VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

10CFR50.92

January 19, 2007

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Serial No.	06-1021
NLOS/GDM	R0
Docket Nos.	50-280, 281
License Nos.	DPR-32, 37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED LICENSE AMENDMENT REQUEST
CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS
TECHNICAL SPECIFICATION IMPROVEMENT
REGARDING STEAM GENERATOR TUBE INTEGRITY
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

In a letter dated May 26, 2006 (Serial No. 06-024), Virginia Electric and Power Company (Dominion) requested amendments in the form of changes to the Technical Specifications (TS) to Facility Operating License Numbers DPR-32 and DPR-37 for Surry Power Station Units 1 and 2, respectively. The proposed amendment would revise the TS requirements related to Reactor Coolant System leakage definitions and requirements and steam generator tube integrity consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity" (TSTF-449, Rev. 4). The availability of this TS improvement was announced in the Federal Register on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process (CLIIP). In a letter dated November 6, 2006, the NRC staff requested additional information to facilitate their review of the proposed license amendment. Dominion's response to the staff's request is included in Attachment 1. As noted in Attachment 1, revisions to the previously submitted license amendment request were required to address the NRC's comments. Consequently, the previously submitted license amendment request has been revised to resolve the NRC comments. These changes are indicated by a single revision bar in the right hand margin in Attachment 2, *Description and Assessment*, and by double revision bars in the right hand margin in Attachment 3, *Proposed Technical Specifications Pages (Mark-Up)*. Attachment 4, *Proposed Technical Specifications Pages (Typed)*, incorporates the indicated revisions.

We have evaluated the changes to the proposed amendment and have determined that the previously provided no significant hazards consideration as defined in 10 CFR 50.92 and the environmental assessment are not affected. Although unchanged, the original evaluations are included in Attachment 2 for completeness. The revised license amendment request has been reviewed and approved by the Station Nuclear Safety and Operating Committee.

Dominion's previous request for approval of the license amendments by March 31, 2007 with a 180-day implementation period is unchanged.

If you have any questions or require additional information, please contact Mr. Gary D. Miller at (804) 273-2771.

Sincerely,

G. T. Bischof
Vice President – Nuclear Engineering

Attachments

1. Resolution of NRC Comments on License Amendment Request Dated May 26, 2006 (Serial No. 06-024)
2. Description and Assessment
3. Proposed Technical Specifications Pages (Mark-Up)
4. Proposed Technical Specifications Pages (Typed)

Commitments made in this letter: None

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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Gerald T. Bischof, who is Vice President – Nuclear Engineering, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 19th day of January, 2007.

My Commission Expires: August 31, 2008.

Margaret B. Bennett
Notary Public

(SEAL)

ATTACHMENT 1

Resolution of NRC Comments on License Amendment Request Dated May 26, 2006 (Serial No. 06-024)

**Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2**

Resolution of NRC Comments on License Amendment Request Dated May 26, 2006 (Serial No. 06-024)

Surry Power Station Units 1 And 2

Virginia Electric and Power Company (Dominion) submitted a License Amendment Request (LAR) to the NRC in a letter dated May 26, 2006 (Serial No. 06-024). The proposed amendment would revise the Technical Specifications (TS) requirements related to Reactor Coolant System leakage definitions and requirements and steam generator tube integrity. The change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity" (TSTF-449, Rev. 4). In a letter dated November 6, 2006, the NRC requested additional information to support their review of the LAR. Dominion's response to the NRC request for additional information is provided below.

1. *Please discuss your plans to make the language in proposed Technical Specification (TS) Sections 3.1.C.2.b and 3.1.H.3 consistent with the language used in the associated bases sections.*

   Response - The TS Bases sections associated with proposed TS 3.1.C.2.b and 3.1.H.3 have been revised to be consistent with the language used in their associated TS. Specifically, the term "...COLD SHUTDOWN within 36 hours." has been changed to "...COLD SHUTDOWN within the following 30 hours."

2. *In proposed Bases Sections 3.1.C.2.b and 3.1.C.3, and Surveillance Requirement (SR) 4.13.A, the term "unit" is used, however, in proposed Bases Section 3.1.H.3 the term "plant" is used. Please discuss your plans to modify the bases and SRs to be consistent by using either "unit" or "plant."*

   Response - The phrase "plant conditions" has been changed to "unit conditions" in TS Bases Section 3.1.H.3, and the term "plant operation" was changed to "unit operation" in TS Bases Section 3.1.H.2.a and b for TS consistency. In addition, the term "plant life" was changed to "unit life" in the BACKGROUND section of the TS 3.1.C Bases.

3. *You made several changes to Bases Sections 3.1.C and 4.13 that go beyond TSTF-449. Please confirm that all of the proposed changes are consistent with your current Nuclear Regulatory Commission (NRC) approved design and licensing basis. If they are not consistent, please provide the technical justification or discuss your plans to remove them. In addition, discuss why the statement concerning General Design Criterion 30 and Regulatory Guide 1.45 was not included in the proposed bases for TS Section 3.1.C.*

   Response - The portions of the TS Bases Sections 3.1.C and 4.13 wording that are different than TSTF-449 reflect the current Surry design and licensing bases, which are based on the full implementation of the alternative source term (AST) for Surry

Power Station Units 1 and 2. Dominion implemented the AST as the design basis for Surry by License Amendments 230/230, respectively, issued by the NRC on March 8, 2002. The approval was based on reanalysis of the Loss of Coolant Accident and the Fuel Handling Accident. Subsequently, the Locked Rotor, Main Steam Line Break, and Steam Generator Tube Rupture accidents were also reanalyzed with the AST methodology as a follow on to the previously approved AST analyses in accordance with 10 CFR 50.59. The safety analysis for design basis accidents is also discussed in Chapter 14 of the Surry Updated Final Safety Analysis Report (UFSAR). The Surry TS Bases revisions are consistent with the UFSAR discussion. The wording used in the Surry TS Bases sections is also similar to that used in Dominion's North Anna license amendment request, which implemented TSTF-449, Rev. 4, as North Anna has also implemented the AST. The North Anna license amendment request was approved by the NRC in Amendments 248/228 for Units 1 and 2 dated October 16, 2006.

In addition, the statement in TSTF-449 concerning General Design Criterion (GDC) 30 and Regulatory Guide (RG) 1.45 was not included in the proposed Bases for TS Section 3.1.C because Surry Units 1 and 2 were licensed prior to the issue of these documents and is therefore not committed to them. During the initial plant licensing of Surry Units 1 and 2, it was demonstrated that the design of the reactor coolant pressure boundary met the regulatory requirements in place at that time. The GDC included in Appendix A to 10 CFR Part 50 did not become effective until May 21, 1971. However, the Construction Permits for Surry Units 1 and 2 were issued prior to May 21, 1971; consequently, these units were not subject to GDC requirements. (Reference SECY-92-223 dated September 18, 1992.) Nevertheless, Surry meets the GDC 30 requirements. In addition, Surry received its operating license in 1972, prior to the issue of RG 1.45 in May 1973, and consequently did not commit to the RG requirements. As a result, it was determined that it would be inappropriate to include the statement concerning GDC 30 and RG 1.45 in the proposed TS 3.1.C Bases.

4. *In several TSs (e.g., TS Section 3.1.C, Applicability Section of TS Bases Sections 3.1.C, 3.1.H, 3.1.H.2.b, Bases Sections 3.1.H.2.a and b, 4.19.B), terminology such as "whenever REACTOR OPERATION exceeds COLD SHUTDOWN conditions" is used. This terminology is not clear since both refueling shutdown and intermediate shutdown conditions "exceed" cold shutdown conditions (depending on your perspective). Please discuss your plans to more clearly specify the requirement.*

   For example, in proposed TS Section 3.1.C: "The following specifications are applicable to RCS [reactor coolant system] operational LEAKAGE during the following REACTOR OPERATION conditions: INTERMEDIATE SHUTDOWN, HOT SHUTDOWN, REACTOR CRITICAL, and POWER OPERATION." Another example would be: "The following specifications are applicable to RCS operational LEAKAGE whenever Tavg (average temperature) exceeds 200°F (200 degrees Fahrenheit)."

   In proposed TS Section 3.1.H.2.b, it may be acceptable to replace "exceeding" with "exiting" consistent with the wording in proposed TS Section 6.6.A.3.

Response - The following changes have been made to the proposed TS sections to include less confusing REACTOR OPERATION condition terminology to resolve the NRC's noted concern:

- TS 3.1.C Applicability

  The phrase "...RCS operational LEAKAGE whenever REACTOR OPERATION exceeds COLD SHUTDOWN conditions."

  has been changed to

  "...RCS operational LEAKAGE whenever Tavg (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit)."

- TS Bases Section 3.1.C - APPLICABILITY

  The phrase "In operating modes above COLD SHUTDOWN,..."

  has been changed to

  "In REACTOR OPERATION conditions where Tavg exceeds 200°F,..."

- TS 3.1.H Applicability

  The phrase "...whenever REACTOR OPERATION exceeds COLD SHUTDOWN conditions."

  has been changed to

  "...whenever Tavg (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit)."

- TS 3.1.H.2.b

  The phrase "...exceeding COLD SHUTDOWN conditions..."

  has been changed to

  "...Tavg exceeding 200°F..."

- TS Bases Section 3.1.H - APPLICABILITY

  The phrase "...when operating above COLD SHUTDOWN conditions."

  has been changed to

  "...when Tavg exceeds 200°F."

- TS Bases Section 3.1.H.2.a and b

  The phrase "...exceeding COLD SHUTDOWN conditions..."

  has been changed to

  "...Tavg exceeding 200°F..."

- TS 4.19.B

  The phrase "...exceeding COLD SHUTDOWN conditions..."

  has been changed to

  "...Tavg exceeding 200°F..."

- TS Bases Section SR 4.19.B

  The phrase "...exceeding COLD SHUTDOWN conditions..."

  has been changed to

  "...Tavg exceeding 200°F..."

- TS 6.6.A.3

  The phrase "...after exiting COLD SHUTDOWN conditions..."

  has been changed to

  "...after Tavg exceeds 200°F..."

5. *Please provide justification for removing "...to be met until the next refueling outage..." from proposed Bases Sections 3.1.H.2.a and b or alternatively discuss your plans to modify these bases sections to state "...to be met until the next refueling outage or SG tube inspection."*

   Response - The phrase *"to be met until the next refueling outage..."* has been incorporated into the proposed TS Bases Section 3.1.H.2.a and b.

6. *In the second paragraph of proposed SR 4.13.A, there appears to be a typographical error, "met" should be "performed." Please discuss your plans to correct this typographical error.*

   Response - The word "met" has been changed to "performed" in the second paragraph of the proposed TS Bases Section SR 4.13.A.

7. *You deleted the existing Bases for proposed TS Section 3.1.C which describes various leak detection instruments. Please discuss whether this information is captured elsewhere in your TSs. The NRC staff notes that in the Bases for TS Section 4.13 you did not propose to incorporate the following sentence from the standard TSs: "These leakage detections systems are specified in LCO (Limiting Condition for Operation) 3.4.15, "RCS Leakage Detection Instrumentation." If the information concerning the leakage detection instruments is not captured elsewhere, please discuss your plans to leave it in TS Section 3.1.C and to incorporate a sentence into the Bases for TS Section 4.13 indicating that the leakage detection instruments are specified in the Bases of TS Section 3.1.*

   Response - The previously deleted information concerning RCS leakage detection instruments will be retained in the TS Section 3.1.C Basis, and a sentence has been incorporated into the TS Bases Section SR 4.13.A indicating that the leakage detection instruments are specified in the Bases Section of TS Section 3.1.

8. *In proposed TS Section 3.1.H, it would appear that TS Section 3.1.H.2 would permit you to elect not to plug a tube provided the conditions in TS Section 3.1.H.2.a and TS Section 3.1.H.2.b were met. This is not consistent with the Technical Specification Task Force (TSTF)-449. In TSTF-449, the required actions are intended to apply only in the event that a tube that should have been plugged was inadvertently not plugged rather than electing not to plug a tube. Please discuss your plans to clarify your TSs in this regard. For example, terminology such as, "If the requirements of 3.1.H.1 are not met for one or more SG tubes, then perform the following:"*

   Response - Proposed TS 3.1.H.2 has been revised to include the following introductory phrase, "If the requirements of 3.1.H.1 are not met for one or more SG tubes, then perform the following:" to ensure that it is clear that TS Section 3.1.H.2 would *not* permit you to elect not to plug a tube provided the conditions in TS Sections 3.1.H.2.a and 3.1.H.2.b were met.

9. *In proposed TS Section 4.13.A, you proposed relaxing the surveillance frequency for verifying RCS operational leakage from daily to once every 72 hours. Please discuss your plans to modify this requirement to be consistent with your existing TSs since this change is not consistent with TSTF-449.*

   Response - The TS SR 4.13.A frequency for verifying that the operational LEAKAGE is within limits has been changed from "once every 72 hours" to "once every 24 hours." The associated change has also been made in the TS 4.13.A Bases section.

10. *In several sections of your proposed TSs, there are "Applicability" and "Objective" sections. Please discuss why these sections are not included in every proposed TS section.*

Response - "Applicability" and "Objective" sections have been added to TS 4.13. Objective sections were not added to TS 3.1.C and 3.1.H since they are subsections of TS 3.1, Reactor Coolant System, which currently includes an Objective section that is applicable to its subsections. However, even though TS 3.1 also includes an Applicability section, separate Applicability sections were added to TS 3.1.C and 3.1.H to provide greater specificity regarding the conditions for which these two TS apply.

ATTACHMENT 2

DESCRIPTION AND ASSESSMENT

**Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2**

DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

The proposed license amendment revises the requirements in the Surry Power Station Units 1and 2 Technical Specifications (TS) related to steam generator tube integrity. The changes are consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4. The availability of this TS improvement was announced in the Federal Register on May 6, 2005 as part of the consolidated line item improvement process (CLIIP).

2.0 DESCRIPTION OF PROPOSED AMENDMENT

Consistent with the NRC-approved Revision 4 of TSTF-449, the proposed TS changes include the following:

- Revise TS Table of Contents
- Add new TS 1.X – Definition of LEAKAGE
- Revise TS 3.1.C, "Leakage"
- Add new TS 3.1.H, "Steam Generator (SG) Tube Integrity"
- Revise TS Table 4.1-2A – Delete Item 10 and revise a reference in Item 19
- Add new TS 4.13 Surveillance, "RCS Operational Leakage"
- Replace existing TS 4.19, "Steam Generator Inservice Inspection" with new TS 4.19, "Steam Generator (SG) Tube Integrity"
- Add new TS 6.4.Q, "Steam Generator (SG) Program"
- Add new TS 6.6.A.3, "Steam Generator Tube Inspection Report"

Proposed revisions to the TS Bases are also included in this application. As discussed in the NRC's model safety evaluation, adoption of the revised TS Bases associated with TSTF-449, Revision 4, is an integral part of implementing this TS improvement. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program.

3.0 BACKGROUND

The background for this application is adequately addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The applicable regulatory requirements and guidance associated with this application are adequately addressed by the NRC Notice of Availability published on May 6, 2005

(70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

5.0 TECHNICAL ANALYSIS

Dominion has reviewed the safety evaluation (SE) published on March 2, 2005 (70 FR 10298) as part of the CLIIP Notice for Comment. This included the NRC staff's SE, the supporting information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. Dominion has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to Surry Power Station Units 1 and 2 and justify this amendment for the incorporation of the changes to the Surry Units 1 and 2 TS.

6.0 REGULATORY ANALYSIS

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

6.1 Verification and Commitments

The following information is provided to support the NRC staff's review of this amendment application:

Plant Name, Unit No.	Surry Power Station (SPS) Units 1 and 2	
Steam Generator Model(s)	Westinghouse Model 51-F; 3-Loop	
Effective Full Power Years (EFPY) of service for currently installed SGs	SPS 1 19.5 EFPY at last inspection in April 2006 SPS 2 20.6 EFPY at last inspection in October 2006	
Tubing Material	Alloy 600TT	
Number of tubes per SG	3342 tubes/SG	
Number and percentage of tubes plugged in each SG	SPS 1 SG A – 29 (0.87%) SG B – 21 (0.63%) SG C – 20 (0.60%)	SPS 2 SG A – 20 (0.60%) SG B – 10 (0.30%) SG C – 25 (0.75%)
Number of tubes repaired in each SG	None	

Degradation mechanism(s) identified	SPS 1 • Outer diameter (OD) wear at anti-vibration bar intersections • OD wear due to transient foreign objects • OD wear caused by sludge lance equipment used during outage SPS 2 • Same as SPS 1 above • Inactive RCS Cold Leg pit indications
Current primary to secondary leakage limits	Limit — TS / Admin. Control Per SG: 500 gpd / ≥75 gpd for ≥1 hr Total: 1 gpm 1 gpm total SG leakage at room temperature (≅70°F) and normal atmospheric pressure (14.7 psi)
Approved Alternate Tube Repair Criteria (ARC)	None
Approved SG Tube Repair Methods	None
Performance criteria for accident leakage	1 gpm total SG leakage at room temperature (≅70°F) and normal atmospheric pressure (14.7 psi)

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

7.1 Incorporation of TSTF-449, Revision 4

Dominion has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIIP. Dominion has concluded that the proposed determination presented in the notice is applicable to Surry Power Station Units 1 and 2 and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91 (a).

7.2 Changes to Address Improved Technical Specifications Format

Dominion is proposing minor variations and/or deviations from the TS changes described in TSTF-449, Revision 4, to provide consistent terminology and format within Surry's custom TS, as well as consistency with Surry's design and licensing bases. For example, Surry TS do not use the NUREG-1431 Improved Standard TS (ITS) defined terms, such as, Condition, Action and Frequency or their associated Table format, as the Surry TS typically use a more narrative format. However, the intent of the CLIIP wording has been maintained in the proposed TS change and has been used verbatim to the extent possible. Also, Surry TS do not currently include a definition for LEAKAGE; consequently, the ITS definition for LEAKAGE, as modified by the CLIIP, is being incorporated into Surry's TS in this license amendment request. In addition, the Surry TS format separates Limiting Conditions for Operation (LCOs) and Action Statements (ASs) from Surveillance Requirements (SRs) by placing them in different TS sections (Sections 3 and 4, respectively). Also, Surry TS do not use the ITS MODE terminology convention for reactor operating conditions. Surry TS use specific definitions for each operating condition instead, e.g., POWER OPERATION, HOT SHUTDOWN, INTERMEDIATE SHUTDOWN, REACTOR CRITICAL, COLD SHUTDOWN and REFUELING SHUTDOWN. While not identical, the reactor operating MODES specified in the CLIIP are generally consistent with the defined REACTOR OPERATION conditions used in the Surry license amendment request. The corresponding REACTOR OPERATION conditions were used instead of the specified ITS MODES where appropriate. Also, the Applicability of several TS sections was described using words similar to "... whenever Tavg exceeds 200°F..." in lieu of MODES or listing out numerous REACTOR OPERATION conditions, and the SR frequency for certain TS SR was written using words similar to "prior to exceeding 200°F".

The minor variations and/or deviations from the specific wording/format provided in the CLIIP do not change the meaning, intent or applicability of the CLIIP. A table summarizing the minor variations and/or deviations from the TS changes described in TSTF-449, Revision 4, is provided in Attachment A.

A significant hazards consideration determination has been performed for the TS changes associated with terminology and format differences between the Surry TS and the ITS to facilitate incorporation of the changes described in TSTF-449, Revision 4. Dominion has concluded that the proposed changes do not involve a significant hazards consideration because the changes do not:

1. <u>Involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

The proposed changes involve adding a new definition of RCS leakage and rewording certain Technical Specifications (TS) for consistency with NUREG-1431, Revision 3. These changes do not involve any physical plant modifications or changes in plant operation; consequently, no technical changes are being made to the existing TS. As such, these changes are administrative in nature and do not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes involve adding a new definition of RCS leakage and rewording certain Technical Specifications (TS) for consistency with NUREG-1431, Revision 3. These administrative changes do not involve physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed changes involve adding a new definition of RCS leakage and rewording certain Technical Specifications (TS) for consistency with NUREG-1431, Revision 3. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

8.0 ENVIRONMENTAL EVALUATION

Dominion has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298) as part of the CLIIP. Dominion has concluded that the staff's findings presented in that evaluation are applicable to Surry Power Station Units 1 and 2, and the evaluation is hereby incorporated by reference for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. Dominion is not proposing significant variations and/or deviations from the TS changes described in TSTF-449, Revision 4, or the NRC staff's model SE published on March 2, 2005 (70 FR 10298). However, since Surry's TS are custom TS, as opposed to ITS, the changes proposed by the CLIIP have been implemented such that they are consistent with the existing Surry TS format requirements. Consequently, these minor variations and/or deviations do not conflict with the applicability of the NRC's model safety evaluation to the proposed changes.

10.0 REFERENCES

Federal Register Notices:

Notice for Comment published on March 2, 2005 (70 CFR 10298)

Notice of Availability published on May 6, 2005 (70 FR 24126)

Attachment A

Variations from the TS Changes Described in TSTF-449, Revision 4 for Surry Power Station Units 1 and 2 TS			
ITS Section	**CLIIP/TSTF 449 TS Revision**	**Surry TS Section**	**Inclusion of Proposed Change into Surry Custom TS**
TOC	None	TOC	The Surry TS Table of Contents has been revised to reflect the new/revised TS sections as appropriate.
1.1 Definitions	Revises the LEAKAGE definition in TS to include the parenthetical phrase "(primary to secondary LEAKAGE)" in item a.3 and item c and deletes the term "(SG)" in both items.	1.X	Surry TS do not currently include a definition for LEAKAGE. The proposed change incorporates a definition for LEAKAGE into the Surry TS that is identical to the ITS Definition including the proposed TSTF change.
B3.4.4	Deletes the term "in accordance with the Steam Generator Tube Surveillance Program." in the RCS Loops – MODES 1 and 2 LCO Bases section.	N/A	SPS TS do not include this TS/wording; therefore, no change is required.
B3.4.5	Deletes the term "in accordance with the Steam Generator Tube Surveillance Program." in the RCS Loops – MODE 3 LCO Bases section.	N/A	SPS TS do not include this TS/ wording; therefore, no change is required.
B3.4.6	Deletes the term "in accordance with the Steam Generator Tube Surveillance Program." in the RCS Loops – MODE 4 LCO Bases section.	N/A	SPS TS do not include this TS/ wording; therefore, no change is required.
B3.4.7	Deletes the term "in accordance with the Steam Generator Tube Surveillance Program." in the RCS Loops – MODE 5, Loops Filled LCO Bases section.	N/A	SPS TS do not include this TS/ wording; therefore, no change is required.

Attachment A

Variations from the TS Changes Described in TSTF-449, Revision 4 for Surry Power Station Units 1 and 2 TS			
ITS Section	**CLIIP/TSTF 449 TS Revision**	**Surry TS Section**	**Inclusion of Proposed Change into Surry Custom TS**
LCO 3.4.13	Revises RCS Operational LEAKAGE for primary to secondary LEAKAGE to ≤ 150 gallons per day primary to secondary LEAKAGE through any one SG. Includes primary to secondary LEAKAGE in the CONDITIONS column of the LCO ACTIONS.	3.1.C.1 through 3.1.C.3	Surry TS do not use the ITS MODE terminology; therefore, the applicability of the TS has been stated as follows: *The following specifications are applicable to RCS operational LEAKAGE whenever Tavg (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit)*, to address operating conditions above COLD SHUTDOWN. Surry TS 3.1.C.2 through .6 LCOs and ACTIONS associated with RCS Operational LEAKAGE have been replaced with TS 3.1.C.1 through 3.1.C.3 specifications that are consistent with the revised ITS Section 3.4.13. Surry's TS format and reactor operating condition terminology is retained vs. the format and MODE terminology used in ITS. Specifically, MODE 3 is changed to HOT SHUTDOWN and MODE 5 is changed to COLD SHUTDOWN. The existing RCS LEAKAGE detection specification has been renumbered from 3.1.C.1 to 3.1.C.4 and the existing RCS pressure isolation valves' TS has been renumbered from 3.1.C.7 to 3.1.C.5 to accommodate the TSTF changes.
SR 3.4.13	Added new Note to Surveillance Requirement (SR) 3.4.13.1 indicating SR not applicable to primary to secondary LEAKAGE. Revised SR 3.4.13.2 to verify primary to secondary LEAKAGE every 72 hours. Added a new Note stating "Not required to be performed until 12 hours after establishment of steady state operation."	4.13	Surry TS do not use the ITS MODE terminology; therefore the applicability of the TS has been stated as follows: *The following specifications are applicable to RCS operational LEAKAGE whenever Tavg (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit)*, to address operating conditions above COLD SHUTDOWN. New SPS TS 4.13.A includes the revised ITS SR 3.4.13.1 for verification that RCS operational LEAKAGE is within limits by performance of an RCS water inventory balance with the exception that the frequency has been specified as once every 24 hours instead of once every 72 hours. The 24 hour frequency is consistent with the

Attachment A

Variations from the TS Changes Described in TSTF-449, Revision 4 for Surry Power Station Units 1 and 2 TS			
ITS Section	**CLIIP/TSTF 449 TS Revision**	**Surry TS Section**	**Inclusion of Proposed Change into Surry Custom TS**
			current Surry licensing basis. TS 4.13.A also includes the associated ITS Notes as revised by the TSTF with the exception that the word "performed" in Note 1 has been changed to "completed" to preclude confusion regarding when the verification is to be performed/completed. New SPS TS 4.13.B includes the revised ITS SR 3.4.13.2 for verification of SG tube integrity every 72 hours by verifying primary to secondary LEAKAGE is $\leq$150 gallons per day through any one SG. TS 4.13.B also includes the new Note with the exception that the word "performed" has been changed to "completed" to preclude confusion regarding when the verification is to be performed/ completed.
B3.4.13	Revise the Bases for the RCS Operational LEAKAGE TS to address TSTF-449, Rev. 4 changes.	3.1.C.1 and 4.13 Bases	The existing TS Basis for SPS TS 3.1.C is being replaced with Bases wording consistent with the ITS B3.4.13 Bases wording, as revised by TSTF-449, and Surry's alternate source term licensing basis, with the exception of the SRs Bases portion. The SRs Bases discussion section is included in the TS 4.13 Bases, which includes the RCS Operational LEAKAGE SRs. Consequently, the Background, Applicable Safety Analyses, Limiting Conditions for Operation, Applicability, Actions and References sections are included in the TS 3.1.C Basis, and the Surveillance Requirements and References (repeated) sections are included in the TS 4.13 Bases for consistency with SPS custom TS format (i.e., separate TS sections for LCOs/AS and their associated SRs). An additional UFSAR reference has also been added to both Bases sections.
LCO 3.4.20	New TS added for SG Tube Integrity.	3.1.H	New SPS TS 3.1.H, SG Tube Integrity, has been added and is consistent with ITS 3.4.20. (Note: SPS TS LCOs/ASs and SRs are contained in different TS sections.)

Attachment A

Variations from the TS Changes Described in TSTF-449, Revision 4 for Surry Power Station Units 1 and 2 TS			
ITS Section	**CLIIP/TSTF 449 TS Revision**	**Surry TS Section**	**Inclusion of Proposed Change into Surry Custom TS**
			Surry TS do not use the MODE terminology; therefore, the applicability of the TS has been stated as follows: *The following specifications are applicable to RCS operational LEAKAGE whenever Tavg (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit)*, to address operating conditions above COLD SHUTDOWN. The ITS Note, "Separate Condition entry is allowed for each SG tube," has been revised to read, "A separate TS action entry is allowed for each SG tube," for consistency with SPS custom TS terminology. The ITS COMPLETION TIME "prior to entering MODE 4" has been changed to "prior to Tavg exceeding 220°F" for consistency with SPS TS terminology.
SR 3.4.20	SG Tube Integrity Surveillance Requirements.	4.19	Existing SPS TS 4.19, Steam Generator Inservice Inspection, has been replaced in its entirety by new TS 4.19, SG Tube Integrity. The new TS 4.19 SRs are consistent with ITS TS 3.4.20 SRs. The ITS phrase "prior to entering MODE 4" has been changed to "prior to Tavg exceeding 200°F" for consistency with SPS TS terminology.
B3.4.20	New Bases for the new SG Tube Integrity TS in accordance with TSTF-449, Rev. 4.	3.1.H and 4.19 Bases	As discussed above, TS 3.4.20 is divided into two parts to address the SPS TS format. Specifically, the LCO/AS portion is included in TS 3.1.H, and the SRs are included in TS 4.19. Consequently, the associated Bases B3.4.20 has been divided between the two SPS TS sections accordingly. TS references to other ITS sections have been changed to match the applicable SPS TS sections. The Background, Applicable Safety Analyses, Limiting Conditions for Operation, Applicability, Actions and References sections are included with the TS 3.1.H Bases, and the Surveillance Requirements and References (repeated) are included in the TS 4.19 Bases.

Attachment A

Variations from the TS Changes Described in TSTF-449, Revision 4 for Surry Power Station Units 1 and 2 TS			
ITS Section	**CLIIP/TSTF 449 TS Revision**	**Surry TS Section**	**Inclusion of Proposed Change into Surry Custom TS**
			Also, the sentence in the TS 3.1.H Bases that states: "The accident analysis assumes that accident induced leakage does not exceed 1 gpm per SG." has been changed to read, "The accident analysis assumes that accident induced leakage does not exceed 1 gpm." The words "per SG" have been deleted to accurately reflect Surry accident analysis assumptions. Additional references (10 CFR 50.67 and RG 1.183) have been included and one reference deleted (10 CFR 100) in the TS Bases to reflect Surry's approved licensing and design bases regarding the dose consequences associated with design basis accidents.
5.5.9	New Steam Generator (SG) Program description/criteria.	6.4.Q	New TS 6.4.Q, Steam Generator Program, has been incorporated into SPS TS.
5.6.9	New Steam Generator Tube Inspection Report description/criteria.	6.6.A.3	New SPS TS 6.6.A.3, Steam Generator Tube Inspection Report, has been incorporated into SPS TS. The TS 6.6.A.3 text is identical to the revised ITS 5.6.9 text with the exception of the use of the term "after the initial entry into MODE 4," since Surry's TS do not use the MODE plant condition terminology. This phrase has been revised to "after Tavg exceeds 200°F" for consistency with SPS TS reactor operating condition terminology.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS PAGES (MARK-UP)

Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2

TECHNICAL SPECIFICATION
TABLE OF CONTENTS

TECHNICAL SPECIFICATION
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W. STAGGERED TEST BASIS

A staggered test basis shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

INSERT 1

Insert 1 (in TS Section 1.0 DEFINITIONS – New Item X)

X. LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank,

2. 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

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE, and

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

C. ~~Leakage~~ RCS Operational LEAKAGE

~~Specifications~~ {Relocate & renumber as TS 3.1.C.4}

1. Detected or suspected leakage from the Reactor Coolant System shall be investigated and evaluated. At least two means shall be available to detect reactor coolant system leakage. One of these means must depend on the detection of radionuclides in the containment.

2. ~~If the leakage rate, from other than controlled leakage sources, such as the Reactor Coolant Pump Controlled Leakage Seals, exceeds 1 gpm and the source of the leakage is not identified within four hours of leak detection, the reactor shall be brought to hot shutdown. If the source of leakage is not identified within an additional 48 hours, the reactor shall be brought to a cold shutdown condition.~~

3. ~~If the sources of leakage are identified and the results of the evaluations are that continued operation is safe, operation of the reactor with a total leakage, other than leakage from controlled sources, not exceeding 10 gpm shall be permitted except as specified in C.4 below.~~

4. ~~If it is determined that leakage exists through a non-isolable fault which has developed in a Reactor Coolant System component body, pipe well, vessel wall, or pipe weld, the reactor shall be brought to a cold shutdown condition and corrective action taken prior to resumption of unit operation.~~

5. ~~If the total leakage, other than leakage from controlled sources, exceeds 10 gpm the reactor shall be placed in the cold shutdown condition.~~

{Replace with INSERT 2}

Insert 2 (in TS Section 3.1.C – Replace existing TS 3.1.C.1 through .6)

Applicability

The following specifications are applicable to RCS operational LEAKAGE whenever Tavg (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Specifications

1. RCS operational LEAKAGE shall be limited to:

   a. No pressure boundary LEAKAGE,

   b. 1 gpm unidentified LEAKAGE,

   c. 10 gpm identified LEAKAGE, and

   d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

2. a. If RCS operational LEAKAGE is not within the limits of 3.1.C.1 for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE, reduce LEAKAGE to within the specified limits within 4 hours.

   b. If the LEAKAGE is not reduced to within the specified limits within 4 hours, the unit shall be brought to HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

3. If RCS pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within the limit specified in 3.1.C.1.d, the unit shall be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

Relocate TS 3.1.C.1 here as 3.1.C.4

~~6. If the primary-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System exceeds 1 gpm total and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System, reduce the leakage rate to within limits within 4 hours or be in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.~~

5. ~~7~~a. Prior to going critical all primary coolant system pressure isolation valves listed below shall be functional as a pressure isolation device, except as specified in 3.1.C.~~7~~ 5.b. Valve leakage shall not exceed the amounts indicated.

	Unit 1	Unit 2	Max. Allowable Leakage (see note (a) below)
Loop A, Cold Leg	1-SI-79, 1-SI-241	2-SI-79, 2-SI-241	≤5.0 gpm for each valve
Loop B, Cold Leg	1-SI-82, 1-SI-242	2-SI-82, 2-SI-242	
Loop C, Cold Leg	1-SI-85, 1-SI-243	2-SI-85, 2-SI-243	

b. If Specification 3.1.C.~~7~~ 5.a cannot be met, an orderly shutdown shall be initiated and the reactor shall be in ~~hot shutdown~~ within 6 hours and in ~~the cold shutdown condition~~ within the following 30 hours.

Notes

(a)
1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.
2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
4. Leakage rates greater than 5.0 gpm are considered unacceptable.

Replace with INSERT 3

Basis

Leakage from the Reactor Coolant System is collected in the containment or by other systems. These systems are the Main Steam System, Condensate and Feedwater System, the Gaseous and Liquid Waste Disposal Systems, the Component Cooling System, and the Chemical and Volume Control System.

Detection of leaks from the Reactor Coolant System is by one or more of the following:

1. An increased amount of makeup water required to maintain normal level in the pressurizer.
2. A high temperature alarm in the leakoff piping provided to collect reactor head flange leakage.
3. Containment sump water level indication.
4. Containment pressure, temperature, and humidity indication.

If there is significant radioactive contamination of the reactor coolant, the radiation monitoring system provides a sensitive indication of primary

Replace with INSERT 3

~~system leakage. Radiation monitors which indicate primary system leakage include the containment air particulate and gas monitoring, the condenser air ejector monitor, the component cooling water monitor, and the steam generator blowdown monitor.~~

~~References~~

~~FSAR, Section 4.2.7 - Reactor Coolant System Leakage~~

~~FSAR, Section 14.3.2 - Rupture of a Main Steam Pipe~~

D. Maximum Reactor Coolant Activity

Specifications

1. The total specific activity of the reactor coolant due to nuclides with half-lives of more than 15 minutes shall not exceed $100/\bar{E}$ µCi/cc whenever the reactor is critical or the average temperature is greater than 500°F, where $\bar{E}$ is the average sum of the beta and gamma energies, in Mev, per disintegration. If this limit is not satisfied, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection. Should this limit be exceeded by 25%, the reactor shall be made subcritical and cooled to 500°F or less within 2 hours after detection. ~~71~~

Insert 3 (in TS 3.1.C Basis)

BASES

BACKGROUND - Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During unit life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE limiting condition for operation (LCO) is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

Leakage from the RCS is collected in the containment or by other systems. These systems are the Main Steam System, Condensate and Feedwater System, the Gaseous and Liquid Waste Disposal Systems, the Component Cooling System, and the Chemical and Volume Control System.

Detection of leaks from the RCS is by one or more of the following:

1. An increased amount of makeup water required to maintain normal level in the pressurizer.
2. A high temperature alarm in the leakoff piping provided to collect reactor head flange leakage.
3. Containment sump water level indication.
4. Containment pressure, temperature, and humidity indication.

If there is significant radioactive contamination of the reactor coolant, the radiation monitoring system provides a sensitive indication of primary system leakage. Radiation monitors which indicate primary system leakage include the containment gas and particulate monitors, the condenser air ejector monitor, the component cooling water monitor, and the steam generator blowdown monitor.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

<u>APPLICABLE SAFETY ANALYSES</u> - Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 1 gpm or increases to 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a main steam line break (MSLB) accident. Other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 2) analysis for SGTR assumes the contaminated secondary fluid is released via power operated relief valves or safety valves. The source term in the primary system coolant is transported to the affected (ruptured) steam generator by the break flow. The affected steam generator discharges steam to the environment for 30 minutes until the generator is manually isolated. The 1 gpm primary to secondary LEAKAGE transports the source term to the unaffected steam generators. Releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service.

The MSLB is less limiting for site radiation releases than the SGTR. The safety analysis for the MSLB accident assumes 1 gpm total primary to secondary LEAKAGE, including 500 gpd leakage into the faulted generator. The dose consequences resulting from the MSLB and the SGTR accidents are within the limits defined in the plant licensing basis.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

<u>LIMITING CONDITIONS FOR OPERATION</u> - RCS operational LEAKAGE shall be limited to:

a. <u>Pressure Boundary LEAKAGE</u>

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 3). 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 SG 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.

APPLICABILITY - In REACTOR OPERATION conditions where Tavg exceeds 200°F, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In COLD SHUTDOWN and REFUELING SHUTDOWN, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.1.C.5 measures leakage through each individual pressure isolation valve (PIV) and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

3.1.C.2.a

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This completion 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 RCPB.

3.1.C.2.b and 3.1.C.3

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 SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

REFERENCES

1. UFSAR, Chapter 4, Surry Units 1 and 2.
2. UFSAR, Chapter 14, Surry Units 1 and 2.
3. NEI 97-06, “Steam Generator Program Guidelines.”
4. EPRI, “Pressurized Water Reactor Primary-to-Secondary Leak Guidelines.”

by a bubble in the pressurizer and/or pressurizer safety valves. A single PORV has adequate relieving capability to protect the Reactor Vessel from overpressurization when the transient is limited to either (1) the start of an idle Reactor Coolant Pump with the secondary water temperature of a steam generator ≤ 50°F above the RCS cold leg temperature or (2) the start of a charging pump and its injection into a water solid RCS.

The limitation for a maximum of one charging pump allowed OPERABLE and the surveillance required to verify that two charging pumps are inoperable below 350°F provide assurance that a mass addition pressure transient can be relieved by the operation of a single PORV, or equivalent. The Safety Injection accumulators are not considered a credible mass input mechanism for RCS low temperature overpressurization concerns. There are administrative controls to ensure isolation, including de-energizing the Safety Injection (SI) accumulator isolation valves, during plant shutdown conditions (RCS pressure less than 1000 psig) to prevent inadvertent SI accumulator discharge into the RCS for low temperature overpressurization concerns. An undesired pressurizer PORV lift due to inadvertent SI accumulator discharge is not possible when SI accumulator pressure is less than the low temperature PORV lift setpoint specified in TS 3.1.G. Therefore, SI accumulator isolation, and verification of such isolation is not necessary when SI accumulator pressure is less than the low temperature PORV setpoint.

A maximum pressurizer narrow range level of 33% has been selected to provide sufficient time, approximately 10 minutes, for operator response in case of a malfunction resulting in maximum charging flow from one charging pump (530 gpm). Operator action would be initiated by at least two alarms that would occur between the normal operating level and the maximum allowable level (33%). When both PORVs are inoperable and it is impossible to manually open at least one PORV, additional administrative controls shall be implemented to prevent a pressure transient that would exceed the limits of Appendix G to 10 CFR Part 50.

The requirements of this specification are only applicable when the Reactor Vessel head is bolted. When the Reactor Vessel head is unbolted, a RCS pressure of < 100 psig will lift the head, thereby creating a relieving capability equivalent to at least one PORV.

New TS 3.1.H - INSERT 4

Insert 4 (New TS 3.1.H)

H. Steam Generator (SG) Tube Integrity

Applicability

The following specifications are applicable whenever Tavg (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Specifications

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

2. If the requirements of 3.1.H.1 are not met for one or more SG tubes, then perform the following[1]:

    a. Within 7 days, verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection; and

    b. Plug the affected tube(s) in accordance with the Steam Generator Program prior to Tavg exceeding 200°F following the next refueling outage or SG tube inspection.

3. If the required actions of Specification 3.1.H.2 are not completed within the specified completion time, or SG tube integrity is not maintained, the unit shall be brought to HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

Note:

1. A separate TS action entry is allowed for each SG tube.

BASES

BACKGROUND - Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.1.A.2.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience 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. The SG performance criteria are used to manage SG tube degradation.

Specification 6.4.Q, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.4.Q, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 6.4.Q. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

<u>APPLICABLE SAFETY ANALYSES</u> - The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate of 1 gpm, which is conservative with respect to the operational LEAKAGE rate limits in Specification 3.1.C, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The UFSAR analysis for SGTR assumes the contaminated secondary fluid is released via power operated relief valves or safety valves. The source term in the primary system coolant is transported to the affected (ruptured) steam generator by the break flow. The affected steam generator discharges steam to the environment for 30 minutes until the generator is manually isolated. The 1 gpm primary to secondary LEAKAGE transports the source term to the unaffected steam generators. Releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service.

The analyses for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be within Specification 3.1.D, "Maximum Reactor Coolant Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 50.67 (Ref. 3) or Regulatory Guide 1.183 (Ref. 4), as appropriate.

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

<u>LIMITING CONDITIONS FOR OPERATION</u> - 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 6.4.Q, "Steam Generator 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 "significantly" 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." 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 (Ref. 5) and Draft Regulatory Guide 1.121 (Ref. 6).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm. 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 Specification 3.1.C, "RCS 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.

<u>APPLICABILITY</u> - Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced when Tavg exceeds 200°F.

RCS conditions are far less challenging in COLD SHUTDOWN and REFUELING SHUTDOWN than during INTERMEDIATE SHUTDOWN, HOT SHUTDOWN, REACTOR CRITICAL and POWER OPERATION. In COLD SHUTDOWN and REFUELING SHUTDOWN, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

<u>ACTIONS</u> - The actions are modified by a Note clarifying that the conditions may be entered independently for each SG tube. This is acceptable because the required actions provide appropriate compensatory actions for each affected SG tube. Complying with the required actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent condition entry and application of associated required actions.

3.1.H.2.a and b

Specification 3.1.H.2 applies 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 as required by SR 4.19. 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. If it is determined that tube integrity is not being maintained, Specification 3.1.H.3 applies.

A completion time of 7 days is sufficient to complete the evaluation while minimizing the risk of unit operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, required action 3.1.H.2.b allows unit operation 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 Tavg exceeding 200°F following the next refueling outage or SG inspection. This completion time is acceptable since operation until the next inspection is supported by the operational assessment.

3.1.H.3

If the required actions and associated completion times of Specification 3.1.H.2 are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

The allowed completion times are reasonable, based on operating experience, to reach the desired unit conditions from full power conditions in an orderly manner and without challenging plant systems.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 50.67.

4. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.

5. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

6. Draft Regulatory Guide 1.121, “Basis for Plugging Degraded Steam Generator Tubes,” August 1976.

7. EPRI, “Pressurized Water Reactor Steam Generator Examination Guidelines.”

TABLE 4.1-2A

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
1. Control Rod Assemblies	Rod drop times of all full length rods at hot conditions	Prior to reactor criticality: a. For all rods following each removal of the reactor vessel head b. For specially affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and c. Once per 18 months	7
2. Control Rod Assemblies	Partial movement of all rods	Quarterly	7
3. Refueling Water Chemical Addition Tank	Functional	Once per 18 months	6
4. Pressurizer Safety Valves	Setpoint	Per the Inservice Testing Program	4
5. Main Steam Safety Valves	Setpoint	Per the Inservice Testing Program	10
6. Containment Isolation Trip	* Functional	Once per 18 months	5
7. Refueling System Interlocks	* Functional	Prior to refueling	9.12
8. Service Water System	* Functional	Once per 18 months	9.9
9. Deleted			
10. ~~Primary System Leakage~~ Deleted	~~* Evaluate~~	~~Daily~~	~~4~~
11. Diesel Fuel Supply	* Fuel Inventory	5 days/week	8.5
12. Deleted			
13. Main Steam Line Trip Valves	Functional (Full Closure)	Before each startup (TS 4.7) The provisions of Specification 4.0.4. are not applicable	10

TABLE 4.1-2A(CONTINUED)
MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	UFSAR SECTION REFERENCE
19.	Primary Coolant System	Functional	1. Periodic leakage testing(a)(b) on each valve listed in Specification ~~3.1.C.7a~~ 3.1.C.5a shall be accomplished prior to entering POWER OPERATION after every time the plant is placed in COLD SHUTDOWN for refueling, after each time the plant is placed in COLD SHUTDOWN for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.	
20.	Containment Purge MOV Leakage	Functional	Semi-Annual (Unit at power or shutdown) if purge valves are operated during interval(c)	
21.	Deleted			
22.	RCS Flow	Flow $\geq$ 273,000 gpm	Once per 18 months	14
23.	Deleted			

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(b) Minimum differential test pressure shall not be below 150 psid.

(c) Refer to Section 4.4 for acceptance criteria.

* See Specification 4.1.D.

Insert 5 (New TS 4.13)

4.13 RCS OPERATIONAL LEAKAGE

Applicability

The following specifications are applicable to RCS operational LEAKAGE whenever Tavg (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Objective

To verify that RCS operational LEAKAGE is maintained within the allowable limits.

Specifications

A. Verify RCS operational LEAKAGE is within the limits specified in TS 3.1.C by performance of RCS water inventory balance once every 24 hours.[1, 2]

B. Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG once every 72 hours.[1]

Notes:

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

2. Not applicable to primary to secondary LEAKAGE.

BASES

SURVEILLANCE REQUIREMENTS (SR)

SR 4.13.A

Verifying RCS LEAKAGE to be within the Limiting Condition for Operation (LCO) limits ensures the integrity of the reactor coolant pressure boundary (RCPB) 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 balance must be performed with the reactor at steady state operating conditions (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The surveillance is modified by two notes. Note 1 states that this SR is not required to be completed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit 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 RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in the TS 3.1.C Bases.

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 24 hour frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 4.13.B

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.1.H, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 4. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG.

If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG. The surveillance is modified by a Note, which 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 RCP 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 the EPRI guidelines (Ref. 4).

REFERENCES

1. UFSAR, Chapter 4, Surry Units 1 and 2.

2. UFSAR, Chapter 14, Surry Units 1 and 2.

3. NEI 97-06, "Steam Generator Program Guidelines."

4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

4.19 STEAM GENERATOR INSERVICE INSPECTION

Applicability

Applies to the periodic inservice inspection of the steam generators.

Objective

To provide assurance of the continued integrity of the steam generator pressure boundaries.

Specifications

A. Each steam generator shall be demonstrated operable pursuant to Specification 3.1.A.2 by performance of the following augmented inservice inspection program and the requirement of Specification 4.2.A. 71

B. Steam Generator Sample Selection and Inspection - Each steam generator shall be determined operable during shutdown by selecting and inspection at least the minimum number of steam generators specified in Table 4.19-1.

C. 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

Replace with new TS 4.19 - INSERT 6

Table 4.19-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.19.D and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.19.E. 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 sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include: 71

  1. All nonplugged tubes that previously had detectable wall penetrations > 20%.

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

  3. A tube inspection pursuant to Specification 4.19.E.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 4.19-2) 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. 71

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 (>10%) further wall penetrations to be included in the above percentage calculations.

D. Inspected Frequencies - The above 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. 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. 71

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.19-2 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 4.19.D.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 4.19-2 during the shutdown subsequent to any of the following conditions: 71

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.1.C.6.
2. A seismic occurrence greater than the Operating Basis Earthquake.
3. A loss-of-coolant accident requiring actuation of the engineered safeguards.
4. A major main steam line or feedwater line break.

E. 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. 71

3. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused be 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 because it may become unserviceable prior to the next inspection 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 is 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 4.19.D.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. 71

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 to establish a baseline condition of the tubing. This inspection shall be performed 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 4.19-2.

F. 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 reported on an annual basis 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 and require prompt notification of the Commission shall be reported by special report prior to resumption of plant operation. The report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Basis

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 71
that corrective measures can be taken.

The unit is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitatin of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage of 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.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop inservice, it will be found during scheduled inservice steam generator tube examination. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.19.E.a, if 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability of reliably detecting degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission by special report prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations tests, additional eddy current inspection, and revision of the Technical Specification, if necessary.

TABLE 4.19-1
MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

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

71

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.

TABLE 4.19-2
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 addit- 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 & inspect 2S tubes in each other S.G. Special Report	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 Special Report	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

~~Amendment Nos. 104 and 104~~

Insert 6 (in TS 4.19 - Replace existing TS 4.19 in its entirety)

4.19 STEAM GENERATOR (SG) TUBE INTEGRITY

Applicability

Applies to the verification of SG tube integrity in accordance with the Steam Generator Program.

Objective

To provide assurance of SG tube integrity.

Specifications

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

B. Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to Tavg exceeding 200°F following a SG tube inspection.

BASES

SURVEILLANCE REQUIREMENTS (SR)

SR 4.19.A

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), 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, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.19.A. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 7). 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 6.4.Q contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 4.19.B

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 6.4.Q are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw 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). Reference 1 and Reference 7 provide 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 Tavg exceeding 200°F 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.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 50.67.
4. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.
5. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
6. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
7. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

O. 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 this 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.

P. Secondary Water Chemistry Monitoring Program

A secondary water chemistry monitoring program shall be provided to inhibit steam generator tube degradation. This program shall include the following:

1) Identification of a sampling schedule for the critical parameters and control points for these parameters;

2) Identification of the procedures used to quantify parameters that are critical to control points;

3) Identification of process sampling points;

4) Procedure for the recording and management of data;

5) Procedures defining corrective actions for off control point chemistry conditions; and

6) A procedure for identifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.

Q. INSERT 7

Insert 7 (new TS 6.4.Q)

Q. Steam Generator (SG) Program

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

1. 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.

2. 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.

   a. 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.

   b. 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 gpm for all SGs.

   c. The operational LEAKAGE performance criterion is specified in TS 3.1.C and 4.13, "RCS Operational LEAKAGE."

3. 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.

4. 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 4.a, 4.b, and 4.c 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.

   a. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

   b. 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.

   c. 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.

5. Provisions for monitoring operational primary to secondary LEAKAGE.

b. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.1.D.4. In addition, the information itemized in Specification 3.1.D.4 shall be included in this report.

3. ~~deleted~~ INSERT 8

Insert 8 (new TS 6.6.A.3)

3. Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after Tavg exceeds 200°F following completion of an inspection performed in accordance with the Specification 6.4.Q, 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, and

h. The effective plugging percentage for all plugging in each SG.

ATTACHMENT 4

PROPOSED TECHNICAL SPECIFICATIONS PAGES (TYPED)

Virginia Electric and Power Company
(Dominion)
Surry Power Station Units 1 and 2

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W. <u>STAGGERED TEST BASIS</u>

A staggered test basis shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

X. <u>LEAKAGE</u>

LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank,

2. 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

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. <u>Unidentified LEAKAGE</u>

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE, and

c. <u>Pressure Boundary LEAKAGE</u>

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

C. RCS Operational LEAKAGE

Applicability

The following specifications are applicable to RCS operational LEAKAGE whenever T_{avg} (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Specifications

1. RCS operational LEAKAGE shall be limited to:

   a. No pressure boundary LEAKAGE,

   b. 1 gpm unidentified LEAKAGE,

   c. 10 gpm identified LEAKAGE, and

   d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

2.a. If RCS operational LEAKAGE is not within the limits of 3.1.C.1 for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE, reduce LEAKAGE to within the specified limits within 4 hours.

   b. If the LEAKAGE is not reduced to within the specified limits within 4 hours, the unit shall be brought to HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

3. If RCS pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within the limit specified in 3.1.C.1.d, the unit shall be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

4. Detected or suspected leakage from the Reactor Coolant System shall be investigated and evaluated. At least two means shall be available to detect reactor coolant system leakage. One of these means must depend on the detection of radionuclides in the containment.

5.a. Prior to going critical all primary coolant system pressure isolation valves listed below shall be functional as a pressure isolation device, except as specified in 3.1.C.5.b. Valve leakage shall not exceed the amounts indicated.

	Unit 1	Unit 2	Max. Allowable Leakage (see note (a) below)
Loop A, Cold Leg	1-SI-79, 1-SI-241	2-SI-79, 2-SI-241	≤ 5.0 gpm for each valve
Loop B, Cold Leg	1-SI-82, 1-SI-242	2-SI-82, 2-SI-242	
Loop C, Cold Leg	1-SI-85, 1-SI-243	2-SI-85, 2-SI-243	

b. If Specification 3.1.C.5.a cannot be met, an orderly shutdown shall be initiated and the reactor shall be in HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours.

Notes

(a)

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

BASES

BACKGROUND - Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During unit life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE limiting condition for operation (LCO) is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

Leakage from the RCS is collected in the containment or by other systems. These systems are the Main Steam System, Condensate and Feedwater System, the Gaseous and Liquid Waste Disposal Systems, the Component Cooling System, and the Chemical and Volume Control System.

Detection of leaks from the RCS is by one or more of the following:

1. An increased amount of makeup water required to maintain normal level in the pressurizer.

2. A high temperature alarm in the leakoff piping provided to collect reactor head flange leakage.

3. Containment sump water level indication.

4. Containment pressure, temperature, and humidity indication.

If there is significant radioactive contamination of the reactor coolant, the radiation monitoring system provides a sensitive indication of primary system leakage. Radiation monitors which indicate primary system leakage include the containment gas and particulate monitors, the condenser air ejector monitor, the component cooling water monitor, and the steam generator blowdown monitor.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES - Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 1 gpm or increases to 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a main steam line break (MSLB) accident. Other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 2) analysis for SGTR assumes the contaminated secondary fluid is released via power operated relief valves or safety valves. The source term in the primary system coolant is transported to the affected (ruptured) steam generator by the break flow. The affected steam generator discharges steam to the environment for 30 minutes until the generator is manually isolated. The 1 gpm primary to secondary LEAKAGE transports the source term to the unaffected steam generators. Releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service.

The MSLB is less limiting for site radiation releases than the SGTR. The safety analysis for the MSLB accident assumes 1 gpm total primary to secondary LEAKAGE, including 500 gpd leakage into the faulted generator. The dose consequences resulting from the MSLB and the SGTR accidents are within the limits defined in the plant licensing basis.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LIMITING CONDITIONS FOR OPERATION - RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

   No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 3). 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 SG 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.

APPLICABILITY - In REACTOR OPERATION conditions where T_{avg} exceeds 200°F, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In COLD SHUTDOWN and REFUELING SHUTDOWN, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.1.C.5 measures leakage through each individual pressure isolation valve (PIV) and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

3.1.C.2.a

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This completion 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 RCPB.

3.1.C.2.b and 3.1.C.3

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 SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

REFERENCES

1. UFSAR, Chapter 4, Surry Units 1 and 2.
2. UFSAR, Chapter 14, Surry Units 1 and 2.
3. NEI 97-06, "Steam Generator Program Guidelines."
4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

D. Maximum Reactor Coolant Activity

Specifications

1. The total specific activity of the reactor coolant due to nuclides with half-lives of more than 15 minutes shall not exceed $100/\bar{E}$ µCi/cc whenever the reactor is critical or the average temperature is greater than 500°F, where $\bar{E}$ is the average sum of the beta and gamma energies, in Mev, per disintegration. If this limit is not satisfied, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection. Should this limit be exceeded by 25%, the reactor shall be made subcritical and cooled to 500°F or less within 2 hours after detection.

2. The specific activity of the reactor coolant shall be limited to ≤ 1.0 µCi/cc DOSE EQUIVALENT I-131 whenever the reactor is critical or the average temperature is greater than 500°F.

3. The requirements of D-2 above may be modified to allow the specific activity of the reactor coolant > 1.0 µCi/cc DOSE EQUIVALENT I-131 but less than 10.0 µCi/cc DOSE EQUIVALENT I-131. Following shutdown, the unit may be restarted and/or operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. With the specific activity of the reactor coolant > 1.0 µCi/cc DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 10.0 µCi/cc DOSE EQUIVALENT I-131, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection.

4. If the specific activity of the reactor coolant exceeds 1.0 µCi/cc DOSE EQUIVALENT I-131 or $100/\bar{E}$ µCi/cc, a report shall be prepared and submitted to the Commission pursuant to Specification 6.6.A.2. This report shall contain the results of the specific activity analysis together with the following information:

   a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

   b. Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded,

H. Steam Generator (SG) Tube Integrity

Applicability

The following specifications are applicable whenever T_{avg} (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Specifications

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

2. If the requirements of 3.1.H.1 are not met for one or more SG tubes, then perform the following:[1]

   a. Within 7 days, verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection; and

   b. Plug the affected tube(s) in accordance with the Steam Generator Program prior to T_{avg} exceeding 200°F following the next refueling outage or SG tube inspection.

3. If the required actions of Specification 3.1.H.2 are not completed within the specified completion time, or SG tube integrity is not maintained, the unit shall be brought to HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

Note:

1. A separate TS action entry is allowed for each SG tube.

BASES

BACKGROUND - Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.1.A.2.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience 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. The SG performance criteria are used to manage SG tube degradation.

Specification 6.4.Q, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.4.Q, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 6.4.Q. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

<u>APPLICABLE SAFETY ANALYSES</u> - The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate of 1 gpm, which is conservative with respect to the operational LEAKAGE rate limits in Specification 3.1.C, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The UFSAR analysis for SGTR assumes the contaminated secondary fluid is released via power operated relief valves or safety valves. The source term in the primary system coolant is transported to the affected (ruptured) steam generator by the break flow. The affected steam generator discharges steam to the environment for 30 minutes until the generator is manually isolated. The 1 gpm primary to secondary LEAKAGE transports the source term to the unaffected steam generators. Releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service.

The analyses for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be within Specification 3.1.D, "Maximum Reactor Coolant Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 50.67 (Ref. 3) or Regulatory Guide 1.183 (Ref. 4), as appropriate.

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LIMITING CONDITIONS FOR OPERATION - 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 6.4.Q, "Steam Generator 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 "significantly" 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." 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 (Ref. 5) and Draft Regulatory Guide 1.121 (Ref. 6).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm. 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 Specification 3.1.C, "RCS 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.

<u>APPLICABILITY</u> - Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced when T_{avg} exceeds 200°F.

RCS conditions are far less challenging in COLD SHUTDOWN and REFUELING SHUTDOWN than during INTERMEDIATE SHUTDOWN, HOT SHUTDOWN, REACTOR CRITICAL and POWER OPERATION. In COLD SHUTDOWN and REFUELING SHUTDOWN, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

<u>ACTIONS</u> - The actions are modified by a Note clarifying that the conditions may be entered independently for each SG tube. This is acceptable because the required actions provide appropriate compensatory actions for each affected SG tube. Complying with the required actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent condition entry and application of associated required actions.

3.1.H.2.a and b

Specification 3.1.H.2 applies 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 as required by SR 4.19. 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. If it is determined that tube integrity is not being maintained, Specification 3.1.H.3 applies.

A completion time of 7 days is sufficient to complete the evaluation while minimizing the risk of unit operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, required action 3.1.H.2.b allows unit operation 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 T_{avg} exceeding 200°F following the next refueling outage or SG inspection. This completion time is acceptable since operation until the next inspection is supported by the operational assessment.

3.1.H.3

If the required actions and associated completion times of Specification 3.1.H.2 are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

The allowed completion times are reasonable, based on operating experience, to reach the desired unit conditions from full power conditions in an orderly manner and without challenging plant systems.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 50.67.
4. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.
5. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
6. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
7. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

TABLE 4.1-2A

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
1. Control Rod Assemblies	Rod drop times of all full length rods at hot conditions	Prior to reactor criticality: a. For all rods following each removal of the reactor vessel head b. For specially affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and c. Once per 18 months	7
2. Control Rod Assemblies	Partial movement of all rods	Quarterly	7
3. Refueling Water Chemical Addition Tank	Functional	Once per 18 months	6
4. Pressurizer Safety Valves	Setpoint	Per the Inservice Testing Program	4
5. Main Steam Safety Valves	Setpoint	Per the Inservice Testing Program	10
6. Containment Isolation Trip	* Functional	Once per 18 months	5
7. Refueling System Interlocks	* Functional	Prior to refueling	9.12
8. Service Water System	* Functional	Once per 18 months	9.9
9. Deleted			
10. Deleted			
11. Diesel Fuel Supply	* Fuel Inventory	5 days/week	8.5
12. Deleted			
13. Main Steam Line Trip Valves	Functional (Full Closure)	Before each startup (TS 4.7) The provisions of Specification 4.0.4. are not applicable	10

TABLE 4.1-2A(CONTINUED)
MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	UFSAR SECTION REFERENCE
19.	Primary Coolant System	Functional	1. Periodic leakage testing(a)(b) on each valve listed in Specification 3.1.C.5.a shall be accomplished prior to entering POWER OPERATION after every time the plant is placed in COLD SHUTDOWN for refueling, after each time the plant is placed in COLD SHUTDOWN for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.	
20.	Containment Purge MOV Leakage	Functional	Semi-Annual (Unit at power or shutdown) if purge valves are operated during interval(c)	
21.	Deleted			
22.	RCS Flow	Flow $\geq$ 273,000 gpm	Once per 18 months	14
23.	Deleted			

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(b) Minimum differential test pressure shall not be below 150 psid.

(c) Refer to Section 4.4 for acceptance criteria.

* See Specification 4.1.D.

4.13 RCS OPERATIONAL LEAKAGE

Applicability

The following specifications are applicable to RCS operational LEAKAGE whenever T_{avg} (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Objective

To verify that RCS operational LEAKAGE is maintained within the allowable limits.

Specifications

A. Verify RCS operational LEAKAGE is within the limits specified in TS 3.1.C by performance of RCS water inventory balance once every 24 hours.[1, 2]

B. Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG once every 72 hours.[1]

Notes:

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

2. Not applicable to primary to secondary LEAKAGE.

BASES

SURVEILLANCE REQUIREMENTS (SR)

SR 4.13.A

Verifying RCS LEAKAGE to be within the Limiting Condition for Operation (LCO) limits ensures the integrity of the reactor coolant pressure boundary (RCPB) 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 balance must be performed with the reactor at steady state operating conditions (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The surveillance is modified by two notes. Note 1 states that this SR is not required to be completed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit 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 RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in the TS 3.1.C Bases.

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 24 hour frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 4.13.B

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.1.H, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 4. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG.

If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG. The surveillance is modified by a Note, which 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 RCP 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 the EPRI guidelines (Ref. 4).

REFERENCES

1. UFSAR, Chapter 4, Surry Units 1 and 2.
2. UFSAR, Chapter 14, Surry Units 1 and 2.
3. NEI 97-06, "Steam Generator Program Guidelines."
4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

4.19 STEAM GENERATOR (SG) TUBE INTEGRITY

Applicability

Applies to the verification of SG tube integrity in accordance with the Steam Generator Program.

Objective

To provide assurance of SG tube integrity.

Specifications

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

B. Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to T_{avg} exceeding 200°F following a SG tube inspection.

BASES

SURVEILLANCE REQUIREMENTS (SR)

SR 4.19.A

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), 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, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.19.A. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 7). 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 6.4.Q contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 4.19.B

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 6.4.Q are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw 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). Reference 1 and Reference 7 provide 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 T_{avg} exceeding 200°F 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.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 50.67.

4. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

5. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

6. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

7. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

Q. Steam Generator (SG) Program

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

1. 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.
2. 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.
   a. 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.
   b. 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 gpm for all SG.

c. The operational LEAKAGE performance criterion is specified in TS 3.1.C and 4.13, "RCS Operational LEAKAGE."

3. 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.

4. 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 4.a, 4.b, and 4.c 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.

   a. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

   b. 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.

   c. 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.

5. Provisions for monitoring operational primary to secondary LEAKAGE.

b. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.1.D.4. In addition, the information itemized in Specification 3.1.D.4 shall be included in this report.

3. Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after T_{avg} exceeds 200°F following completion of an inspection performed in accordance with the Specification 6.4.Q, 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, and
h. The effective plugging percentage for all plugging in each SG.