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PSEG
Nuclear LLC

January 19, 2007

10 CFR 50.90

LR-N07-0010

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

SALEM GENERATING STATION – UNIT 2
FACILITY OPERATING LICENSE NO. DPR-75
NRC DOCKET NO. 50-311

Subject: **RESPONSE TO RAIs ON LCR S06-01 REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS TO ADD STEAM GENERATOR TUBE INTEGRITY PROGRAM**

References: (1) Letter from PSEG to NRC: LCR S06-01, "Request for Change to Technical Specifications to Add Requirements for Steam Generator Tube Integrity, Steam Generator Program, and Steam Generator Tube Inspection Report and to Revise Reactor Coolant System Operational Leakage Requirements, Salem Nuclear Generating Station - Unit 2, Docket No. 50-311, Facility Operating License DPR-75," dated April 6, 2006

(2) Letter from NRC to PSEG: Salem Nuclear Generating Station, Unit 2, Request for Additional Information Re: Amendment Request to Implement TSTF-449 (TAC No. MD1193), dated December 20, 2006

In accordance with the requirements of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) previously submitted License Change Request (LCR) S06-01, dated April 6, 2006, to amend the Technical Specifications (TS) for Salem Generating Station Unit 1 and Unit 2 (Reference 1). LCR S06-01 proposed to replace the existing steam generator (SG) tube surveillance program with the program being proposed by the Technical Specification Task Force (TSTF) in TSTF-449, Revision 4. The proposed change revises the Technical Specifications (TS) definitions of "Identified Leakage" (TS 1.19) and "Pressure Boundary Leakage," (TS 1.21) as well as the TS and Associated Bases for Specifications 3/4.4.6, "Steam Generator (SG) Tube Integrity," and 3/4.4.7.2, "Operational Leakage." The change also adds Specifications 6.8.4.i, "Steam Generator (SG) Program," and 6.9.1.10, "Steam Generator Tube Inspection Report."

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PSEG received a Request for Additional Information (RAI) on LCR S06-01, which was formally transmitted via Reference 2. The response to the RAI is provided in Attachment 1. Additional changes to the TS are provided in Attachment 2. Note that in addition to the RAI responses, one additional change is included to address an administrative reporting correction for TS 6.9.1.5.b.

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

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

Executed on 1/19/07
(Date)

Sincerely,

Thomas P. Joyce

Thomas P. Joyce
Site Vice President
Salem Generating Station

Attachments 2

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

Mr. R. Ennis, Licensing Project Manager - Salem
U. S. Nuclear Regulatory Commission
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RESPONSE TO LCR S06-01 RAIs

By letter dated April 6, 2006 (ML061560393), PSEG Nuclear LLC (the licensee) requested a license amendment for Salem Generating Station, Unit 2 to replace the existing steam generator (SG) tube surveillance program with the program being proposed by the Technical Specification Task Force (TSTF) in TSTF-449, Revision 4. The proposed change revises the Technical Specifications (TS) definitions of "Identified Leakage" (TS 1.19) and "Pressure Boundary Leakage," (TS 1.21) as well as the TS and Associated Bases for Specifications 3/4.4.6, "Steam Generator (SG) Tube Integrity," and 3/4.4.7.2, "Operational Leakage." The change also adds Specifications 6.8.4.i, "Steam Generator (SG) Program," and 6.9.1.10, "Steam Generator Tube Inspection Report."

In order to complete the review, the staff needs the additional information requested below:

1. Proposed TS 6.8.4.i, "Steam Generator (SG) Program," lists "WEXTEX expanded region inspection methodology (W* Methodology)" as an alternative repair option (TS 6.8.4.i.c.1). An associated note indicates the details of the W* methodology will be incorporated into the TS if the amendment is approved by the NRC.

   The NRC cannot approve the TS as written because they do not include the details for the W* alternative repair criteria. Please provide modified TS and Bases pages that include the details necessary for implementing the W* criteria (e.g., tube inspection, tube repair criteria, and reporting requirements), or discuss your plans for removing all references to W* from the TSTF-449 submittal.

A W* placeholder was provided in the April 6, 2006 submittal since the W* Amendment had not yet been approved. Subsequently, on September 28, 2006, Amendment 256 was issued approving the W* Methodology for Salem Unit 2. Appropriate wording from Amendment 256 has now been incorporated into the markup of TS 6.8.4.i and Bases section 3/4.4.6 (Attachment 2).

2. Attachment 3 of your submittal is a regulatory commitment to implement the Steam Generator Program in accordance with NEI 97-06 at the same time the TSTF-449 amendment is implemented. It is the staff's understanding that each plant already implements the NEI 97-06 guidelines, whether or not it has received an amendment to adopt the TSTF-449 steam generator tube integrity program. Please confirm the staff's understanding that NEI 97-06 is currently implemented at Salem 2.

NEI 97-06 is currently implemented at Salem Unit 2.

3. The sixth paragraph of Insert 4, proposed Bases section 3/4.4.6, "Steam Generator (SG) Tube Integrity," discusses the structural integrity performance criterion. Although similar to the corresponding paragraph in TSTF-449, it adds the following sentence:

> The determination of whether thermal loads are primary or secondary loads is based on the ASME definition in which secondary loads are self-limiting and will not cause failure under single load application.

Since the TSTF-449 wording explains that, for circumferential cracking, the classification of loads as primary or secondary will be evaluated on a case-by-case basis, the intent of the added statement above is unclear. Please discuss the reason for adding this statement.

This wording was inadvertently included in our Submittal. The specific wording is a carryover from an earlier draft, based on a previous version of the TSTF-449. The wording has been removed in Attachment 2 to this letter.

4. The proposed Bases section 3/4.4.7.2, "Operational Leakage," Insert 5, omits applicable safety analysis information that is in your current TS Bases and consistent with TSTF-449. Please discuss why this information was deleted from your Bases.

Only one paragraph from the current Bases section 3/4.4.7.2 was deleted in our submittal markup; the deleted paragraph relates to the leakage limit of 1 gpm for all steam generators and the 500 gpd leakage limit for any steam generator. This paragraph is replaced by the TSTF-449 criteria (the limit of 150 gallons per day per steam generator), as indicated in Insert 5 of our submittal markup. The other paragraphs of the current Bases section 3/4.4.7.2 will remain as augmented by Insert 5. This is consistent with TSTF-449, and with the Salem Unit 1 SG Program as approved by Amendment 268.

5. In Action b listed under proposed TS 3.4.7.2, the requirements are to "reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN **during within** the following 30 hours." (Emphasis added by the staff.) Please discuss your plans to correct this apparent typographical error.

The error has been corrected in Attachment 2.

6. In the proposed Index, pages V and XII list the title of 3/4.4.6 as "Steam Generators (SG) Tube Integrity." The staff notes that the singular form, "Steam Generator (SG) Tube Integrity," is actually the title for these sections.

The error has been corrected in Attachment 2.

Administrative Correction to TS 6.9.1.5.b

TS 6.9.1.5.b currently states:

[Report required on an annual basis shall include:] "The complete results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.5.b)."

TS 6.9.1.5.b will be deleted; this reporting requirement is superceded by the reporting requirements of TSTF-449. TS 4.4.5.5.b is being deleted and TS 6.9.1.10 is being added for SG inspection reporting. This additional change to 6.9.1.5.b is reflected in Attachment 2.

Additional Changes to TS

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 Iondine-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

INSERT 2

INSERT 2

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 provisions:

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 and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

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

   3. The operational leakage performance criterion is specified in LCO 3.4.7.2, "Reactor Coolant System Operational Leakage."

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

The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria:

1. WEXTEX expanded region inspection methodology (W* Methodology). -Note: excluded portion of the tube defined by W* is ONLY applicable to Westinghouse installed steam generators, Amendment 256.

Implementation of the steam generator WEXTEX expanded region inspection methodology (W*), as described in Amendment 256 (-LCR S05-07), requires a 100 percent inspection of the inservice tubes for the entire hot leg tubesheet W* Distance. Tubes with service induced degradation identified in the W* Distance shall be removed from service on detection by tube plugging. The following definitions apply to W*:

Bottom of WEXTEX transition (BWT) is the highest point of contact between the tube and the tubesheet at, or below the top-of-tubesheet, as determined by eddy current testing.

W* Length is defined as the length of tubing below the bottom of the WEXTEX transition (BWT) that must be demonstrated to be non-degraded in order for the tube to maintain structural and leakage integrity. For the hot leg, the W* Length is 7.0 inches, which represents the most conservative hot leg length defined in WCAP-14797, Revision 2.

W* Distance is defined in WCAP-14797, Revision 2, as the non-degraded distance from the top of the tubesheet to the bottom of the W* Length, including the distance from the top-of-tubesheet to the bottom of the WEXTEX transition (BWT) and Non-Destructive Examination (NDE) measurement uncertainties (i.e., W* Distance = W* Length + distance to BWT + NDE uncertainties). The W* Distance is the larger of the following two distances as measured from the top-of-the-tubesheet (TTS): (a) 8-inches below the TTS or (b) the non-degraded distance from the TTS to the bottom of the W* Length, including the distance from the TTS to the bottom of the WEXTEX transition (BWT) and Non-Destructive Examination (NDE) measurement uncertainties (i.e., W* Distance = W* Length + distance to BWT + NDE uncertainties)

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 (Excluding the portion of the tube within the tubesheet below the W* Distance, Amendment 256) to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair 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. An assessment of degradation 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 replacement.

Note: Step 2 has two separate requirements (a and b), depending on the type of SG tubes installed.

2a. Original SGs with Alloy 600MA tubes: Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

2b Replacement SGs with Alloy 690 TT tubes: 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.

3. 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). 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.

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

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

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

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

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

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

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

INSERT 3

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 Violations of the requirements of the fire protection program described in the Updated Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report System within 30 days.

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.

INSERT 3

6.9.1.10 STEAM GENERATOR TUBE INSPECTION REPORT

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

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

h. Information regarding the application of W* inspection methodology; including the number of indications, the location of indications (relative to the BWT and TTS), the orientation (axial, circumferential, volumetric), the severity of each indication (e.g., near through-wall or not through wall), the tube side where the indication initiated (inside or outside diameter), the cumulative number of indications detected in the tubesheet region as a function of elevation within the tubesheet, the condition monitoring and operational assessment main steam line leak rate (including aggregate calculated main steam line break leak rate from all other sources), and an assessment of whether the results were consistent with expectations regarding the number of flaws and flaw severity (and if not consistent, a description of the proposed corrective action).

A notification to the NRC shall be provided prior to unit restart if the estimated main steam line leak rate exceeds the design and licensing basis.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. ~~1 GPM total primary-to-secondary leakage through all steam generators and~~ 150~~0~~ gallons per day **primary-to-secondary leakage** through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. NOT USED

f. 1 GPM leakage at a Reactor Coolant System pressure of 2230 ±20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, **or primary-to-secondary leakage not within limit**, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, **and primary-to-secondary leakage**, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN ~~during~~ within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

a. Monitoring the containment sump inventory at least once per 12 hours.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION | PAGE

6.9.1.5 Reports required on an annual basis shall include:

a. DELETED

b. DELETED ~~The complete results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.5.b).~~

c. The results of any specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

6.9.1.6 DELETED

3/4.4.5 RELIEF VALVES (continued)

B. Automatic control of PORVs to control reactor coolant system pressure. This is a function that reduces challenges to the code safety valves for overpressurization events, including an inadvertent actuation of the Safety Injection System.

C. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

D. Manual control of the block valve to : (1) unblock an isolated PORV to allow it to be used for manual and automatic control of Reactor Coolant System pressure (Items A & B), and (2) isolate a PORV with excessive seat leakage (Item C).

E. Manual control of a block valve to isolate a stuck-open PORV.

3/4.4.6 STEAM GENERATOR (SG) **TUBE INTEGRITY**

~~The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.~~

~~The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.~~

INSERT 4

3/4.4.6 STEAM GENERATORS (continued)

~~Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.~~

License Change Request (LCR) S05-07 (LR-N05-0397, LR-N06-0277, LR-N06-0338) provides requirements for limited tubesheet inspection that is only applicable within the hot leg WEXTEX expanded region of the tubesheet for the Salem Unit 2 Westinghouse Series 51 Steam Generators. LCR S05-07 is supported by, but not limited to, the guidance provided in WCAP-14797, Revision 2, "Generic W* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEX Expansions" and supporting information provided from Westinghouse Letter Report LTR-CDME-05-30, "W* Integrity Evaluation for Salem Unit 2 Limited SG Tube RPC Examination (Based on WCAP-14797, Revision 2). In accordance with LCR S05-07, the W* Length is the undegraded length of tubing into the tubesheet below the bottom of the WEXTEX transition (BWT) that precludes tube pullout in the event of a complete circumferential separation of the tube below the W* Length. The W* Distance is the larger of the following two distances as measured from the top-of-the-tubesheet (TTS): (a) 8-inches below the TTS or (b) the non-degraded distance from the TTS to the bottom of the W* Length, including the distance from the TTS to the bottom of the WEXTEX transition (BWT) and Non-Destructive Examination (NDE) measurement uncertainties (i.e., W* Distance = W* Length + distance to BWT + NDE uncertainties). Non-Destructive Examination determines the distance to the BWT for each tube. The nondestructive examination (NDE) measurement uncertainty is provided from LCR S05-07, as supported by WCAP-14797 Revision 2. Tubes with indications detected within the W* Distance will be removed from service by tube plugging.

Tube degradation of any type or extent below the W* Distance, including a complete circumferential separation of the tube, is acceptable and therefore may remain in service. As applied at Salem Unit 2, LCR S05-07 is used to define the required tube inspection depth into the tubesheet, and is not used to permit degradation in the W* Distance to remain in service. Furthermore, potential primary to secondary leakage in the W* Distance, and below the W* Distance, can be conservatively evaluated in accordance with LCR S05-07. The leak rate potential for axial, circumferential, and volumetric indications detected within 12 inches from the top of the tubesheet can be conservatively calculated using the constrained crack model as delineated in LCR S05-07 (supported by Westinghouse LTR-CDME-05-30).

The postulated leakage during a steam line break shall be equal to the following equation, as supported by LCR S05-07:

Postulated SLB Leakage = Assumed Leakage $_{0''-8''\ <TTS}$ + Assumed Leakage $_{8''-12''\ <TTS}$ + Assumed Leakage $_{>12''\ <TTS}$

Where: **Assumed Leakage $_{0''-8''\ <TTS}$ is the postulated leakage for indications that are deemed via flaw depth estimation techniques to be 100% throughwall, and therefore present a potential leak path. This term is applicable to detected indications during an in-service inspection and potentially undetected indications in the steam generator tubes left in service between 0 inches and 8 inches below the top of the tubesheet (TTS). Since tubes with indications detected between 0 and 8 inches below the TTS are plugged upon detection, the calculation of this term for the assessment of SLB leakage for the subsequent operation cycle following an in-service inspection only requires consideration of potentially undetected indications. The calculation of this term for the assessment of SLB leakage for the previous operation cycle, following an in-service inspection, requires consideration of both detected and potentially undetected indications.**

Assumed Leakage $_{''8-12''\ <TTS}$ is the conservatively projected leakage in steam generator tubes between 8 and 12 inches below the top of the tubesheet. Implementation of LCR S05-07 does not require tube inspection below the W* Distance, therefore the methodology for conservatively calculating the population of indications between 8 and 12 inches below the TTS is provided by fitting a regression line to the cumulative inspection data (detected indications) from all SGs and projecting the number of indications (to minus 12 inches below TTS) using a 95-percent probability prediction bound. The cumulative indications from all steam generators are conservatively assumed to occur in one SG (similar to figure 16 of Westinghouse LTR-CDME-05-30). The conservative leakage rate for the indications between 8 and 12 inches is 0.0033 gpm multiplied by the number of projected indications (as discussed in LCR S05-07 submittals LR-N06-0277 and LR-N06-0338). The leak rate of indications detected between 8 and 12 inches are bounded by the projected total discussed above, assuming that the inspection results for detected indications do not contradict the calculated population as described previously.

Assumed Leakage $_{>12'' <TTS}$ is the calculated leakage from the steam generator tubes left in service below 12 inches from the top of the tubesheet. This is 0.00009 gpm times number of tubes left in service in the steam generator.

Each SG is assessed for Main Steam Line Break (MSLB) leakage individually in accordance with the discussion above, and the SG with the most calculated leakage is conservatively assigned as the affected SG.

The calculated MSLB leakage provided above, including MSLB leakage from all other sources, shall be reported to the NRC in accordance with applicable Technical Specifications. The Calculated MSLB Leakage must be less than the maximum allowable MSLB leak rate limit in any one steam generator in order to maintain doses within 10 CFR 50.67 guideline values and within GDC-19 values during a postulated main steam line break event.

3/4.4.6 STEAM GENERATORS (SG) **TUBE INTEGRITY** (Continued)

~~Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be evaluated for reportability to the Commission pursuant to the applicable sections of 10 CFR 50.72 and 10 CFR 50.73.~~

3/4.4.7 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.7.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.7.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet, excluding the portion of the tube within the tubesheet below the W* Distance (excluded portion of the tube defined by W* is ONLY applicable to Westinghouse installed steam generators, Amendment 256). The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.i, "Steam Generator (SG) Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that significantly affect burst or collapse. In that context, the term "significant" is defined as, "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." ~~The determination of whether thermal loads are primary or secondary loads is based on the ASME definition in which secondary loads are self-limiting and will not cause failure under single load application~~. For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB and draft Reg. Guide 1.121.

The accident induced leakage performance criterion ensures that the primary-to-secondary leakage caused by a design basis accident, other than a steam generator tube rupture (SGTR), is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm per SG. The accident induced leakage rate includes any primary-to-secondary leakage existing prior to the accident in addition to primary-to-secondary leakage induced during the accident.

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.7.2, "Operational Leakage," and limits primary-to-secondary leakage through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

The ACTION requirements are modified by a Note clarifying that the Actions may be entered independently for each SG tube. This is acceptable because the Action requirements provide appropriate compensatory actions for each affected SG tube. Complying with the Action requirements may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Action requirements.

If it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program, an evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. An action time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity. If the evaluation determines that the affected tube(s) have tube integrity, plant operation is allowed to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering HOT SHUTDOWN following the next refueling outage or SG inspection. This allowed outage time is acceptable since operation until the next inspection is supported by the operational assessment.

If SG tube integrity is not being maintained or the Action requirements are not met, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. The action times are reasonable based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

During shutdown periods the SGs are inspected as required by surveillance requirements and the Steam Generator Program. NEI 97-06, "Steam Generator Program Guidelines," and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period. The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities and inspection locations. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines. The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.4.i contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.8.4.i are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in size measurement and future growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). NEI 97-06 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria. The Frequency of prior to entering HOT SHUTDOWN following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.