December 15, 2005

WBN-TS-05-10

10 CFR 50.90

U. S. Nuclear Regulatory Commission
Mail Stop: OFWN P1-35
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket No. 50-390
Tennessee Valley Authority	)	

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - TECHNICAL SPECIFICATIONS (TS) CHANGE WBN-TS-05-10 - APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

Pursuant to 10 CFR 50.90, TVA is submitting a request for a TS change (WBN-TS-05-10) to License NPF-90 for WBN Unit 1. The proposed TS change revises the Unit 1 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).

Enclosure 1 provides a description of the proposed change and confirmation of applicability. Enclosure 2 provides the existing TS pages marked-up to show the proposed change. These changes are consistent with TSTF-449, Revision 4. Enclosure 3 provides the applicable TS Bases pages associated with the TS change.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the TS change qualifies for categorical exclusion from environmental review pursuant to the provisions of

DEC 1 5 2005

10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Tennessee State Department of Public Health.

TVA requests approval of the proposed license amendment by September 2006, to support implementation for replacement steam generators that will be installed during the Unit 1 Cycle 7 outage. TVA requests that implementation of the revised TS be prior to Mode 4 during startup from the Unit 1 Cycle 7 outage.

There are no regulatory commitments associated with this submittal. If you have any questions concerning this matter, please contact me at (423) 365-1824.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 15th day of December.2005.

Sincerely,

P. L. Pace
Manager, Site Licensing
and Industry Affairs

Enclosures:
1. TVA Evaluation of the Proposed Changes
2. Proposed Technical Specifications Changes (mark-up)
3. Changes to Technical Specifications Bases pages

cc: See Page 2

Enclosures
cc (Enclosures):

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
DOCKET NO. 50-390

PROPOSED LICENSE AMENDMENT REQUEST WBN-TS-05-10
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-90 for WBN Unit 1.

The proposed changes would revise the Operating License to change the requirements in the Technical Specifications (TSs) related to steam generator tube integrity. The TS changes are consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, *Steam Generator Tube Integrity*, Revision 4. The availability of this TS improvement was announced in the *Federal Register* on May 6, 2005, (70 FR 24126) as part of the Consolidated Line Item Improvement Process (CLIIP).

2.0 PROPOSED CHANGE

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

- Revised TS definition of "LEAKAGE"
- Revised TS 3.4.13, "*RCS Operational LEAKAGE,*"
- Added new TS 3.4.17, "*Steam Generator Tube Integrity,*"
- Revised TS 5.7.2.12, "*Steam Generator (SG) Tube Surveillance Program,*"
- Revised TS 5.9.9, "*SG Tube Inspection Report,*"

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

3.0 BACKGROUND

The background for this application is adequately addressed by the NRC Notice of Availability published on May 6, 2005

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

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The applicable regulatory requirements and guidance associated with this application are adequately addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

5.0 TECHNICAL ANALYSIS

TVA 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. TVA has concluded that the justifications, presented in the TSTF proposal and the SE prepared by NRC, are applicable to WBN Unit 1 and justify this amendment for the incorporation of the changes to the WBN TSs.

6.0 REGULATORY SAFETY 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 Technical Specification Task Force (TSTF) 449, Revision 4.

6.1 Verification and commitments

The following information concerning the replacement steam generators is provided to support the NRC staff’s review of this amendment application:

Plant Name, Unit No.	Watts Bar Nuclear Plant Unit 1
Steam Generator (SG) Model(s):	Westinghouse Electric Company Model 68AXP
Effective Full Power Years (EFPY) of service for currently installed SGs	TVA is replacing the current SGs in the WBN Unit 1 Cycle 7 Refueling Outage in September 2006. Therefore, the TS changes addressed by this amendment request will be in affect when the SGs are initially placed in service and the EFPY for the new SGs will be Zero (as of September 2006).
Tubing Material (e.g., 600M, 600TT, 660TT)	690TT
Number of tubes per SG	5,128
Number and percentage of tubes plugged in each SG	SG 1: 0, 0 SG 2: 1, 0.02% SG 3: 0, 0 SG 4: 0, 0
Number of tubes repaired in each SG	None
Degradation mechanism(s)	None
Current primary-to-secondary leakage limits:	In accordance with TS 3.4.13: • 150 gallons per day through any one SG Leak rates are evaluated at 70 degrees Fahrenheit
Approved Alternate Tube Repair Criteria (ARC)	None. Voltage Based ARC for Outside Diameter Stress Corrosion Cracking in the Tube Support Plate and the use of the F-star (F*) criteria will be deleted as part of this TS change.
Approved SG Tube Repair Methods	None. Sleeving repair method will be deleted as part of this TS change.
Performance criteria for accident leakage	Primary-to-secondary leak rate values assumed in WBN's licensing basis accident analysis is 0.1 gpm for the non-faulted SG and 3 gpm for the faulted SG (assumed at 70 degrees Fahrenheit temperature condition). The 3 gpm leakage limit is the approved limit for ODSCC ARC. WBN Unit 1 replacement SGs will no longer apply the ARC. Accordingly, the accident induced leakage is conservatively limited to 1 gpm for the faulted SG.

6.2 No Significant Hazards Consideration

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

7.0 ENVIRONMENTAL EVALUATION

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

8.0 PRECEDENT

This application is being made in accordance with the CLIIP. TVA is not proposing variations or deviations from the TS changes described in TSTF-449, Revision 4, or the NRC staff's model SE published on March 2, 2005 (70 FR 10298).

8.0 REFERENCES

1. *Federal Register* Notice - Notice for Comment published on March 2, 2005 (70 CFR 10298).

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

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
PROPOSED LICENSE AMENDMENT REQUEST WBN-TS-05-10

PROPOSED TECHNICAL SPECIFICATIONS CHANGES (MARK-UP)

I. Affected Page List

1.1-4
3.4-30
3.4-31
3.4-43
3.4.44
5.0-15
5.0-16
5.0-17
5.0-18
5.0-19
5.0-19a
5.0-19b
5.0-19c
5.0-20
5.0-20a
5.0-21
5.0-35
5.0-35a

II. MARKED PAGES

See Attached.

1.1 Definitions

LEAKAGE (continued)

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

INSERT BOLDED & DELETE STRIKE THROUGH

b. <u>Unidentified LEAKAGE</u>

All LEAKAGE (except RCP seal water injection or leakoff) t is not identified LEAKAGE;

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

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

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE-OPERABILITY

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

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

INSERT BOLDED AND DELETE STRIKE THROUGH

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; **and**

~~d. 600 gallons per day total primary to secondary LEAKAGE through all steam generators (SGs);~~

d.~~e~~. 150 gallons per day primary to secondary LEAKAGE through any one **steam generator (**SG~~); and~~

~~f. Four SGs shall be OPERABLE.~~

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS **operational** LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE **or primary-to-secondary LEAKAGE.**	A.1 Reduce LEAKAGE to within limits.	4 hours
~~B. Steam Generator Tube Surveillance Program not met.~~	~~B.1 Determine steam generator tube integrity is acceptable for continued operation~~	~~4 hours~~

INSERT BOLDED AND DELETE STRIKE THROUGH

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B.~~C.~~ Required Action and associated Completion Time of Condition A ~~or B~~ not met. OR Pressure boundary LEAKAGE exists. OR **Primary-to-secondary LEAKAGE not within limit.**	**B**.~~C.~~1 Be in MODE 3. AND **B**.~~C.~~2 Be in MODE 5.	6 hours 36 hours

INSERT BOLDED AND DELETE STRIKE THROUGH

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	-----------------------------NOTE**S**----------------------- **1.** Not required to be performed ~~in MODE 3 or 4~~ until 12 hours **after establishment** of steady state operation. **2. Not applicable to primary-to-secondary LEAKAGE** --- **Verify RCS operational LEAKAGE is within limits by performance of** ~~Perform~~ **RCS water inventory balance.**	~~----------NOTE----------~~ ~~Only required to be performed during steady state operation~~ ----------------------------- 72 hours
SR 3.4.13.2	**----------------------------NOTE---------------------------** **Not required to be performed until 12 hours after establishment or steady state operation.** **---** Verify **primary-to-secondary LEAKAGE is ≤ 150 gallons per day through any one SG.** ~~steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program~~ .	~~In accordance with the Steam Generator Tube Surveillance Program~~ **72 hours**

INSERT BOLDED

DELETE STRIKE THROUGH

Add new Technical Specification 3.4.17

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 STEAM GENERATOR (SG) TUBE INTERITY

LCO 3.4.17 **SG tube integrity shall be maintained**

AND

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

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

--NOTE--
Separate Condition entry is allowed for each SG tube.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	**Verify steam generator tube integrity in accordance with the Steam Generator Program.**	**In accordance with the Steam Generator Program**
SR 3.4.17.2	**Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.**	**Prior to entering MODE 4 following a SG tube inspection.**

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.12 ~~Steam Generator (SG) Tube Surveillance Program~~ **INSERT A**

Each steam generator (SG) shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the provisions for inservice inspection of ASME Code Class 1, 2, and 3 components which shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a. See Specification 5.7.2.11 for applicable inspection Frequencies.

a. SG Sample Selection and Inspection - Each SG shall be determined OPERABLE during shutdown by selecting the number of steam generators according to Table 5.7.2.12-3 and inspecting at least the minimum number of SG tubes specified in Tables 5.7.2.12-1 and 5.7.2.12-2.

b. SG Tube Sample Selection and Inspection - The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Tables 5.7.2.12-1 and 5.7.2.12-2. The inservice inspection of SG tubes shall be performed at the frequencies specified in Specification 5.7.2.12.f and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.7.2.12.g. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all SGs; the tubes selected for these inspections shall be selected on a random basis except:

   1. 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 the critical areas;

   2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each SG shall include:

      a) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

      b) Tubes in those areas where experience has indicated potential problems,

      c) A tube inspection (pursuant to Specification 5.7.2.12.g) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection, and

(continued)

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

d) In addition to the samples required in 5.7.2.12.b.2.a) through c), all tubes which have had the F* criterion applied will be inspected in the tubesheet region. These F* tubes may be excluded from 5.7.2.12.b.2.a, provided the only previous wall penetration of greater than 20% was located below the F* distance of 1.40 inches (which includes NDE uncertainty) extending from either the bottom of the steam generator tube roll transition or the top of the tubesheet, whichever is lower in elevation.

e) Indications left in service at the flow distribution baffles and tube support plate elevations as a result of the application of the tube support plate voltage repair criteria shall be inspected by bobbin probe during all future refueling outages.

c. Examination Results - The results of each sample inspection shall be classified into one of the following three categories:

C-1 Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

C-2 One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.

C-3 More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

------------------------NOTE-------------------------
In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.
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d. Supplemental Sampling Requirements - The tubes selected as the second and third samples (if required by Tables 5.7.2.12-1 and 5.7.2.12-2) may be subjected to a partial tube inspection provided:

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

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

(continued)

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

e. Supplemental Inspection Requirements - Implementation of the steam generator tube to tube support plate repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections (including the flow distribution baffles) down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

f. Inspection frequency - The above required inservice inspections of the SG tubes shall be performed at the following frequencies:

1. The first inservice inspection shall be performed after 6 effective full power months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

2. If the results of the inservice inspection of a SG conducted in accordance with Tables 5.7.2.12-1 and 5.7.2.12-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.7.2.12.f.1; the interval may then be extended to a maximum of once per 40 months; and

3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Tables 5.7.2.12-1 and 5.7.2.12-2 during the shutdown subsequent to any of the following conditions:

   a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13, or

   b) A seismic occurrence greater than the Operating Basis Earthquake, or

(continued)

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

d) A main steam line or feedwater line break.

g. Acceptance Criteria

1. Terms as used in this specification will be defined as follows:

a) Degradation - A service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;

b) Degraded Tube - A tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

c) % Degradation - The percentage of the tube wall thickness affected or removed by degradation;

d) Defect - An imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

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

f) Plugging Limit - The imperfection depth at or beyond which the tube shall be removed from service by plugging, or repaired by sleeving in the affected area. The plugging and repair criteria are specified as follows:

A sleeved tube shall be plugged if an imperfection is detected in a Westinghouse Alloy 800 leak-limiting sleeve.

A sleeved tube shall be plugged if an imperfection is detected in the pressure boundary portion of the original tube wall in the Westinghouse Alloy 800 leak-limiting sleeve/tube assembly (i.e., at the sleeve-tube joint(s)).

A tube shall be plugged or repaired if the depth of an imperfection in the original tube wall is greater than or equal to 40% of the nominal wall. This definition does not apply to imperfections detected in the non-pressure boundary portion

(continued)

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

of the original tube wall associated with a sleeve. This definition does not apply to the portion of the original tube in the tubesheet below the F* distance provided the tube does not have a sleeve installed in the tubesheet region and the tube is not degraded within the F* distance.

For tubes to which the F* criteria is applied, a minimum of 1.5 inches of the tube into the tubesheet from the top of the tubesheet or from the bottom of the roll transition, whichever is lower in elevation, shall be inspected using rotating pancake coil eddy current technique or an inspection method shown to give equivalent or better information on the orientation and length of cracking. A minimum of 1.40 inches (which includes NDE uncertainty) of continuous, sound expanded tube must be established, extending from either the bottom of the roll transition or the top of the tubesheet, whichever is lower in elevation, to the uppermost extent of the indication.

This definition does not apply to flow distribution baffles and tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 5.7.2.12.g.1.l for repair limit applicable to these intersections.

g) Preservice Inspection - An inspection of the full length of each tube in each SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial MODE 1 operation using the equipment and techniques expected to be used during subsequent inservice inspections.

h) Tube Inspection - An inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

i) Unserviceable - The condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operational Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break accident as specified in Specification 5.7.2.12.f.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

j) F* Distance is the distance into the tubesheet from the bottom of the steam generator tube roll transition or the top of the tubesheet, whichever is lower in elevation (further into the tubesheet), that has been conservatively chosen to be 1.40 inches (which includes NDE uncertainty).

k) F* Tube is the tube with degradation equal to or greater than 40%, below the F* distance and not degraded (i.e., no indications of degradation) within the F* distance.

l) The Tube Support Plate Repair Limit - The Tube Support Plate Repair Limit is used for the disposition of Alloy 600 steam generator tubes for continued service that are experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates and flow distribution baffle (FDB). At tube support plate intersections (and FDB), the repair limit is based on maintaining steam generator tube serviceability as described below:

1. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the flow distribution baffles and tube support plates with bobbin voltages less than or equal to the lower voltage repair limit of 1.0 volt will be allowed to remain in service.

2. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the flow distribution baffles and tube support plates with the bobbin voltage greater than the lower voltage repair limit of 1.0 volt, will be repaired, except as noted in Specification 5.7.2.12.g.1.l.3 below.

3. Steam generator tubes with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the flow distribution baffles

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

and tube support plates with a bobbin voltage greater than the lower voltage repair limit of 1.0 volt but less than or equal to the upper voltage limit*, may remain inservice if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit* will be plugged or repaired.

4. Certain intersections as identified in Attachment 2 of WAT-D-10709 ("Tennessee Valley Authority, Watts Bar Nuclear Power Plant Unit 1, Application for Implementation of Voltage Based Repair Criteria, Westinghouse Steam Generator Tubes Affected by ODSCC at TSPs," J. W. Irons, Revision 0, 1/12/00) will be excluded from application of the voltage-based repair criteria as it is determined that these intersection may collapse or deform following a postulated LOCA + SSE event.

5. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.7.2.12.g.1.I.1, 5.7.2.12.g.1.I.2, and 5.7.2.12.g.1.I.3.

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{VSL}{1.0 + NDE + Gr\left[\frac{CL - \Delta t}{CL}\right]}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left[\frac{CL - \Delta t}{CL}\right]$$

where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e. a value of 20-percent has been approved by the NRC).

Implementation of these mid-cycle repair limits should follow the same approach used in specifications 5.7.2.12.g.1.I.1, 5.7.2.12.g.1.I.2, and 5.7.2.12.g.1.I.3.

* The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented. V_{URL} will differ at the tube support plates and flow distribution baffle.

m) Tube Repair refers to a process that reestablishes tube serviceability. Tube repair of defective tubes will be performed by installation of the Westinghouse Alloy 800 leak-limiting repair sleeve as described in the proprietary Westinghouse Report WCAP-15918-P, Revision 00, (Draft CEN-633-P, Revision 05-P), “Steam Generator Tube Repair For Combustion Engineering and Westinghouse Designed Plants with ¾ Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves.”

2. The SG shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Tables 5.7.2.12-1 and 5.7.2.12-2.

h. Reports - The content and frequency of written reports shall be in accordance with Specification 5.9.9.

(continued)

5.7 Procedures, Programs, and Manuals (continued)

TABLE 5.7.2.12-1

STEAM GENERATOR TUBE INSPECTION SUPPLEMENTAL SAMPLING REQUIREMENTS

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per SG	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or Repair defective tubes and inspect an additional 2S tubes in this SG.	C-1	None	N/A	N/A
			C-2	Plug or Repair defective tubes and inspect an additional 4S tubes in this SG.	C-1	N/A
					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.	N/A	N/A
	C-3	Inspect all tubes in this SG, plug or repair defective tubes and inspect 2S tubes in each other SG. Notification to NRC pursuant to 10CFR50.72	All other SGs C-1	None	N/A	N/A
			Some SGs C-2 but no other is C-3	Perform action for C-2 result of second sample.	N/A	N/A
			Additional SG is C-3	Inspect all tubes in each SG and plug or repair defective tubes. Notification to NRC pursuant to 10CFR50.72.	N/A	N/A

S = 3 N/n % Where N is the number of SGs in the unit and n is the number of S.G.s inspected during an inspection.

(continued)

5.7 Procedures, Programs, and Manuals (continued)

TABLE 5.7.2.12-2

STEAM GENERATOR REPAIRED TUBE INSPECTION
SAMPLING REQUIREMENTS

1st Sample Inspection			2nd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of repaired tubes	C-1	None	N/A	N/A
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this SG	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample.
	C-3	Inspect all repaired tubes in this SG, plug defective repaired tubes and inspect 20% of the repaired tubes in each other SG. Notification to NRC pursuant to 10CFR50.72	All other SGs C-1	None
			Some SGs C-2 but no other is C-3	Perform action for C-2 result of first sample.
			Additional SG is C-3	Inspect all repaired tubes in each SG and plug defective repaired tubes. Notification to NRC pursuant to 10CFR50.72.

(continued)

5.7 Procedures, Programs, and Manuals (continued)

TABLE 5.7.2.12-3

MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	All
First Inservice Inspection	Two
Second and Subsequent Inservice Inspections	One[1]

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

   One of the other two SGs not inspected during the first inservice inspections shall be inspected during the second inspection period and the remaining SG shall be inspected during the third inspection period. The fourth and subsequent inspections shall follow the instructions described above.

5.7.2.13 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

(continued)

5.9 Reporting Requirements (continued)

5.9.7 EDG Failures Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.4, or existing Regulatory Guide 1.108 reporting requirement.

5.9.8 PAMS Report

When a Report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.9.9 SG Tube Inspection Report

INSERT B

~~Following each inservice inspection of steam generator (SG) tubes, in accordance with the SG Tube Surveillance Program, the number of tubes plugged and tubes sleeved in each SG shall be reported to the NRC within 15 days.~~

~~The complete results of the SG tube inservice inspection shall be submitted to the NRC within 12 months following the completion of the inspection. The report shall include:~~

1. ~~Number and extent of tubes inspected,~~
2. ~~Location and percent of wall-thickness penetration for each indication of an imperfection, and~~
3. ~~Identification of tubes plugged.~~

~~Results of SG tube inspections that fall into Category C-3 shall be reported to the NRC in accordance with 10 CFR 50.72. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

~~The results of the inspection of F* tubes shall be reported to the Commission in accordance with 10 CFR 50.4, prior to the restart of the unit. This report shall include:~~

1. ~~Identification of F* tubes.~~
2. ~~Uppermost elevation of the degradation and extent of the degradation.~~

~~NRC approval of this report is not required prior to restart.~~

~~For implementation of the voltage based repair criteria to tube support plate (and flow distribution baffle) intersections, notify the NRC prior to returning the steam generators to service should any of the following conditions arise:~~

(continued)

5.9 Reporting Requirements

5.9.9 SG Tube Inspection Report (continued)

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leakage limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersection and flow distribution baffles.

3. If indications are identified that extend beyond the confines of the tube support plate and flow distribution baffles.

4. If indications are identified at the tube support plate and flow distribution baffle elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

5.7.2.12 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 and/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, cooldown, 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. The accident induced leakage is not to exceed 1.0 gpm for the faulted SG.

3. The operational leakage performance criterion is specified in LCO 3.4.13, “RCS Operational LEAKAGE.”

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

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

INSERT A (continued)

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 SGs 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.

INSERT B

5.9 Reporting Requirements

5.9.9 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.7.2.12, 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.

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
PROPOSED LICENSE AMENDMENT REQUEST WBN-TS-05-10

PROPOSED TECHNICAL SPECIFICATIONS BASES CHANGES

I. AFFECTED PAGE LIST

B3.4-19
B3.4-23
B3.4-29
B3.4-35
B3.4-74a
B3.4-74b
B3.4-75
B3.4-76
B3.4-77
B3.4-78
B3.4-79
B3.4-80
B3.4-80a
B3.4-99
B3.4-100
B3.4-101
B3.4-102
B3.4-103
B3.4-104

II. ANNOTATED PAGES

See Attached.

BASES

LCO (continued)

OPERABLE SG. ~~in accordance with the Steam Generator Tube Surveillance Program.~~

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

BASES

LCO (continued)

If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG ~~**in accordance with the Steam Generator Tube Surveillance Program**~~, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with RTBs in the closed position. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the RTBs open.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";

(continued)

BASES

LCO (continued)

The Note requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ 350°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG **~~in accordance with the Steam Generator Tube Surveillance Program~~**, which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

(continued)

BASES

LCO (continued)

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. ~~**An OPERABLE**~~ **A** SG can perform as a heat sink when it has an adequate water level and is OPERABLE ~~**in accordance with the Steam Generator Tube Surveillance Program.**~~

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be ≥ 6% narrow range.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side water levels < 6% narrow range redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE, all

(continued)

~~**BACKGROUND (continued)**~~

~~**The voltage based repair limit of Specification 5.7.2.12.g.1.I implements the guidance of Generic Letter (GL) 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections and flow distribution baffles. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate and flow distribution baffles. Refer to GL 95-05 for additional description of the degradation morphology.**~~

~~**Implementation of Specification 5.7.2.12.g.1.I requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this Specification).**~~

~~**The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation at the 95% prediction interval curve reduced to account for the lower 95/95% tolerance bound for tubing material properties at 650°F (i.e. the 95% lower tolerance limit (LTL) curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit,**~~ $\cancel{V_{URL}}$, ~~**is determined from the structural voltage limit by applying the following equation:**~~

$$\cancel{V_{URL} = V_{SL} - V_{GR} - V_{NDE}}$$

~~**where**~~ $\cancel{V_{GR}}$ ~~**represents the allowance for flaw growth between inspections and**~~ $\cancel{V_{NDE}}$ ~~**represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.**~~

~~**The mid-cycle equation in Specification 5.7.2.12.g.1.I should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates and flow distribution baffles.**~~

(continued)

~~**BACKGROUND (continued)**~~

~~**Specification 5.9.9 implements several reporting requirements recommended by GL 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected end-of-cycle (EOC) voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL Section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL Section 6.b(c) criteria.**~~

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety **analysis for a main steam line break (MSLB) assumes that the pre-accident primary to secondary LEAKAGE from three steam generators is 150 gallons per day (gpd) per steam generator and 1 gallon per minute (gpm) from one steam generator. This leakage assumption remains the same after the accident. For a SGTR accident, the accident analysis assumes a primary to secondary leakage of 150 gpd per steam generator prior to the accident. Subsequent to the SGTR a leakage of 150 gpd is assumed in each of three intact steam generators and RCS blowdown flow through the ruptured tube in the faulted steam generator. Consequently, the LCO requirement to limit primary to secondary LEAKAGE through any one steam generator to less than or equal to 150 gpd is acceptable.**

~~**The SLB is more limiting for site radiation releases.**~~ The safety analysis for the SLB accident assumes **the entire ~~3~~ 1** gpm primary to secondary LEAKAGE ~~**in the**~~ **is through the affected ~~faulted~~** steam generator ~~**(during the accident)**~~ **as an initial condition.** ~~**and 150 gpd in the intact steam generators (Ref. 4).**~~ The dose consequences resulting from the SLB accident are within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of ~~**the NRC Policy Statement.**~~**10 CFR 50.36(c)(2)(ii).**

`

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

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

c. Identified LEAKAGE

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

~~**d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)**~~

~~**The LEAKAGE limits incorporated into LCOs 3.4.13.d and 3.4.13.e are more restrictive than the standard operating LEAKAGE limits and are intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shut down in a timely manner.**~~

~~**The 600 gallons per day total primary to secondary LEAKAGE through all SGs ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 limits in the event of either a steam generator tube rupture or steam line break. The limit is consistent with, or conservative to, the assumptions used in the analysis of these accidents.**~~

(continued)

LCO (continued)

~~e~~d. Primary to Secondary LEAKAGE through Any One SG

~~The 150 gallons per day primary to secondary LEAKAGE through any one SG ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions. The limit is consistent with, or conservative to, the assumptions used in the analysis of these accidents.~~
The limit of 150 gallons per day (gpd) per SG (600 gpd total for all SG) is based on the operational LEAKAGE performance criteria in NEI 97-06, Steam Generator Program Guidelines (Reference 4). 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.

~~f~~ ~~Steam Generator OPERABILITY~~

~~Four SGs are also required to be OPERABLE. This requirement is met by satisfying the augmented inservice inspection requirements of the Steam Generator Tube Surveillance Program (Specification 5.7.2.12).~~

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE **or** identified LEAKAGE, **~~or primary to secondary LEAKAGE~~** in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

(continued)

BASES

ACTIONS (continued)

~~**B.1**~~

~~**With the Steam Generator Tube Surveillance Program (Specification 5.7.2.12) not met, integrity of the steam generator tubes must be determined to be acceptable for continued operation within 4 hours. This Condition specifically addresses the appropriate actions to be taken in the event a non-significant Program discrepancy is discovered with the plant operating in MODES 1, 2, 3, or 4. Examples of this type of discrepancy include administrative (e.g., documentation of inspection results) or similar deviation which do not result in inadequate tube integrity. The 4 hour Completion Time allows a reasonable period of time for the correction of administrative-only problems or for the plant to contact the NRC to discuss appropriate action. The 4 hour Completion Time is based on engineering judgement.**~~

~~**This Condition does not supersede the ACTIONS of Condition A in the event LEAKAGE from one or more steam generators exceeds the LCO limit. In the event this occurs, the LEAKAGE must be restored to within limit within 4 hours, or a unit shutdown commenced. This Condition is also not applicable to a situation in which integrity of the tube is questionable. In the event integrity of the tube is determined to be inadequate, this Condition is no longer applicable and Condition C of this LCO should be entered immediately.**~~

~~C.1 and C.2~~ B.1 and B.2

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 within limits** ~~**if the Required Actions of Conditions A or B cannot be completed**~~ within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

(continued)

BASES

ACTIONS

~~**C.1 and C.2**~~ **B.1 and B.2** (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. ~~**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.**~~

The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. ~~**Therefore, this SR**~~ **The SR is modified by 2 Notes. Note 1 states that this SR** is not required to be performed until 12 hours **after establishing** steady state operation. ~~**near operating pressure have been established**~~. **The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.**

Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful. ~~**and a Note requires the Surveillance to be met when steady state is established**~~. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment pocket sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

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

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. ~~**A Note under the Frequency column states that this SR is required to be performed during steady state operation.**~~

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17 “Steam Generator Tube Integrity,” should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Ref. 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 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with EPRI guidelines (Ref. 5).

(continued)

BASES

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criteria 30, "Quality of Reactor Coolant Boundary."

2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3. Watts Bar FSAR, Section 15.4, "Condition IV - Limiting Faults."

~~**4. NRC Safety Evaluation for Watts Bar Nuclear Plant Unit 1, Amendment 38, for Steam Generator Tubing Voltage Based Alternate Repair Criteria for Outside Diameter Stress Corrosion Cracking (ODSCC), dated February 26, 2002.**~~

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

5. EPRI Pressurized Water Reactor Primary-to-Secondary Leak Guidelines.

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4 "RCS Loops – MODES 1and 2," LCO 3.4.5, "RCS Loops – MODE 3," LCO 3.4.6, "RCS Loops – MODE 4," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled."

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

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.7.2.12 "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.7.2.12, 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 5.7.2.12. 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).

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APPLICABLE SAFETY ANALYSES

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 LCO 3.4.13, “RCS Operational LEAKAGE,” plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

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 150 gallons per day (gpd) per steam generator and 1 gallon per minute (gpm) in the faulted steam generator. 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 LCO 3.4.16 “RCS Specific Activity,” limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), and 10 CFR 100 (Ref. 3).

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

LCO

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 5.7.2.12, “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.

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LCO
(continued)

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

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, “The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation.” Tube collapse is defined as, “For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero.” The structural integrity performance criterion provides guidance on assessing loads that 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 in the faulted 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.

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LCO (continued)	The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.
APPLICABILITY	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 in MODE 1, 2, 3, or 4. RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.
ACTIONS	The ACTIONS are modified by a Note 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. <u>A.1 and A.2</u> Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged, has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained until the next SG inspection, Condition B applies.

(continued)

BASES

ACTIONS (continued)

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 A.2 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 entering MODE 4 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.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A 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 5 within the 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.
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SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the “as found” condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1 (continued)

The Steam Generator Program defines the Frequency of SR 3.4.17.1. 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 5.7.2.12 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

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 5.7.2.12 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 entering MODE 4 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, Control Room.
3. 10 CFR 100, Reactor Site Criteria.
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.”