~~**Proprietary Information Withhold from Public Disclosure Under 10 CFR § 2.390**~~.
This letter is decontrolled when separated from Enclosure 6.

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-15-060

December 15, 2015

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
Facility Operating License No. NPF-96
Docket No. 50-391

Subject: **Technical Specifications Change No. WBN2-TS-15-16 - Revise Technical Specifications for Use of Steam Generator Alternate Repair Criterion F***

References:

1. TVA Letter to NRC, “Watts Bar Nuclear Plant (WBN) - Unit 1 -Technical Specification (TS) Change No. WBN-TS-99-013 - Alternate Steam Generator Tubesheet Region Plugging Criterion (F*),” dated April 10, 2000 (ML003703068)

2. WCAP-13084-P, "Tubesheet Region Tube Alternate Plugging (F*) Criterion for the Tennessee Valley Authority Watts Bar Units 1 and 2 Nuclear Power Plant Steam Generators," dated October, 1991

3. NRC Letter to TVA, “Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment Regarding Steam Generator Tubing Alternate Repair Criteria (ARC) (TAC No. MA8635), dated February 26, 2002 (ML003748725)

In accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.90, "Application for amendment of license, construction permit, or early site permit," the Tennessee Valley Authority (TVA) is submitting a request for an amendment to Facility Operating License No. NPF-96 for Watts Bar Nuclear Plant (WBN), Unit 2.

This amendment request proposes to revise Technical Specifications (TS) 3.4.17, “Steam Generator (SG) Tube Integrity,” 5.7.2.12, “Steam Generator (SG) Program,” and 5.9.9, “Steam Generator Tube Inspection Report,” to exclude portions of the steam generator tubes below the top of the tube sheet from needing to be plugged. The proposed license amendment requests approval to use the Alternate Repair Criterion (ARC) F* repair limit of 1.64 inches (including bounding nondestructive examination (NDE) uncertainty of 0.34 inches).

TVA submitted a License Amendment Request for approval to use the ARC F* for WBN Unit 1 on April 10, 2000 (Reference 1). Attached to that License Amendment Request was the technical justification developed by Westinghouse, WCAP-13084-P (Reference 2). The NRC issued the License Amendment on September 8, 2000 (Reference 3) that allowed WBN Unit 1 to use the requested ARC (1.06 inches) specified in WCAP-13084-P, but required the combination of the F* distance with the NDE uncertainty of 0.34 inches, for an F* distance of 1.40 inches. That submittal and subsequent approval was based upon the Model D3 Westinghouse Steam Generators (SGs) which have subsequently been replaced for WBN U1. WBN Unit 2 SGs are the same Westinghouse Model D3 as the WBN Unit 1 original SGs.

Enclosure 1 provides a description, technical evaluation, regulatory evaluation, and environmental consideration of the proposed change. Attachment 1 to Enclosure 1 identifies a commitment to revise the Final Safety Analysis Report to include a reference to this letter for implementing the alternate repair criteria once approved by the NRC.

Enclosure 2 provides the existing TS pages marked-up to show the proposed changes. Enclosure 3 provides the existing TS pages retyped to show the proposed changes. Enclosure 4 provides the existing TS Bases pages marked-up to show the proposed changes. Enclosure 5 provides the existing TS Bases pages retyped to show the proposed changes. Enclosures 4 and 5 are provided for information only.

WBN Unit 2 requested Westinghouse to review industry data on similar SG designs to verify that the ARC could still be justified for use in the WBN Unit 2 SGs. Westinghouse documented their review in report SG-SGMP-13-15-P, Revision 0, “Watts Bar Nuclear Plant Unit 2 F* Alternate Repair Criterion Technical Support Document,” dated March 2014 (Enclosure 6). Report SG-SGMP-13-15-P provides technical justification for the use of the ARC F* of 1.64 inches (including bounding NDE uncertainty of 0.34 inches). A copy of SG-SGMP-13-15-P is provided in Enclosure 6.

As SG-SGMP-13-15-P contains information proprietary to Westinghouse Electric Company, it is supported by an affidavit signed by Westinghouse, the owner of the information (Enclosure 7). The affidavit sets forth the basis on which the information should be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission’s regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission’s regulations. Correspondence with respect to the copyright or proprietary aspects of the technical support document or the supporting Westinghouse affidavit should reference SG-SGMP-13-15-P and CAW-15-4140, and should be addressed to James A. Gresham, Manager Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

Enclosure 8 contains report SG-SGMP-13-15-NP, a non-proprietary version of SG-SGMP-13-15-P.

TVA requests that the NRC approve this amendment before the first refueling outage for WBN Unit 2, currently scheduled for September 5, 2016 with implementation within 60 days of issuance.

TVA has determined that there are no significant hazard considerations associated with the proposed change and that the change qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

The WBN Plant Operations Review Committee and the WBN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of WBN Unit 2 in accordance with the proposed change will not endanger the health and safety of the public.

Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and the enclosures to the Tennessee Department of Environment and Conservation.

If you have any questions regarding this submittal, please contact Gordon Arent at 423-365-2004.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 15th day of December 2015.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:

1. Evaluation of Proposed Change Regarding Watts Bar Nuclear Plant, Unit 2 License Amendment Request to Revise Technical Specifications for Use of Alternate Repair Criterion F*
2. Proposed Technical Specifications (Markups)
3. Retyped Proposed Technical Specifications
4. Proposed Technical Specifications Bases (Markups)
5. Retyped Proposed Technical Specifications Bases
6. Westinghouse Report SG-SGMP-13-15-P, Revision 0, "Watts Bar Nuclear Plant Unit 2 F* Alternate Repair Criterion Technical Support Document," March 2014
7. Westinghouse Affidavit CAW-15-4140 Supporting SG-SGMP-13-15-P
8. Westinghouse Report SG-SGMP-13-15-NP, Revision 0, "Watts Bar Nuclear Plant Unit 2 F* Alternate Repair Criterion Technical Support Document," March 2015

cc: See Page 4

cc (Enclosures):

NRC Regional Administrator – Region II
NRC Project Manager – Watts Bar Nuclear Plant
NRC Senior Resident Inspector – Watts Bar Nuclear Plant
Director, Division of Radiological Health – Tennessee State Department of Environment and Conservation (w/o Enclosure 6)

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 2

EVALUATION OF PROPOSED CHANGE

Subject: **Watts Bar Nuclear Plant, Unit 2 License Amendment Request to Revise Technical Specifications for Use of Alternate Repair Criterion F***

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1. List of Commitments

1.0 SUMMARY DESCRIPTION

In accordance with Title 10 of the *Code of Federal Regulations* (10CFR) 50.90, TVA requests an amendment to Facility Operating License No. NPF-96 for Watts Bar Nuclear Plant (WBN) Unit 2. The proposed amendment will revise WBN Unit 2 Technical Specifications (TS) 3.4.17 and 5.7.2.12 to permit the use of F* alternate repair criterion (ARC) in addition to tube plugging in the area of the Steam Generator (SG) below the top of the tube sheet. The proposed amendment will also revise the reporting requirements in TS 5.9.9.

This change is supported by Westinghouse report SG-SGMP-13-15-P, "Watts Bar Nuclear Plant Unit 2 F* Alternate Repair Criterion Technical Support Document," dated March 2014 (Reference 1). This report, provided in Enclosure 6, updates the information from Westinghouse topical report WCAP-13804-P (Reference 2) submitted in support of a WBN Unit 1 license amendment request (Reference 3) that was approved by the NRC on September 8, 2000 (Reference 4). TVA is proposing to use an F* value of 1.64 inches including uncertainty which is more conservative than the F* value of 1.4 inches approved for WBN Unit 1.

Approval of this amendment application is requested by September 5, 2016 to support SG inspection activities during the WBN Unit 2 refueling outage scheduled for the fall of 2016.

2.0 DETAILED DESCRIPTION

2.1 Proposed Changes

The proposed license amendment requests NRC's approval to use an ARC F* in the tubesheet region of the SG. The F* criterion addresses the action required when degradation has been detected in the top of the mechanically expanded portion of SG tubes within the SG tubesheet (see attached Figure 1). Existing tube repair criteria do not take into account the effect of the tubesheet on the external surface of the tube. The presence of the tubesheet will enhance the integrity of potentially degraded tubes in this region by precluding tube deformation beyond the expanded outside diameter. An axial length of sound roll expansion equal to the F* length at the top of the tube/tubesheet roll expansion provides sufficient structural integrity to preclude pull out of the tube due to pressure effects, even assuming that the tube has experienced a complete circumferential severance within the tubesheet region, at or below the bottom of the F* distance. This same axial length of roll expansion of the tube into the tubesheet also provides a barrier to significant leakage for through wall cracking of the tube in the expanded region.

The proposed change designates a portion of the tube for which tube degradation of a defined type does not necessitate remedial action, except as dictated for compliance with tube leakage limits as set forth in the WBN Unit 2 TS. As noted above, the region of the tube subject to this change is in the expanded portion of the tube within the tubesheet of the SGs. The length of mechanical expansion required to prevent pullout for all normal operating and postulated accident conditions, designated F*, has been conservatively determined to be 1.64 inches (including bounding nondestructive examination (NDE) uncertainty of 0.34 inches).

The proposed amendment would modify TS 3.4.17, “Steam Generator (SG) Tube Integrity,” 5.7.2.12, “Steam Generator (SG) Program,” and 5.9.9, “Steam Generator Tube Inspection Report,” which provides tube inspection requirements and acceptance criteria to determine the level of degradation below which the tube may remain in service. The proposed amendment would prescribe the portion of the tube subject to the F* ARC criterion.

Attachment 1 to this enclosure provides a list of commitments made by TVA in this submittal. Enclosure 2 provides the existing WBN Unit 2 TS pages marked-up to show the proposed changes. Enclosure 3 provides the existing TS pages retyped to show the proposed changes. Enclosure 4 provides the existing WBN Unit 2 TS Bases pages marked-up to show the proposed changes. Enclosure 5 provides the existing TS Bases pages retyped to show the proposed changes. The proposed Bases changes are provided to the NRC for information only.

2.2 Need for Proposed Changes

The amendment has been proposed to address potential eddy current indications of tube degradation which may occur in the roll expanded portion of the tubes within the tubesheet in the SGs at WBN Unit 2. These SGs were fabricated with a full depth roll expansion in the tubesheet. Interpretation of eddy current data from similar plants has shown a potential for primary water stress corrosion cracking (PWSCC) in the roll expanded portion of the tube within the tubesheet.

It can be shown that tube plugging is not required in many cases to maintain SG tube integrity. Using existing TS tube repair criteria, many of the tubes with potential indications would have to be removed from service. The proposed amendment would reduce the occupational radiation exposure that would be incurred by plant workers involved in tube plugging operations. The proposed amendment would minimize the loss of margin in the reactor coolant flow through the SG assumed in the loss-of-coolant-accident (LOCA) analyses and therefore, assist in assuring that minimum flow rates are maintained in excess of that required for operation at full power. Reductions in the amount of tube plugging required can reduce the length of plant outages and reduce the time that the SG is open to the containment environment during an outage.

2.3 Implementation

The TVA process governing the preparation and submittal of TS changes and License Amendment Requests (LAR) requires that the appropriate organizations (e.g., Operations, Training, Engineering, Maintenance, Chemistry, Radiation Protection, and Work Control) identify the documents that are affected by each proposed change to the TS and Operating License. Among the items that are considered are training, plant modifications, procedures, special implementation constraints, design documents, and surveillance instructions associated with TS Surveillance Requirements (SRs), Technical Requirements Manual, TS Bases, and the Updated Final Safety Analysis Report (UFSAR). The process requires that procedures and design document changes necessary to support TS Operability are approved prior to implementation of an NRC approved license amendment. The process also provides assurance that the remaining changes, if any, are scheduled and tracked for configuration control.

3.0 TECHNICAL EVALUATION

3.1 System Description

WBN Unit 2 contains four Westinghouse Model D3 recirculating pre-heater type SGs. Each SG contains 4674 mill annealed Alloy 600 tubes that have an outer diameter of 0.75 inches thick with a 0.043 inch nominal wall thickness. These SGs are the same design as the original WBN Unit 1 SGs. The SGs have a vertical shell and U-tube evaporator with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the SG. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tubesheet. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

The tubesheet is approximately 21 inches thick and the tubes are roller expanded for the full depth of the tubesheet after the ends are seal welded to the tubesheet cladding.

No significant general corrosion of the Inconel tubing is expected during the lifetime of the unit. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel-600 has excellent resistance to general and pitting type corrosion in severe operating water conditions, hence its selection for use in the SG.

Primary water stress corrosion cracking (PWSCC) of mill annealed Inconel-600 tubing has been identified as having a potentially significant impact on plant availability. The conditions necessary for this type of cracking are primarily in two locations in the SGs. The first location is in the narrow radius bends in the row 1 and 2 tubes, where the residual stresses from tube bending may be high enough to lead to PWSCC. The second location is in the area where the tube is roller-expanded into the tubesheet, where the residual stresses from rolling can lead to PWSCC.

To provide additional margin against PWSCC, procedures were performed for reduction of residual tensile stresses at or near the inner surface of the tube. The row 1 and 2 U-bends, including both tangent points, have had a thermal stress relief cycle applied by use of a resistance heater. The roller-expanded portion of the tubes within the hot leg portion of the tubesheet and at the transition between the expanded and non-expanded portion at the top of the hot leg portion of the tubesheet have had a mechanical stress modification by the application of the process known as rotopeening. The identical rotopeening (hot leg) process was also applied on the original WBN Unit 1 Model D3 SGs. The tube expansion processes used for WBN Unit 1 and 2 are identical and the rotopeening process was implemented for both units prior to operation.

3.2 Evaluation

WCAP-13084-P, “Tubesheet Region Tube Alternate Plugging (F*) Criterion for the Tennessee Valley Authority Watts Bar Units 1 and 2 Nuclear Power Plant Steam Generators,” (Reference 2) summarizes the results of analysis and testing programs that address the residual radial preload of Westinghouse SG tubes hardrolled into the tubesheet to determine the length of hardroll engagement necessary to resist tube

pullout forces during normal and faulted condition loadings. SG tube integrity requires a safety factor of 3 be maintained at normal operating pressure differential and that a safety factor of 1.4 be maintained at faulted conditions. WCAP-13084-P assumed a normal operating pressure differential of 1400 psi and a faulted condition pressure differential of 2650 psi.

When tubes have been hardrolled into the tubesheet, any axial loads developed by pressure and/or mechanical forces acting on the tubes are resisted by frictional forces developed by the residual radial preload that exists between the tube and the tubesheet. For some specific length of engagement of the hardroll, no significant axial forces will be transmitted further along the tube, and that length of tubing (i.e., F*), will be sufficient to anchor the tube in the tubesheet. The amount of interference (i.e., residual radial preload) was determined by installing tube specimens in collars specifically designed to simulate the tubesheet radial stiffness. A hardroll process representative of that used during SG manufacture was used in order to obtain specimens which would exhibit installed preload characteristics like the tubes in the tubesheet. After hardrolling, the test collars were removed from the tube specimens and the spring-back of the tube was measured. The amount of spring-back was used in an analysis to determine the magnitude of the interference fit, which is representative of the residual tube-to-tubesheet radial load in Westinghouse Model D SGs. Tube and collar measurements were performed before and after hardrolling and used to determine the radial residual preload at room temperature. Plant operating pressures and temperatures along with tubesheet bowing were considered to determine the normal operating and faulted condition radial preload forces at the top-of-tubesheet. The tubesheet hole diameter was used to determine the forces acting to remove the tube from the tubesheet at normal operating and faulted conditions. The F* engagement length was determined utilizing an assumed static coefficient of friction. In order to validate the assumed static coefficient of friction, dry pullout tests and hydraulic proof tests were performed. Dry pull-out tests were performed on rolled joints (i.e., tubes in collars) at room temperature via tensile testing machine in order to provide an indirect measurement of the static coefficient of friction between the tube and the tubesheet. The dry pull-out tests indicated the assumed static coefficient of friction was conservative by a factor of 2.7 relative to a dry interface between the tube and collar. Hydraulic proof tests were conducted on rolled joints at room temperature. Retention forces were discovered to be greater than the burst pressure of the tubing. No rolled joint greater than 0.50 inches were expelled from the collars despite some samples being subjected to pressure as high as 23,500 psi. The tubes expelled from the collars indicate the assumed static coefficient of friction was conservative by a factor greater than 2. WCAP-13084-P states that hydraulic proof test specimens did not leak until applied pressures were significantly above pressure associated with accident conditions.

The information provided in WCAP-13084-P was reviewed and supplemented by the Westinghouse report (Reference 1), provided in Enclosure 6 of this submittal. The report evaluates the continued applicability of the original F* analysis to WBN Unit 2. The report also considered and dispositioned pertinent NRC RAIs from the more recent H* licensing reviews for hydraulically expanded tubes. The original input, assumptions, and analysis tools were verified and information updated when required. The likelihood of tube pullout and the impact of increased leakage from cracks in the tubesheet region must be considered. The main steam line break is the event that produces the highest primary to secondary pressure and would be the most likely to result in tube pullout. The main steam line break (MSLB) also provides the limiting plant condition for tube leakage.

The large break LOCA is the event that produces the highest secondary side to primary side differential pressure. The original analysis of Reference 2 was recreated in Reference 1 to verify the input assumptions and analysis tools. The original WCAP analysis used material property values at a slightly lower differential temperature than the actual hot leg conditions, and made the judgment that the analysis would equally apply to both hot and cold legs. Reference 1 recalculates the F* lengths for normal operating conditions utilizing the specific differential temperature condition for both hot and cold legs and for faulted conditions utilizing the applied temperature differential assumed for both legs. The report considered the potential impact of the F* length as a function of changes in the reported material property values in three ASME Code years. The ASME Code material property values have changed over time. Three specific ASME Code versions were compared: SG Code of Record (1971), Code applicable at the time of the WCAP-13084-P analysis (1989), and the most recent ASME Code version approved by the NRC (2007 Code through 2008 Addenda). The material property values in the latest Code results in a more conservative F* value than reported in WCAP-13084-P.

Reference 1 evaluated the NDE Measurement Uncertainty of the Reference 2. The NRC has placed increased emphasis on the probabilistic evaluation of uncertainties as demonstrated by the RAIs received during the H* licensing process. Because of this, an alternate uncertainty evaluation was developed that shows that application of accepted probabilistic techniques to the eddy current uncertainty would result in a shorter distance for F*. However, for added conservatism, the original NDE measurement uncertainty wil be applied. Reference 1 evaluated the accident initiated leakage associated with the F*alternate repair criteria. When the WCAP was developed, sensitivity to very small quantities of primary-to-secondary leakage was not as great as at present. The WCAP concluded that the F* leakage from postulated tube degradation below the F* distance would be negligible, and could be ignored. Since the WCAP was developed, the industry has realized that when all the tubes in a SG contribute a very small amount to primary-to-secondary leakage, the total leakage can be significant. Leakage values and leakage probabilities were extracted from Reference 2 for various lengths. However it was concluded that postulated indications that reside below 2 inches of SRE (sound roll expansion) of the tubing will not significantly contribute to primary-to-secondary leakage during a postulated accident and therefore may be neglected. Reference 1 evaluated the effects of tubesheet bow and contact pressure loss. Roll expanded tubes create an interference fit with the tubesheet hole. That is, if the tubesheet were removed, the tube OD would enlarge due to the residual radial stresses inherent to the roll expansion process. During some accident scenarios, the tubesheet is expected to bow which would create a tubesheet hole bore enlargement. The WCAP utilized a two dimensional model for the original F* analysis. A more refined, three-dimensional tubesheet bow model was developed for the H* program. The results of the three dimensional model were compared with the results of the two dimensional model for H*. This comparison showed that the two-dimensional model yields conservative H* distance results compared to the three-dimensional model. Based on this, it is concluded that the tubesheet bow is conservatively modeled in the WCAP F* technical justification. The report evaluated the leakage prediction model and the impact of crevice flow phase assumptions. Application of the same equations used to describe the H* tests were applied to the F* leakage data and would require crevice widths of approximately 9E-6 inches for the 0.50 roll engagement length specimens which leaked. This suggests that any leakage from F* tubes would be due to surface imperfections of the tubesheet hole. Considering the extremely limited crevice width condition for F* roll

expanded tubing, any primary fluid pushed through the tube-to-tubesheet crevice would be expected to remain in a liquid state until the fluid exits the crevice at the top-of-tubesheet. Enclosure 6 evaluated a variance in the tube and tubesheet thermal expansion coefficient for sensitivity of the F* distance. An F* sensitivity case was run using the 2007 through 2008 Addenda thermal expansion coefficient data where the tube thermal expansion coefficient was reduced by 2.33% and the tubesheet thermal expansion coefficient was increased by 1.44%. This evaluation showed an increase in the F* distance of 0.14 inches. The WBN Unit 2 pressurizer power operated relief valves (PORVs) are qualified for operation during postulated accident conditions, and therefore, would be expected to be available during plant recovery from a MSLB event. Generic Letter 95-05, Attachment 1, Section 2 allows the use of PORV set-point plus 3 percent as the limiting reactor coolant system pressure. As such, the maximum reactor coolant system pressure attained during recovery from a MSLB would be approximately 2405 psig. At the maximum primary to secondary pressure differential of 2405 psig during the limiting accident conditions, with a safety factor 1.43, the proposed F* distance is conservative.

Based on the above summary of Reference 1 and Reference 2, this License Amendment Request reflects the conservative value of 1.64 inches for the F* distance which, includes a bounding NDE uncertainty of 0.34 inches.

3.3 Summary

TVA submitted a License Amendment request for approval to use the ARC F* for WBN Unit 1 on April 10, 2000 (Reference 3). Attached to that License Amendment Request was the technical justification developed by Westinghouse, WCAP-13084-P (Reference 2). The NRC issued the License Amendment on September 8, 2000 (Reference 4) that allowed WBN Unit 1 to use the requested ARC (1.06 inches) specified in WCAP-13084-P, but required the combination of the F* distance with the NDE uncertainty of 0.34 inches, for an F* distance of 1.40 inches.

More recently, WBN Unit 2 requested Westinghouse to look at industry data on similar SG designs to verify that the ARC could still be justified for use in the WBN Unit 2 SGs. Westinghouse documented their review in report SG-SGMP-13-15-P, dated March 2014 (Reference 1) and provided technical justification for the use of the ARC F* of 1.64 inches (including bounding NDE uncertainty of 0.34 inches).

3.4 Conclusion

This evaluation examined the effects of using ARC F* for analyzing tube plugging or repair criteria. The bases surrounding the establishment of the F* methods show that these methods are conservatively developed. Application of the criterion will not represent a condition where tube structural performance criteria will be jeopardized at the end of future operating cycles. Leakage testing has shown that the likelihood of primary-to-secondary leakage is negligible, and that tubes permitted to remain in service via application of the criterion will not effectively contribute to calculated primary-to-secondary leakage during postulated accident conditions. Therefore, application of the criterion does not represent a condition where the applicable leakage performance criteria will be challenged at the end of future operating cycles.

TVA's conclusion is that a F* distance of 1.64 inches, including a bounding NDE uncertainty of 0.34 inches, which is the length of sound (non-degraded) roll expansion as measured from the top-of-tubesheet or bottom of roll expansion transition, whichever is lower, will be applied for both the hot and cold leg regions of the WBN Unit 2 SGs.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements and Criteria

A review of 10 CFR 50, Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," was conducted to assess the potential impact associated with the proposed changes. The SGs at WBN Unit 2 are designed to comply with the following applicable regulations and requirements.

- Title 10 Code of Federal Regulations (CFR) Part 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," specifies that the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture.

- Title 10 CFR Part 50, Appendix A, GDC 15, "Reactor Coolant System Design," specifies that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- Title 10 CFR Part 50, Appendix A, GDC 16, "Reactor containment design," specifies that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

- Title 10 CFR Part 50, Appendix A, GDC 19, ""Control room," specifies that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures".

- Title 10 CFR Part 50, Appendix A, GDC 30, “Quality of reactor coolant pressure boundary,” specifies that components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

- Title 10 CFR Part 50, Appendix A, GDC 31, “Fracture prevention of reactor coolant pressure boundary,” specifies that the reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

- Title 10 CFR Part 50, Appendix A, GDC 32, “Inspection of reactor coolant pressure boundary,” specifies that components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

The reactor coolant pressure boundary, containment boundary and tube-bundle integrity will not be adversely affected by expansion of the implementation of the F* tube inspection scope to the cold-leg tubesheet region. SG tube burst or collapse cannot occur within the confines of the tubesheet; therefore, the tube burst and collapse criteria of Regulatory Guide (RG) 1.121 are inherently met. Any degradation below the F* length is shown by analyses and test results to be acceptable, thereby precluding an event with consequences similar to a postulated tube rupture event. SG tube SRs continue to ensure that degraded tubes will be repaired or removed from service upon detection. Postulated leakage in the limiting SG shall be less than the bounding faulted condition leakage necessary to ensure that offsite doses remain a small fraction of the 10 CFR Part 100 reactor site criteria and that control room doses remain within 10 CFR 50 Appendix A, GDC 19 limits. Therefore, conformance with all applicable GDCs remains valid.

In conclusion, with the implementation of the proposed changes, WBN Unit 2 continues to meet the applicable regulations and requirements.

4.2 Precedent

The ARC F* has previously been applied at numerous plants, without experiencing normal operating condition leakage or tube failure with tubes left in service due to application of the F* criterion. A listing of those plants which have applied the F* criterion is provided below:

- Farley 2
- South Texas 1
- Beaver Valley 2
- Comanche Peak 1
- Watts Bar 1
- V. C. Summer 1
- Prairie Island 1 and 2
- Cook 1 and 2
- Kewaunee
- McGuire 1 and 2
- Catawba 1

All of the above plants have replaced SGs with the exception of Beaver Valley 2.

4.3 Significant Hazards Consideration

The proposed change would modify Watts Bar Nuclear Plant (WBN), Unit 2, Technical Specifications 3.4.17, “Steam Generator (SG) Tube Integrity,” 5.7.2.12 “Steam Generator (SG) Program,” and 5.9.9, “Steam Generator Tube Inspection Report.” The proposed change requests approval to use the Alternate Repair Criterion (ARC) F* repair limit of 1.64 inches (including bounding nondestructive examination (NDE) uncertainty of 0.34 inches) as technically justified in Westinghouse WCAP-13084-P, "Tubesheet Region Tube Alternate Plugging (F*) Criterion for the Tennessee Valley Authority Watts Bar Units 1 and 2 Nuclear Power Plant Steam Generators," dated October 1991 and as supplemented by Westinghouse Report SG-SGMP-13-15-P, “Watts Bar Nuclear Plant Unit 2 F* Alternate Repair Criterion Technical Support Document,” dated March 2014.

TVA has concluded that the proposed change does not involve a significant hazards consideration. TVA’s conclusion is based on its evaluation in accordance with 10 CFR 50.91(a)(1) of the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   Response: No

   Allowing the use of an alternate repair criteria as proposed in this amendment request does not involve a significant increase in the probability or consequence of an accident previously evaluated.

The presence of the tubesheet enhances the tube integrity in the region of the hardroll by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and tube collapse is strengthened by the presence of the tubesheet in that region. Hardrolling of the tube into the tubesheet results in an interference fit between the tube and the tubesheet. Tube rupture cannot occur because the contact between the tube and tubesheet does not permit sufficient movement of tube material. In a similar manner, the tubesheet does not permit sufficient movement of tube material to permit buckling collapse of the tube during postulated loss-of-coolant-accident (LOCA) loadings.

The type of degradation for which the F* has been developed (cracking with a circumferential orientation) can theoretically lead to a postulated tube rupture event, provided that the postulated through-wall circumferential crack exists near the top of the tubesheet. An evaluation including analysis and testing has been performed to determine the resistive strength of roll expanded tubes within the tubesheet. That evaluation provides the basis for the acceptance criteria for tube degradation subject to the F* criterion.

The F* length of roll expansion is sufficient to preclude tube pullout from tube degradation located below the F* distance, regardless of the extent of the tube degradation. The existing technical specification leakage rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. As noted above, tube rupture and pullout are not expected for tubes using the ARC. Any leakage out of the tube from within the tubesheet at any elevation in the tubesheet is fully bounded by the existing Main Steam Line Break (MSLB) analysis included in the WBN Unit 2 Final Safety Analysis Report (FSAR).

Therefore, the proposed ARC does not adversely impact any other previously evaluated design basis accident.

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

Response: No.

Implementation of the proposed ARC does not introduce any significant changes to the plant design basis. Use of the criterion does not provide a mechanism to result in an accident initiated outside of the region of the tubesheet expansion. A hypothetical accident as a result of any tube degradation in the expanded portion of the tube would be bounded by the existing tube rupture accident analysis. Tube bundle structural integrity and leak tightness are expected to be maintained.

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

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

Response: No.

The use of the ARC has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," for indications in the free span of tubes and the primary to secondary pressure boundary under normal and postulated accident conditions. Acceptable tube degradation for the F* criterion is any degradation indication in the tubesheet region, more than the F* distance below either the bottom of the transition between the roll expansion and the unexpanded tube, or the top of the tubesheet, whichever is lower. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code used in SG design. The F* distance has been verified by testing to be greater than the length of roll expansion required to preclude both tube pullout and significant leakage during normal and postulated accident conditions. Resistance to tube pullout is based upon the primary to secondary pressure differential as it acts on the surface area of the tube, which includes the tube wall cross-section, in addition to the inside diameter-based area of the tube. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the technical specifications and the leakage assumptions used in the UFSAR accident analyses. Implementation of the ARC will decrease the number of tubes which must be taken out of service with tube plugs. Plugs reduce the RCS flow margin; thus, implementation of the ARC will maintain the margin of flow that would otherwise be reduced in the event of increased plugging.

Based on the above, it is concluded that the proposed change does not result in a significant reduction in or a loss of margin with respect to plant safety as defined in the FSAR or the bases of the WBN Unit 2 technical specifications.

4.4 Conclusion

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.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

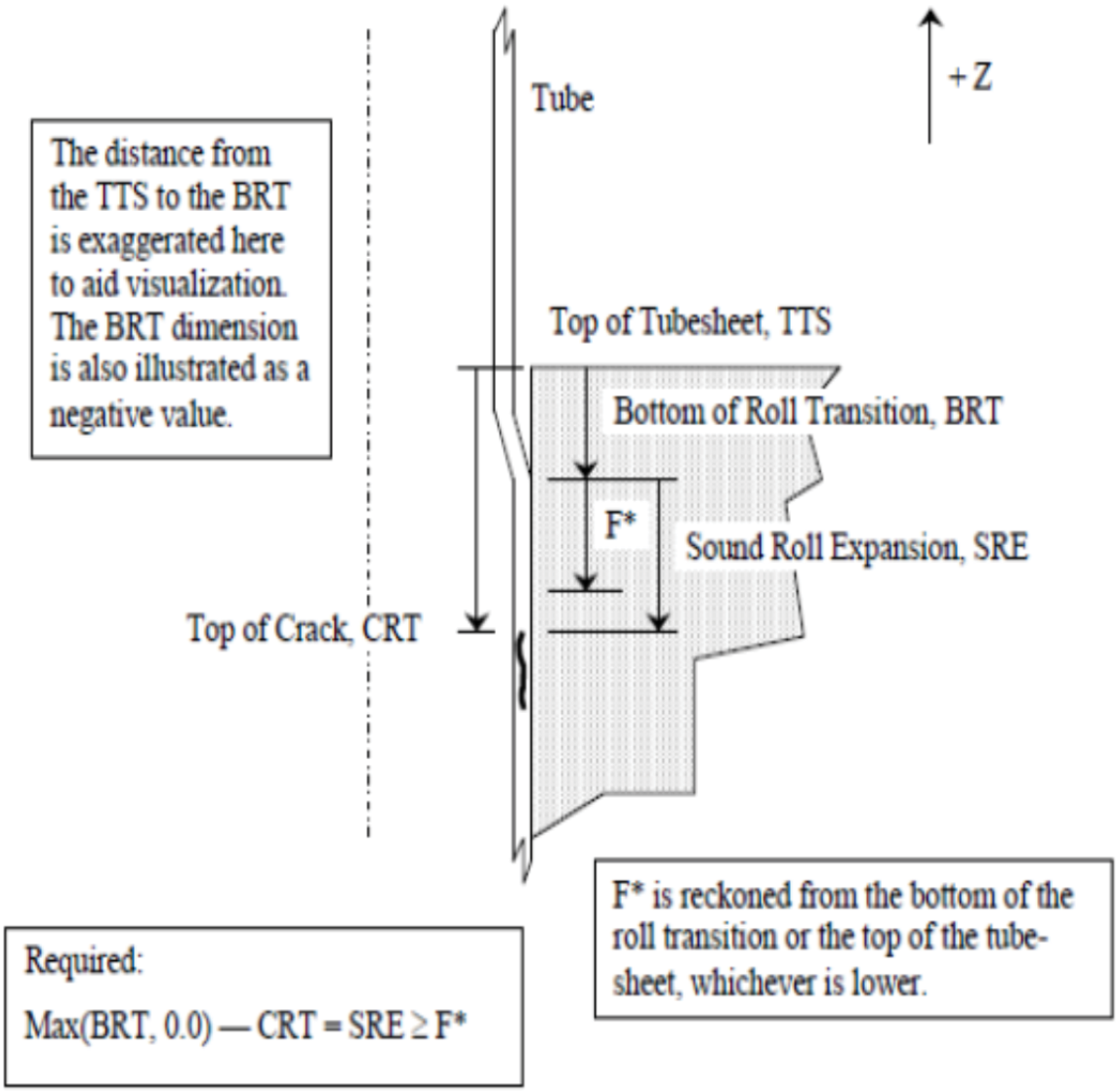
1. Westinghouse Report SG-SGMP-13-15-P, "Watts Bar Nuclear Plant Unit 2 F* Alternate Repair Criterion Technical Support Document," dated March 2014. (Enclosure 6 of this cover letter)

2. WCAP-13084-P, "Tubesheet Region Tube Alternate Plugging (F*) Criterion for the Tennessee Valley Authority Watts Bar Units 1 and 2 Nuclear Power Plant Steam Generators," dated October, 1991 (Enclosure 4 of ML003703068)

3. TVA Letter to NRC, "Watts Bar Nuclear Plant (WBN) - Unit 1 - Technical Specification (TS) Change No. WBN-TS-99-013 - Alternate Steam Generator Tubesheet Region Plugging Criterion (F*)," dated April 10, 2000 (ML003703068)

4. NRC Letter to TVA, "Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment Regarding Steam Generator Tubing Alternate Repair Criteria (ARC) (TAC No. MA8635)," dated February 26, 2002 (ML003748725)

Figure 1


Tube
+Z
The distance from the TTS to the BRT is exaggerated here to aid visualization. The BRT dimension is also illustrated as a negative value.
Top of Tubesheet, TTS
Bottom of Roll Transition, BRT
F*
Sound Roll Expansion, SRE
Top of Crack, CRT
F* is reckoned from the bottom of the roll transition or the top of the tube-sheet, whichever is lower.
Required:
Max(BRT, 0.0) — CRT = SRE ≥ F*

ATTACHMENT 1 to ENCLOSURE 1

List of Commitments

Commitment	Due Date
TVA will revise the Updated Final Safety Analysis Report to include a reference to this letter for implementing the F* Alternate Repair Criteria	12/23/16

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATIONS (MARK-UPS)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained

Insert: "(or repair)"

AND

All SG tubes satisfying the tube plugging 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.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program. Insert: "(or repair)"	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 plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection.

Insert: "(or repair)"

5.7 Procedures, Programs, and Manuals (continued)

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:

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 a 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 cooldown), 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.

(continued)

5.7.2.12 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than an 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.

Insert: "(or repair)"

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

c. Provisions for SG tube plugging 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.

Insert A

Insert: "(or repair)"

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 plugging 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. A degradation assessment 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 installation.

(continued)

Insert A:

----------REVIEWER'S NOTE----------
Alternate tube repair criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube repair criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.

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

1. The F* methodology, as described below, may be applied to the expanded portion of the tube in the hot-leg or cold-leg tubesheet region as an alternative to the 40% depth based criteria of Specification 5.7.2.12.c:

    a) Tubes shall be plugged upon detection of any flaw identified within 1.64 inches below the bottom of roll transition or top of tubesheet whichever is lower. Flaws located below this elevation may remain in service regardless of size.

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Program (continued)

2. After the first refueling outage following SG installation, inspect each SG at least every 24 effective full power months or at least every refueling outage (whichever results in more frequent inspections). In addition, inspect 100% of the tubes at sequential periods of 60 effective full power months beginning after the first refueling outage inspection following SG installation. Each 60 effective full power month inspection period may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Insert: "(or repair)"

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). 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.

5.9 Reporting Requirements (continued)

5.9.7 DG Failures Report

If an individual diesel generator (DG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that DG in that time period shall be reported within 30 days. Reports on DG 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 F 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 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. 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 degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG,

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

Insert: ", and"

Insert: "h. Repair method utilized and the number of tubes repaired by each repair method."

ENCLOSURE 3

REVISED PROPOSED TECHNICAL SPECIFICATIONS CHANGES

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained

AND

All SG tubes satisfying the tube plugging (or 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.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging (or 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.	B.1	Be in MODE 3.	6 hours
	AND		
OR	B.2	Be in MODE 5.	36 hours
SG tube integrity not maintained.			

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 plugging (or repair) criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection.

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:

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 a 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 cooldown), 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.

(continued)

5.7.2.12 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than an 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 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging (or 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.

----------REVIEWER'S NOTE----------
Alternate tube repair criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube repair criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.

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

1. The F* methodology, as described below, may be applied to the expanded portion of the tube in the hot-leg or cold-leg tubesheet region as an alternative to the 40% depth based criteria of Specification 5.7.2.12.c:

   a) Tubes shall be plugged upon detection of any flaw identified within 1.64 inches below the bottom of roll transition or top of tubesheet whichever is lower. Flaws located below this elevation may remain in service regardless of size.

(continued)

5.7.2.12 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 plugging (or 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. A degradation assessment 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 installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 24 effective full power months or at least every refueling outage (whichever results in more frequent inspections). In addition, inspect 100% of the tubes at sequential periods of 60 effective full power months beginning after the first refueling outage inspection following SG installation. Each 60 effective full power month inspection period may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging (or repair) criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of

(continued)

degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). 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.

(continued)

5.9 Reporting Requirements (continued)

5.9.7 DG Failures Report

If an individual diesel generator (DG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that DG in that time period shall be reported within 30 days. Reports on DG 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 F 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 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. 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 degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG,

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

h. Repair method utilized and the number of tubes repaired by each repair method.

ENCLOSURE 4

PROPOSED TECHNICAL SPECIFICATIONS BASES (MARK-UPS)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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 1 and 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).

BASES (continued)

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 an 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 an 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 of 150 gallons per day (gpd) per unfaulted 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) or the NRC approved licensing basis.

Delete paragraph

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

Insert: "(or repair)"

LCO

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

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

In the context of this Specification, an 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.

Insert: "(or repair)"

Insert: "(or repair)"

(continued)

BASES

LCO (continued)

An 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.

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 Regulatory Guide 1.121 (Ref. 5).

(continued)

BASES

LCO (continued)

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than an 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.

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 an 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.

(continued)

BASES

ACTIONS (continued)

A.1 and A.2

Insert: "(or repair)"

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging 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 plugging 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 an 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, Condition B applies.

Insert: "(or repair)"

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 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.

BASES (continued)

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 plugging 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.

Insert: "(or repair)"

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. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.7.2.12 until subsequent inspections support extending the inspection interval.

(continued)

Insert: "(or repair)"

Insert: "(or repair)"

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging 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 plugging 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 an SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

Insert: "(or repair)"

Insert: "(or repair)"

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. Regulatory Guide 1.121, “Basis for Plugging Degraded Steam Generator Tubes,” August 1976.
6. EPRI, “Pressurized Water Reactor Steam Generator Examination Guidelines.”

ENCLOSURE 5

REVISED PROPOSED TECHNICAL SPECIFICATION BASES CHANGES

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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 1 and 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).

BASES (continued)

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 an 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 an 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 of 150 gallons per day (gpd) per unfaulted 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) or the NRC approved licensing basis.

LCO

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

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

In the context of this Specification, an 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.

(continued)

BASES

LCO (continued)

An 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.

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 Regulatory Guide 1.121 (Ref. 5).

(continued)

BASES

LCO (continued)

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than an 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.

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 an 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.

(continued)

BASES

ACTIONS (continued)

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging (or 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 plugging (or 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 an 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, Condition B 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 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 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.

BASES (continued)

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

The 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. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.7.2.12 until subsequent inspections support extending the inspection interval.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging (or repair) criteria is removed from service by plugging. The tube plugging (or 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 plugging (or 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 an SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging (or 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. Regulatory Guide 1.121, “Basis for Plugging Degraded Steam Generator Tubes,” August 1976.
6. EPRI, “Pressurized Water Reactor Steam Generator Examination Guidelines.”

(continued)

ENCLOSURE 6

WESTINGHOUSE REPORT SG-SGMP-13-15-P, REVISION 0, "WATTS BAR NUCLEAR PLANT UNIT 2 F* ALTERNATE REPAIR CRITERION TECHNICAL SUPPORT DOCUMENT," MARCH 2014

Proprietary Information – Withhold from Public Disclosure Under 10 CFR § 2.390.

ENCLOSURE 7

WESTINGHOUSE AFFIDAVIT CAW-15-4140
SUPPORTING SG-SGMP-13-15-P

Westinghouse

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CAW-15-4140

March 25, 2015

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: SG-SGMP-13-15-P, Revision 0, "Watts Bar Nuclear Plant Unit 2 F* Alternate Repair Criterion Technical Support Document" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-15-4140 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Tennessee Valley Authority.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-15-4140, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

CAW-15-4140
March 25, 2015

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in SG-SGMP-13-15-P, Revision 0, "Watts Bar Nuclear Plant Unit 2 F* Alternate Repair Criterion Technical Support Document" (Proprietary), for submittal to the Commission, being transmitted by Tennessee Valley Authority letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with steam generator tube repair, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to:

(i) Provide input to Tennessee Valley Authority to submit to the U. S. Nuclear regulatory Commission information related to steam generator repair criteria.

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of repairing steam generators.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of documents furnished to the NRC in connection with steam generator tube repair, and may be used only for that purpose.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

ENCLOSURE 8

**WESTINGHOUSE REPORT SG-SGMP-13-15-NP, REVISION 0,
"WATTS BAR NUCLEAR PLANT UNIT 2 F* ALTERNATE REPAIR CRITERION
TECHNICAL SUPPORT DOCUMENT,"
MARCH 2015
(non-proprietary version)**

Westinghouse Non-Proprietary Class 3

SG-SGMP-13-15-NP
Revision 0

March 2015

Watts Bar Nuclear Plant Unit 2 F* Alternate Repair Criterion Technical Support Document

Westinghouse

SG-SGMP-13-15-NP
Revision 0

Watts Bar Nuclear Plant Unit 2 F* Alternate Repair Criterion Technical Support Document

William K. Cullen*
Steam Generator Management Programs

March 2015

Reviewer: Hermann O. Lagally*
Steam Generator Management Programs

Approved: Ryan R. Lowes*, Acting Manager
Steam Generator Management Programs

*Electronically approved records are authenticated in the electronic document management system.

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List of Abbreviations

Abbreviation	Definition
AILPC	Accident Induced Leakage Performance Criterion
ASME	American Society of Mechanical Engineers
BRT	Bottom of the Roll Transition
CRT	Crack Tip (distance from the TTS to the top of the flaw)
EDM	Electrical Discharge Machining
EPRI	Electric Power Research Institute
F*	Criterion distance for intact tub-to-tubesheet interface (F-Star)
F* Length	Physical sound roll expansion length required to prevent tube pullout
F* Distance	F* Length to which NDE uncertainty has been applied
ID	Inside Diameter
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OD	Outside Diameter
PORV	Power Operated Relief Valve
RAI	Request for Additional Information
RPC	Rotating Pancake Coil
SG	Steam Generator
SLB	Steam Line Break
SRE	Sound Roll Expansion
TTS	Top-of-Tubesheet
TVA	Tennessee Valley Authority
TW	Through-wall
WBNP	Watts Bar Nuclear Plant
WBN Unit 1	Watts Bar Nuclear Plant Unit 1
WBN Unit 2	Watts Bar Nuclear Plant Unit 2
+Pt™[1]	Plus Point ™ probe

[1] +Point is a trademark of Zetec, Inc.

1.0 Introduction

In October 1991, WCAP-13084, "Tubesheet Region Tube Alternate Plugging (F*) Criterion for the Tennessee Valley Authority Watts Bar Units 1 and 2 Nuclear Power Plant Steam Generators," was prepared for TVA, and provided the bases for the establishment of an F* (pronounced F-star) repair limit of 1.06 inch (excluding NDE length measurement uncertainty). This value represents the length of sound (non-degraded) roll engagement between the tube and tubesheet required to preclude tube pull-out during either normal operation or accident conditions, assuming a complete circumferential separation of the tube below the F* distance, with appropriate safety factors applied. This document evaluates the applicability of the original F* analysis, documented in WCAP-13084 to the WBN Unit 2, Model D3 SGs and provides updated information where necessary. Figure 1 provides a representation of the F* criterion. Each of the elements which describe the criterion, and are used in the field verification of continued operability, the TTS, BRT, F*, and SRE, are depicted on Figure 1.

The WBN Unit 2 plant utilizes Westinghouse Model D3 SGs, with mill annealed Alloy 600 tubing, full depth mechanical (roll) expanded through the tubesheet thickness, and drilled hole design TSPs. The materials of construction for the components important to the F* analysis: the tubes, tubesheet, channelhead, and lower stub barrel, are consistent between the original WBN Unit 1 SGs and the WBN Unit 2 SGs.

Operating histories of WBN Unit 1 and similar plants (SGs with Alloy 600 mill annealed tubing and roll expanded tubes) have shown a potential for both axially and circumferentially oriented tube stress corrosion cracking degradation to occur predominantly at the tube TTS expansion transition and within the roll expanded region within the tubesheet. The F* criterion establishes a length of sound, i.e., non-degraded, tube roll engagement, starting from the bottom of the roll expansion transition near the TTS, and extending down into the tubesheet, below which postulated tube degradation, including a complete circumferential separation of the tube, would not represent a condition under which satisfaction of SG tube structural and leakage integrity performance criteria would be jeopardized.

Subsequent to the licensing of F* to the original WBN Unit 1 SGs, the H* alternate repair criterion has been approved for plants with hydraulically expanded tube-in-tubesheet joints. This document herein considers significant and pertinent NRC RAIs received during the H* licensing process to show that these RAIs do not affect the F* analysis for WBN Unit 2.

2.0 Summary of F* Criterion for WBNP

2.1 Original (Circa 1991) F* Evaluation

In the event of degradation of a steam generator tube at an elevation within the tubesheet region of a full (through the tubesheet) depth roll expanded tube, structural and leakage integrity for the tube can be established by measurement of the sound roll distance extending from either the BRT or the TTS, whichever is physically lower in elevation, to the uppermost extent of the tube degradation. Both axial and circumferentially oriented degradation are addressed by application of the criterion. WCAP-13084 established an F* length of 1.06 inch for WBN Unit 1 and WBN Unit 2. The criterion is evaluated for both normal and postulated accident conditions. The limiting F* length of 1.06 inch was established based on the postulated accident conditions associated with a SLB or FLB event. The original accident condition analysis utilizes the FLB primary-to-secondary pressure differential (as it bounds the SLB pressure differential), and uses the SLB thermal conditions (as they represent a more stringent condition compared to the FLB event) where the applied temperature was equal to the operating temperature at the time of the event. A limiting F* length of 1.04 inch was established for normal operating conditions. Safety factors of three for normal operating conditions and 1.43 for SLB conditions were applied, and are therefore consistent with the safety factor requirements provided in NEI 97-06 (latest revision).

The limiting primary-to-secondary pressure differential used for the normal operating condition analysis was 1400 psi. In comparison, the steam pressure upon startup of the WBN Unit 2 SGs (Reference 2) is expected to be approximately 1000 psia (1250 psi differential), while the expected steam pressure at 10% average tube plugging and 3475 MWt reactor power is predicted to be approximately 946 psia (1304 psi differential); thus, the applied pressure differential is conservative compared to expected plant operating conditions.

As described in WCAP-13084, the F* length was conservatively established based on the results of load testing of the tube/tubesheet specimens where the tube was subjected to internal pressurization. Samples were rolled into tubesheet simulant collars consistent with the original SG manufacturing process, subjected to a [ ]^{a,c}, and then circumferentially separated by machining at various roll engagement lengths. The thermal soak simulates the effects of the tubesheet to lower stub barrel post-weld heat treatment. The post-weld heat treatment is applied at [ ]^{a,c}.

The original analysis of Reference 1 was recreated to verify the input assumptions and analysis tools. Table 1 identifies the input parameters used in the original analysis. As seen from Table 1, the parameters applied in the original WCAP-13084 analyses are similar to hot leg conditions. The original F* application to WBN1 was approved for both the hot and cold legs. The original WCAP analysis used material property values at a slightly lower differential temperature than the actual hot leg conditions, and made the judgment that the analysis would equally apply to both hot and cold legs. The effort described in Section 2.2 recalculates the F* lengths for the specific differential temperature condition for both hot and cold legs for normal operating conditions. For faulted conditions the applied temperature differential is assumed equal for both legs.

2.2 2013 Update of F* Distances for WBN Unit 2

For the analyses performed in 2013 (Reference 10), F* length is calculated for both hot and cold legs, using the specific hot and cold leg temperature conditions (and thus material property values). Thus the 2013 analysis is a more detailed recreation of the original effort. The 2013 analysis also considers the potential impact of the F* length as a function of changes in the reported material property values in relation to updates to the ASME material property values.

Table 2 identifies the plant parameters important to the F* analysis; these values are taken from Reference 2. To support F* application on the cold leg, the analysis was repeated using cold leg conditions.

Table 3 presents the input parameters for a bounding set of hot and cold leg conditions for the 3475 MWt NSSS power level.

Calculated F* lengths are provided in Table 4 for various versions of the ASME Code as material property values may have changed with time. Table 4 uses input from three specific ASME Code versions; the Code of Record (1971), the Code applicable at the time of the WCAP-13084 analysis (1989), and the most recent ASME Code version approved by the NRC, the 2007 Code through 2008 Addenda. The applied values for Young's Modulus and thermal expansion coefficients are interpolated at the hot and cold leg temperatures of 620 and 557°F. Table 4 shows that there is minimal difference between the hot and cold leg F* values, that the faulted condition analysis is limiting, and that the value of F* could vary, depending upon the version of the ASME Code used. As the most recent ASME Code material property values results in the most conservative F* value, it is recommended that the applied F* value be based on the 2007 Code through 2008 Addenda.

For simplicity, it is recommended that the same F* value be applied to both the hot and cold legs.

3.0 Evaluation of Eddy Current Measurement Uncertainties

The practical application of the F* criterion involves the establishment of an NDE-based F* distance defined as the sum of the F* distance based on analysis and the NDE measurement uncertainty. The NDE measurement uncertainty involves measurement of the length of sound roll expansion (SRE) between the bottom of the roll transition (BRT), which represents the point of contact between the tube and tubesheet in the expansion transition region, or the top of the tubesheet (TTS) if the BRT is located above the TTS, to the uppermost extent of the tip of the flaw. NDE measurement uncertainties for F* application have been developed in Westinghouse document DAT-UNC-001, Revision 0, specifically for F* application at a plant with 7/8 inch x 0.050 inch wall thickness tubing. As the process involves measurement using rotating coil techniques, the difference in tubing geometries are judged not to have an impact upon transference of the uncertainties to ¾ inch OD tubing. The pitch of the RPC probe, or axial translation related to each probe rotation, is controlled by the applied rotational speed and probe pull speed; thus, tube OD does not influence NDE performance.

3.1 Summary of Test Data and Analysis Techniques

Test samples were prepared by rolling sections of tubing into carbon steel tubesheet simulants with ID axial flaws produced by EDM following rolling of the tubes into the simulants. The test data set included configurations where the roll transition was located both above and below the TTS. RPC data was collected using a probe that contained a 115 mil mid-range pancake coil, a Plus Point coil, and an 80 mil high frequency pancake coil. All data was collected while the probe was pushed into the tube. Each sample had NDE data collected for three separate runs, and five separate analysts participated in the program. The individual results from all five analysts were combined to form a single data set.

The set-up parameters include a maximum probe pitch of 0.040 inch, with a minimum sample rate of 30 data points per helical inch of probe travel. The TTS is located using the 115 mil pancake coil, using either the 10 kHz or 20 kHz frequencies (or functionally equivalent test frequencies), and all RPC based measurements are performed using this reference. If the BRT cannot be located using the bobbin probe, the BRT is located using the 115 mil pancake coil using a minimum frequency of 300 kHz. It should be noted that the preferred method of locating the BRT is with the bobbin coil.

3.2 Analysis Results

The practical application of the F* criterion involves the combination of the BRT location as measured by bobbin coil with the measurement of the uppermost portion of the crack tip with reference to the top of the tubesheet. BRTs located above the TTS are given a positive sign convention while BRTs located below the TTS are given a negative sign convention. Similarly, since the crack tip (CRT) measurement extends from the TTS to the uppermost portion of the flaw, it also is given a negative sign convention. Therefore, for an assumed F* distance including NDE uncertainty of 1.5 inches, with CRT measurement of -2.0 inches and BRT located at 0.25 inch below the TTS, the SRE distance is computed as:

SRE = BRT - CRT or,

SRE = (-0.25) – (-2.0) = 1.75 inches.

As the SRE length in this example is greater than or equal to the F* distance including NDE uncertainty, the tube may remain in service.

Conversely, for an assumed F* distance including NDE uncertainty of 1.5 inches, with a CRT measurement of -1.6 inches and BRT located at 0.2 inch above the TTS, the SRE distance is computed as:

SRE = 0.0 – (-1.6) = 1.6 inches.

As the SRE length in this example is greater than or equal to the F* distance including NDE uncertainty, the tube may remain in service. In this example, the BRT is located above the TTS and the length of sound roll expansion above the TTS does not contribute to the total SRE length. Therefore, the positive BRT location is set to 0. It should be emphasized the SRE is calculated using either the BRT or TTS, whichever is lower, to ensure that the SRE distance is calculated correctly.

As the calculation of the SRE distance uses both the bobbin measurement of BRT location and CRT measurement from the TTS to the uppermost portion of the crack tip, the combined NDE uncertainty value must include both contributions from the bobbin measurement of BRT, and the RPC measurement of the CRT distance. To perform this combination, the mean error values for the bobbin measurement of BRT and CRT measurement relative to the TTS are added. The standard deviation of the bobbin and RPC measurement errors are combined by a square root of the sum of squares method, with this value multiplied by a statistical constant based on the sample data size and selected confidence level. The constant selected was based on a probability of 95% and confidence level of 95%, for the sample size. Using the test results, the combined NDE uncertainty values were obtained and are listed in Table 5. Therefore, the F* distances, including NDE uncertainty are as listed in Table 6 for the three available coils.

3.3 Alternate Uncertainty Analysis

The NRC has placed increased emphasis on the probabilistic evaluation of uncertainties as demonstrated by the RAIs received during the H* licensing process. Because of this, an alternate uncertainty evaluation was developed that shows that application of accepted probabilistic techniques to the eddy current (EC) uncertainty would result in a shorter distance for F*. However, the original uncertainty analysis is recommended to be applied as additional margin to the performance criterion will be provided.

Westinghouse technical report DAT-UNC-001 includes RPC based measurements of the SRE, for the 115 mil pancake, 80 mil pancake, and +Pt coils. The uncertainty values provided in Table 5 are developed in a deterministic fashion. That is, the uncertainty is established by combination of a mean error and standard deviation of error, times a factor to provide a representation at an elevated probability level.

Prediction and confidence intervals can be developed for a regression of the measured SRE length on true SRE length. Figure 2 provides the regression of measured SRE on true SRE. This plot can be used to define the range of SRE measurements which are representative of a true SRE dimension. For a true SRE dimension of 1.16 inches, mean SRE measurement is [

]^{b,c,e}. Thus, this evaluation of the SRE measurement uncertainty would reduce the overall F* distance based on +Pt coil inspection from 1.50 (1.16 + 0.34) inches to 1.35 inches.

Also, the combined uncertainty associated with the bobbin coil measurement of the BRT and the +Pt based measurement of the crack tip relative to the top-of-tubesheet were combined using a Monte Carlo simulation. The mean and standard deviation of the BRT and CRT measurements were input as normally distributed populations, described by a mean and a standard deviation. As the BRT measurement is performed using the bobbin coil and the CRT measurement is performed using a RPC coil, the measurements are independent. Thus, the combination of both is a random process described by two independent variables. The Monte Carlo simulation was performed by random sampling from a BRT error and from a CRT error; each of the samplings was then simply added to obtain the total uncertainty. This sampling was performed 10,000 times, with the combined errors then ordered smallest to largest. At the 95th percentile the combined BRT/CRT measurement error is [

]^{a,b,c,e} for a SRE required length of 1.16 inches.

It is recommended that for conservatism, the accepted uncertainty allowance for the +Pt coil of 0.34 inch be applied. The conservatism associated with the applied uncertainty can be used to offset other variances.

4.0 Accident Condition Leakage Allowance

4.1 Limiting Conditions

The AILPC specified within the NEI 97-06 Steam Generator Program Guidelines (Reference 8) provides guidance related to the magnitude of permissible primary-to-secondary leakage during accident conditions where direct steam release to the environment may occur. In general, "Leakage is not to exceed 1 gpm per steam generator, except for specific types of degradation at specific locations when implementing alternate repair criteria as documented in the Steam Generator Program technical specifications." Meeting the performance criteria provides a reasonable assurance that the WBN Unit 2 steam generator tubing remains capable of fulfilling its specific safety function of maintaining the primary pressure boundary, and that off-site dose estimates will remain within the analyzed limits. As tubes with potential through-wall defects may be justified for continued service by application of the F* criterion the potential for primary-to-secondary leakage must be evaluated. A review of Section 15.5 of the WBN Unit 2 Final Safety Analysis Report indicates that only two accidents within the current licensing basis evaluate the effects of the release of steam from the secondary system; Loss of AC Power to the Plant Auxiliaries and a postulated SLB event. Of the two accidents identified, only the SLB event is a design basis event; Loss of AC Power to Plant Auxiliaries is a Category II event, thus only the SLB condition needs to be considered in the development of the F* alternate repair criterion.

As the degradation potentially justified for continued operation is located at greater than the F* distance below the top-of-tubesheet or bottom of roll expansion transition, whichever is lower, there is no potential for such degradation to develop into a condition in which the observed normal operating conditions leak rate can be considered a precursor to a tube rupture. This concept forms the basis of the leak rate evaluation criteria of the Electric Power Research Institute (EPRI) Primary-to-Secondary Leakage Guidelines (Reference 9). For the F* case, axial degradation located below the F* distance (actually any location below the BRT) is restrained from radial opening of the crack face due to the proximity of the tubesheet, and the maintenance of the F* distance precludes a circumferential separation of a tube below the F* distance from causing displacement of a postulated circumferentially separated tube such that tube rupture type leak rates could be experienced.

4.2 Evaluation of Available Leakage Data

The historical inspection data from other plants with full depth roll expanded tube-to-tubesheet joints indicates that axial cracking within the roll transition and roll expanded portion of the tube in the tubesheet has an extremely low probability of leakage. The cold work imparted to the tube and the radial preloads provided by the roll expansion process effectively limit leakage for through-wall axial crack lengths at the expansion transition of less than about ¼ inch to ½ inch while in the installed condition. WCAP-13084 concludes that the roll expanded joint is leak tight under both normal and faulted plant conditions for roll expansion lengths of greater than 0.5 inch, much less than the recommended F* length. This statement (present in all F* WCAPs) was developed at a point in time when sensitivity to very small quantities of primary-to-secondary leakage was not as great as the current. Considering this increased sensitivity, the available F* leakage test data was reviewed with the intent of confirming the statement that leakage from postulated tube degradation below the F* distance will remain negligible, and can be ignored.

During the development phase of the F* criterion, leakage tests were conducted to determine the likelihood of primary-to-secondary leakage. Alloy 600 tubing of the same geometry as the Watts Bar tubing was installed in tubesheet simulant collars, and [

]a,b,c,e.

If the average leak rate for the samples with 0.25 inch roll lengths, tested at 619°F is used, and it is assumed that 100 transitions per SG progress from a NDD condition to a condition with through-wall axial degradation starting at 0.25 inch below the BRT, the postulated leak rate is found to be [ ]a,c,e gpm/SG. This example suggests that should postulated degradation occur during operation, and should said degradation progress to a 100%TW condition that end of cycle leakage contribution will be negligible, and will not challenge leakage related performance criteria.

[

]a,b,e.

For the samples with one inch roll lengths for pressure differentials ranging from 2250 psi to 2650 psi tested at 619°F, the probability of leakage was [

]a,b,c,e. It should be noted that the F* distance including NDE measurement uncertainty is >1 inch, and this additional sound roll engagement will further reduce both the likelihood of leakage and leak rate as compared

to the above estimation. It should also be noted that degradation located at >1 inch below the TTS or BRT, whichever is lower, but <F* would be detected during normal eddy current inspections. Thus, the above example represents a theoretical condition only, since it is not plausible that all hot leg tubes would develop 100% TW degradation between 1 inch below TTS or BRT but <F* in one operating cycle. Given the tube material, expansion process, and operating history of WBN Unit 1 and other similar units, RPC inspection of the expanded tube-in-tubesheet region above F* will likely be inspected at each refueling outage.

For the samples with two inch roll lengths for pressure differentials ranging from 2250 psi to 2650 psi tested at 619°F, the probability of leakage was [

]^{a,b,c,e} gpm. This value is so small that it may be neglected when considering postulated primary-to-secondary leakage during postulated accident conditions.

The leakage test data also shows that for the samples tested at pressure differentials of 2250 psi to 2650 psi, the likelihood of leakage and the leak rate increased for the 2650 psi differential pressure condition. If the availability of the WBNP pressurizer PORVs during postulated accident conditions can be established, the leakage estimations provided above are conservative estimations of the postulated maximum SLB leakage contribution from F* tubes as SLB conditions pressure differential will effectively be limited to approximately 2350 psi.

Further support for the leakage integrity of the hard-rolled tube joint is provided by operating plant data. At a plant with 7/8 inch OD tubing, four tubes were pulled during a 1996 outage. Three of these tubes contained through-wall axial degradation within the roll transition. The first tube had three through-wall axial ID cracks. The through-wall lengths and total crack lengths for these three flaws were, 0.07 inch TW; 0.138 inch total, 0.06 inch TW; 0.112 inch total, and 0.06 inch TW; 0.128 inch total, with all flaws located above the TTS. The second tube had two through-wall axial ID cracks with through-wall lengths and total crack lengths of 0.374 inch TW; 0.44 inch total, and 0.12 inch TW; 0.188 inch total, with approximately ½ of the total crack length existing above the TTS for the first flaw while the second flaw existed entirely above the TTS. The third tube had two through-wall axial ID cracks with through-wall lengths and total crack lengths of 0.08 inch TW; 0.14 inch total, and 0.092 inch TW; 0.144 inch total, with both flaws located above the TTS. All of these tubes were in situ pressure tested at a pressure differential of 2900 psi with no reported leakage. In the laboratory at room temperature conditions, all three tubes leaked at the SLB pressure differential, with leak rates of $1.9 \cdot 10^{-4}$ gpm, $5.9 \cdot 10^{-2}$ gpm, and $1 \cdot 10^{-4}$ gpm, respectively. If only the longest of the 100% through-wall cracks of each tube is considered, the calculated leak rates (using EPRI equations) are substantially larger than the measured leak rates. For a pressure differential of 2500 psi, Young's Modulus of $3.1 \cdot 10^{4}$ ksi, and flow stress of 75.5 ksi, the calculated leak rate for each tube is $6.5 \cdot 10^{-3}$ gpm, 0.28 gpm, and $1.9 \cdot 10^{-3}$ gpm. For a flow stress of 100 ksi (based on the fact the tube was cold worked due to roll expansion), the calculated leak rates are $5.8 \cdot 10^{-3}$ gpm, $1.9 \cdot 10^{-1}$ gpm, and $1.7 \cdot 10^{-3}$ gpm, respectively, which still exceeds the measured leak rates. Thus, it can be concluded that the estimates of leak rate based on observed crack lengths is conservative, and that the F* leakage testing, described below, represents a conservative assessment of leakage potential.

It should be noted that this area will be inspected at least once every 24 effective full power months. As such, it is unlikely that through-wall (or near through-wall) flaws will develop between inspections.

5.0 Impact of Recent ARC Programs on F*

The F* ARC has previously been applied at numerous plants, without experiencing normal operating condition leakage with tubes left in service due to application of the F* criterion. A listing of those plants which have applied the F* criterion is provided below:

- Farley 2
- South Texas 1
- Beaver Valley 2
- Comanche Peak 1
- Watts Bar 1
- V. C. Summer
- Prairie Island 1 and 2
- Cook 1 and 2
- Kewaunee
- McGuire 1 and 2
- Catawba 1

All of the above plants have replaced steam generators with the exception of Beaver Valley Unit 2. The most recent F* application included responses to NRC requests for additional information (RAI) based on the NRC review of the H* program.

Recently, permanent approval of the H* ARC has been granted for those plants requesting permanent application. The H* ARC is similar to F* in that it addresses degradation within the tubesheet region. However, there is a large fundamental difference between the tube expansions processes; F* applies to hard-roll expanded tubes and H* applies to hydraulically expanded tubes. The hard-roll expansion process results in a significantly more robust expansion joint. During the development of H*, RAIs were received regarding the (potential) loss or reduction of contact pressure due to tubesheet bow, leakage prediction, material property variance, and inspection practices. The following information summarizes why these subject matter have no impact upon the F* criterion.

5.1 Tubesheet Bow and Contact Pressure Loss

Roll expanded tubes create an interference fit with the tubesheet hole. That is, if the tubesheet were removed, the tube OD would enlarge due to the large residual stresses inherent to the roll expansion process. Testing has shown this enlargement, or springback, is approximately [

]a,b,c,e.

The hydraulic expansion process results in lower residual contact pressure due to the expansion process compared to roll expansion. For the condition where no tubesheet bow effects are included, the tubes will experience a contact pressure with the tubesheet hole due to thermal and pressure

expansion of the tube for both F* and H*. The tubesheet bore enlargement due to bowing was calculated for another plant using the original two-dimensional structural model of the lower tubesheet complex and was found to be substantially less than the available tube springback from the roll expansion process. Therefore, ignoring the effects of differential thermal expansion and tube internal pressure expansion, the hardroll expansion process retains positive contact pressure between the tube and tubesheet hole during all operating and accident conditions.

A more refined, three-dimensional tubesheet bow model was developed for the H* program. The results of this model were compared with the results of the original two-dimensional model developed for the original F* analyses and applied during the initial development of H*. This comparison showed that the two-dimensional model yields conservative H* distance results compared to the three-dimensional model (Reference 4). The two-dimensional model was applied for the Watts Bar F* analysis, thus the influences of tubesheet bow are conservatively modeled in the current F* technical justification. Reference 1 utilized a top-of-tubesheet, tubesheet bow contact pressure loss of [ ]^{a,c,e} psi per unit length at normal operating conditions and [ ]^{a,c,e} psi per unit length for faulted conditions. The two-dimensional model was rerun for the WBN Unit 2 inputs of a bounding primary-to-secondary pressure differential of 1400 psi during normal operating conditions, hot leg and cold leg tube temperatures, shell ΔT based on applicable steam saturation temperatures (no downcomer mixing due to preheater design), and material properties at the various locations. The contact pressure losses were found to be bounding for the cold leg conditions, and also bounded by the assumed values applied in the original analysis. Additionally, the combined tubesheet bow effects change sign at approximately a 38 inch radius about the center of the tubesheet x-y plane. That is, for radii of 38 inches or less, the combined effects of all contributing inputs results in a loss of contact pressure, but bounded by the Reference 1 assumptions while for radii greater than 38 inches, the combined effects of all contributing inputs results in an increase in contact pressure.

On June 14 and 15 of 2010, the NRC Staff conducted an audit of the H* program at the Westinghouse Waltz Mill Site (Reference 3). The purpose of the audit was to gain a better understanding of the H* analysis pertaining to tubesheet bore eccentricity resulting from tubesheet bending. Reference 5 documents the Westinghouse evaluation of this issue. Based on the audit, the NRC Staff concluded that eccentricity does not appear to be a significant variable affecting either average tube-to-tubesheet contact pressure at a given elevation or calculated values for H*. The NRC found that average contact pressure at a given elevation is primarily a function of average bore diameter change at that elevation associated with temperature and pressure loading of the tubesheet. Accordingly, the NRC concluded that no adjustment of computed average bore diameter is needed to account for eccentricities computed by the 3-dimensional Finite Element Analysis for the lower assembly of a steam generator. Thus, as the existing WBN analysis for estimation of contact pressure reductions is based on the more conservative 2-dimensional model, and the NRC has concluded that eccentricity is not a relevant parameter, the existing normal operating condition and accident condition contact pressure reductions are consistent and adequate for the F* analysis.

5.2 Leakage Prediction and Impact of Crevice Flow Phase Assumptions

Room temperature leakage testing of H* specimens showed that tube-in-tubesheet engagement lengths as long as 16.5 inches included measurable leakage. Prediction of the leak rate using fluid mechanics formulas which include only the effects of crevice length, flow area, and fluid viscosity

show a very good correlation with the test data, indicating that relatively low residual contact pressure between the tube and tubesheet is generated by the hydraulic expansion process. For example, the average of the measured leak rates for the H*, 16.5 inch joint length tests at room temperature with a pressure differential of 2650 psi is [ ]^{a,b,c,e}. For an assumed crevice (tube-to-tubesheet radial direction) width of 0.0005 inch, the calculated leak rate is 0.006 gpm, and the equivalent crevice width which provides for a calculated leak rate of 0.004 gpm is then 0.0004 inch. At elevated temperature with a pressure differential of 2650 psi, the average leak rate is also 0.004 gpm. The calculated equivalent crevice width which provides for a calculated leak rate of 0.004 gpm is 0.0002 inch.

In comparison, room and elevated temperature leakage testing of roll expanded tube samples show that roll engagement lengths as short as 0.50 inch produced a leak tight condition at both normal operating and faulted conditions (for a portion of the specimens). Application of the same equations used to describe the H* tests to the F* leakage data would require crevice widths of approximately 9 x 10^{-6} inch for the 0.50 inch roll engagement length specimens which leaked, suggesting that any leakage from F* tubes would be due to surface imperfections of the tubesheet hole, and would therefore be exceptionally small such that any contribution to primary-to-secondary leakage estimates from F* tubes can be neglected. The ¾ inch OD tube F* leak tests with 2 inch roll length include two tests; [ ]^{a,b,c,e}.

Considering the extremely limited crevice width condition for roll expanded tubing, any primary fluid pushed through the tube-to-tubesheet crevice would be expected to remain in a liquid state until the fluid exits the crevice at the top-of-tubesheet. Therefore, two-phase liquid conditions will not exist in the tube-to-tubesheet crevice.

5.3 Material Property Variance

Other tubesheet region F* and W* ARC submittals presented to the NRC, during the review phase of the H* program, utilized a 5% variance in tube thermal expansion coefficient for evaluation of the sensitivity of the calculated (X)* (i.e., F* and W* ARCs) value. During the H* development, coefficient of thermal expansion tests were performed and the results were published in Reference 6. These tests supported the conclusion that the use of a singular (one element, either the tube or tubesheet thermal expansion properties were adjusted by 5%) 5% variance of the ASME Code properties is acceptable. Subsequent analysis of the thermal expansion coefficient data showed that the apparent variability was essentially completely attributable to measurement error and that the actual property variability is significantly less than the 5% variance around the ASME Code properties (Reference 7).

Reference 6 identifies a standard deviation on tube thermal expansion coefficient of 2.33% and a standard deviation on tubesheet thermal expansion of 1.44%. A F* sensitivity case was run using the 2007 through 2008 Addenda thermal expansion coefficient data where the tube thermal expansion coefficient was *reduced* by 2.33% and the tubesheet thermal expansion coefficient was *increased* by 1.44%. This condition represents a conservative adjustment of thermal expansion coefficient for both the tube and tubesheet. The F* distances were found to be 1.23 inches for normal conditions, 1.30 inches for faulted conditions, for both the hot and cold legs, or an increase in the recommended F* distance of 0.14 inches. Considering the margin in the NDE uncertainty of

0.10 inch, and the fact that the above calculation applies a simultaneous conservative adjustment of the thermal expansion coefficients, the variance in the calculated F* distance is judged to be bounded by the inherent conservatisms in the F* analysis. If it is selected by TVA to implement an F* value (F* distance + NDE uncertainty allowance) based on the above sensitivity case, the developed F* value is then 1.64 inches.

5.4 Inspection Practices

The proposed field F* application includes verification of a minimum sound roll expansion length of 1.50 inches from the top-of-tubesheet, or bottom of expansion transition, whichever is lower. TVA will perform an evaluation of BRT locations for all WBN Unit 2 tubes prior to restart. This information will be applied to the inspection distance to be utilized in field inspections. The minimum (lowest absolute value) BRT dimension will be added to the recommended F* distance to ensure that all tubes are inspected to an adequate distance. In addition, standard industry practice is to "over collect" the +Pt data at the top-of-tubesheet region to ensure that the required inspection length has been achieved such that retesting of the tube to acquire sufficient data is not required. Thus, in practice, the tube length at the top-of-tubesheet region that is inspected using the +Pt coil will exceed the minimum required F* inspection distance. Such actions only strengthen the judgments that tube pull-out resistance will be provided for all tubes during all plant conditions and leakage integrity will be provided by all tubes during all plant conditions.

Those plants implementing the H* criterion have committed to monitor "tube slippage," or axial displacement of the tube in the tubesheet hole using the bobbin coil. It has been theorized within the H* evaluation by the NRC, that the end cap load applied to the tubes could cause the tube to axially displace in the hole in the event of a postulated circumferential separation of a tube below the H* distance. It is not necessary for TVA to monitor tube slippage as the unit length resistive load capabilities of the roll expansion process far exceed that of the hydraulic expansion process. Additionally, tube slippage has never been reported in the field at any plant. Any postulated circumferential separation of a SG tube would be expected to be readily observable using the bobbin coil.

5.5 Recommended Inspection Distance

A recommended F* inspection distance of 2 inches below the top-of-tubesheet is established for WBN Unit 2. Based on the BRT measurements from another plant, this nominal inspection distance will satisfy the F* requirements for approximately 99.8% of all tubes, assuming that TVA applies the more limiting F* distance of 1.64 inches developed in Section 5.3. The BRT data shall then be reviewed; any BRT location which does not satisfy the F* distance requirement for a nominal 2-inch inspection distance from the TTS shall be inspected to a depth which then satisfies F*. For example, a tube is found to have a BRT dimension of -0.75 inch. For a F* distance of 1.64 inches, the required inspection depth is the absolute sum of the F* distance and the BRT dimension, or 2.39 inches. The inspection depth for this tube shall be adjusted to ensure that a minimum of 2.39 inches below the top-of-tubesheet, or 1.64 inches below the bottom of the roll expansion transition shall be inspected.

For an applied F* distance of 1.50 inches, approximately 99.85% of all tubes will be inspected to an adequate depth using a nominal 2 inches deep inspection distance, based on the BRT location information from another plant.

Due to the temperature dependence of SCC mechanisms, 100% of the hot leg tubes shall be inspected upon implementation of the criteria, and at all subsequent inspections.

RPC based inspection history of the cold leg TTS region is limited. A plant with Model 51 SGs using partial depth (2.75 inches up from the tube end) roll expansion in the tubesheet region has an extensive inspection history for the cold leg roll expanded region. This plant (590°F T-hot) began to see numerous PWSCC indications on the hot leg at approximately 17 EFPY. At approximately 20 EFPY, +Point sampling (20% minimum) of the cold legs was implemented. PWSCC was not reported on the cold leg side up to the final inspection before SG replacement (approximately 31 EFPY). If the inspection data from this plant and an estimate of the initiation potential between the cold legs of WBN Unit 2 and this plant are considered, PWSCC degradation would not be expected on the cold leg side at WBN Unit 2 prior to about 9 EFPY. As observations of degradation on the hot leg side due to ODSCC may prompt inspection of the cold leg TTS region at WBN Unit 2 prior to 9 EFPY, the recommendation for implementation of +Point sampling of the cold leg TTS region will be defined by the degradation assessment. Once sampling of the cold leg TTS region is implemented, the minimum sample size shall be 20% of active tubes.

5.6 AILPC Considerations as Related to F* Application

Any crack-like indication reported below the F* distance (but observed within the inspected region) shall be evaluated for their effect on the leakage integrity of the SG using methods consistent with the current revision to the EPRI Integrity Assessment Guidelines. Any indication(s) determined to represent a leakage potential will be considered in the condition monitoring assessment when validating that the AILPC was satisfied. The sum of the leakage contribution for all such indications must be considered in the operational assessment of SG tubing, performed following each SG inspection. Such indications are located below the F* distance, and therefore justified for continued operation, but within the inspected length (by virtue of being detected). Postulated indications that reside below the nominal recommended inspected length of tubing (2 inches below TTS) will not significantly contribute to primary-to-secondary leakage during a postulated SLB event, and therefore can be neglected.

6.0 Conclusion

The bases surrounding the establishment of the F* methods show that these methods are conservatively developed. Application of the criterion will not represent a condition where tube structural performance criteria will be jeopardized at the end of future operating cycles. Leakage testing has shown that the likelihood of primary-to-secondary leakage is negligible, and tubes permitted to remain in service via application of the criterion will not effectively contribute to calculated primary-to-secondary leakage during postulated accident conditions. Therefore, application of the criterion does not represent a condition where the applicable leakage performance criteria will be challenged at the end of future operating cycles.

It is recommended that a single F* length of 1.16 inch, with bounding NDE uncertainty of 0.34 inch, for a final F* distance of 1.50 inches, which is the length of sound (non-degraded) roll expansion as measured from the top-of-tubesheet or bottom or roll expansion transition, whichever is lower, be applied for both the hot and cold leg regions of the WBN Unit 2 SGs.

Section 5.3 describes the impact upon the F* distance for extreme combination of variances in tube and tubesheet thermal expansion coefficient values. As Reference 6 has concluded that the observed variance in thermal expansion coefficient is likely attributed to measurement system repeatability, the results of Section 6.3 are used for information purposes only to show that extreme variance in these parameters has a negligible impact upon application of the criteria, and that the impact upon the F* distance under these conditions is balanced by the conservatism in the F* analysis methodology, including NDE uncertainty application. If TVA should choose to apply a F* distance based on this extreme consideration of thermal expansion coefficient variance, the result is that the recommended F* distance is increased from 1.50 inches to 1.64 inches.

7.0 References

1. WCAP-13084, "Tubesheet Region Tube Alternate Plugging (F*) Criterion for the Tennessee Valley Authority Watts Bar Units 1 and 2 Nuclear Power Plant Steam Generators," October 1991.
2. PCWG-08-27, Rev. 1, "Watts Bar Unit 2 (WBT): Approval of Category III (for Contract) PCWG Parameters to Support the Completion Program – Feedwater Temperature Range and Additional Reduced Power Level," September 2009.
3. U.S. Nuclear Regulatory Commission, "Wolf Creek Generating Station – Issuance of Amendment Re: Steam Generator Tube Permanent Alternate Repair Criteria," December 11, 2012.
4. LTR-SGMP-10-151, Rev. 1, "Westinghouse Input to FENOC RAI Responses for Cold Leg F* Program," November 24, 2010.
5. LTR-SGMP-10-78 P-Attachment, "Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and their Relative Importance to H*," September 7, 2010.
6. WEST-13-402, Rev. 1, "An Evaluation of the Statistical Variability in the Coefficient of Thermal Expansion Properties of SA-508 and Alloy 600,"Structural Integrity Associates, December 2008.
7. LTR-SGDA-11-157, "Response to RAI on MPR Comment on CTE Values Used in H*," June 16, 2011.
8. NEI 97-06, Revision 3, "Steam Generator Program Guidelines," Nuclear Energy Institute, Washington, D. C., January 2011.
9. *PWR Primary-to-Secondary Leak Guidelines – Revision 3*, EPRI, Palo Alto, CA: 2004. 1008219
10. Calculation Note CN-SGMP-13-7, "Watts Bar Unit 2 F* Calculation Update," March 2013.

Table 1 Verification of WCAP-13084 Analysis	
Thermal Design Parameter	MWt: N/A 0% Plugging
Vessel Outlet Temperature	620°F
Vessel Average Temperature	N/A
SG Outlet Temperature	N/A
Steam Pressure	850 psia
Normal Operating Pressure Differential	1400 psi
Faulted Pressure Differential	2650 psi
Hot Leg ΔT	550°F
Cold Leg ΔT	N/A
Tube Young's Modulus	2.87E07 psi
Tube Thermal Expansion Coefficient	7.82E-06 in/in/^{o}F
Tubesheet Young's Modulus	2.64E07 psi
Tubesheet Thermal Expansion Coefficient	7.28E-06 in/in/^{o}F
Normal Operation F* (WCAP-13084)	1.04 inch
Faulted F* (WCAP-13084)	1.06 inch
Normal Operation F* (verification)	1.04 inch
Faulted F* (verification)	1.06 inch

Table 2 WBN Unit 2 Best Estimate Operating Parameters				
Thermal Design Parameter	3427 MWt: 0% Plugging	3427 MWt: 10% Plugging	3475 MWt: 0% Plugging	3475 MWt: 10% Plugging
Vessel Outlet Temperature (^{o}F)	618.6	618.6	619.1	619.1
Vessel Average Temperature (^{o}F)	588.2	588.2	588.2	588.2
SG Outlet Temperature (^{o}F)	557.4	557.4	557.0	557.0
Steam Pressure	985	953	978	946
Normal Operating Pressure Differential (psi)	1265	1297	1272	1304
Faulted Pressure Differential (psi)	2650	2650	2650	2650
Hot Leg ΔT (^{o}F)	548.6	548.6	549.1	549.1
Cold Leg ΔT (^{o}F)	487.4	487.4	487.0	487.0

Table 3 Bounding Hot and Cold Leg Analysis Inputs (Normal Operating Conditions)		
Thermal Design Parameter	MWt: 3475	MWt: 3475
Vessel Outlet Temperature (^oF)	620	620
Vessel Average Temperature (^oF)	N/A	N/A
SG Outlet Temperature (^oF)	N/A	N/A
Steam Pressure (psi)	850	850
Normal Operating Pressure Differential (psi)	1400	1400
Faulted Pressure Differential (psi)	2650	2650
Hot Leg ΔT (^oF)	550	N/A
Cold Leg ΔT (^oF)	N/A	487

Table 4 F* Length for Various Material Property Conditions (inch)				
	Hot Leg Norm. Op.	Hot Leg Faulted	Cold Leg Norm. Op.	Cold Leg Faulted
Code of Record (1971)	0.98	1.00	0.98	0.99
1989 ASME Code	1.05	1.07	1.04	1.06
2008 ASME Code	1.10	1.14	1.12	1.16

Table 5: Combined Bobbin and RPC NDE Measurement Uncertainty Values			
	115 mil Pancake Coil	80 mil Pancake Coil	Plus Point Coil
NDE Uncertainty	0.28 inch	0.30 inch	0.34 inch

Table 6: F* Distance (Including NDE Measurement Uncertainty)			
	115 mil Pancake Coil	80 mil Pancake Coil	Plus Point Coil
F* Distance	1.44 inch	1.46 inch	1.50 inch

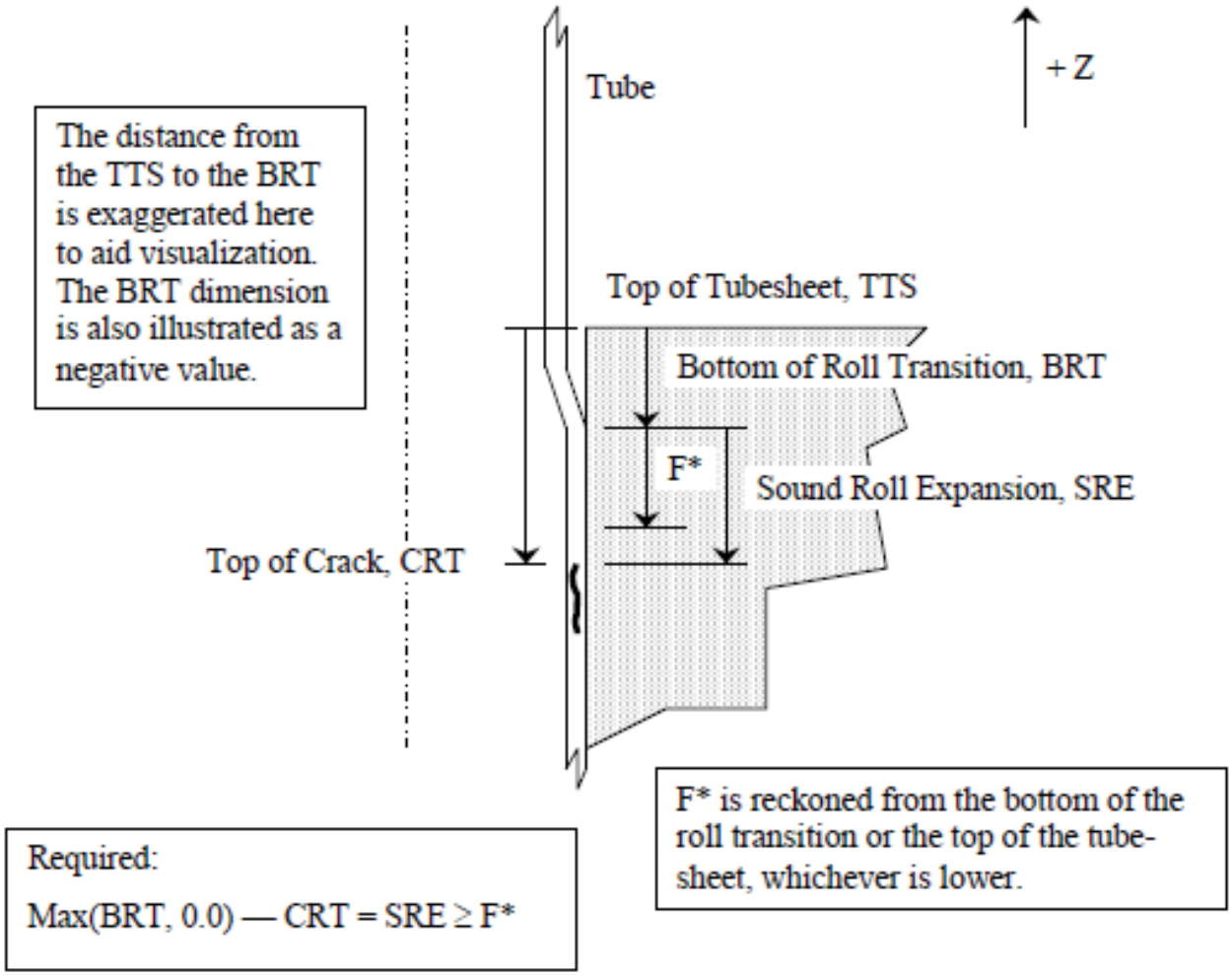

Figure 1: Representation of F* Criterion

a,b,c,e

Figure 2: Monte Carlo Simulation of SRE Measurement Performance Using +Pt Coil