Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

August 30, 2006
LIC-06-0087

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

Reference: 1) Docket No. 50-285
2) Letter from Jeffrey A. Reinhart (OPPD) to Document Control Desk (NRC) dated May 30, 2006, Fort Calhoun Station Unit No. 1 License Amendment Request, "Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process and Deletion of Sleeving as a Steam Generator Tube Repair Method" (LIC-06-0002) (ML061510203)

SUBJECT: Fort Calhoun Station Unit No. 1 Revised License Amendment Request, "Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process and Deletion of Sleeving as a Steam Generator Tube Repair Method" (TAC# MD2188)

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby proposes to make changes to the Fort Calhoun Station Unit No. 1 (FCS) Technical Specifications (TS). The proposed amendment would revise the TS requirements related to 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." 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).

OPPD also proposes to change the FCS TS by deleting the sleeving repair alternative to plugging for steam generator tubes. The FCS replacement steam generators (RSGs) to be installed during the fall of 2006 are manufactured by Mitsubishi Heavy Industries, Ltd. (MHI). The change is being requested because OPPD has determined that the sleeving repair alternative to plugging can not be used for the MHI RSGs at this time.

Reference 2 contained OPPD's original submittal with respect to TSTF-449. OPPD is resubmitting this amendment request to incorporate various revisions to the TS as a result of a Request for Additional Information (RAI) from the NRC staff. Attachment 5 provides the RAI questions and responses as discussed in a phone call on July 19, 2006. This revised amendment request does not change Attachment 1 as previously submitted, including the No Significant Hazards Consideration and the Environmental Evaluation.

Attachment 1 provides a description of the proposed change, the requested confirmation of applicability, and plant-specific verifications. Attachment 2 provides the existing TS pages marked up to show the proposed change. Attachment 3 provides revised (clean) TS pages. The proposed FCS TS changes are consistent with NRC approved Revision 4 of TSTF-449. Since FCS has custom TS, the numbering of the proposed TS and the location of information differ in some instances from TSTF-449. As a result, a cross reference table is provided in Attachment 4 to identify the location of TSTF-449 revisions in the FCS TS.

OPPD requests approval of the proposed amendment by November 1, 2006, to support scheduled implementation during the 2006 refueling outage. OPPD requests that the effective date for this TS change be the end of the 2006 refueling outage to allow for implementation of these proposed changes. No new commitments are made to the NRC in this letter.

I declare under penalty of perjury that the foregoing is true and correct. (Executed on August 30, 2006).

If you have any questions or require additional information, please contact Thomas R. Byrne at (402) 533-7368.

Sincerely,

Jeffrey A. Reinhart
Site Director
Fort Calhoun Station

JAR/TRB/trb

Attachments:
1. Omaha Public Power District Evaluation
2. Markup of Technical Specification Pages
3. Proposed Technical Specifications (clean)
4. Location of TSTF-449 Requirements in FCS TS
5. Responses to Requests for Additional Information related to the May 30, 2006 TSTF-449 Submittal

c: Director of Consumer Health Services, Department of Regulation and Licensure, Nebraska Health and Human Services, State of Nebraska

ATTACHMENT 1
Omaha Public Power District
Fort Calhoun Station
Application For Technical Specification Improvement Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process and Deletion of Sleeving as a Steam Generator Tube Repair Method

Omaha Public Power District
Fort Calhoun Station
Application For Technical Specification Improvement Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process and Deletion of Sleeving as a Steam Generator Tube Repair Method

1.0 DESCRIPTION

The Omaha Public Power District (OPPD) proposes to revise the requirements in the Fort Calhoun Station Unit No. 1 (FCS) 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 technical specification improvement was announced in the Federal Register on May 6, 2005 as part of the consolidated line item improvement process (CLIIP). The proposed change also deletes steam generator tube sleeving as an alternative to plugging for steam generator tube repairs.

2.0 ASSESSMENT

The proposed FCS TS changes are consistent with NRC approved Revision 4 of TSTF-449. Since FCS has custom TS, however, the numbering of the proposed TS and the location of information differs in some instances from that of TSTF-449. Attachment 4 identifies the location of TSTF-449 revisions in the FCS TS.

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.

The sleeving repair method for Combustion Engineering steam generators was added to the FCS TS by Amendment 195 (Reference 10.3). During the 2006 Refueling Outage, replacement steam generators (RSGs) will be installed at FCS. Because the RSGs are manufactured by Mitsubishi Heavy Industries, Ltd. (MHI), the RSG tubes do not have an approved sleeving repair alternative to plugging. Therefore, OPPD is requesting the deletion of the sleeving capability from the TS because analyses supporting the existing specification are not applicable to the MHI steam generators.

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.

The proposed TS changes to delete surveillance requirements for a steam generator tube repair alternative (sleeving) are being requested since they will no longer be utilized or credited. Likewise, the analyses supporting the tube sleeving repair alternative are not applicable and are not planned to be applied to the RSGs. Thus, the proposed changes eliminate requirements not applicable to the RSGs. The TS will still contain the steam generator tube surveillance requirements which existed before tube sleeving was incorporated by Amendment 195 (Reference 10.3). Other TS changes, not related to tube sleeving, made by Amendment 195 are not being affected or altered. In accordance with the FCS general design criteria (Reference 10.4), the FCS TS are being revised as a result of a change in facility equipment.

5.0 TECHNICAL ANALYSIS

OPPD 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. OPPD has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to FCS and justify this amendment for the incorporation of the changes to the FCS TS.

The proposed changes remove provisions added to the TS in Amendment 195 (Reference 10.3) related to the surveillance requirements of leak tight sleeves, and the reference to the Electric Power Research Institute (EPRI) Pressurized Water Reactor (PWR) Steam Generator Examination Guidelines in the basis. Other TS changes, not related to tube sleeving, implemented by Amendment 195 are not being affected or altered.

The proposed changes remove the capability for repairing degraded steam generator tubes using sleeving which was approved for the original Combustion Engineering steam generators. These generators are being replaced in the fall of 2006 with steam generators manufactured by MHI which are not expected to use the tube sleeving alternative to tube plugging. The proposed changes eliminate tube sleeving as a repair for the RSG tubes. These changes are necessary, not as a result of new or enhanced analyses or evaluations, but as a consequence of MHI's steam generator operating experience which has not resulted in the need to develop the sleeving repair option.

The proposed changes do not alter, degrade, or prevent actions described or assumed in any accident analysis. They will not change any assumptions previously made in evaluating radiological consequences or affect any fission product barriers, nor do they increase any challenges to safety systems. They do not create any new systems interactions. Therefore, the proposed change does not increase or have any impact on the consequences of events described and evaluated in Chapter 14 of the Fort Calhoun Updated Safety Analysis Report (USAR).

5.1 Accident Induced Leakage Performance Criterion

The Nuclear Energy Institute (NEI) provided members the following information and status report in a letter dated September 2, 2005. NEI stated that this was offered for use in license amendment requests. NEI stated that the following information and status report has been reviewed with the NRC.

The industry is currently evaluating a technical issue related to the Accident Induced Leakage Performance Criterion (AILPC) specified in Section 5.23 of the proposed Technical Specifications. The issue concerns the consideration of non-pressure (bending) loads on the accident induced leak rates of steam generator tubes (axial differential thermal loads are routinely considered in assessing accident induced leakage). The EPRI Steam Generator Management Program (SGMP) is conducting a study to determine if bending loads are significant, and if they are, to define how to account for the loads in steam generator tube integrity assessments. In the interim, as this study is being completed, EPRI has completed a preliminary impact assessment. The assessment *(Preliminary Assessment of the Impact of Non-Pressure Loads on Leakage Integrity of Steam Generator Tubing)* found that the effect of the loads in question may, in certain circumstances, initiate primary-to-secondary leakage, or increase pre-existing primary-to-secondary leakage during and after load application. The effort also assessed the effect of such loads in combination with the applicable design basis accident. The results indicate that these circumstances are expected to be limited to the presence of significant circumferential cracks located in high bending stress regions of tubing. As of this date, such degradation has not been observed in the industry.

The structural integrity impact of non-pressure loads on degraded steam generator tubes has been well-documented in a previous EPRI report (NRC accession number ML050760208) related to the revised Structural Integrity Performance Criterion (SIPC). Experimental results indicated that neither axial loads nor bending loads have a significant effect on the burst pressure of tubing with axial degradation. Similarly, these loads are considered inconsequential for axially oriented degradation with respect to localized pop-through conditions and corresponding accident leakage. As such, industry experience indicates that the only meaningful impact of non-pressure loads with respect to leakage is due to the application of bending moments on circumferential cracking.

The EPRI Preliminary Assessment found that high bending loads that could affect the leakage analysis are only present in the top span region in the original design of once-through steam generators (OTSGs) and in the U-bend region of large-radius tubes in some recirculating steam generators. The high bending loads in the OTSGs are a consequence of crossflow during a steam line break whereas the high bending loads in the recirculating steam generators are a result of a seismic event.

After review of available analysis and experimental data, the EPRI Assessment concluded that the effect of high bending loads is only noteworthy for large 100% or near through-wall circumferential degradation. From a degradation assessment perspective, the EPRI study also reported that current industry experience indicates that there have been no observed stress corrosion circumferential cracks that are both capable of leaking and located in high bending stress regions. The industry's preliminary impact assessment and the plans for further technical study and experimental testing were presented to the NRC Staff in meetings on August 12, 2005 and July 12, 2006. The NRC Staff did not have any significant comments on the results presented.

Based on the above, OPPD believes that the effect of bending loads is not safety significant for FCS with respect to leakage integrity given the expected effect and existing margins with respect to degradation type, susceptible location and allowable flaw size.

If upon completion of EPRI's technical study, it is concluded that the effect of non-pressure loads, including bending loads, should be specifically accounted for in integrity assessments, the industry will revise the applicable steam generator program guideline documents to reflect the means developed to account for the loads.

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.	Fort Calhoun Station Unit No. 1
Steam Generator Model(s):	Mitsubishi Model: 49TT-1
Effective Full Power Years (EFPY) of service for currently installed SGs	0 EFPY
Tubing Material	Alloy 690TT
Number of tubes per SG	5200
Number and percentage of tubes plugged in each SG	1 tube in SG RC-2B (0.019%)
Number of tubes repaired in each SG	0
Degradation mechanism(s) identified	None
Current primary-to-secondary leakage limits:	150 gallons per day through any one steam generator at standard temperature.
Approved Alternate Tube Repair Criteria (ARC):	None
Approved SG Tube Repair Methods	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.
Performance criteria for accident Leakage	1.0 gallon per minute at standard temperature.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

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

OPPD has evaluated whether or not a significant hazards consideration is involved with the proposed amendment to delete steam generator tube sleeving as an alternative to plugging for steam generator tube repairs by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   The elimination from the TS surveillance requirements of leak tight sleeves as a repair method alternative to plugging defective steam generator tubes does not introduce an initiator to any previously evaluated accident. The frequency or periodicity of performance of the remaining surveillance requirements for steam generator tubes (including plugged tubes) is not affected by this change. Elimination of the tube repair method has no effect on the consequences of any previously evaluated accident. The proposed changes will not prevent safety systems from performing their accident mitigation function as assumed in the safety analysis.

   Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

   Response: No.

   The proposed change only affects the TS surveillance requirements. The proposed change is a result of installation of RSGs. The proposed change will eliminate a steam generator tube repair alternative which cannot be utilized or credited for the RSGs. This change will not alter assumptions made in the safety analysis and licensing bases and will not create new or different systems interactions.

   Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No.

   The proposed change deletes surveillance requirements for a steam generator tube repair alternative which will no longer be necessary or applicable. The remaining

TS steam generator tube surveillance requirements, including inspection and plugging requirements, will continue to maintain the applicable margin of safety.

Therefore, this TS change does not involve a significant reduction in the margin of safety.

Based on the above, Omaha Public Power District concludes that the proposed amendment to delete steam generator tube sleeving as an alternative to plugging for steam generator tube repairs presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

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

8.0 ENVIRONMENTAL EVALUATION

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

Based on the above considerations, the proposed amendment to delete steam generator tube sleeving as an alternative to plugging for steam generator tube repairs does not involve and will not result in a condition which significantly alters the impact of Fort Calhoun Station on the environment. Thus, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51.22(c)(9), and, pursuant to 10 CFR Part 51.22(b), no environmental assessment need be prepared.

9.0 PRECEDENT

The TS changes addressed in TSTF-449, Revision 4, are being made in accordance with the CLIIP. OPPD is not proposing variations or deviations from those TS changes, or the NRC staff's model SE published on March 2, 2005 (70 FR 10298) with the exception that the Basis for the applicable safety analyses has been changed to reflect the FCS-specific Steam Generator Tube Rupture (SGTR) assumptions.

OPPD is also requesting deletion from the FCS TS of the surveillance requirements for a steam generator tube repair alternative which will no longer be necessary or applicable due to installation of RSGs. Removal of unnecessary, inapplicable provisions from TS is an accepted practice.

10.0 REFERENCES

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

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

10.3 Letter from NRC (L. R. Wharton) to OPPD (S. K. Gambhir) dated March 1, 2001, Issuance of Amendment Re: Leak Tight Sleeves as an Alternative Tube Repair Method To Plugging Defective Steam Generator Tubes (TAC No. MA9653) (NRC-01-012)

10.4 Fort Calhoun Station Updated Safety Analysis Report, Responses to 70 Criteria, Appendix G.

ATTACHMENT 2

Markup of Technical Specification Pages

(NOTE: Additions are indicated by *italic* font; deletions are indicated by strikethrough.)

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TECHNICAL SPECIFICATIONS – TABLES

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DEFINITIONS

Ē - Average Disintegration Energy

Ē is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in MEV, for isotopes, other than iodines, with half lives greater than 15 minutes making up at least 95% of the total non-iodine radioactivity in the coolant.

Offsite Dose Calculation Manual (ODCM)

The document(s) that contain the methodology and parameters used in the calculations of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent radiation monitoring Warn/High (trip) Alarm setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:

1) The Radiological Effluent Controls and the Radiological Environmental Monitoring Program required by Specification 5.16.

2) Descriptions of the information that should be included in the Annual Radiological Environmental Operating Reports and Annual Radioactive Effluent Release Reports required by Specifications 5.9.4.a and 5.9.4.b.

Unrestricted Area

Any area at or beyond the site boundary access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

Core Operating Limits Report (COLR)

The Core Operating Limits Report (COLR) is a Fort Calhoun Station Unit No. 1 specific document that provides core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Section 5.9.5. Plant operation within these operating limits is addressed in the individual specifications.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE
   1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal 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 ~~(SG)~~ to the Secondary System *(primary to secondary LEAKAGE),*

DEFINITIONS

b. Unidentified LEAKAGE

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

c. Pressure Boundary LEAKAGE

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

RCS Pressure-Temperature Limits Report (PTLR)

The PTLR is a fluence dependent document that provides Limiting Conditions for Operation (LCO) in the form of pressure-temperature (P-T) limits to ensure prevention of brittle fracture. In addition, this document establishes power operated relief valve setpoints which provide low temperature overpressure protection (LTOP) to assure the P-T limits are not exceeded during the most limiting LTOP event. The P-T limits and LTOP criteria in the PTLR are applicable through the effective full power years (EFPYs) specified in the PTLR. NRC approved methodologies are used as the bases for the information provided in the PTLR.

References

(1) USAR, Section 7.2
(2) USAR, Section 7.3

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.1 Operable Components (Continued)

(5) DELETED

(6) Both steam generators shall be filled above the low steam generator water level trip set point and available to remove decay heat whenever the average temperature of the reactor coolant is above 300°F. ~~Each steam generator shall be demonstrated operable by performance of the inservice inspection program specified in Section 3.17 prior to exceeding a reactor coolant temperature of 300°F.~~

(7) Maximum reactor coolant system hydrostatic test pressure shall be 3125 psia. A maximum of 10 cycles of 3125 psia hydrostatic tests are allowed.

(8) Reactor coolant system leak and hydrostatic test shall be conducted within the limitations of the pressure and temperature limit Figure(s) shown in the PTLR.

(9) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum measured temperature of 73°F is required. Only 10 cycles are permitted.

(10) Maximum steam generator steam side leak test pressure shall not exceed 1000 psia. A minimum measured temperature of 73°F is required.

(11) Low Temperature Overpressure Protection (LTOP)

(a) The LTOP enable temperature and RCP operations shall be maintained in accordance with the PTLR.

(b) The unit can not be placed on shutdown cooling until the RCS has cooled to an indicated RCS temperature of less than or equal to 300°F.

(c) If no reactor coolant pumps are operating, a non-operating reactor coolant pump shall not be started while T_c is below the LTOP enable temperature stated in the PTLR unless there is a minimum indicated pressurizer steam space of at least 50% by volume.

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~~161,188, 207, 221~~

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits

Applicability

Applies to the leakage rates of the reactor coolant system whenever the reactor coolant temperature (T_{cold}) is greater than 210 °F.

Objective

To specify limiting conditions of the reactor coolant system leakage rates.

Specifications

To assure safe reactor operation, the following limiting conditions of the reactor coolant system leakage rates must be met:

(1) RCS operational LEAKAGE shall be limited to:

a. No Pressure Boundary LEAKAGE,
b. 1 gpm unidentified LEAKAGE,
c. 10 gpm identified LEAKAGE,
d. 150 gallons per day primary to secondary LEAKAGE through any one *steam generator (*SG*)*.

(2) If RCS *operational* LEAKAGE limits of (1), above, are not met for reasons other than Pressure Boundary LEAKAGE *or primary to secondary LEAKAGE*, then reduce LEAKAGE to meet limits within 4 hours.

(3) If the Required Action and associated completion time of (2), above, is not met, OR Pressure Boundary LEAKAGE exists, *or primary to secondary LEAKAGE is not within limits*, then be in MODE 3, Hot Shutdown, within 6 hours AND be in MODE 4, Cold Shutdown, within 36 hours.

(4) To determine leakage to the containment, a containment atmosphere radiation monitor (gaseous or particulate) or dew point instrument, and a containment sump level instrument must be operable.

a. With no containment sump level instrument operable, verify that a containment atmosphere radiation monitor is operable, and restore the containment sump level instrument to operable status within 30 days.

b. With no containment atmosphere radiation monitor and no dewpoint instrument operable, restore either a radiation monitor or dewpoint instrument to operable status within 30 days.

c. With only the dewpoint instrument operable, or with no operable instruments, enter Specification 2.0.1 immediately.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits (Continued)

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 150 gallon per day *operational* limit on primary to secondary LEAKAGE through any one SG is based upon guidance in NEI 97-06, Steam Generator Program Guidelines. *The Steam Generator Program operational LEAKAGE performance Criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with 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

The potential for RCPB LEAKAGE is greatest when the RCS is pressurized, that is, when the reactor coolant temperature (T_{cold}) is greater than 210°F.

In MODES 4 and 5, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

REQUIRED ACTIONS (2).
Unidentified LEAKAGE~~,~~ *or* identified LEAKAGE~~, or primary to secondary LEAKAGE~~ in excess of the LCO limits must be reduced to meet 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.

REQUIRED ACTIONS (3)
If any pressure boundary LEAKAGE exists *or primary to secondary LEAKAGE is not within limits,* or if unidentified~~,~~ *or* identified~~, or primary to secondary~~ LEAKAGE cannot be reduced to meet limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3, Hot Shutdown, within 6 hours and to MODE 4, Cold Shutdown, within 36 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 conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

2.0 ***LIMITING CONDITIONS FOR OPERATION***

2.23 *Steam Generator (SG) Tube Integrity*

Applicability

Applies whenever the reactor coolant temperature (T_{cold}) is greater than 210°F.

Objective

To ensure that SG tube integrity is maintained.

Specification

NOTE: Separate Condition entry is allowed for each SG Tube.

(1) The following conditions shall be maintained:

(a) SG tube Integrity shall be maintained, and

(b) All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

(2) If the requirements of (1)(b) above are not met for one or more SG tubes, then perform the following.

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

(b) Plug the affected tube(s) in accordance with the Steam Generator Program prior to exceeding 210°F reactor coolant temperature (T_{cold}) following the next refueling outage or SG tube inspection.

(3) If the Required Action and associated completion time of (2), above, is not met, or if SG tube integrity is not maintained, then be in MODE 3, Hot Shutdown, within 6 hours AND be in MODE 4, Cold Shutdown, within 36 hours.

Basis

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 Technical Specification 2.1.1, "Operable Components."

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.

2.0 ***LIMITING CONDITIONS FOR OPERATION***

2.23 *Steam Generator (SG) Tube Integrity (continued)*

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 5.23, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.23, 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 5.23. 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).

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 a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in Technical Specification 2.1.4, "Reactor Coolant System Leakage Limits," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes releases of activity occur from the faulted steam generator to the environment via the condenser air ejector and Main Steam Safety Valves (MSSVs) and Atmospheric Dump Valves (ADVs). The release via the condenser air ejector starts at the initiation of the event and continues to the reactor trip, while the release via the MSSVs/ADVs starts at the reactor trip and continues for the duration of the event."

The analysis 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 equal to the Technical Specification 2.1.3, "Reactor Coolant Radioactivity," 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 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). 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 5.23, "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.

2.0 ***LIMITING CONDITIONS FOR OPERATION***
2.23 *Steam Generator (SG) Tube Integrity (continued)*

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 have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." 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. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

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 per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in Technical Specification 2.1.4. Reactor Coolant System Leakage Limits," 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.

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_{cold} is > 210°F.

RCS conditions are far less challenging in MODES 4 and 5 than during MODES 1, 2, and 3. In MODES 4 and 5, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

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.

Specification 2.23(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 Technical Specification 3.17. 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.

2.0 *LIMITING CONDITIONS FOR OPERATION*

2.23 Steam Generator (SG) Tube Integrity (continued)

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 2.23(3) applies.

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

If the evaluation determines that the affected tube(s) have tube integrity, Required Action 2.23(2)b allows plant 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 exceeding 210°F reactor coolant temperature (T_{cold}) 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.

If the Required Actions and associated Completion Times of Technical Specification 2.23(2) are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 4 within 36 hours.

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

References

1. *NEI 97-06, "Steam Generator Program Guidelines."*
2. *10 CFR 50 Appendix A, GDC 19.*
3. *10 CFR 100.*
4. *ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.*
5. *Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.*
6. *EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."*

3.0 **SURVEILLANCE REQUIREMENTS**

3.2 Equipment and Sampling Tests (continued)

The Safety Injection (SI) pump room air treatment system consists of charcoal adsorbers which are installed in normally bypassed ducts. This system is designed to reduce the potential release of radioiodine in SI pump rooms during the recirculation period following a DBA. The in-place and laboratory testing of charcoal adsorbers will assure system integrity and performance.

Pressure drops across the combined HEPA filters and charcoal adsorbers, of less than 9 inches of water for the control room filters (VA-64A & VA-64B) and of less than 6 inches of water for each of the other air treatment systems will indicate that the filters and adsorbers are not clogged by amounts of foreign matter that would interfere with performance to established levels. Operation of each system for 10 hours every month will demonstrate operability and remove excessive moisture build-up in the adsorbers.

The hydrogen purge system provides the control of combustible gases (hydrogen) in containment for a post-LOCA environment. The surveillance tests provide assurance that the system is operable and capable of performing its design function. VA-80A or VA-80B is capable of controlling the expected hydrogen generation (67 SCFM) associated with 1) Zirconium - water reactions, 2) radiolytic decomposition of sump water and 3) corrosion of metals within containment. The system should have a minimum of one blower with associated valves and piping (VA-80A or VA-80B) available at all times to meet the guidelines of Regulatory Guide 1.7 (1971).

If significant painting, fire or chemical release occurs such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals or foreign materials, testing will be performed to confirm system performance.

Demonstration of the automatic and/or manual initiation capability will assure the system's availability.

Verifying Reactor Coolant System (RCS) leakage to be within the 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. Unidentified leakage is determined by performance of an RCS water inventory balance. Identified leakage is then determined by isolation and/or inspection. *Since Primary to Secondary Leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance, note "***" for line item 8a on Table 3-5 states that the Reactor Coolant System Leakage surveillance is not applicable to Primary to Secondary Leakage.* Primary to secondary leakage is ~~also~~ measured by performance of ~~an RCS water inventory balance in conjunction with~~ effluent monitoring within the secondary steam and feedwater systems.

3.0 **SURVEILLANCE REQUIREMENTS**

3.2 Equipment and Sampling Tests (continued)

Table 3-5, Item 8b verifies that primary to secondary LEAKAGE is less 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 surveillance requirement is not met, compliance with LCO 2.23, "Steam Generator (SG) Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. 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 daily 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. 5).

References

1) USAR, Section 9.10
2) ASTM D4057-95(2000), ASTM D975-98b, ASTM D4176-93, ASTM D129-00, ASTM D2622-87, ASTM D287-82, ASTM 6217-98, ASTM D2709-96
3) ASTM D975-98b, Table 1
4) Regulatory Guide 1.137
5) *EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."*

TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

		Test	Frequency	USAR Section Reference
1.	Control Element Assemblies	Drop times of all full-length CEA's	Prior to reactor criticality after each removal of the reactor vessel closure head	7.5.3
2.	Control Element Assemblies	Partial movement of all CEA's (Minimum of 6 in)	Q	7
3.	Pressurizer Safety Valves	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be 2485 psig ±1% and 2530 psig ±1% respectively.	R	7
4.	Main Steam Safety Valves	Set Point	R	4
5.	DELETED			
6.	DELETED			
7.	DELETED			
8a.	Reactor Coolant System Leakage***	Evaluate	D*	4
8b	*Primary to Secondary Leakage *****	*Continuous process radiation monitors or radiochemical grab sampling*	*D**	*4*
9a	Diesel Fuel Supply	Fuel Inventory	M	8.4
9b.	Diesel Lubricating Oil Inventory	Lube Oil Inventory	M	8.4
9c.	Diesel Fuel Oil Properties	Test Properties	In accordance with the Diesel Fuel Oil Testing Program	8.4
9d.	Required Diesel Generator Air Start Receiver Bank Pressure	Air Pressure	M	8.4

TABLE 3-5 (continued)
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

*****Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.*
This surveillance is not required to be performed until 12 hours after establishment of steady state operation.

3.0 **SURVEILLANCE REQUIREMENTS**

3.17 Steam Generator (SG) Tube~~s~~ *Integrity*

Applicability

Applies to in-service surveillance of steam generator tubes.

Objective

To ensure the integrity of the steam generator tubes.

Specifications

Each steam generator shall be demonstrated OPERABLE by performance of the following*:* ~~in-service inspection program.~~

(1) Verify SG Tube Integrity in accordance with the Steam Generator Program.

(2) Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to exceeding 210°F reactor coolant temperature (Tcold).

~~(1) Steam Generator Sample Selection and Inspection Methods~~

~~The in-service inspection shall be performed on each steam generator on a rotating schedule. Under some circumstances, the operating conditions in one steam generator may be found to be more severe than those in the second steam generator. Under such circumstance, the sample sequence shall be modified to inspect the steam generator with the most severe conditions.~~

~~(2) Steam Generator Tube Sample Selection and Inspection~~

~~The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 3-13. The in-service inspection of steam generator tubes shall be performed according to Specification 3.17(4)(i), "Tube Inspection," and at the frequencies specified in Specification 3.17(3). The inspected tubes shall be verified acceptable per the acceptance criteria of Specification 3.17(4). When applying the exceptions of (i), (ii) and (iii) below, previous degradiation, imperfections or defects in the area of the tube repaired by sleeving are not considered an area requiring reinspection or inspection of adjacent tubes. The tubes selected for each in-service inspection shall include at least 3% of the total tubes in the steam generators and the tubes selected for these inspections shall be selected on a random basis, except:~~

~~(i) If the tube is recorded as a degraded tube, then an adjacent tube shall be inspected.~~

~~(ii) The first sample inspection during each in-service inspection of each steam generator shall include all non-plugged tubes that previously had detectable wall penetrations (>20%) and shall also include tubes in those areas where experience has indicated potential problems.~~

~~(iii) The second and third sample inspections, if required, may be less than an entire tube length inspection provided the inspection concentrates on those areas of the tube~~

~~sheet array and on those portions of the tubes where defects were previously detected.~~

~~(iv)~~ ~~To the extent practical, 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.~~

~~The results of each sample inspection shall be classified into one of the following three categories (this classification shall apply to the inspection of tubes and is exclusive of the sleeve inspection requirements in Specification 3.17(2a)).~~

~~Category~~	~~Inspection Results~~
~~C-1~~	~~No more than 5% of the tubes inspected are degraded and none of the inspected tubes are defective.~~
~~C-2~~	~~No more than 1% of the tubes inspected are defective, or between 5% and 10% of the tubes inspected are degraded.~~
~~C-3~~	~~More than 1% of the tubes inspected are defective, or more than 10% of the tubes inspected are degraded.~~

~~**NOTE:** In all inspections, previously degraded tubes must exhibit growth of greater than 10% through wall or growth of greater than 25% of the repair limit to be included in the above calculations.~~

~~(2a)~~ ~~Steam Generator Tube Sleeve Sample Selection and Inspection~~

~~The steam generator tube sleeve minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 3-14. The in-service inspection of steam generator tube sleeves shall be performed according to Specification 3.17(4)(i), "Tube Sleeve Inspection," and at the frequencies specified in Specification 3.17(3). The inspected tube sleeves shall be verified acceptable per the acceptance criteria of Specification 3.17(4). The tube sleeves selected for each in-service inspection shall include at least 20% of the total number of tube sleeves in the steam generators and the tube sleeves selected for these inspections shall be selected on a random basis, except:~~

~~(i)~~ ~~If the tube sleeve is recorded as a degraded tube sleeve and an adjacent tube sleeve exists, then an adjacent tube sleeve shall be inspected.~~

~~(ii)~~ ~~The first sample inspection during each in-service inspection of each steam generator shall include all tube sleeves in non-plugged tubes that previously had detectable wall penetrations (>20%) and shall also include tube sleeves in those areas where experience has indicated potential problems.~~

3.0 **SURVEILLANCE REQUIREMENTS**
3.17 Steam Generator Tubes (Continued)

~~(iii) To the extent practical, where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tube sleeves inspected shall be from these critical areas. Where the number of sleeves in the critical areas represent less than 50% of the initial sample, all sleeves in the critical areas shall be inspected.~~

~~The results of each sample inspection shall be classified into one of the following three categories (this classification shall apply to the inspection of sleeves and is exclusive of the tube inspection requirements in Specification 3.17(2)).~~

~~Category~~	~~Inspection Results~~
~~C-1~~	~~No more than 5% of the tube sleeves inspected are degraded and none of the inspected tube sleeves are defective.~~
~~C-2~~	~~No more than 1% of the tube sleeves inspected are defective, or between 5% and 10% of the tube sleeves inspected are degraded.~~
~~C-3~~	~~More than 1% of the tube sleeves inspected are defective, or more than 10% of the tube sleeves inspected are degraded.~~

~~**NOTE**: In all inspections, previously degraded tube sleeves must exhibit growth of greater than 10% through wall or growth of greater than 25% of the repair limit to be included in the above calculations.~~

~~(3) Inspection Frequencies~~

~~The above required in-service inspections of steam generator tubes and tube sleeves shall be performed at the following frequencies (inspections shall be performed, unless otherwise specified, coincident with refueling outages or any scheduled cold shutdown for plant repair and maintenance):~~

~~(i) In-service inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection, subject to the following clarifications and exceptions.~~

~~1. If a plant operating cycle is less than 12 months, inspections may be performed at the end of that cycle.~~

~~2. If two consecutive tube inspections following service under all volatile treatment conditions result in all inspection results falling into the C-1 category or if two consecutive tube inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the tube inspection interval may be extended to a maximum of once per 40 months.~~

3.0 **SURVEILLANCE REQUIREMENTS**

3.17 Steam Generator Tubes (Continued)

~~3. The inspections of tube sleeves shall be configured to ensure that each individual tube sleeve is inspected at least once in 60 months, with the following exception: if the 60 month time frame falls during an operating cycle, completion of that cycle is acceptable prior to meeting this requirement.~~

~~(ii) Increased Inspection Frequencies~~

~~1. If results of the in-service inspection of the steam generator tubes conducted in accordance with Table 3-13 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 a subsequent inspection meets the conditions specified in Section 3.17(3)(i)2 above, at which time the interval can be extended to a 40-month period.~~

~~2. If results of the inservice inspection of tube sleeves conducted in accordance with Table 3-14 fall into Category C-3, the inspection frequency shall be increased such that 100% of the tube sleeves in the affected steam generator are inspected during subsequent inspections. The increase in inspection frequency shall apply until two consecutive tube sleeve inspections meet the conditions for Category C-1 or two consecutive tube sleeve inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, at which time the inspection frequency of Specification 3.17(3)(i)3 shall again apply.~~

~~(iii) Unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Tables 3-13 and 3-14 during the shutdown subsequent to any of the following conditions:~~

~~1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Section 2.1.4 of the Technical Specifications,~~

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

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

~~4. A main steam line or main feedwater line break.~~

~~(4) Acceptance Criteria~~

~~(i) As used in this specification:~~

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

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

~~Degraded Tube or Sleeve means a tube or sleeve containing imperfections $\geq$20% of the nominal wall thickness caused by degradation. Any tube which does not permit the passage of the eddy-current inspection probe through its entire length and U-bend shall be deemed a degraded tube. Any tube sleeve which does not permit the passage of the eddy current inspection probe through its entire length shall be deemed a degraded sleeve.~~

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

~~Defect means an imperfection of such severity that it exceeds the plugging or repair limit.~~

~~Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. Plugging or repair limit is equal to 40% of the nominal tube wall thickness for the original tube wall. Sleeved tubes shall be plugged upon detection of unacceptable degradiation in the pressure boundary region of the sleeve.~~

~~Unserviceable describes the condition of a tube or sleeve if it leaks in excess of analyzed limits or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break.~~

~~Tube or Tubing means that portion of the tube which forms the primary system to the secondary system pressure boundary.~~

~~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, excluding any areas defined under "Tube Sleeve Inspection".~~

3.0 **SURVEILLANCE REQUIREMENTS**

3.17 Steam Generator Tubes (Continued)

~~Tube Repair or Sleeve refers to a process that re-establishes tube serviceability. Acceptable tube repairs will be performed using the Combustion Engineering, Inc. Leak Tight Sleeve as described in the propietary Combustion Engineering, Inc. Report, CEN-630-P, Revision 02, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," June 1997.~~

~~Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure for the purpose of sleeving the tube. A tube inspection as defined herein is required prior to returning previously plugged tubes to service.~~

~~Tube Sleeve Inspection refers to inspection of the section of the steam generator tube repaired by sleeving. This includes the pressure retaining portions of the parent tube in contact with the sleeve, the sleeve-to-tube weld, and the pressure retaining portion of the sleeve.~~

~~(ii) The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks, plug all tubes with sleeves containing defects) required by Tables 3-13 and 3-14.~~

(3) Reporting Requirements

A report shall be submitted within 180 days after exceeding 210°F reactor coolant temperature (Tcold) following completion of an inspection performed in accordance with the Specification 5.23, 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.

~~(i) Following each in-service inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission within 30 days.~~

~~(ii) The complete results of the steam generator tube in-service inspection shall be reported to the Commission within 6 months following completion of the inspection. This report shall include:~~

~~1. Number and extent of tubes and tube sleeves inspected.~~

~~2. Location and percent of wall thickness penetration for each imperfection.~~

~~3. Identification of tubes plugged.~~

~~4. Identification of tubes repaired by sleeving.~~

~~3.0 SURVEILLANCE REQUIREMENTS~~

~~3.17 Steam Generator Tubes (Continued)~~

~~(iii) Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

~~TABLE 3-13~~

~~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 300 tubes per S.G.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~	~~N/A~~	~~N/A~~
	~~C-2~~	~~Plug or repair defective tubes and inspect additional 600 tubes in this S.G.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~C-2~~	~~Plug or repair defective tubes and inspect additional 1200 tubes in this S.G.~~	~~C-1~~	~~None~~
					~~C-2~~	~~Plug or repair defective tubes~~
			~~C-3~~	~~Perform action for C-3 result of first sample~~	~~C-3~~	~~Perform action for C-3 result of first sample~~
	~~C-3~~	~~Inspect all tubes in this S.G., plug or repair defective tubes and inspect 600 tubes in other S.G.~~	~~The second S.G. is C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~The second S.G. is C-2~~	~~Perform action for C-2 result of second sample~~	~~N/A~~	~~N/A~~
			~~The second S.G. is C-3~~	~~Inspect all tubes in the second S.G. and plug or repair defective tubes.~~	N/A	~~N/A~~

~~N/A Not applicable~~

~~TABLE 3-14~~

~~STEAM GENERATOR TUBE SLEEVE INSPECTION~~

~~1st Sample Inspection~~			~~2nd Sample Inspection~~	
~~Sample Size~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~
~~A minimum of 20% of the installed tube sleeves~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
	~~C-2~~	~~Plug tubes containing defective sleeves and inspect all remaining installed sleeves in this S.G.~~	~~C-1~~	~~None~~
			~~C-2~~	~~Plug tubes containing defective sleeves~~
			~~C-3~~	~~Perform action for C-3 result of first sample~~
	~~C-3~~	~~Inspect all installed sleeves in this S.G., plug tubes containing defective sleeves, and inspect a minimum of 20% of the installed sleeves in other S.G.~~ ~~Add the tubes with defective sleeves to the number of defective tubes list for NRC notification per Table 3-13~~	~~The second S.G. is C-1~~	~~None~~
			~~The second S.G. is C-2~~	~~Perform action for C-2 result of first sample~~
			~~The second S.G. is C-3~~	~~Inspect all sleeves in the second S.G. and plug tubes containing defective sleeves. Add the tubes with defective sleeves to the number of defective tubes list for NRC notification per Table 3-13~~

~~N/A - Not applicable~~

3.0 **SURVEILLANCE REQUIREMENTS**

3.17 Steam Generator Tubes (Continued)

Basis

During shutdown periods the SGs are inspected as required by this Surveillance requirement (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 3.17(1). The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). 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 5.23 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.23 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 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to exceeding 210°F reactor coolant temperature (T_{cold}) 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 100.

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

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

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

3.0 **SURVEILLANCE REQUIREMENTS**

3.17 Steam Generator Tubes (Continued)

~~The surveillance requirements for inspection of the steam generator tubes and tube sleeves ensure that the structural integrity of this portion of the RCS will be maintained. The program for in-service inspection of the steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1, dated July 1975. The program for in-service inspection of steam generator tube sleeves is based on a modification of EPRI PWR Steam Generator Examination Guidelines, Revision 5, Dated September 1997 In-service inspection of steam generator tubing and tube sleeves is essential in order to maintain surveillance of the conditions of the tubes and sleeves in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that lead to corrosion.~~

~~In-service inspection of steam generator tubing and tube sleeves also provides a means of characterizing the nature and cause of any tube or sleeve degradation so that corrective measures can be taken.~~

~~Tubes with defects may be repaired by a Combustion Engineering, Inc. Leak Tight Sleeve. The technical bases for sleeving repair are described in the Proprietary Combustion Engineering, Inc. Report CEN-630-P, Revision 02, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," June 1997.~~

~~Whenever the results of any steam generator tubing in-service inspection fall into Category C-3, these results will be promptly reported to the Commission prior to the 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 Specifications, if necessary.~~

5.0 ADMINISTRATIVE CONTROLS

5.23 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:

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

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

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

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

3. The operational LEAKAGE performance criterion is specified in LCO 2.1.4, "Reactor Coolant System Leakage Limits."

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

5.0 ADMINISTRATIVE CONTROLS

5.23 Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

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

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

ATTACHMENT 3

Proposed Technical Specification Pages (clean)

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DEFINITIONS

$\bar{E}$ - Average Disintegration Energy

$\bar{E}$ is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in MEV, for isotopes, other than iodines, with half lives greater than 15 minutes making up at least 95% of the total non-iodine radioactivity in the coolant.

Offsite Dose Calculation Manual (ODCM)

The document(s) that contain the methodology and parameters used in the calculations of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent radiation monitoring Warn/High (trip) Alarm setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:

1) The Radiological Effluent Controls and the Radiological Environmental Monitoring Program required by Specification 5.16.

2) Descriptions of the information that should be included in the Annual Radiological Environmental Operating Reports and Annual Radioactive Effluent Release Reports required by Specifications 5.9.4.a and 5.9.4.b.

Unrestricted Area

Any area at or beyond the site boundary access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

Core Operating Limits Report (COLR)

The Core Operating Limits Report (COLR) is a Fort Calhoun Station Unit No. 1 specific document that provides core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Section 5.9.5. Plant operation within these operating limits is addressed in the individual specifications.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE
    1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal 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),

DEFINITIONS

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal 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.

RCS Pressure-Temperature Limits Report (PTLR)

The PTLR is a fluence dependent document that provides Limiting Conditions for Operation (LCO) in the form of pressure-temperature (P-T) limits to ensure prevention of brittle fracture. In addition, this document establishes power operated relief valve setpoints which provide low temperature overpressure protection (LTOP) to assure the P-T limits are not exceeded during the most limiting LTOP event. The P-T limits and LTOP criteria in the PTLR are applicable through the effective full power years (EFPYs) specified in the PTLR. NRC approved methodologies are used as the bases for the information provided in the PTLR.

References

(1) USAR, Section 7.2
(2) USAR, Section 7.3

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.1 Operable Components (Continued)

(5) DELETED

(6) Both steam generators shall be filled above the low steam generator water level trip set point and available to remove decay heat whenever the average temperature of the reactor coolant is above 300°F.

(7) Maximum reactor coolant system hydrostatic test pressure shall be 3125 psia. A maximum of 10 cycles of 3125 psia hydrostatic tests are allowed.

(8) Reactor coolant system leak and hydrostatic test shall be conducted within the limitations of the pressure and temperature limit Figure(s) shown in the PTLR.

(9) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum measured temperature of 73°F is required. Only 10 cycles are permitted.

(10) Maximum steam generator steam side leak test pressure shall not exceed 1000 psia. A minimum measured temperature of 73°F is required.

(11) Low Temperature Overpressure Protection (LTOP)

(a) The LTOP enable temperature and RCP operations shall be maintained in accordance with the PTLR.

(b) The unit can not be placed on shutdown cooling until the RCS has cooled to an indicated RCS temperature of less than or equal to 300°F.

(c) If no reactor coolant pumps are operating, a non-operating reactor coolant pump shall not be started while T_c is below the LTOP enable temperature stated in the PTLR unless there is a minimum indicated pressurizer steam space of at least 50% by volume.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits

Applicability

Applies to the leakage rates of the reactor coolant system whenever the reactor coolant temperature (T_{cold}) is greater than 210 °F.

Objective

To specify limiting conditions of the reactor coolant system leakage rates.

Specifications

To assure safe reactor operation, the following limiting conditions of the reactor coolant system leakage rates must be met:

(1) RCS operational LEAKAGE shall be limited to:

a. No Pressure Boundary LEAKAGE,
b. 1 gpm unidentified LEAKAGE,
c. 10 gpm identified LEAKAGE,
d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

(2) If RCS operational LEAKAGE limits of (1), above, are not met for reasons other than Pressure Boundary LEAKAGE or primary to secondary LEAKAGE, then reduce LEAKAGE to meet limits within 4 hours.

(3) If the Required Action and associated completion time of (2), above, is not met, OR Pressure Boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limits, then be in MODE 3, Hot Shutdown, within 6 hours AND be in MODE 4, Cold Shutdown, within 36 hours.

(4) To determine leakage to the containment, a containment atmosphere radiation monitor (gaseous or particulate) or dew point instrument, and a containment sump level instrument must be operable.

a. With no containment sump level instrument operable, verify that a containment atmosphere radiation monitor is operable, and restore the containment sump level instrument to operable status within 30 days.

b. With no containment atmosphere radiation monitor and no dewpoint instrument operable, restore either a radiation monitor or dewpoint instrument to operable status within 30 days.

c. With only the dewpoint instrument operable, or with no operable instruments, enter Specification 2.0.1 immediately.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits (Continued)

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 150 gallon per day operational limit on primary to secondary LEAKAGE through any one SG is based upon guidance in NEI 97-06, Steam Generator Program Guidelines. The Steam Generator Program operational LEAKAGE performance Criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with 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

The potential for RCPB LEAKAGE is greatest when the RCS is pressurized, that is, when the reactor coolant temperature (T_{cold}) is greater than 210°F.

In MODES 4 and 5, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

REQUIRED ACTIONS (2).

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to meet 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.

REQUIRED ACTIONS (3)

If any pressure boundary LEAKAGE exists or primary to secondary LEAKAGE is not within limits, or if unidentified or identified LEAKAGE cannot be reduced to meet limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3, Hot Shutdown, within 6 hours and to MODE 4, Cold Shutdown, within 36 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 conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

2.0 LIMITING CONDITIONS FOR OPERATION

2.23 Steam Generator (SG) Tube Integrity

Applicability

Applies whenever the reactor coolant temperature (T_{cold}) is greater than 210°F.

Objective

To ensure that SG tube integrity is maintained.

Specification

NOTE: Separate Condition entry is allowed for each SG Tube.

(1) The following conditions shall be maintained:

(a) SG tube Integrity shall be maintained, and

(b) All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

(2) If the requirements of (1)(b) above are not met for one or more SG tubes, then perform the following:

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

(b) Plug the affected tube(s) in accordance with the Steam Generator Program prior to exceeding 210°F reactor coolant temperature (Tcold) following the next refueling outage or SG tube inspection.

(3) If the Required Action and associated completion time of (2), above, is not met, or if SG tube integrity is not maintained, then be in MODE 3, Hot Shutdown, within 6 hours AND be in MODE 4, Cold Shutdown, within 36 hours.

Basis

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 Technical Specification 2.1.1, "Operable Components."

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.

2.0 LIMITING CONDITIONS FOR OPERATION

2.23 Steam Generator (SG) Tube Integrity (continued)

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 5.23, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.23, 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 5.23. 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).

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 a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in Technical Specification 2.1.4, "Reactor Coolant System Leakage Limits," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes releases of activity occur from the faulted steam generator to the environment via the condenser air ejector and Main Steam Safety Valves (MSSVs) and Atmospheric Dump Valves (ADVs). The release via the condenser air ejector starts at the initiation of the event and continues to the reactor trip, while the release via the MSSVs/ADVs starts at the reactor trip and continues for the duration of the event."

The analysis 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 equal to the Technical Specification 2.1.3, "Reactor Coolant Radioactivity," 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 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). 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 5.23, "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.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.23 Steam Generator (SG) Tube Integrity (continued)

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 have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." 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. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

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 per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in Technical Specification 2.1.4. Reactor Coolant System Leakage Limits," 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.

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_{cold} is > 210°F.

RCS conditions are far less challenging in MODES 4 and 5 than during MODES 1, 2, and 3. In MODES 4 and 5, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

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.

Specification 2.23(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 Technical Specification 3.17. 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 the Steam Generator Program.

2.0 LIMITING CONDITIONS FOR OPERATION
2.23 Steam Generator (SG) Tube Integrity (continued)

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 2.23(3) applies.

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

If the evaluation determines that the affected tube(s) have tube integrity, Required Action 2.23(2)b allows plant 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 exceeding 210°F reactor coolant temperature (T_{cold}) 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.

If the Required Actions and associated Completion Times of Technical Specification 2.23(2) are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 4 within 36 hours.

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

References

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 100.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

3.0 **SURVEILLANCE REQUIREMENTS**
3.2 Equipment and Sampling Tests (continued)

The Safety Injection (SI) pump room air treatment system consists of charcoal adsorbers which are installed in normally bypassed ducts. This system is designed to reduce the potential release of radioiodine in SI pump rooms during the recirculation period following a DBA. The in-place and laboratory testing of charcoal adsorbers will assure system integrity and performance.

Pressure drops across the combined HEPA filters and charcoal adsorbers, of less than 9 inches of water for the control room filters (VA-64A & VA-64B) and of less than 6 inches of water for each of the other air treatment systems will indicate that the filters and adsorbers are not clogged by amounts of foreign matter that would interfere with performance to established levels. Operation of each system for 10 hours every month will demonstrate operability and remove excessive moisture build-up in the adsorbers.

The hydrogen purge system provides the control of combustible gases (hydrogen) in containment for a post-LOCA environment. The surveillance tests provide assurance that the system is operable and capable of performing its design function. VA-80A or VA-80B is capable of controlling the expected hydrogen generation (67 SCFM) associated with 1) Zirconium - water reactions, 2) radiolytic decomposition of sump water and 3) corrosion of metals within containment. The system should have a minimum of one blower with associated valves and piping (VA-80A or VA-80B) available at all times to meet the guidelines of Regulatory Guide 1.7 (1971).

If significant painting, fire or chemical release occurs such that the HEPA filters or charcoal adsorbers could become contaminated from the fumes, chemicals or foreign materials, testing will be performed to confirm system performance.

Demonstration of the automatic and/or manual initiation capability will assure the system's availability.

Verifying Reactor Coolant System (RCS) leakage to be within the 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. Unidentified leakage is determined by performance of an RCS water inventory balance. Identified leakage is then determined by isolation and/or inspection. Since Primary to Secondary Leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance, note "***" for line item 8a on Table 3-5 states that the Reactor Coolant System Leakage surveillance is not applicable to Primary to Secondary Leakage. Primary to secondary leakage is measured by performance of effluent monitoring within the secondary steam and feedwater systems.

3.0 **SURVEILLANCE REQUIREMENTS**

3.2 Equipment and Sampling Tests (continued)

Table 3-5, Item 8b verifies that primary to secondary LEAKAGE is less 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 surveillance requirement is not met, compliance with LCO 3.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. 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 daily 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. 5).

References

1) USAR, Section 9.10
2) ASTM D4057-95(2000), ASTM D975-98b, ASTM D4176-93, ASTM D129-00, ASTM D2622-87, ASTM D287-82, ASTM 6217-98, ASTM D2709-96
3) ASTM D975-98b, Table 1
4) Regulatory Guide 1.137
5) EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

		Test	Frequency	USAR Section Reference
1.	Control Element Assemblies	Drop times of all full-length CEA's	Prior to reactor criticality after each removal of the reactor vessel closure head	7.5.3
2.	Control Element Assemblies	Partial movement of all CEA's (Minimum of 6 in)	Q	7
3.	Pressurizer Safety Valves	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be 2485 psig ±1% and 2530 psig ±1% respectively.	R	7
4.	Main Steam Safety Valves	Set Point	R	4
5.	DELETED			
6.	DELETED			
7.	DELETED			
8a.	Reactor Coolant System Leakage***	Evaluate	D*	4
8b	Primary to Secondary Leakage ****	Continuous process radiation monitors or radiochemical grab sampling	D*	4
9a	Diesel Fuel Supply	Fuel Inventory	M	8.4
9b.	Diesel Lubricating Oil Inventory	Lube Oil Inventory	M	8.4
9c.	Diesel Fuel Oil Properties	Test Properties	In accordance with the Diesel Fuel Oil Testing Program	8.4
9d.	Required Diesel Generator Air Start Receiver Bank Pressure	Air Pressure	M	8.4

TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

****Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.
This surveillance is not required to be performed until 12 hours after establishment of steady state operation.

TABLE 3-5
<u>MINIMUM FREQUENCIES FOR EQUIPMENT TESTS</u>

		Test	Frequency	USAR Section Reference
9e.	Check for and Remove Accumulated Water from Each Fuel Oil Storage Tank	Check for Water and Remove	Q	8.4
10a.	Charcoal and HEPA Filters for Control Room	1. <u>In-Place Testing</u>** Charcoal adsorbers and HEPA filter banks shall be leak tested and show ≥99.95% Freon (R-11 or R-112) and cold DOP particulates removal, respectively.	On a refueling frequency or every 720 hours of system operation or after each complete or partial replacement of the charcoal adsorber/HEPA filter banks, or after any major structural maintenance on the system housing or following significant painting, fire or chemical releases in a ventilation zone communicating with the system.	9.10
		2. <u>Laboratory Testing</u>** Verify, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows methyliodide penetration less than 0.175% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.	On a refueling frequency <u>or</u> every 720 hours of system operation or after any structural maintenance on the HEPA filter or charcoal adsorber housing <u>or</u> following significant painting, fire <u>or</u> chemical release in a ventilation zone communicating with the system.	

**Tests shall be performed in accordance with applicable section(s) of ANSI N510-1980.

TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Test	Frequency	USAR Section Reference
10a. (continued)	3. Overall System Operation		
	a. Each circuit shall be operated.	Ten hours every month.	
	b. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 9 inches of water at system design flow rate.	R	
	c. Fan shall be shown to operate within ± 10% design flow.	R	
	4. Automatic and manual initiation of the system shall be demonstrated.	R	
10b. Charcoal Adsorbers for Spent Fuel Storage Pool Area	1. In-Place Testing** Charcoal adsorbers shall be leak tested and shall show ≥99% Freon (R-11 or R-112) removal.	On a refueling frequency or every 720 hours of system operation, or after each complete or partial replacement of the charcoal adsorber bank, or after any major structural maintenance on the system housing or following significant painting, fire or chemical release in a ventilation zone communicating with the system.	6.2 9.10
	2. Laboratory Testing Verify, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows methyliodide penetration less than 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 95%.	On a refueling frequency or every 720 hours of system operation or after any structural maintenance on the HEPA filter or charcoal adsorber housing or following significant painting, fire or chemical release in a ventilation zone communicating with the system.	

**Tests shall be performed in accordance with applicable section(s) of ANSI N510-1980.

TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

		Test	Frequency	USAR Section Reference
10b.	(continued)	3. Overall System Operation		
		a. Operation of each circuit shall be demonstrated.	Ten hours every month.	
		b. Volume flow rate through charcoal filter shall be shown to be between 4500 and 12,000 cfm.	R	
		4. Manual initiation of the system shall be demonstrated.	R	
10c.	Charcoal Adsorbers for S.I. Pump Room	1. In-Place Testing** Charcoal adsorbers shall be leak tested and shall show $\geq$99% Freon (R-11 or R-112) removal.	On a refueling frequency or every 720 hours of system operation, or after each complete or partial replacement of the charcoal adsorber bank, or after any major structural maintenance on the system housing or following significant painting, fire or chemical release in any ventilation zone communicating with the system.	9.10 6.2
		2. Laboratory Testing Verify, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows methyliodide penetration less than 10% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 95%.	On a refueling frequency <u>or</u> following 720 hours of system operation <u>or</u> after any structural maintenance on the HEPA filter or charcoal adsorber housing <u>or</u> following significant painting, fire <u>or</u> chemical release in a ventilation zone communicating with the system.	
		3. Overall System Operation		
		a. Operation of each circuit shall be demonstrated.	Ten hours every month.	
		b. Volume flow rate shall be shown to be between 3000 and 6000 cfm.	R	

**Tests shall be performed in accordance with applicable section(s) of ANSI N510-1980

TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

		Test	Frequency	USAR Section Reference
10c.	(continued)	4. Automatic and/or manual initiation of the system shall be demonstrated.	R	
11.	Containment Ventilation System Fusible Linked Dampers	1. Demonstrate damper action.	1 year, 2 years, 5 years, and every 5 years thereafter.	9.10
		2. Test a spare fusible link.		
12.	Diesel Generator Calibrate Under-Voltage Relays		R	8.4.3
13.	Motor Operated Safety Injection Loop Valve Motor Starters (HCV-311, 314, 317, 320, 327, 329, 331, 333, 312, 315, 318, 321)	Verify the contactor pickup value at ≤85% of 460 V.	R	
14.	Pressurizer Heaters	Verify control circuits operation for post-accident heater use.	R	
15.	Spent Fuel Pool Racks	Test neutron poison samples for dimensional change, weight, neutron attenuation change and specific gravity change.	1, 2, 4, 7, and 10 years after installation, and every 5 years thereafter.	
16.	Reactor Coolant Gas Vent System	1. Verify all manual isolation valves in each vent path are in the open position.	During each refueling outage just prior to plant start-up.	
		2. Cycle each automatic valve in the vent path through at least one complete cycle of full travel from the control room. Verification of valve cycling may be determined by observation of position indicating lights.	R	
		3. Verify flow through the reactor coolant vent system vent paths.	R	

TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

		Test	Frequency
17.	Hydrogen Purge System	1. Verify all manual valves are operable by completing at least one cycle.	R
		2. Cycle each automatic valve through at least one complete cycle of full travel from the control room. Verification of the valve cycling may be determined by the observation of position indicating lights.	R
		3. Initiate flow through the VA-80A and VA-80B blowers, HEPA filter, and charcoal adsorbers and verify that the system operates for at least (a) 30 minutes with suction from the auxiliary building (Room 59) (b) 10 hours with suction from the containment	 a) M b) R
		4. Verify the pressure drop across the VA-82 HEPAs and charcoal filter to be less than 6 inches of water. Verify a system flow rate of greater than 80 scfm and less than 230 scfm during system operation when tested in accordance with 3b. above.	R
18.	Shutdown Cooling	1. Verify required shutdown cooling loops are OPERABLE and one shutdown cooling loop is IN OPERATION.	S (when shutdown cooling is required by TS 2.8).
		2. Verify correct breaker alignment and indicated power is available to the required shutdown cooling pump that is not IN OPERATION.	W (when shutdown cooling is required by TS 2.8).

TABLE 3-5
MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Test	Frequency
19. Refueling Water Level	Verify refueling water level is ∃ 23 ft. above the top of the reactor vessel flange.	Prior to commencing, and daily during CORE ALTERATIONS and/or REFUELING OPERATIONS inside containment.
20. Spent Fuel Pool Level	Verify spent fuel pool water level is ∃ 23 ft. above the top of irradiated fuel assemblies seated in the storage racks.	Prior to commencing, and weekly during REFUELING OPERATIONS in the spent fuel pool.
21. Containment Penetrations	Verify each required containment penetration is in the required status.	Prior to commencing, and weekly during CORE ALTERATIONS and/or REFUELING OPERATIONS in containment.
22. Spent Fuel Assembly Storage	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-10.	Prior to storing the fuel assembly in Region 2 (including peripheral cells).
23. P-T Limit Curve	Verify RCS Pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified by the P-T limit Figure(s) shown in the PTLR.	This test is only required during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. While these operations are occurring, this test shall be performed every 30 minutes.
24. Spent Fuel Cask Loading	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-11.	Prior to placing the fuel assembly in a spent fuel cask in the spent fuel pool.

3.0 **SURVEILLANCE REQUIREMENTS**

3.17 Steam Generator (SG) Tubes~~s~~ Integrity

Applicability

Applies to in-service surveillance of steam generator tubes.

Objective

To ensure the integrity of the steam generator tubes.

Specifications

Each steam generator shall be demonstrated OPERABLE by performance of the following:

(1) Verify SG **T**ube Integrity in accordance with the Steam Generator Program.

(2) Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to exceeding 210°F reactor coolant temperature (Tcold).

(3) Reporting Requirements

A report shall be submitted within 180 days after exceeding 210°F reactor coolant temperature (Tcold) following completion of an inspection performed in accordance with the Specification 5.23, 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.

3.0 **SURVEILLANCE REQUIREMENTS**

3.17 Steam Generator Tubes (Continued)

Basis

During shutdown periods the SGs are inspected as required by this Surveillance Requirement (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 3.17(1). The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). 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 5.23 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.23 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 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to exceeding 210°F reactor coolant temperature (T_{cold}) 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 100.

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

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

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

5.0 ADMINISTRATIVE CONTROLS

5.23 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:

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

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

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

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

   3. The operational LEAKAGE performance criterion is specified in LCO 2.1.4, "Reactor Coolant System Leakage Limits."

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

5.0 ADMINISTRATIVE CONTROLS

5.23 Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

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

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

ATTACHMENT 4

Location of TSTF-449 Requirements in FCS TS

Location of TSTF-449 Revisions in FCS TS	
TSTF-449	FCS TS
Revised TS definition of LEAKAGE	Revised TS definition of LEAKAGE.
Revised TS [3.4.13], RCS [Reactor Coolant System] Operational Leakage	Revised TS 2.1.4, Reactor Coolant System Operational Leakage Limits, addresses the LCO and ACTION revisions of TS 3.4.13. Revised TS 3.2 addresses the SURVEILLANCE REQUIREMENTS revisions of TS 3.4.13. Specifically, line 8 of Table 3-5 has been replaced by lines 8a and 8b of Table 3-5, and new footnotes designated "***" and "****" have been added.
New TS [3.4.18], Steam Generator Tube Integrity	New TS 2.23, Steam Generator (SG) Tube Integrity, addresses the LCO, APLICABILITY, and ACTION portions of TS 3.4.18. Revised TS 3.17, Steam Generator Tube Integrity, Specifications 1 and 2, address the SURVEILLANCE REQUIREMENTS portion of TS 3.4.18.
Revised TS [5.5.9], Steam Generator (SG) Program	New Administrative Controls TS 5.23, Steam Generator (SG) Program
Revised TS [5.6.9], Steam Generator Tube Inspection Report	Revised TS 3.17, Steam Generator Tube Integrity, Specification (5).
Insert B 3.4.13 A	First sentence was incorporated in the Basis for TS 2.23. Second sentence was not incorporated because the licensing basis main steam line break analysis for FCS assumes 0 gpm leakage into the intact steam generator.
Insert B 3.4.13 B	Incorporated in the Basis for FCS TS 2.1.4.
Insert B 3.4.13 C	Incorporated in the Basis for FCS TS 3.2.
Insert B 3.4.13 D (CEOG)	Incorporated in the Basis for FCS TS 3.2.
Insert B 3.4.13 E	Reference 4 is incorporated in the Basis for FCS TS 3.17. Reference 5 is incorporated in the Basis for FCS TS 3.2.
CEOG STS page B 3.4.4-2	Not applicable because the Bases markup deletes text in the Standard TS Bases that is not included in the FCS TS Bases.
CEOG STS page B 3.4.5-2	Not applicable because the Bases markup deletes text in the Standard TS Bases that is not included in the FCS TS Bases.
CEOG STS page B 3.4.6-2	Not applicable because the Bases markup deletes text in the Standard TS Bases that is not included in the FCS TS Bases.
CEOG STS page B 3.4.7-3	Not applicable because the Bases markup deletes text in the Standard TS Bases that is not included in the FCS TS Bases.
CEOG STS page B 3.4.13-2	Insert B 3.4.13 A on CEOG STS page B 3.4.13-2 is addressed as noted above. The remaining markups are not applicable because they revise text that is not in the FCS TS Bases.
CEOG STS page B 3.4.13-3	Deletion of item "d." is not applicable because the deleted text is not included in the FCS TS Bases. Insert B 3.4.13 B on CEOG STS page B 3.4.13-3 is addressed as described above.

Location of TSTF-449 Revisions in FCS TS	
TSTF-449	FCS TS
CEOG STS page B 3.4.13-4	The revisions in the ACTION section were incorporated in FCS TS Basis 2.1.4. In the SURVEILLANCE REQUIREMENTS section, deletion of the text regarding measurement of primary to secondary leakage using a water balance method is not applicable because the FCS TS do not include that method for determining primary to secondary leakage. The essence of the markups explaining the notes is incorporated in the Basis for FCS TS 3.2.
CEOG STS page B 3.4.13-5	Inserts B 3.4.13 C, B 3.4.13 D (CEOG) and B 3.4.13 E on CEOG STS page 3.4.13-4 are addressed above.
CEOG STS pages B 3.4.18-1 through B 3.4.18-5, excluding the SURVEILLANCE REQUIREMENTS discussion on page B 3.4.18-5.	Incorporated in the Basis for FCS TS 2.23.
CEOG STS page B 3.4.18-5, beginning with the SURVEILLANCE REQUIREMENTS discussion, through page B 3.4.18-6.	Incorporated in the Basis for FCS TS 3.17.
CEOG STS page B 3.4.18-7	The cited references are incorporated in the Bases for FCS TS 2.23 and 3.17, as appropriate.

ATTACHMENT 5

Responses to Requests for Additional Information Related to the May 30, 2006 TSTF-449 Submittal

Responses to Requests for Additional Information Related to the May 30, 2006 TSTF-449 Submittal

NRC Request #1

In Technical Specification (TS) 2.1.1(6), you proposed that "each steam generator shall be demonstrated operable by the performance of the requirements specified in Section 3.17 prior to exceeding a reactor coolant temperature of 300°F."

In addition, you indicate that steam generator operability can be achieved by verifying tube integrity in accordance with the Steam Generator Program and by verifying that each inspected tube that satisfies the tube repair criteria is plugged prior to exceeding a cold leg temperature of 210°F. As currently written, steam generator operability can be demonstrated by simply performing the surveillance requirement.

TSTF-449 expanded the definition of steam generator operability by deleting the phrase that operability is in accordance with the steam generator tube surveillance program. Please discuss your plans for making your proposal consistent with TSTF-449 (in TS 2.1.1 and TS 3.17).

In addition, given the unique structure of your TS, please clarify why TS 2.1.1(6) is still needed in light of the proposed addition of TS 2.23 which will establish new limiting conditions for operation for the steam generators.

OPPD Response:

The sentence in TS 2.1.1(6), concerning demonstrating that each SG is operable by performance of an inservice inspection program, is duplicate information to TS 2.23, which specifies the requirements for steam generator tube integrity as part of the reactor coolant pressure boundary in more detail, and will be deleted.

NRC Request #2

There are inconsistencies between the temperature in TS 2.1.1(6) (temperature of 300°F), TS 3.17(2) (cold-leg temperature of 210°F), and TSTF-449 (average coolant temperature of 200°F). In addition, there are no limits on the reactivity condition in your proposed TS requirements; unlike TSTF-449. Please discuss your plans for modifying your proposal to make it consistent with TSTF-449.

OPPD Response:

The T_{cold} 210°F requirement is used in other technical specifications. TS 2.1.1(6) will be deleted in response to question 1.

NRC Request #3

It would appear that the title for entry 8b in Table 3-5 would more appropriately be "primary- to-secondary leakage" rather than steam generator tube integrity. Please discuss why steam generator tube integrity was chosen rather than primary-to-secondary leakage. Alternatively, modify your proposal to indicate primary-to-secondary leakage for item 8b. The staff notes that the leakage limit will not ensure steam generator tube integrity.

OPPD Response:

The table entry will be revised to labeled primary-to-secondary leakage per the Staff request.

NRC Request #4

In Table 3-5, the frequency is labeled as "D." Please discuss where "D" is defined within the specification. Alternatively, discuss your plans for specifying that the frequency is "daily."

OPPD Response:

TS 3.0.2 defines the letter designations.

NRC Request #5

In Table 3-5, the "Test" for "primary to secondary leakage" is listed as "evaluate". The meaning of this term is not clear. Isn't the "Test" for primary to secondary leakage, continuous monitoring of the effluent (steam and feedwater systems) for radioactive isotopes or performing radiochemical analyses of grab samples of the steam and feedwater systems? Similarly, isn't the "Test" for reactor coolant system leakage, a water inventory balance? Please clarify.

OPPD Response:

The technical basis is provided in the last paragraph of TS 3.2: "The primary to secondary LEAKAGE is determined using the continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines." The table 3.5 test will be modified to change "evaluate" by replacing it with "Continuous process radiation monitors or radiochemical grab sampling".

NRC Request #6

In the second paragraph on page 5 of TS 3.2, it would appear that the reference to the limiting condition for operation should be TS 2.23 rather than TS 3.17 since TS 3.17 are the surveillance requirements. Please discuss your plans for modifying this reference.

OPPD Response:

OPPD agrees and Page 5 of TS 3.2, will be modified to reference the limiting condition for operation TS 2.23.

NRC Request #7

Proposed TS 2.23(2)(b) indicates that a tube should be plugged prior to entering Mode 4 following the next refueling outage or SG tube inspection. Given that Mode 4 is Cold Shutdown at your plant, this would require you to be in a refueling outage (i.e., prior to entering Mode 4). To be more consistent with the applicability section of TS 2.23 and TSTF-449, would it be more appropriate to indicate that the tube(s) should be plugged "prior to exiting Mode 4." Please discuss. If this specification is changed, the bases on 2.23-Page 4 will also need to be modified.

OPPD Response:

OPPD agrees and Page 5 of TS 2.23(b) including the bases on 2.23-Page 4 will be modified to indicate that the tube(s) should be plugged prior to a cold-leg temperature of 210°F. This change is also consistent with the statement in the last paragraph in the TS 3.17 basis.

NRC Request #8

In proposed TS 2.23, it would appear that TS 2.23(2) would permit you to elect not to plug a tube provided the conditions in TS 2.23(2)(a) and TS 2.23(2)(b) were met. This is not consistent with TSTF-449. In TSTF-449, the required actions are intended to apply only in the event a tube was inadvertently identified as not being plugged rather than electing not to plug a tube. TS 2.23 should be worded such that the plugging of SG tubes that meet the repair criteria cannot be interpreted as an elective action. Suggested wording for TS 2.23(2) that is also consistent with TS 2.1.4 would be, "If the requirements of (1)(b) above are not met for one or more SG tubes, then perform the following." Please discuss your plans to clarify your technical specifications in this regard. In addition, discuss your plans to clearly indicate that "separate condition entry" is only allowed for TS 2.23(2).

OPPD Response:

OPPD agrees and TS 2.23(2) will be reworded to: "If the requirements of (1)(b) above are not met for one or more SG tubes, then perform the following."

NRC Request #9

In your Basis for TS 2.23, you indicate that "large differential pressures across SG tubes can only be experienced in MODE 1, 2, or 3". It is not clear that this is a true statement for your facility since MODE 3 is when the average temperature is greater than 515°F and MODE 4 is when the cold-leg temperature is less than 210°F. It would appear that large differential pressures could occur when the temperature is 510°F (which is before MODE 4). Please discuss whether large differential pressures could occur when the reactor coolant temperatures are "between" those defined in MODES 3 and 4. If so, discuss your plans to modify your basis.

OPPD Response:

OPPD agrees and Bases for TS 2.23(2) will be reworded to indicate that large differential pressures across SG tubes can only be experienced when the Tcold is > 210°F.

NRC Request #10

TS 3.17(1) indicates that tube integrity will be verified "at a frequency defined by the Steam Generator Program." The phrase "at a frequency defined by the Steam Generator Program" is not included in TSTF-449. The frequency specified in the Steam Generator Program (TS 5.23) is a maximum inspection interval and the actual intervals may need to be less to ensure tube integrity is maintained. Please discuss your plans to remove this phrase from your proposal (to avoid the possibility that this phrase could be referring to the maximum interval specified in TS 5.23).

OPPD Response:

OPPD agrees to truncate the sentence to delete the phrase to match the TSTF-449.

NRC Request #11

NRC Request #12

In the last sentence of TS 5.23a, there appears to be a typographical error. The sentence should read, "...tubes are inspected or plugged to confirm....".

OPPD Response:

This change will be made.

NRC Request #13

In TS 5.23.b.3, TS 2.1.4 is referred to as "Reactor Coolant System Operational Leakage.". The actual title is "Reactor Coolant System Leakage Limits." Please discuss your plans for correcting this typographical error. The staff notes that the correct title is used in TS 2.23 - page 3 (although without the opening quote marks).

OPPD Response:

This change will be made.

NRC Request #14

In TS 5.23.d.2, the parentheses ("]") at the end of the sentence should be deleted. Please discuss your plans to correct this apparent typographical error.

OPPD Response:

This change will be made.