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United States Nuclear Regulatory Commission
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Washington, DC 20555

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

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear, LLC (PSEG) hereby transmits a request for amendment of the Technical Specifications (TS) for Salem Generating Station Unit 2. Pursuant to the requirements of 10CFR50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

The proposed amendment would revise the Unit 2 Technical Specification (TS) requirements related to steam generator (SG) tube integrity. The proposed changes are consistent with those in NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity."[1] These changes will provide a programmatic framework for monitoring and maintaining the integrity of steam generator tubes consistent with 10 CFR 50, Appendices A and B, and the guidance provided in NEI 97-06, "Steam Generator Program Guidelines." The proposed changes are similar to changes previously approved in Amendment 268 for Salem Unit 1, dated October 14, 2005.

The proposed amendment specifies inspection requirements both for SGs containing Alloy 600 mill annealed (MA) tubes and SGs containing Alloy 690 thermally treated (TT) tubes. The Unit 2 original SGs, containing Alloy 600 MA tubes, are scheduled to be replaced with Alloy 690 TT-tube SGs in 2008. Since this proposed amendment specifies inspection requirements for SGs of specific tube material, it can be

1 This Unit 2 submittal fulfills our GL 2006-01 commitment provided in PSEG letter LR-N06-0054, dated February 15, 2006

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implemented at one time for Salem Unit 2. Unit 2 will comply with the inspection requirements for the specific SG tube material type at the time of the inspection.

The proposed amendment also includes an alternate repair criteria that was previously submitted for approval via PSEG License Change Request LCR S05-07, dated September 21, 2005. LCR S05-07 requested approval of the WEXTEX expanded region inspection methodology (W* methodology). If LCR S05-07 is approved, then an equivalent description of the W* methodology will be incorporated into this proposed change as well.

Attachment 1 provides an evaluation of the proposed changes. Attachment 2 provides the existing TS pages marked-up to show the proposed changes. Attachment 3 summarizes the regulatory commitments made in this submittal. Attachment 4 provides the TS Bases pages marked-up to show changes consistent with the proposed TS changes.

PSEG requests a 60-day implementation period after amendment approval. Approval of this change is requested by March 31, 2007.

Should you have any questions regarding this request, please contact Mr. Paul Duke at (856) 339-1466.

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

Executed on 4/6/06
(Date)

Sincerely,

Thomas P. Joyce

Thomas P. Joyce
Site Vice President
Salem Generating Station

Attachments (4)

C

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

ADDITION OF REQUIREMENTS FOR STEAM GENERATOR TUBE INTEGRITY, STEAM GENERATOR PROGRAM, AND STEAM GENERATOR TUBE INSPECTION REPORT AND REVISION OF REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE REQUIREMENTS

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CHANGES TO TECHNICAL SPECIFICATIONS

1. DESCRIPTION

The purpose of this amendment is to replace the steam generator (SG) detailed programmatic requirements contained in Technical Specifications (TS) with a SG Tube Integrity TS and Bases, revise the TS for reactor coolant system (RCS) Operational Leakage, and add a SG Program and SG Tube Inspection Report. The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97-06, "Steam Generator Program Guidelines," (Reference 1). The changes proposed are based on Technical Specification Task Force (TSTF) Traveler TSTF-449, Revision 4, "Steam Generator Tube Integrity," which was transmitted by letter dated April 14, 2005.

2. PROPOSED CHANGE

The detailed, prescriptive requirements in existing Salem Unit 2 TS 3/4.4.6 are replaced by requirements for a new Limiting Condition for Operation (LCO), "Steam Generator (SG) Tube Integrity," a new program 6.8.4.i, "Steam Generator (SG) Program," and a new reporting requirement 6.9.1.10, "Steam Generator Tube Inspection Report." The amendment replaces a large amount of prescriptive, outdated details on SG inspection requirements with a requirement to implement a state of the art performance-based program that is supported by a NEI SG initiative (NEI 97-06), extensive industry guidance, and an active industry Technical Advisory Group. TS 6.8.4.i requires a Steam Generator Program to be established and implemented to ensure that SG tube integrity is maintained, and to describe SG condition monitoring, performance criteria, repair methods, repair criteria, and inspection intervals. TS 6.9.1.10 requires a report within 180 days of initial entry into MODE 4 following a steam generator inspection. These changes are a significant improvement over the existing outdated TS requirements. The TS for SG Tube Integrity contains surveillance requirements (SR) for tube integrity verification and repair and actions necessary should tube integrity not be maintained. The proposed changes to Salem Unit 2 TS 3/4.4.7.2, "Reactor Coolant System Operational Leakage," reduce the allowable leakage from any one SG from 500 to 150 gallons per day. The proposed changes to Salem Unit 2 TS 3/4.4.7.2 also revise the LCO, ACTION requirements and Surveillances to clarify the requirements related to primary-to-secondary leakage. The proposed changes to TS 1.15, "IDENTIFIED LEAKAGE," and TS 1.21, "PRESSURE BOUNDARY LEAKAGE," are conforming changes to clarify primary-to-secondary leakage. Title changes are proposed to TS INDEX pages V and XII, and changes are proposed to TS 1.19, "OPERATIONAL MODE – MODE," and

TS 6.8.4.g.9 to correct typographical errors. TS Bases changes are made to reflect the corresponding changes proposed to the TS.

The above changes are shown on the attached marked-up TS pages (Attachment 2). Changes to be inserted in the Bases to reflect the proposed TS changes are included in Attachment 4 for informational purposes.

3. BACKGROUND

The SG tubes in pressurized water reactors have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the RCPB, the SG tubes are unique in that they act as a heat transfer surface between the primary and secondary systems to remove heat from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system.

SG tube integrity is necessary in order to satisfy the tubing's safety functions. Maintaining tube integrity ensures that the tubes are capable of performing their intended safety functions consistent with the plant licensing basis, including applicable regulatory requirements.

Concerns relating to the integrity of the tubing stem from the fact that the SG tubing is subject to a variety of degradation mechanisms. SG tubes have experienced tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. When the degradation of the tube wall reaches a prescribed repair criterion, the tube is considered defective and corrective action is taken.

The criteria governing structural integrity of SG tubes were developed in the 1970s and assumed uniform tube wall thinning. This led to the establishment of a through wall SG tube repair criterion (e.g., 40%) that has historically been incorporated into most pressurized water reactor TS and has been applied, in the absence of other repair criteria, to all forms of SG tube degradation where sizing techniques are available. Since the basis of the through wall depth criterion was 360° wastage, it is generally considered to be conservative for other mechanisms of SG tube degradation. The repair criterion does not allow licensees the flexibility to manage different types of SG tube degradation. Licensees must either use the through wall criterion for all forms of degradation or obtain

approval for use of more appropriate repair criteria that consider the structural integrity implications of the given mechanism.

For the last several years, the industry, through the Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP), has developed a generic approach to improving SG performance referred to as "Steam Generator Degradation Specific Management" (SGDSM). Under this approach, different methods of inspection and different repair criteria may be developed for different types of degradation. A degradation specific approach to managing SG tube integrity has several important benefits. These include:

- Improved scope and methods for SG inspection,
- Industry incentive to continue to improve inspection methods, and
- Development of plugging and repair criteria based on appropriate nondestructive examination (NDE) parameters.

As a result, the assurance of SG tube integrity is improved and unnecessary conservatism is eliminated.

Over the course of this effort, the SGMP has developed a series of EPRI guidelines that define the elements of a successful SG program. These guidelines include:

- "Steam Generator Examination Guideline", (Reference 2),
- "Steam Generator Integrity Assessment Guideline", (Reference 3),
- "Steam Generator In-situ Pressure Test Guideline", (Reference 4),
- "PWR Primary-to-Secondary Leak Guideline", (Reference 5),
- "Primary Water Chemistry Guideline", (Reference 6), and
- "Secondary Water Chemistry Guideline", (Reference 7).

These EPRI guidelines, along with NEI 97-06, "Steam Generator Program Guidelines," (Reference 1) tie the entire SG program together, while defining a comprehensive, performance based approach to managing SG performance.

In parallel with the industry efforts, the NRC pursued resolution of SG performance issues. In December of 1998, the NRC Staff acknowledged that the Steam Generator Program described by NEI 97-06 and its referenced EPRI Guidelines provides an acceptable starting point to use in the resolution of differences between it and the staff's proposed Generic Letter and draft Regulatory Guide (DG-1074). Since then the industry and the NRC have participated in a series of meetings to resolve the differences and develop the regulatory framework necessary to implement a comprehensive Steam Generator Program.

Revising the existing regulatory framework to accommodate degradation specific management is the most appropriate way to address the issues of regulatory stability, resource expenditure, use of state-of-the-art inservice inspection techniques, repair criteria, and enforceability. The NRC staff has stated that an integrated approach for addressing SG tube integrity is essential and that materials, systems, and radiological issues that pertain to tube integrity need to be considered in the development of the new regulatory framework.

4. TECHNICAL ANALYSIS

The proposed changes do not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The primary coolant activity limit and its assumptions are not affected by the proposed changes to these TS. The proposed changes are an improvement to the existing SG inspection requirements and provide additional assurance that the plant licensing basis will be maintained between SG inspections.

A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of Salem's licensing basis. The analysis of a SGTR event assumes a bounding primary-to-secondary leakage rate equal to the operational leakage rate limits in the licensing basis plus the leakage rate associated with a double-ended rupture of a single tube.

For design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor, the SG tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses typically assume that primary-to-secondary leakage for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident-induced stresses. For accidents that do not involve fuel damage, the reactor coolant activity levels are at the TS allowable limits. For accidents that do involve fuel damage, the primary coolant activity values are a function of the amount of activity released from the damaged fuel.

The consequences of these design basis accidents are, in part, functions of the radioactivity levels in the primary coolant and the accident primary-to-secondary leakage rates. As a result, limits are included in the Salem TS for operational leakage and for DOSE EQUIVALENT I-131 in primary coolant to ensure that Salem is operated within its analyzed condition.

The proposed TS change includes a reduction in the current TS Reactor Coolant System operational leakage limit from 500 gallons per day to 150 gallons per day. The new limit of 150 gallons per day of primary-to-

secondary leakage through any one SG is based on operating experience as an indication of one or more tube leaks. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

The other TS changes proposed are a significant improvement over current requirements. They replace an outdated prescriptive TS with one that references Steam Generator Program requirements that incorporate the latest knowledge of SG tube degradation morphologies and the techniques developed to manage them.

The requirements being proposed are more effective in detecting SG degradation and prescribing corrective actions than those required by current TS. As a result, these proposed changes will result in added assurance of the function and integrity of SG tubes.

The table below and associated sections describe in detail and provide the technical justification for the proposed changes.

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Section
Operational primary-to-secondary leakage	$\leq$1 gpm total through all SGs and $\leq$ 500 gallons per day through any one SG	RCS Operational leakage TS $\leq$ 150 gallons per day through any one SG.	1
RCS primary-to-secondary leakage through any one SG not within limits	Reduce LEAKAGE to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours	RCS Operational leakage TS - Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.	2
RCS LEAKAGE determined by water inventory balance (Unit 2 SR 4.4.7.2.1.d)	Modifying notes not specified	Added new Notes indicating SR not applicable to primary-to-secondary leakage and not required to be performed until 12 hours after establishment of steady state operation.	3
SR for primary-to-secondary leakage	Not specified	RCS Operational leakage TS: Added SR to verify primary-to-secondary leakage every 72 hours. Added Note stating "Not required to be performed until 12 hours after establishment of steady state operation."	4

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Section
Frequency of verification of tube integrity (Unit 2 SR 4.4.6.3)	6 to 40 months depending on SG category defined by previous inspection results.	SG Tube Integrity TS – Requires Surveillance Frequency in accordance with TS 6.8.4.i, Steam Generator Program. Frequency is dependent on tubing material and the previous inspection results and the anticipated defect growth rate. Steam Generator Program – Establishes maximum inspection intervals	5
Tube sample selection (Unit 2 SR 4.4.6.2)	Based on SG Category, industry experience, random selection, existing indications, and results of the initial sample set - 3% times the number of SGs at the plant as a minimum	Steam Generator Program and implementing procedures - Dependent on a pre-outage evaluation of actual degradation locations and mechanisms, and operating experience – 20% of the active tube population as a minimum.	6

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Section
Inspection techniques	Not specified	SG Tube Integrity TS – Unit 2 SR 4.4.6.1 requires that tube integrity be verified in accordance with the Steam Generator Program. Steam Generator Program and implementing procedures – Establishes requirements for using qualified NDE techniques. Requires use of qualified techniques in SG inspections. Requires a pre-outage evaluation of potential tube degradation morphologies and locations and an identification of NDE techniques capable of finding the degradation.	7
Inspection Scope (Unit 2 SR 4.4.6.4.a.8)	Hot leg point of entry to the top support plate on the cold leg side of the U-bend	Steam Generator Program procedures – Inspection scope is defined by the degradation assessment that considers existing and potential degradation morphologies and locations. Explicitly requires consideration of entire length of tube from tube-sheet weld to tube-sheet weld. (The tube-to-tubesheet weld is not part of the tube.)	8

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Section
Performance criteria	Operational leakage ≤ 1 gpm total or ≤ 500 gallons per day through any one SG. No criteria specified for structural integrity or accident induced leakage.	RCS Operational leakage TS – Unit 2 LCO 3.4.7.2 requires Operational leakage ≤ 150 gallons per day through any one SG. SG Tube Integrity TS – Unit 2 TS 3/4.4.6 requires that tube integrity be maintained. TS 6.8.4.i – Defines structural integrity and accident induced leakage performance criteria which are dependent on design basis limits. Provides provisions for condition monitoring assessment to verify compliance.	9
Repair criteria (Unit 2 SR 4.4.6.4.a.6)	Plug tubes with imperfections extending ≥ 40% nominal tube wall thickness.	TS 6.8.4.i – Criteria unchanged	10
ACTIONS (Unit 2 LCO 3.4.7.2.c)	Performance Criteria not defined. Primary-to-secondary leakage limit and actions included in TS. Plug tubes exceeding plugging limit.	RCS Operational leakage TS and SG Tube Integrity TS – Contains primary-to-secondary leakage limit, SG tube integrity requirements and ACTIONS required upon failure to meet performance criteria. Plug tubes satisfying repair criteria.	11

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Section
Repair methods (Unit 2 SR 4.4.6.4.a.6)	Methods (except plugging) require previous approval by the NRC. No alternate repair criteria has been approved by NRC.	TS 6.8.4.i – Requirements unchanged	12
Reporting requirements (Unit 2 SR 4.4.6.5)	Plugging report required 15 days after each inservice inspection, 12-month report documenting inspection results, and reports in accordance with §50.72 when the inspection results fall into category C-3.	CFR - Serious SG tube degradation (i.e., tubing fails to meet the structural integrity or accident induced leakage criteria) requires reporting in accordance with 50.72 or 50.73. TS 6.9.1.10 - 180 days after the initial entry into MODE 4 after performing a SG inspection	13
Defining SG Terminology	Normal TS definitions (i.e., Definitions Section) did not address SG Program issues. The Definitions Section uses the term "steam generator tube leakage."	TS 6.8.4.i, TS Bases, Steam Generator Program procedures – Includes Steam Generator Program terminology applicable only to SGs. The Definitions Section is revised to use the term "primary-to-secondary leakage."	14

Section 1: Operational Leakage

The primary-to-secondary leakage limit has been reduced to ≤ 150 gallons per day through any one SG. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures. This together with the allowable accident induced leakage limit helps to ensure that the dose contribution from tube leakage will be limited to less than the 10 CFR 50.67 and GDC 19 dose limits, or other NRC approved licensing basis, for postulated faulted events.

This limit also contributes to meeting the GDC 14 requirement that the reactor coolant pressure boundary "have an extremely low probability of abnormal leakage, of rapidly propagating to failure, and of gross rupture." The proposed Surveillance ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. The Steam Generator Program uses the EPRI Primary-to-Secondary Leak Guideline (Ref. 5) to establish sampling requirements for determining primary-to-secondary leakage and plant shutdown requirements if leakage limits are exceeded. The guidelines ensure leakage is effectively monitored and timely action is taken before a leaking tube exceeds the performance criteria.

The proposed revision to the technical specification requirement to limit primary-to-secondary leakage through any one SG to less than or equal to 150 gallons per day is significantly more conservative than the existing technical specification limit of 1 gpm total primary-to-secondary leakage through all SGs that is based on an initial condition of the safety analysis.

Section 2: Operational Leakage Actions

If primary-to-secondary leakage exceeds 150 gallons per day through any one SG, a plant shutdown must be commenced. The existing technical specifications allow 4 hours to reduce primary-to-secondary leakage to less than the limit. HOT STANDBY must be achieved within the next 6 hours and COLD SHUTDOWN within the following 30 hours. The proposed technical specification removes this allowance.

The removal of the 4-hour period during which primary-to-secondary leakage can be reduced to avoid a plant shutdown results in a technical specification that is significantly more conservative than the existing RCS Operational Leakage specification. This change is consistent with the Steam Generator Program that also does not allow 4 hours before commencing a plant shutdown.

Section 3: RCS Operational Leakage Determined by Water Inventory Balance

The proposed change adds Notes to Unit 2 SR 4.4.7.2.1.d that make the water inventory balance method not applicable to determining primary-to-secondary leakage and allows the SR to not be performed until 12 hours after establishment of steady state operation. This change is proposed because primary-to-secondary leakage as low as 150 gallons per day through any one SG cannot be measured accurately by an RCS water inventory balance. This change is necessary to make the surveillance requirement appropriate for the proposed LCO.

Section 4: SG Tube Integrity Verification

Unit 2 SR 4.4.7.2.1.c has been added to verify the LCO requirement on primary-to-secondary leakage, separate from the water inventory balance of SR 4.4.7.2.1.d. Steam generator tube integrity is verified in accordance with a SR in the SG Tube Integrity Specification.

The Steam Generator Program and the EPRI "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines" (Ref. 5) provide guidance on leak rate monitoring. During normal operation the program depends upon continuous process radiation monitors and/or radiochemical grab sampling in accordance with the EPRI guidelines. The monitoring and sampling frequency increases as the amount of detected leakage increases or if there are no continuous radiation monitors available.

Determination of primary-to-secondary leakage is required every 72 hours. The SR is modified by a Note stating the SR is not required to be performed until 12 hours after establishment of stable operating conditions. As stated above, additional monitoring of primary-to-secondary leakage is also required by the Steam Generator Program based upon guidance provided in Reference 5.

Section 5: Frequency of Verification of SG Tube Integrity

The current technical specifications contain prescriptive inspection intervals which depend on the condition of the tubes as determined by the last SG inspection. The tube condition is classified into one of three categories based on the number of tubes found degraded and defective. The minimum inspection interval is no less than 12 and no more than 24 months unless the results of two consecutive inspections are in the best category (no additional degradation), and then the interval can be extended to 40 months.

The surveillance Frequency in the proposed Steam Generator Tube Integrity specification is governed by the requirements in the Steam Generator Program and specifically by References 2 and 3. The proposed Frequency is also prescriptive, but has a stronger engineering basis than the existing technical

specification requirements. The interval is dependent on tubing material and whether any active degradation associated with cracking is found. The interval is limited by existing and potential degradation mechanisms and their anticipated growth rate. In addition, a maximum inspection interval is established in TS 6.8.4.i.

The maximum inspection interval requirement for Alloy 600 mill annealed tubing (600MA) is "Inspect 100% of the tubes at sequential periods of 60 effective full power months[2]. 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." This frequency is at least as conservative as the current technical specification requirement.

The maximum inspection interval for Alloy 600 thermally treated tubing is "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 maximum inspection interval for Alloy 690 thermally treated tubing is "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." Even though the maximum interval for Alloy 600 thermally treated tubing and Alloy 690 thermally treated tubing is slightly longer than allowed by current technical specifications, it is only applicable to SGs with advanced materials, it is only achievable early in SG life and only if the SGs are free from active degradation. In addition, the interval must be supported by an evaluation that shows that the performance criteria will continue to be met at the next SG inspection. Taken in total, the proposed inspection intervals provide a larger margin of safety than the current requirements because they are based on an engineering evaluation of the tubing condition and potential degradation mechanisms and growth rates, not only on the previous inspection results. As an added safety measure, the Steam Generator Program requires a minimum sample size at each inspection that is significantly larger than that required by current technical specifications (20

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2 Salem Unit 2 original SGs contain Alloy 600 mill annealed tubing. The original SGs are scheduled to be replaced in 2008 with SGs containing Alloy 690 thermally treated tubing.

percent versus 3 percent times the number of SGs in the plant); thus providing added assurance that any degradation within the SGs will be detected and accounted for in establishing the inspection interval.

The proposed maximum inspection intervals are based on the historical performance of advanced SG tubing materials. Reference 8 shows that the performance of Alloy 600TT is significantly better than the performance of 600MA tubing, the material used in SG tubing at the time that the current technical specifications were written. There have been very few instances of cracking in 600TT tubes in a U.S. SG and this degradation appears to be limited to a small number of tubes in specific SGs that were left with high residual stress as a result of a problem in their manufacturing process. The mechanism is not a result of operational degradation. There are no known instances of cracking in 690TT tubes in either the U.S. or international SGs.

In summary, the proposed change is an improvement over the current technical specification. The current technical specification bases inspection intervals on the results of previous inspections; it does not require an evaluation of expected performance. The proposed technical specification uses information from previous plant inspections as well as industry experience to evaluate the length of time that the SGs can be operated and still provide reasonable assurance that the performance criteria will be met at the next inspection. The actual interval is the shorter of the evaluation results and the requirements in Reference 3. Allowing plants to use the proposed inspection intervals maximizes the potential that plants will use improved techniques and knowledge since better knowledge of SG conditions supports longer intervals.

<u>Section 6: SG Tube Sample Selection</u>

The current technical specifications base tube selection on SG conditions and industry and plant experience. The minimum sample size is 3% of the tubes, times the number of SGs in the plant. The proposed change refers to the Steam Generator Program degradation assessment guidance for sampling requirements. The minimum sample size is 20% of the active tube population inspected.

The Steam Generator Program requires the preparation of a degradation assessment. The degradation assessment is the key document used for planning a SG inspection, where inspection plans and related actions are determined, documented, and communicated. The degradation assessment addresses the various reactor coolant pressure boundary components within the SG (e.g., plugs, sleeves, tubes, and components that support the pressure boundary.) In a degradation assessment, tube sample selection is performance based and is dependent upon actual SG conditions and plant operational experience and of the industry in general. Existing and potential degradation mechanisms and their locations are evaluated to determine which tubes will be

inspected. Tube sample selection is adjusted to minimize the possibility that tube integrity might degrade during an operating cycle beyond the limits defined by the performance criteria. The EPRI Steam Generator Examination Guidelines (Ref. 2) and EPRI Steam Generator Integrity Assessment Guidelines (Ref. 3) provide guidance on degradation assessment.

In general, the sample selection considerations required by the current technical specifications and the requirements in the Steam Generator Program as proposed by this change are consistent, but the Steam Generator Program provides more guidance on selection methodologies and incorporation of industry experience and requires more extensive documentation of the results. Therefore, the sample selection method proposed by this change is more conservative than the current technical specification requirements. In addition, the minimum sample size in the proposed requirements is larger.

Section 7: SG Inspection Techniques

The Surveillance Requirements proposed in the Steam Generator Tube Integrity specification require that tube integrity be verified in accordance with the requirements of the Steam Generator Program. The Steam Generator Program uses the EPRI Steam Generator Examination Guidelines (Ref. 2) to establish requirements for qualifying NDE techniques and maintains a list of qualified techniques and their capabilities.

The Steam Generator Program requires the performance of a degradation assessment and refers utilities to EPRI Steam Generator Examination Guidelines (Ref. 2) and EPRI Steam Generator Integrity Assessment Guidelines (Ref. 3) for guidance on its performance. The degradation assessment will identify current and potential degradation locations and mechanisms and NDE techniques that are effective in detecting their existence. Tube inspection techniques are chosen to reliably detect flaws that might progress during an operating cycle beyond the limits defined by the performance criteria.

The current technical specifications contain no requirements on NDE inspection techniques. The proposed change is an improvement over the current technical specifications that contained no similar requirement.

Section 8: SG Inspection Scope

The current technical specifications include a definition of tube inspection that specifies the end points of the eddy-current examination of each tube. An inspection is required from the point of entry of the tube on the hot leg side to the top support plate on the cold leg side of the tube after the U-bend. This definition is overly prescriptive and simplistic and has led to interpretation questions in the past.

The Steam Generator Program states, "The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. 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." The Steam Generator Program provides extensive guidance and a defined process, the degradation assessment, for determining the extent of a tube inspection. This guidance takes into account industry and plant specific history to determine potential degradation mechanisms and the location that they might occur within the SG. This information is used to define a performance based inspection scope targeted on plant specific conditions and SG design.

The proposed change is an improvement over the current technical specifications because it focuses the inspection effort on the areas of concern, thereby minimizing the unnecessary data that the NDE analyst must review to identify indication of tube degradation.

Section 9: SG Performance Criteria

The proposed change adds a performance-based Steam Generator Program to the Technical Specifications. A performance-based approach has the following attributes:

- measurable parameters,
- objective criteria to assess performance based on risk-insights,
- deterministic analysis and/or performance history, and
- licensee flexibility to determine how to meet established performance criteria.

The performance criteria used for SGs are based on tube structural integrity, accident induced leakage, and operational leakage. The structural integrity and accident induced leakage criteria were developed deterministically and are consistent with Salem's licensing basis. The operational leakage criterion was based on providing an effective measure for minimizing the frequency of tube ruptures at normal operating and faulted conditions. The proposed structural integrity and accident induced leakage performance criteria are new requirements. The performance criteria are specified in TS 6.8.4.i. The

requirements and methodologies established to meet the performance criteria are documented in the Steam Generator Program. The current technical specifications contain only the operational leakage criterion; therefore, the proposed change is more conservative than the current requirements.

The SG performance criteria identify the standards against which performance is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity throughout each operating cycle.

The structural integrity performance criterion is:

> "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.

The structural integrity performance criterion is based on providing reasonable assurance that a SG tube will not burst during normal operation or postulated accident conditions.

Adjustments to include contributing loads are addressed in the applicable EPRI guidelines.

Normal steady state full power operation is defined as the conditions existing during MODE 1 operation at the maximum steady state reactor power as defined in the design or equipment specification. Changes in design parameters such as plugging or sleeving levels, primary or secondary modifications, or T_{HOT} should be assessed and included if significant.

The definition of normal steady state full power operation is important as it relates to application of the safety factor of three in the structural integrity performance criterion. The criterion requires "...retaining a safety factor of 3.0 under normal steady state full power operation primary-to-secondary pressure differential...". The application of the safety factor of three to normal steady state full power operation is founded on past NRC positions, accepted industry practice, and the intent of the ASME Code for original design and evaluation of inservice components. The assumption of normal steady state full power operating pressure differential has been consistently used in the analysis, testing and verification of tubes with stress corrosion cracking for verifying a safety factor of three against burst. Additionally, the 3ΔP criterion is measurable through the condition monitoring process.

The actual operational parameters may differ between cycles. As a result of changes to these parameters, reaching the differential pressure in the equipment specification may not be possible during plant operations. Evaluating to the pressure in the design or equipment specification in these cases would be an unnecessary conservatism. Therefore, the definition allows adjustment of the 3ΔP limit for changes in these parameters when necessary. Further guidance on this adjustment is provided in Appendix M of the EPRI Steam Generator Integrity Assessment Guidelines (Ref. 3).

The accident induced leakage performance criterion is:

"The primary-to-secondary accident induced leakage rate for all design basis accidents, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. Leakage is not to exceed 1 gpm per SG."

Primary-to-secondary leakage is a factor in the activity releases outside containment resulting from a limiting design basis accident. The potential dose consequences from primary-to-secondary leakage during postulated design basis accidents must not exceed the radiological limits imposed by 10 CFR Part 50.67 guidelines, or the radiological limits to control room personnel imposed by GDC 19, or other NRC approved licensing basis.

When calculating offsite doses, the safety analysis for the limiting Design Basis Accident, other than a steam generator tube rupture, assumes a total of 1 gpm primary-to-secondary leakage as an initial condition. Recent experience with degradation mechanisms involving tube cracking has revealed that leakage under accident conditions can exceed the level of operating leakage by orders of magnitude. The NRC has concluded (Item Number 3.4 in Attachment 1 to Reference 13) that additional research is needed to develop an adequate methodology for fully predicting the effects of leakage on the outcome of some

accident sequences. Therefore, a separate performance criterion was established for accident-induced leakage. The limit for accident-induced leakage is 1 gpm, which is the plant's design basis.

The operational leakage performance criterion is:

> "The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day."

Plant shutdown will commence if primary-to-secondary leakage exceeds 150 gallons per day at room temperature conditions from any one SG.

The operational leakage performance criterion is documented in the Steam Generator Program and implemented in Unit 2 LCO 3.4.7.2, "Operational LEAKAGE."

Proposed Administrative TS 6.8.4.i contains the performance criteria and is more conservative than the current technical specifications. The current technical specifications do not address the structural integrity and accident induced leakage criteria. In addition, the primary-to-secondary leakage limit (150 gallons per day per SG) included in the proposed changes is more conservative than the primary-to-secondary leakage limit in the current RCS operational leakage specification.

Section 10: SG Repair Criteria

Repair criteria are those NDE measured parameters at or beyond which the tube must be repaired or removed from service by plugging.

Tube repair criteria are established for each active degradation mechanism. Tube repair criteria are the standard through-wall depth-based criterion (i.e., 40% through-wall.) or through-wall depth based criteria for repair techniques approved by the NRC, or other Alternate Repair Criteria (ARC) approved by the NRC. A SG degradation-specific management strategy is followed to develop and implement an ARC.

The surveillance requirements of the proposed Steam Generator Integrity specification require that tubes that satisfy the tube repair criteria be plugged or repaired in accordance with approved methods. SG tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "repaired/plugged-on-detection" and their integrity should be assessed. It cannot be guaranteed that every flaw will be detected with a given eddy-current technique and, therefore, it is possible that some flaws will not be detected during an inspection. If a flaw is discovered and it is determined that this flaw would have satisfied the repair criteria at the time of the last inspection of the affected

tube, this does not mean that the Steam Generator Program was violated. However, it may be an indication of a shortcoming in the inspection program.

Any plant-specific alternate repair criteria approved for a licensee are listed in the Steam Generator Program Technical Specification (TS 6.8.4.i for Salem Unit 2). These are the same criteria that are listed in the existing Technical Specifications. In addition, the Steam Generator Program Technical Specification lists any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.

Section 11: ACTIONS

The RCS Operational Leakage and Steam Generator Tube Integrity specifications require the licensee to monitor SG performance against performance criteria in accordance with the Steam Generator Program.

During plant operation, monitoring is performed using the operational leakage criterion. Exceeding that criterion will lead to a plant shutdown in accordance with Unit 2 LCO 3.4.7.2. Once shutdown, the Steam Generator Program will ensure that the cause of the operational leakage is determined and corrective actions are taken to prevent recurrence. Operation may resume when the requirements of the Steam Generator Program have been met. This requirement is unchanged from the current technical specifications.

Also during plant operation the licensee may discover an error or omission that indicates a failure to implement a required plugging during a previous SG inspection. Under these circumstances, the licensee is expected to take the ACTION requirements required by the Steam Generator Tube Integrity specification. If a performance criterion has been exceeded, a principal safety barrier has been challenged and 10 CFR 50.72 (b) (3) (ii) (A) and 50.73 (a) (2) (ii) (A) require NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence. The Steam Generator Program additionally requires that the report contain information on the performance criteria exceeded and the basis for the planned operating cycle. The current technical specifications only address operational leakage during operations and therefore do not include the proposed requirement.

During MODES 5 and 6, the operational leakage criterion is not applicable, and the SGs will be inspected as required by the surveillance in the Steam Generator Tube Integrity specification. A condition monitoring assessment of the "as found" condition of the SG tubes will be performed to determine the condition of the SGs with respect to the structural integrity and accident leakage performance criteria. If the performance criteria are not met, the Steam Generator Program requires ascertaining the cause and determining corrective actions to prevent recurrence. Operation may resume when the requirements of the Steam Generator Program have been met.

The proposed technical specification's change to the ACTIONS required upon exceeding the operational leakage criterion is conservative with respect to the current technical specifications as explained in Section 2 above.

The current technical specifications do not address ACTIONS required while operating if it is discovered that the structural integrity or accident induced leakage performance criteria or a repair criterion are exceeded, so the proposed change is conservative with respect to the current technical specifications.

If performance or repair criteria are exceeded while shutdown, the affected tubes must be repaired or plugged. A report will be submitted to the NRC in accordance with Technical Specification 6.9.1.10. The changes in the required reports are discussed in Section 13 below.

Section 12: SG Repair Methods

Repair methods are those means used to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a SG tube is not a repair.

The purpose of a repair is typically to reestablish or replace the RCPB. The proposed Steam Generator Tube Integrity surveillance requirements requires that tubes that satisfy the tube repair criteria be plugged in accordance with the Steam Generator Program. Salem Unit 2 does not have any NRC approved repair method established. A separate license change request would be submitted to request approval of a repair method.

Steam generator tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "plugged-on-detection" and their integrity is assessed. This requirement is unchanged by the proposed technical specifications.

Note that SG plug designs do not require NRC review and therefore plugging is not considered a repair in the context of this requirement.

The above approach is not a change to the technical specifications.

Section 13: Reporting Requirements

The current technical specifications require the following reports:

- A report listing the number of tubes plugged or repaired in each SG submitted within 15 days of the end of the inspection.
- A SG inspection results report submitted within 12 months after the inspection.

- Reports required pursuant to 10 CFR 50.73.

The proposed changes to Technical Specifications replace the 15-day and the SG inspection reports with one report required within 180 days. The proposed report also contains more information than the current SG inspection report. This provision expands the report to provide more substantive information and will be sent earlier (180 days versus 12 months). This allows the NRC to focus its attention on the more significant conditions.

The guidance in NUREG-1022, Rev. 2, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," identifies serious SG tube degradation as an example of an event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded. Steam generator tube degradation is considered serious if the tubing fails to meet the structural integrity or accident induced leakage performance criteria. Serious SG tube degradation would be reportable in accordance with 10 CFR 50.72 (b) (3) (ii) (A) and 50.73 (a) (2) (ii) (A) requiring NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence.

The proposed reporting requirements are an improvement as compared to those required by the current technical specifications. The proposed reporting requirements are more useful in identifying the degradation mechanisms and determining their effects. In the unlikely event that a performance criterion is not met, NEI 97-06 (Ref. 1) directs the licensee to submit additional information on the root cause of the condition and the basis for the next operating cycle.

The changes to the reporting requirements are performance based. The new requirements remove the burden of unnecessary reports from both the NRC and the licensee, while ensuring that critical information related to problems and significant tube degradation is reported more completely and, when required, more expeditiously than under the current technical specifications.

Section 14: SG Terminology

The proposed Steam Generator Tube Integrity specification Bases explain a number of terms that are important to the function of a Steam Generator Program. The Technical Specification Bases are controlled by the Technical Specification Bases Control Program, which appears in the Administrative Technical Specifications. Changes are proposed to the TS Definitions Section terms "IDENTIFIED LEAKAGE" and "PRESSURE BOUNDARY LEAKAGE".

The terms are described below.

1. Accident induced leakage rate means the primary-to-secondary leakage rate occurring during postulated accidents other than a steam generator tube rupture. This includes the primary-to-secondary leakage rate existing immediately prior to the accident plus additional primary-to-secondary leakage induced during the accident.

Primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a limiting design basis accident. The potential primary-to-secondary leak rate during postulated design basis accidents must not cause radiological dose consequences in excess of the 10 CFR Part 50.67 guidelines for offsite doses, or the GDC 19 requirements for control room personnel, or other NRC approved licensing basis.

2. The Steam Generator Tube Integrity Bases define the term "burst" 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."

   Since a burst definition is required for condition monitoring, a definition that can be analytically defined and is capable of being assessed via in situ and laboratory testing is necessary. Furthermore, the definition must be consistent with ASME Code requirements, and apply to most forms of tube degradation.

   The definition developed for tube burst is consistent with the testimony of James Knight (Ref. 9), and the historical guidance of draft Regulatory Guide 1.121 (Ref. 10). The definition of burst per these documents is in relation to gross failure of the pressure boundary; e.g., "the degree of loading required to burst or collapse a tube wall is consistent with the design margins in Section III of the ASME B&PV Code (Ref. 11)." Burst, or gross failure, according to the Code would be interpreted as a catastrophic failure of the pressure boundary.

The above definition of burst was chosen for a number of reasons:

- The burst definition supports field application of the condition monitoring process. For example, verification of structural integrity during condition monitoring may be accomplished via in situ testing. Since these tests do not have the capability to provide an unlimited water supply, or the capability to maintain pressure under certain leakage scenarios, opening area may be more a function of fluid reservoir rather than tube strength. Additionally, in situ designs with bladders may not be reinforced. In certain cases, the bladder may rupture when tearing or extension of the

defect has not occurred. This condition may simply mean the opening of the flanks of the defect was sufficient to permit extrusion of the bladder, and that the actual, or true, burst pressure was not achieved during the test. The burst definition addresses this issue.

- The definition does not characterize local instability or "ligament pop-through", as a burst. The onset of ligament tearing need not coincide with the onset of a full burst. For example, an axial crack about 0.5" long with a uniform depth at 98% of the tube wall would be expected to fail the remaining ligament, (i.e., extend the crack tip in the radial direction) due to deformation during pressurization at a pressure below that required to cause extension at the tips in the axial direction. Thus, this would represent a leakage situation as opposed to a burst situation and a factor of safety of three against crack extension in the axial direction may still be demonstrated. Similar conditions have been observed for localized deep wear indications.

3. The Steam Generator Tube Integrity Bases define a SG tube as, "the entire length of the tube, including the tube wall and any repairs to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube."

   This definition ensures that all portions of SG tubes that are part of the RCPB, with the exception of the tube-to-tubesheet weld, are subject to Steam Generator Program requirements. The definition is also intended to exclude tube ends that cannot be NDE inspected by eddy-current. If there are concerns in the area of the tube end, they will be addressed by NDE techniques if possible or by using other methods if necessary.

   For the purposes of SG tube integrity inspection, any weld metal in the area of the tube end is not considered part of the tube. This is necessary since the acceptance requirements for tubing and weld metals are different.

4. The Steam Generator Tube Integrity Bases define the term "collapse" 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."

   In dealing with pure pressure loadings, burst is the only failure mechanism of interest. If bending loads are introduced in combination with pressure loading, the definition of failure must be broadened to encompass both burst and bending collapse. Which failure mode applies depends on the relative magnitude of the pressure and bending loads and also on the nature of any flaws that may be present in the tube. Guidance on assessing applicable failure modes is provided in the EPRI steam generator guidelines.

5. The Steam Generator Tube Integrity Bases define the term "significant" as used in the structural integrity performance criterion 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."

6. The Steam Generator Tube Integrity Bases describes how to determine whether thermal loads are primary or secondary loads. 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.

7. TS Definitions "IDENTIFIED LEAKAGE" and "PRESSURE BOUNDARY LEAKAGE" are revised to clarify that steam generator tube leakage to the secondary system is referred to as primary-to-secondary leakage. An editorial change is made to the definition of OPERATIONAL MODE – MODE to correct punctuation. These changes to TS Definitions are administrative in nature and have no impact on safety.

Conclusion

The proposed changes will provide greater assurance of SG tube integrity than that offered by the current technical specifications. The proposed requirements are performance based and provide the flexibility to adopt new technology as it matures. These changes are consistent with the guidance in NEI 97-06, "Steam Generator Program Guidelines," (Ref. 1).

Adopting the proposed changes will provide added assurance that SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97-06, "Steam Generator Program Guidelines." PSEG Nuclear, LLC (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed changes to Technical Specification 1.15, "Identified Leakage," TS 1.21, "Pressure Boundary Leakage," Salem Unit 2 TS 3/4.4.7.2, "Reactor Coolant System Operational Leakage," and the additions of Salem Unit 2 TS 3/4.4.6, "Steam Generator (SG) Tube Integrity," TS 6.8.4.i, "Steam Generator (SG) Program," and 6.9.1.10, "Steam Generator Tube Inspection Report," by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment" as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change requires a Steam Generator Program that includes performance criteria that will provide reasonable assurance that the steam generator (SG) tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cool down and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational leakage.

The structural integrity performance criterion is:

> *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.

The accident induced leakage performance criterion is:

The primary-to-secondary accident induced leakage rate for any design basis accidents, 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 gpm per SG.

The operational leakage performance criterion is:

The reactor coolant system operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day.

A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a SGTR event, a bounding primary-to-secondary leakage rate equal to the operational leakage rate limits in the licensing basis plus the leakage rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses assume that primary-to-secondary leakage for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident-induced stresses. The accident induced leakage criterion retained by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed as part of these TS changes identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the Steam Generator Program required by the proposed addition of TS 6.8.4.i. The program defined by NEI 97-06 includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I-131 in the primary coolant and the primary-to-secondary leakage rates resulting from an accident. Therefore, limits are included in the Salem TS for operational leakage and for DOSE EQUIVALENT I-131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary-to-secondary leak rate after the accident is 1 gallon per minute with no more than 500 gallons per day through any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I-131 are at the TS values before the accident.

The proposed change that allows SR 4.4.7.2.1.d to not be performed until 12 hours after establishment of steady state operation is consistent with NUREG 1431, "Standard Technical Specifications, Westinghouse Plants", and ensures the surveillance requirement is appropriate for the LCO.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TS and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current TS.

Therefore, the proposed changes do not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the probabilities or consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed performance based requirements are an improvement over the requirements imposed by the current TS.

Implementation of the proposed Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the Steam Generator Program will be an enhancement of SG tube performance. Primary-to-secondary leakage that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed changes do not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

The proposed change that allows SR 4.4.7.2.1.d to not be performed until 12 hours after establishment of steady state operation is consistent with NUREG 1431, "Standard Technical Specifications, Westinghouse Plants", and ensures the surveillance requirement is appropriate for the LCO.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the Steam Generator Program to manage SG tube inspection, assessment, repair and plugging. The requirements established by the Steam Generator Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TS.

The proposed change that allows SR 4.4.7.2.1.d to not be performed until 12 hours after establishment of steady state operation is consistent with NUREG 1431, "Standard Technical Specifications, Westinghouse Plants", and ensures the surveillance requirement is appropriate for the LCO.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed changes to the TS.

Based on the above, PSEG concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The regulatory requirements applicable to SG tube integrity are the following:

10 CFR 50.55a, Codes and Standards – Section (b), ASME Code – c) *Reactor coolant pressure boundary.* (1) Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.

The proposed change and the Steam Generator Program requirements that underlie it are in full compliance with the ASME Code. The proposed TS are more effective at ensuring tube integrity and, therefore, compliance with the ASME Code, than the current TS as described in Section 4.0 (Technical Analysis).

10 CFR 50.65 Maintenance Rule – Each holder of a license to operate a nuclear power plant under §§50.21(b) or 50.22 shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components, as defined in paragraph (b), are capable of fulfilling their intended functions. Such goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken.

Under the Maintenance Rule, PSEG has classified SGs as risk significant components because they are relied on to remain functional during and after design basis events. The performance criteria included in the proposed TS are used to demonstrate that the condition of the SG "is being effectively controlled through the performance of appropriate preventive maintenance" (Maintenance Rule §(a)(2)). If the performance criteria are not met, a root cause determination of appropriate depth is done and the results evaluated to determine if goals should be established per §(a)(1) of the Maintenance Rule.

NEI 97-06, "Steam Generator Program Guidelines," and its referenced EPRI guidelines define a SG program that provides the appropriate preventive maintenance that meets the intent of the Maintenance Rule. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (Reference 12) offers guidance for implementing the Maintenance Rule should a licensee elect to incorporate additional monitoring goals beyond the scope of those documented in NEI 97-06.

10 CFR 50, Appendix A, GDC 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, or rapidly propagating failure, and of gross rupture.

There are no changes to the SG design that impact this general design criterion. The evaluation performed in Section 4.0 (Technical Analysis) concludes that the proposed change will continue to comply with this regulatory requirement.

10 CFR 50, Appendix A, GDC 30 – *Quality of reactor coolant pressure boundary.* Components that 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.

There are no changes to the SG design that impact this general design criterion. The evaluation performed in Section 4.0 (Technical Analysis) concludes that the proposed change will continue to comply with this regulatory requirement.

10 CFR 50, Appendix A, GDC 32 – *Inspection of reactor coolant pressure boundary.* Components which are part of the reactor coolant pressure boundary shall be designed to (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.

There are no changes to the SG design that impact this general design criterion. The evaluation performed in Section 4.0 (Technical Analysis) concludes that the proposed change will continue to comply with this regulatory requirement.

General Design Criteria (GDC) 14, 30, and 32 of 10 CFR Part 50, Appendix A, define requirements for the reactor coolant pressure boundary with respect to structural and leakage integrity. Steam

generator tubing and tube repairs constitute a major fraction of the reactor coolant pressure boundary surface area. Steam generator tubing and associated repair techniques and components, such as plugs and sleeves, must be capable of maintaining reactor coolant inventory and pressure. The Steam Generator Program required by the proposed TS establishes performance criteria, repair criteria, repair methods, inspection intervals and the methods necessary to meet them. These requirements provide reasonable assurance that tube integrity will be met in the interval between SG inspections.

The proposed changes provide requirements that are more effective in detecting SG degradation and prescribing corrective actions. The proposed changes result in added assurance of the function and integrity of SG tubes.

10 CFR 50, Appendix B – Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants. "Quality assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service.

The SG Program required by the proposed TS establishes performance criteria, repair criteria, repair methods, inspection intervals and the methods necessary to meet them. These requirements provide reasonable assurance that the SG will perform satisfactorily in service and meet this regulatory requirement.

Therefore, based on the considerations discussed above:

1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;

2) Such activities will be conducted in compliance with the Commission's regulations; and

3) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6. ENVIRONMENTAL CONSIDERATION

PSEG has determined the proposed amendment relates to changes in a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or relates to changes in an inspection or a surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a

significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment of the proposed change is not required.

7. REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. EPRI, "Steam Generator Examination Guideline."
3. EPRI, "Steam Generator Integrity Assessment Guideline."
4. EPRI, "Steam Generator In-situ Pressure Test Guideline."
5. EPRI, "PWR Primary-to-Secondary Leak Guideline."
6. EPRI, "Primary Water Chemistry Guideline."
7. EPRI, "Secondary Water Chemistry Guideline."
8. EPRI Report R-5515-00-2, "Experience of US and Foreign PWR Steam Generators with Alloy 600TT and Alloy 690TT Tubes and Sleeves," June 5, 2002.
9. Testimony of James Knight Before the Atomic Safety and Licensing Board, Docket Nos. 50-282 and 50-306, January 1975.
10. Draft Regulatory Guide 1.121, "Bases for Plugging Degraded Steam Generator Tubes," August 1976.
11. ASME B&PV Code, Section III, Rules for Construction of Nuclear Facility Components.
12. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3.
13. S. C. Collins memo to W. D. Travers, "Steam Generator Action Plan Revision to Address Differing Professional Opinion on Steam Generator Tube Integrity," May 11, 2001.

14. The NRC has approved a similar license amendment for Farley Nuclear Plant, Units 1 and 2 – Amendments 163 and 156 dated September 10, 2004.

15. The NRC has approved a similar license amendment for South Texas Project, Units 1 and 2 – Amendments 164 and 154 dated November 24, 2004.

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

<u>Salem Unit 2 Affected Page List</u>

Index Page V
Index Page XII

The following Technical Specifications for Salem Unit 2 Facility Operating License DPR-75 are affected by this change request:

Technical Specification	Page
1.15, "Identified Leakage"	1-4
1.19, "Operational MODE – MODE"	1-4
1.21, "Pressure Boundary Leakage"	1-5
3/4.4.6, "Steam Generator (SG) Tube Integrity"	3/4 4-9 through 3/4 4-15a
3/4.4.7.2, "Operational Leakage"	3/4 4-17 and 3/4 4-18
6.8.4.g.9,"Radioactive Effluent Controls Program"	6-19b
6.8.4.i, "Steam Generator (SG) Program"	6-19b
6.9.1.10, "Steam Generator Tube Inspection Report"	6-24a

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION | PAGE

INDEX

BASES

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=

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system (primary-to-secondary leakage).

MEMBER(S) OF THE PUBLIC

1.16 MEMBER(S) OF THE PUBLIC shall be all those persons who are not occupationally associated with the plant. This category does not include employees of PSE&G, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent controls and Radiological Environmental Monitoring programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8 respectively.

OPERABLE - OPERABILITY

1.18 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s), and when all necessary attendant instrumentation, controls, normal or emergency electrical power source, cooling and seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified safety function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

DEFINITIONS

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the Updated FSAR, 2) authorized under the provisions of 10CFR50.59, or 3) otherwise by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary-to-secondary ~~steam generator tube~~ leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt.

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

LIMITING CONDITION FOR OPERATION

3.4.6 **SG tube integrity** ~~Each steam generator~~ shall be ~~OPERABLE~~ **maintained and all SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.**

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

INSERT 1

~~With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.6.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.~~

~~4.4.6.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.~~

~~4.4.6.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.6.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.6.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:~~

~~a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.~~

~~b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:~~

~~1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%), and~~

INSERT 1

a.* With one or more SG tubes satisfying the tube repair 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 repair 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.

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.6.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

4.4.6.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. For Fuel Cycle 10 only, the inspection interval shall begin at criticality. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.6.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.7.2.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

4.4.6.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.6.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.6.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which the inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be evaluated for reportability pursuant to 10CFR50.72 and 10CFR50.73. The evaluation shall be documented, and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to 10CFR50.72 and10CFR50.73, as applicable.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to 10CFR50.72 and 10CFR50.73, as applicable.	N/A	N/A

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

STEAM GENERATOR SURVEILLANCE PERIOD AMENDMENT
FOR SALEM NUCLEAR GENERATING STATION UNIT 2
FUEL CYCLE 2R10

Salem Unit 2 was removed from service in June of 1995 for a comprehensive review of plant methods and policies. In May of 1996, a 100% bobbin coil and additional *specialty examinations inspection of the Salem Unit 2 steam generators was* completed. Permission to restart Unit 2 was given by the NRC in June of 1997 and Mode 2 first achieved on August 17, 1997. After the May 1996 inspection, Unit 2 steam generators were placed in lay-up, using EPRI guidelines, to protect the steam generators from deterioration. PSE&G has a high level of confidence that corrosion growth and new corrosion initiation during the time of lay-up were essentially halted, and the condition of the steam generators has not changed since the May 1996 inspection.

Thus, in order to avoid an unnecessary mid-cycle steam generator inspection forced outage, Technical Specification 3/4.4.6 is hereby amended such that the next steam generator inspection will be required to be performed within 24 months of Mode 2(this would be by August 17, 1999), or during the next scheduled refueling outage, whichever is first for Unit 2 fuel cycle 10. *Subsequent steam generator inspections* will be scheduled accordingly.

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~~ 1500 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.

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

c*. ~~NOT USED~~ Verifying primary-to-secondary leakage is ≤ 150 gallons per day through any one steam generator at least once per 72 hours during steady state operation,

d*. Performance of a Reactor Coolant System water inventory balance** at least once per 72 hours. The water inventory balance shall be performed with the plant at steady state conditions. The provisions of specification 4.0.4 are not applicable for entry into Mode 4, and

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.7.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to returning the valve to service following maintenance repair or replacement work on the valve.

d. For the Residual Heat Removal and Safety Injection Systems hot and cold leg injection valves and accumulator valves listed in Table 3.4-1 the testing will be done within 24 hours following valve actuation due to automatic or manual action or flow through the valve. For all other systems testing will be done once per refueling.

The provisions of specification 4.0.4 are not applicable for entry into MODE 3 or 4.

Not required to be completed until 12 hours after establishment of steady state operation.

*****Not applicable to primary-to-secondary leakage.***

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: PSEG submitted License Change Request LCR S05-07, dated September 21, 2005, requesting approval of the W* methodology. If LCR S05-07 is approved, then an equivalent description of the W* methodology will be included here, including any allowed accident induced leakage rates for specific types of degradation at specific locations associated with the W* methodology, as discussed in TSTF-449, Revision 4.

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

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by PSEG in this document. Any other statements in this submittal are provided for information only purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. Paul Duke at (856) 339-1466.

Regulatory Commitment	Due Date/Event
PSEG will implement the Steam Generator Program in accordance with NEI 97-06, "Steam Generator Program Guidelines"	Concurrent with implementation of the amendment

PROPOSED CHANGES TO TS BASES PAGES

The following Technical Specifications Bases for Salem Unit 2, Facility Operating License No. DPR-75, are affected by this change request:

Salem Unit 2

Technical Specification	Page
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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.

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3/4.4.6 STEAM GENERATORS (SG) **TUBE INTEGRITY** (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.~~

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. 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.

3/4.4.7.2 OPERATIONAL LEAKAGE (Continued)

~~The total steam generator tube leakage limit of 1 GPM for all steam generators (but not more than 500 gpd for any steam generator) ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.~~

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

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Primary to Secondary Leakage Through Any One SG

The primary-to-secondary leakage rate limit applies to leakage through any one Steam Generator. The limit of 150 gallons per day per steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines. The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

Actions

Unidentified leakage or identified leakage in excess of the LCO limits must be reduced to within limits within 4 hours. This action time allows time to verify leakage rates and either identify unidentified leakage or reduce leakage to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the reactor coolant pressure boundary (RCPB). If any pressure boundary leakage exists, or primary-to-secondary leakage is not within limit, or if unidentified or identified leakage cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. The reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary. The action times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Surveillances

Verifying RCS leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. Unidentified leakage and identified leakage are determined by performance of an RCS water inventory balance. The RCS water inventory must be met with the reactor at steady state operating conditions. The surveillance is modified by a Note that the surveillance is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The 72 hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If SR 4.4.7.2.1.c is not met, compliance with LCO 3.4.6, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines). If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one Steam Generator. The Surveillance is modified by a Note that states that the surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary-to-secondary leakage determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The Surveillance Frequency of 72 hours is a reasonable interval to trend primary-to-secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines).