5.9.

These numbering differences do not affect the content of TSTF-510, Revision 2, for the TSs to Salem, Unit Nos. 1 and 2. As a result, the NRC staff finds the differences between what was approved for TSTF-510, Revision 2, and what is being proposed acceptable.

4.2 TS 6.8.4.i, "Steam Generator (SG) Program"

The last sentence of the introductory paragraph of TS 6.8.4.i for Salem, Unit Nos. 1 and 2, currently states, "In addition, the Steam Generator Program shall include the following provisions:."

The sentence is revised to say "In addition, the Steam Generator Program shall include the following:." The reason for this change is that subsequent paragraphs start with "Provisions for" or "Performance criteria" and the word "provisions" in the introductory paragraph is duplicative or unnecessary.

The NRC staff has reviewed the licensee's proposed change to TS 6.8.4.i and has determined that the word "provisions" in the introductory paragraph is duplicative and unnecessary. The NRC staff finds that the change is editorial in nature, and therefore, acceptable.

4.3 TS 6.8.4.i, Paragraph 6.8.4.i.b.1, "Structural integrity performance criterion"

The first sentence of TS 6.8.4.i.b.1 for Salem, Unit Nos. 1 and 2, currently states:

> 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.

The licensee proposed to revise the sentence as follows:

> 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), all anticipated transients included in the design specification, and design basis accidents.

The basis for the change is that this sentence inappropriately includes anticipated transients in the description of normal operating conditions.

The NRC staff has determined that the current wording is incorrect and that anticipated transients should be differentiated from normal operating conditions. Therefore, the NRC staff finds the change acceptable.

4.4 Paragraph 6.8.4.i.c, "Provisions for SG tube repair criteria"; Paragraph 6.8.4.i.d, "Provisions for SG tube inspections"; Salem, Unit No. 1 TS 3.4.5, "Steam Generator (SG) Tube Integrity"; and associated SRs, and Salem, Unit No. 2 TS 3.4.6, "Steam Generator (SG) Tube Integrity," and associated SRs

The license proposed to change all references to "tube repair criteria" to "tube plugging criteria." This change is intended to be consistent with the treatment of SG tube repair throughout TS 6.8.4.i.

The NRC staff determined that the proposed change adds clarity to the specification, because generally, one of two actions must be taken when the criteria are exceeded. One action is to remove the tube from service by plugging the tube at both tube ends. The alternative action is to repair the tube, but only if such a repair is permitted by paragraph 6.8.4.i.c. Salem does not have any approved repair methods, and thus plugging is the only available option if the criteria are exceeded. Therefore, the NRC staff finds the changes acceptable.

4.5 TS 6.8.4.i, Paragraph 6.8.4.i.d, "Provisions for SG tube inspections"

The licensee proposed to change the term "an assessment of degradation" to "a degradation assessment" to be consistent with the terminology used in industry program documents.

The NRC staff agrees that the terminology should be consistent. Furthermore, the proposed wording does not involve a technical change to the specification. Therefore, the NRC staff finds the change acceptable.

4.6 TS 6.8.4.i, Paragraph 6.8.4.i.d.1

The paragraph currently states: "Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement."

The proposed change would replace "SG replacement" with "SG installation."

The change in wording allows for consistency between Salem and other plants. Since this wording modification continues to require 100% inspection of the tubes in each SG during the first refueling outage following SG replacement, the NRC staff finds the change acceptable.

4.7 Salem, Unit No. 1 TS 6.8.4.i, Paragraph 6.8.4.i.d.2 (SGs with alloy 600 thermally treated tubes)

TSTF-510 is written to accommodate plants with several variations of SG tubing material. As described in the Salem, Unit No. 1, UFSAR, Section 5.2.3.2, "Compatibility with Reactor Coolant," Salem, Unit No. 1 SGs employ a thermally treated (TT) alloy 600 tubing design.

Paragraph 6.8.4.i.d.2 currently states:

> Inspect 100% of the tubes at sequential periods of 120, 90, and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

The licensee proposed to replace paragraph 6.8.4.i.d.2 with the following insert:

> After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.
>
> a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;
>
> b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and
>
> c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

The licensee proposed, in part, to move the first two sentences of paragraph 6.8.4.i.d.2 to the inspection periods, as specified in subsections a, b, and c of the revised paragraph, and make editorial changes to improve clarity. The NRC has determined that these changes are clarifying in nature and do not change the current intent of these two sentences. However, the LAR also includes two changes to when inspections are performed as follows:

- The second inspection period would be revised from 90 to 96 effective full power months (EFPM).

- The third and subsequent inspection periods would be revised from 60 to 72 EFPM.

The proposed changes are characterized as marginal increases for consistency with typical fuel cycle lengths that better accommodate the scheduling of inspections. The NRC staff notes that plants with TT alloy 600 SG tubes typically inspect at 18- or 36-month intervals (one or two fuel cycles, respectively) depending on whether stress corrosion crack activity was observed during the most recent inspection. With these intervals, the last scheduled inspection during the first inspection period would occur at 108 months after the first refueling outage following SG installation. This is 12 months before the end of the first 120 EFPM inspection period. However, with the proposed changes to the length of the second and subsequent inspection periods, the NRC staff finds that the last scheduled inspections in the second and subsequent inspection periods will coincide exactly with the end of these periods.

The proposed changes would generally increase the number of inspections in each of the second and subsequent inspection periods by up to one additional inspection. This could reduce the required average minimum sample size during these periods. However, inspection sample sizes will continue to be subject to paragraph 6.8.4.i.d, which states, in part, that in addition to meeting the requirements of paragraphs 6.8.4.i.d.2, "the inspection scope, inspection methods, and inspection intervals shall be such as to ensure SG tube integrity is maintained until the next SG inspection." Therefore, the NRC staff has determined that, with the proposed changes to the length of the second and subsequent inspection periods, compliance with the SG program requirements in TS 6.8.4.i will continue to ensure both adequate inspection scopes and tube integrity.

For each inspection period, paragraph 6.8.4.i.d.2 currently requires that at least 50 percent of the tubes be inspected by the refueling outage nearest to the midpoint of the inspection period and the remaining 50 percent by the refueling outage nearest the end of the inspection period. The NRC staff notes that if there are not an equal number of inspections in the first half and second half of the inspection period, the average minimum sampling requirement may be markedly different for inspections in the first half of the inspection period, compared to those in the second half, even when there are uniform intervals between each inspection.

For example, in a hypothetical plant in the first (120 EFPM) inspection period, a scheduled 36-month interval (two 18-month fuel cycles) between each inspection would currently be required to inspect 50 percent of the tubes by the refueling outage nearest the midpoint of the inspection, which would be the third refueling outage in the period, 6 months before the midpoint. However, since no inspection is scheduled for that outage (because inspections take place every other outage – once every 36 months), then the full 50 percent sample must be performed during the inspection scheduled for the second refueling outage in the period. Two inspections would be scheduled to occur in the second half of the inspection period, at 72 and 108 months into the inspection period. Thus, the current sampling requirement could be satisfied by performing a 25 percent sample during each of these inspections or other combinations of sampling (e.g., 10 percent during one and 40 percent in the other) totaling 50 percent.

The licensee proposed to revise this requirement such that the minimum sample size for a given inspection in a given inspection period is 100 percent divided by the number of scheduled inspections during that inspection period. For the above example, the proposed change would result in a uniform initial minimum sample size of 33.3 percent for each of the three scheduled inspections during the inspection period. The NRC staff has determined that this proposed revision is an improvement to the existing requirement since it provides a more consistent minimum initial sampling requirement.

The proposed third and fourth sentences in paragraph 6.8.4.i.d.2 state:

> If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period.

This addresses the possibility that a degradation assessment in accordance with paragraph 6.8.4.i.d will indicate that the tubing may be susceptible to a type of degradation at a location not previously inspected with a technique capable of detecting that type of degradation at that location. For example, new information from another similar plant becomes available indicating the potential for circumferential cracking at a specific location on the tube. Previous degradation assessments had not identified the potential for this type of degradation at this location. Thus, previous inspections of this location had not been performed with a technique capable of detecting circumferential cracks. However, now that the potential for circumferential cracking has been identified at this location, paragraph 6.8.4.i.d requires an inspection be performed with the objective of detecting circumferential cracks that may be present at this location and which may satisfy the applicable tube repair criteria.

Furthermore, if this inspection is performed for the first time during the third or fourth SG inspections scheduled for the subject 120 EFPM inspection period, the current paragraph 6.8.4.i.d.2 does not specifically specify whether this location needs to be 100 percent inspected by the end of the 120 EFPM inspection period, or whether a prorated approach may be taken. The NRC staff addressed this question in Issue 1 of NRC Regulatory Issue Summary (RIS) 2009-04, "Steam Generator Tube Inspection Requirements," dated April 3, 2009 (ADAMS Accession No. ML083470557), as follows:

> **Issue 1**: *A licensee may identify a new potential degradation mechanism after the first inspection in a sequential period. If this occurs, what are the*

expectations concerning the scope of examinations for this new potential degradation mechanism for the remainder of the period (e.g., do 100 percent of the tubes have to be inspected by the end of the period or can the sample be prorated for the remaining part of the period)?

[NRC Staff Position:] The TS contain requirements that are a mixture of prescriptive and performance-based elements. Paragraph "d" of these requirements indicates that the inspection scope, inspection methods, and inspection intervals shall be sufficient to ensure that SG tube integrity is maintained until the next SG inspection. Paragraph "d" is a performance-based element because it describes the goal of the inspections but does not specify how to achieve the goal. However, paragraph "d.2" is a prescriptive element because it specifies that the licensee must inspect 100 percent of the tubes at specified periods.

If an assessment of degradation performed after the first inspection in a sequential period results in a licensee concluding that a new degradation mechanism (not anticipated during the prior inspections in that period) may potentially occur, the scope of inspections in the remaining portion of the period should be sufficient to ensure SG tube integrity for the period between inspections.

In addition, to satisfy the prescriptive requirements of paragraph "d.2" that the licensee must inspect 100 percent of the tubes within a specified period, a prorated sample for the remaining portion of the period is appropriate for this potentially new degradation mechanism. This prorated sample should be such that if the licensee had implemented it at the beginning of the period, the TS requirement for the 100 percent inspection in the entire period (for this degradation mechanism) would have been met. A prorated sample is appropriate because (1) the licensee would have performed the prior inspections in this sequential period consistently with the requirements, and (2) the scope of inspections must be sufficient to ensure that the licensee maintains SG tube integrity for the period between inspections.

The NRC staff finds that relocation of information in sentences 3 and 4, as described above, clarifies the existing requirement such that it is consistent with the NRC staff's position from RIS 2009-04 quoted above, and therefore, acceptable.

The proposed fifth sentence in paragraph 6.8.4.i.d.2 states, "Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage." Allowing extension of the inspection periods by up to an additional 3 EFPM potentially impacts the average tube inspection sample size to be implemented during a given inspection in the period. For example, if three SG inspections are scheduled to occur within the nominal 120 EFPM period, the minimum sample size for each of the four inspections could average as little as 33.3 percent of the tube population. If a fourth inspection can be included within the period by extending the period by 3 EFPM, then the minimum sample size for each of the four inspections could average as little as 25 percent of the tube population. Since the subsequent period begins at the end of the included SG inspection outage, the proposed change does not impact the required frequency of SG inspections.

Required tube inspection sample sizes are also subject to the performance-based requirement in paragraph 6.8.4.i.d, which states, in part, that in addition to meeting the requirements of paragraph 6.8.4.i.d.2, "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." This requirement remains unchanged under the proposal. The NRC staff has determined that the proposed fifth sentence, by allowing the potential for smaller sample sizes, involves only a relatively minor relaxation to the existing sampling requirements in paragraph 6.8.4.i.d.2. However, the performance-based requirements in 6.8.4.i.d.2 ensure that adequate inspection sampling will be performed to ensure tube integrity is maintained. Thus, the NRC staff finds the proposed change acceptable.

Finally, the first sentence of the proposed revision to paragraph 6.8.4.i.d.2, "After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections)," replaces the last sentence of the current paragraph 6.8.4.i.d.2, "No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected." Both versions establish the minimum allowable SG inspection frequency as at least every 48 EFPM, or at least every other refueling outage (whichever results in more frequent inspections). This minimum inspection frequency is unchanged from the current requirement in Salem, Unit No. 1, TSs. The NRC staff has determined that the wording changes in the sentence are of an editorial and clarifying nature and are not material such that the current intent of the requirement is unchanged. Thus, the NRC staff finds the proposed change acceptable.

4.8 Salem, Unit No. 2 TS 6.8.4.i, Paragraph 6.8.4.i.d.2 (SGs with Alloy 690 Thermally Treated tubes)

As described in the Salem, Unit No. 2, UFSAR, Section 5.1, "Summary Description," Salem, the Unit No 2, replacement SGs employ a TT alloy 690 tubing design.

Paragraph 6.8.4.i.d.2b currently states:

> Inspect 100% of the tubes at sequential periods of 144, 108, 72, and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

The licensee proposed the following insert for revised paragraph 6.8.4.i.d.2:

> After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in

each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

The licensee proposed to delete 6.8.4.i.d.2a and renumber 6.8.4.i.d.2b to 6.8.4.i.d.2. These changes are discussed in Section 4.11 of this SE. In addition, the licensee proposed to delete the first part of the first sentence of paragraph 6.8.4.i.d.2.b, which states, "SGs with Alloy 690 Thermally Treated tubes:." The removal of this wording is administrative. Since the original SGs were replaced and the licensee has proposed to remove all language specific to the original SGs, distinguishing which subparagraph is for which type of SG is no longer needed. As a result, the NRC staff finds the proposed changes acceptable.

The licensee proposed, in part, to move the first two sentences of paragraph 6.8.4.i.d.2b to the inspection periods, as specified in subsections a, b, c, and d of the revised paragraph 6.8.4.i.d.2, and make editorial changes to improve clarity. The NRC staff has determined that these changes are clarifying in nature and do not change the current intent of these two sentences. However, the LAR also includes three changes to when inspections are performed as follows:

- The second inspection period would be revised from 108 to 120 EFPM.
- The third inspection period would be revised from 72 to 96 EFPM.

- The subsequent inspection periods would be revised from 60 to 72 EFPM.

These changes are characterized as marginal increases for consistency with typical fuel cycle lengths that better accommodate the scheduling of inspections. The NRC staff observes that depending on the actual plant inspection schedule, these changes could impact the number of inspections in a given period, as well as the sample size. However, inspection sample sizes will continue to be subject to paragraph 6.8.4.i.d, which states, in part, that in addition to meeting the requirements of paragraph 6.8.4.i.d.2, "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." Therefore, the NRC staff has determined that, with the proposed changes to the length of the second and subsequent inspection periods, compliance with the SG program requirements in TS 6.8.4.i will continue to ensure both adequate inspection scopes and tube integrity.

For each inspection period, paragraph 6.8.4.i.d.2b currently requires that at least 50 percent of the tubes be inspected by the refueling outage nearest to the midpoint of the inspection period and the remaining 50 percent by the refueling outage nearest the end of the inspection period. The NRC staff notes that if there are not an equal number of inspections in the first half and second half of the inspection period, the average minimum sampling requirement may be markedly different for inspections in the first half of the inspection period, compared to those in the second half, even when there are uniform intervals between each inspection.

The licensee proposed to revise this requirement such that the minimum sample size for a given inspection in a given inspection period is 100 percent, divided by the number of scheduled inspections during that inspection period. The NRC staff has determined that this proposed revision is an improvement to the existing requirement since it provides a more consistent minimum initial sampling requirement.

The proposed third and fourth sentences in paragraph 6.8.4.i.d.2 state:

> If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period.

This addresses the possibility that a degradation assessment in accordance with paragraph 6.8.4.i.d will indicate that the tubing may be susceptible to a type of degradation at a location not previously inspected with a technique capable of detecting that type of degradation at that location. For example, new information from another similar plant becomes available indicating the potential for circumferential cracking at a specific location on the tube. Previous degradation assessments had not identified the potential for this type of degradation at this location. Thus, previous inspections of this location had not been performed with a technique capable of detecting circumferential cracks. However, now that the potential for circumferential cracking has been identified at this location, paragraph 6.8.4.i.d requires an inspection be performed with the objective of detecting circumferential cracks that may be present at this location and which may satisfy the applicable tube repair criteria.

Furthermore, if this inspection is performed for the first time during the third of fourth SG inspections scheduled for the subject 144 EFPM inspection period, the current paragraph 6.8.4.i.d.2 does not specifically specify whether this location needs to be 100 percent inspected by the end of the 144 EFPM inspection period, or whether a prorated approach may be taken. The NRC staff addressed this question in Issue 1 of NRC RIS 2009-04. The NRC staff finds that relocation of information in sentences 3 and 4, as described above, clarifies the existing requirement, such that it is consistent with the NRC staff's position from RIS 2009-04 quoted in Section 4.7 of this SE, and is, therefore, acceptable.

The proposed fifth sentence in paragraph 6.8.4.i.d.2 states, "Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage." Allowing extension of the inspection periods by up to an additional 3 EFPM potentially impacts the average tube inspection sample size to be implemented during a given inspection in the period. For example, if four SG inspections are scheduled to occur within the nominal 144 EFPM period, the minimum sample size for each of the four inspections could average as little as 25 percent of the tube population. If a fifth inspection can be included within the period by extending the period by 3 EFPM, then the minimum sample size for each of the five inspections could average as little as 20 percent of the tube population. Since the subsequent period begins at the end of the included SG inspection outage, the proposed change does not impact the required frequency of SG inspections.

Required tube inspection sample sizes are also subject to the performance-based requirement in paragraph 6.8.4.i.d, which states, in part, that in addition to meeting the requirements of paragraph 6.8.4.i.d.2, "the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next scheduled SG inspection." This requirement remains unchanged under the proposal. The NRC staff has determined that by allowing the potential for smaller sample sizes, the proposed fifth sentence, involves only a relatively minor relaxation to the existing sampling requirements in paragraph 6.8.4.i.d.2. However, the performance-based requirements in 6.8.4.i.d.2 ensure that adequate inspection sampling will be performed to ensure tube integrity is maintained. Thus, the NRC staff finds the proposed change acceptable.

Finally, the first sentence of the proposed revision to paragraph 6.8.4.i.d.2, "After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections)," replaces the last sentence of the current paragraph 6.8.4.i.d.2b, "No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected." Both versions establish the minimum allowable SG inspection frequency as at least every 72 EFPM or at least every other refueling outage (whichever results in more frequent inspections). This minimum inspection frequency is unchanged from the current requirement in Salem, Unit No. 2, TSs. The NRC staff has determined that the wording changes in the sentence are of an editorial and clarifying nature and are not material, such that the current intent of the requirement is unchanged. Thus, the NRC staff finds the proposed change acceptable.

4.9 <u>TS 6.8.4.i, Paragraph 6.8.4.i.d.3</u>

The first sentence of paragraph 6.8.4.i.d.3 of Salem, Unit No. 1, currently states:

> If crack indications are found in portions of the SG tube not excluded above, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less).

The licensee proposes to revise the sentence as follows:

> If crack indications are found in portions of the SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections).

The first sentence of paragraph 6.8.4.i.d.3 of Salem, Unit No. 2, currently states:

> If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less).

The licensee proposes to revise the sentence as follows:

> If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections).

The proposed change is replacing the words "for each SG" with the words "for each affected and potentially affected SG" and replacing the words "whichever is less" with the words "whichever results in more frequent inspections" because the existing wording can be misinterpreted.

Paragraph 6.8.4.i.d.2 permits SG inspection intervals to extend over multiple fuel cycles for SGs with TT alloy 600 and 690 tubing, assuming that such intervals can be implemented while ensuring tube integrity is maintained in accordance with paragraph 6.8.4.i.d. However, stress corrosion cracks may not become detectable by inspection until the crack depth approaches the tube repair limit. In addition, stress corrosion cracks may exhibit high growth rates. For these reasons, once cracks have been found in any SG tube, paragraph 6.8.4.i.d.3 restricts the allowable interval to the next scheduled inspection to 24 EFPM or one refueling outage (whichever is less). The intent of this requirement is that it applies to the affected SG and to any other SG, which may be potentially affected by the degradation mechanism that caused the known crack(s).

For example, a root cause analysis in response to the initial finding of one or more cracks might reveal that the crack(s) are associated with a manufacturing anomaly, which causes locally high residual stress, which in turn caused the early initiation of cracks at the affected locations. If it can be established that the extent of condition of the manufacturing anomaly applies only to one SG and not the others, then the NRC staff agrees that only the affected SG needs to be inspected within 24 EFPM or one refueling cycle in accordance with paragraph 6.8.4.i.d.2. Conversely, if it cannot be established that the manufacturing anomaly applies to just one SG, then all potentially affected SGs would have to be inspected. The next scheduled inspections of the other SGs will continue to be subject to all other provisions of paragraph 6.8.4.i.d. The NRC staff finds the proposed change to paragraph 6.8.4.i.d.3 acceptable, because it clarifies the intent of the paragraph.

4.10 <u>Specification 6.9.1.10 "Steam Generator Inspection Report"</u>

Salem, Unit No. 1, TS lists items a. through j. and Salem, Unit No. 2, TS lists a. through h. to be included in a report, which shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.i, "Steam Generator (SG) Program."

Both Unit Nos. 1 and 2, item b. currently reads, "Active degradation mechanisms found."

The proposed revision reads, "Degradation mechanisms found."

Both Units Nos. 1 and 2, item e. currently reads, "Number of tubes plugged during the inspection outage for each active degradation mechanism."

The proposed revision reads, "Number of tubes plugged during the inspection outage for each degradation mechanism."

Unit No. 1, item f. currently reads, "Total number and percentage of tubes plugged to date, and."

The proposed revision reads, "The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator, and."

Unit No. 2, item f. currently reads, "Total number and percentage of tubes plugged to date."

The proposed revision reads, "The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator."

The licensee proposed to delete the word "Active" in items b. and e above. Thus, all degradation mechanisms found, whether deemed to be active or not, would now be reportable. The NRC staff finds the proposed change acceptable because the change requires the licensee to provided additional information to the NRC.

The licensee proposed to add "the effective plugging percentage in each steam generator," to item f. above. This changes provides additional information regarding the SG tube inspection to the NRC in the SG inspection report. The NRC staff finds this change acceptable because it provides additional information to the NRC and does not change or remove any information currently required by the report.

4.11 Salem, Unit No. 2, Revision to Reflect New SGs

The licensee also proposed removing wording from Salem, Unit No. 2, TSs that was applicable to the original SGs, which were replaced in 2008, and is not applicable to the replacement SGs. The wording to be removed is contained in TS 6.8.4.i, "Steam Generator (SG) Program;" and TS 6.9.1.10, "Steam Generator Tube Inspection Report." The wording is associated with implementation of the W* (W-star) repair criteria and inspections for the original SGs.

The licensee's proposed changes to Salem, Unit No 2, TS 6.8.4.i associated with W* and inspections for the original SGs, revise the following:

1. TS 6.8.4.i.c (page 6-19d) removes, in its entirety, the section beginning with "The following alternate tube repair criteria..."

2. TS 6.8.4.i.d, "Provisions for SG tube inspections," (page 6-19e) deletes the third sentence, "The portion of the tube within the hot-leg tubesheet region below the W* distance is excluded."

3. TS 6.8.4.i.d, "Provisions for SG tube inspections," (page 6-19e) deletes, "Note: Step 2 has two separate requirements (a and b), depending on the type of SG tubes installed."

4. TS 6.8.4.i.d.2a (page 6-19e) deletes item 2a.

5. TS 6.8.4.i.d.2b (page 6-19e) revises the numbering format from 2b to 2.

6. TS 6.8.4.i.d.4 (page 6-19f) deletes item 4.

The licensee's proposed change to Salem, Unit No 2, TS 6.9.1.10 associated with W* and inspections for the original SGs, revise the following:

TS 6.9.1.10.h (page 6-24b) deletes item h.

The wording is not applicable to the replacement SGs. As a result, the NRC staff finds the proposed changes acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (79 FR 64227). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Andrew B. Johnson
Carleen J. Parker

Date: July 30, 2015

July 30, 2015

Mr. Robert Braun
President and Chief Nuclear Officer
PSEG Nuclear LLC
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING ADOPTING TECHNICAL SPECIFICATION TASK FORCE TRAVELER (TSTF)-510, REVISION 2, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION FREQUENCIES AND TUBE SAMPLE SELECTION" (TAC NOS. MF4574 AND MF4575)

Dear Mr. Braun:

The Commission has issued the enclosed Amendment Nos. 309 and 291 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. These amendments consist of changes to the technical specifications (TSs) in response to PSEG Nuclear LLC's application dated July 28, 2014, as supplemented by letter dated January 15, 2015.

The amendments revise TS requirements regarding steam generator tube inspections and reporting as described in Technical Specifications Task Force (TSTF) Standard Technical Specifications Change Traveler (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection." In addition, the amendments revise Salem, Unit No. 2 TSs to remove unnecessary information related to the original Salem, Unit No. 2 Westinghouse steam generators.

A copy of our safety evaluation of the amendment is enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/
Carleen J. Parker, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 309 to DPR-70
2. Amendment No. 291 to DPR-75
3. Safety Evaluation

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Amendment Accession No.: ML15153A230 *by e-mail

OFFICE	LPL1-2/PM	LPL1-2/LA	LPL1-2/LA	ESGB/BC*	STSB/BC	OGC	LPL1-2/BC	LPL1-2/PM
NAME	CParker	LRonewicz	ABaxter	GKulesa MColon for	RElliott	DRoth w/comments	DBroaddus (JWhited for)	CParker
DATE	6/02/15	6/8/15	0/1515	6/24/15	6/6/15	7/15/15	7/30/15	7/30/15

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