Westinghouse Non-Proprietary Class 3

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Revision 0

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H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)

Westinghouse

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H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)

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1.0 INTRODUCTION

1.1 H*/B* BACKGROUND INFORMATION

In response to the detection of crack-like indications within the tube expansion region of steam generators (SGs) with Alloy 600 thermally-treated (A600TT) tubing, the NRC issued GL-2004-01 (Reference 1-14) which reiterated the requirement to inspect the full length of the tubes with probes capable of detecting potential degradation in all the areas of the steam generator (SG) unless a technical argument was available to demonstrate that specific types of degradation are not expected. Indications interpreted as primary water system stress corrosion cracking (PWSCC) were reported from the nondestructive, eddy current examination of the SG tubes during the fall 2004 outage at the Catawba Unit 2 Nuclear Power Plant (References 1-1, 1-2, and 1-3). The indications at Catawba Unit 2 were reported about 7.6 inches from the top of the tubesheet in one tube, and just above the tube-to-tubesheet welds in a region of the tube known as the tack expansion (TE) in several other tubes. The Catawba Unit 2 plant has Westinghouse designed, Model D5 SGs fabricated with A600TT tubes of 3/4 inch outside diameter. Subsequently, one indication was reported in each of two SG tubes at the Vogtle Unit 1 Plant (Reference 1-4). The Vogtle Unit 1 SGs are of the Westinghouse Model F design with 11/16 inch outside diameter A600TT tubes. The indication locations in both Catawba Unit 2 and Vogtle Unit 1 were coincident with geometric variations, termed "bulges" (BLG), in the expansion region. It was concluded from those observations that there is the potential for similar tube indications to be reported during future inspections of all SGs with hydraulically expanded A600TT tubes since geometric variations in the tubesheet expansion region are common. Since that time, several plants that have inspected through the entire thickness of the tubesheet with rotating pancake coil (RPC) have reported indications near the tube-to-tubesheet welds, in the tack expansion region.

The findings in the Catawba Unit 2 and Vogtle Unit 1 SG tubes present two distinct issues with regard to future inspections of A600TT SG tubes which have been hydraulically expanded into the tubesheet:

1. Indications may occur at internal bulges (BLG) or overexpansions (OXP) in the tubes within the tubesheet that were created as an artifact of the manufacturing process.

2. Indications may occur at the elevation of the tack expansion transition because it represents a stress riser in the tube.

Although some of the indications at Catawba were reported to be in the tube end weld, subsequent studies using a prototypic tube end test section concluded that the eddy current techniques were not capable of distinguishing the interface between the tube and weld, and further, that the indications likely were in the tube material. However, it could not be ruled out that the indications may extend into the weld. The indications were located within the tack expansion length, which, at Catawba, was made using a hard-rolling process. Thus, it was concluded that the indications that were observed all occurred in areas of potentially elevated residual stress in the tube material.

A technical evaluation is presented in this report that considers the requirements of the American Society of Mechanical Engineers (ASME) Code, Regulatory Guides, NRC Generic Letters, NRC Information Notices, the Code of Federal Regulations, NEI 97-06, and responses to NRC Request for Additional Information (RAI). The two major conclusions of the technical evaluation are that:

1. The structural integrity of the primary-to-secondary pressure boundary is unaffected by tube degradation of any magnitude below a specific depth of 13.31 inches, designated as H*, and,

2. The accident condition leak rate integrity is bounded by an overall leakage increase of 2.03 during the limiting design basis accident (DBA) relative to normal operating plant conditions. This is known as the leakage factor. Although an increase in contact pressure at accident conditions relative to normal operating conditions is not a basis for the leakage evaluation, for conservatism, it is shown that, for the Model 44F SG, the contact pressure between the tube and the tubesheet is greater at accident conditions than at normal operating conditions (NOP) for all relevant accidents.

The determination of the required engagement depth is based on the use of finite element model structural analyses and of a bounding leak rate evaluation for normal operation and postulated accident conditions. The results provide the technical rationale to exempt inspection of the region of the tube below the calculated H* elevation. Such an approach is interpreted as a redefinition of the primary-to-secondary pressure boundary relative to the original design of the SG, which requires the approval of a license amendment by the NRC Staff.

The H* values are determined to assure meeting the structural performance criteria for the operating SG tubes as delineated in NEI 97-06, Revision 2 (Reference 1-5). The leakage factors are determined based on meeting the accident condition leak rate performance criteria for all DBA that model primary-to-secondary leakage. The leakage analysis is based on a first principles application of the Darcy model for leakage through a porous medium, supported by empirical test results that show that there is no correlation between loss coefficient and contact pressure for the conditions of interest. The leakage analysis is supported by the structural analysis (Section 6.0) that shows for the Model 44F SG that the contact pressure between the tubes and tubesheet is always greater at accident conditions than at normal operating conditions.

Tests have shown that all full-depth expanded tube-to-tubesheet joints in Westinghouse-designed SGs have a residual radial preload interface pressure between the tube and the tubesheet. Residual contact pressure is not an essential element for determining a value of H* for hydraulically expanded tubes. The reference approach in this document is to assume zero contribution from residual contact pressure; however, when the existing residual contact pressure is more firmly established through additional testing, the value of H* presented in this report will be significantly smaller. Thus, the assumption of zero residual contact pressure is a conservative assumption.

1.2 DISCUSSION OF THE CALCULATION PROCESSES

The current candidate plants for H* are those plants whose SGs have Alloy 600TT tubes that are hydraulically expanded into the tubesheet. Among these are plants with Model F SGs, Model D5 SGs, Model 44F SGs and Model 51F SGs. Except for the Model 51F SGs, there are multiple plants with each of the other models of SGs. To reduce the analysis burden, a bounding plant was determined for each model of SG as discussed in Section 6.0. The value of H* determined for each of the bounding plants is the recommended H* for each of the models of SG, respectively.

This report is specifically based on the properties of the Model 44F SGs. Separate reports will be provided for the other models of SGs. While specific geometric and operating conditions are different among the various models of SGs, the methodology for the H* calculations are common to all models of SGs represented among the population of H* candidate plants.

1.2.1 Structural Integrity Analysis

The H* technical analysis consists of two essentially independent processes; the structural evaluation to define the value of H*, and the leakage analysis for the tubesheet expansion region. The structural analysis for H* is a complex analysis that involves the use of four different models as shown on the flowchart on Figure 1-1.

- A finite element structural model is used to calculate the deflections and rotations of the tubesheet complex components which include the tubesheet, channelhead, stub barrel and divider plate. The finite element model is a three-dimensional finite element model (3D FEA) using the ANSYS computer code. This model is described in detail in Section 6.0.

- An Excel®[1] (Reference 1-6) spreadsheet model utilizes the deflection and rotation output from the 3D FEA model (Reference 1-7) and a crevice pressure input based on test data to calculate the radially and axially distributed contact pressures between the tube and tubesheet for the various operating conditions. The spreadsheet also axially integrates the forces resisting tube pull out based on the contact pressures and a conservative value of coefficient of friction to define the mean value of H*. H* is defined as the distance from the top of the tubesheet at which the integrated pull out resisting force equals the applied end cap loads. The Excel® model is described in Section 6.0. The end cap force calculation applied to the tubes is described in Section 5.0.

- The third model is an Excel® spreadsheet that calculates the mean residual contact pressure based on pull out test data, and provides the residual contact pressure to the H* integrating spreadsheet discussed above. Residual contact pressure is defined as the contact pressure between the tube and the tubesheet at room temperature that results from the hydraulic expansion process. The use of this model is optional for the justification of H*; the reference calculation in this report assumes that residual contact pressure is zero.

- The variability of the residual contact pressure, also an input to the probabilistic analysis, is determined from a two-dimensional finite element model (2D FEA) (Reference 1-9). The variability of the inputs used to calculate the residual contact pressures are determined individually using an influence factor approach and combined into a single residual contact pressure variability distribution using different approaches including a Monte Carlo sampling technique. This is discussed in Section 7.0.

(1) Microsoft, MSN, and Windows Vista are trademarks of the Microsoft group of companies.

1.2.2 Leakage Integrity Approach

As discussed in Section 9.0 of this report, the expression used to predict the leak rate from tube cracks through the tube-to-tubesheet crevice is the Darcy expression for flow rate, Q, through porous media, i.e.,

$$Q = \frac{\Delta p}{12\mu K l}$$

where

μ	=	the viscosity of the fluid
Δp	=	the driving pressure differential
l	=	the physical dimension in the direction of the flow (effective crevice length)
K	=	the leakage "loss coefficient" which can also be termed the flow resistance.

The leakage analysis utilizes a ratio approach, based on the Darcy equation, to determine the ratio of leakage at accident conditions to that at normal operating conditions. It is shown in Section 9.0 that the loss coefficient is not a function of contact pressure; therefore, the loss coefficient ratio has a value of 1. It is also shown that the tube and the tubesheet are in contact for the total length of the tubesheet thickness. Therefore, the ratio of the length of the porous medium also has a value of 1. The ratio of the viscosity at accident conditions to that at normal operating conditions is also conservatively shown to be 1. Consequently, the leakage ratio is a function of only the ratio of the driving heads, that is, the ratio of the accident condition Δp to that at normal operating conditions.

1.3 SUMMARY OF CHANGES FROM PRIOR H* SUBMITTALS

1.3.1 Structural Integrity Analysis

All prior submittals of the H* technical justification (e.g., Reference 1-9) utilized the same analysis approach summarized in Section 1.2.1. However, since the last submittal by Wolf Creek Nuclear Operating Corporation (Reference 1-8, with Reference 1-9 enclosed) significant changes have been made in the structural models. The original structural model utilized a two-dimensional (2D) axisymmetric model for the tubesheet complex. A number of RAIs were issued by the NRC (see Section 2.1) that questioned the details of the application of this model. Further, questions were raised regarding the efficacy of the superpositioning approach employed with this model because it was noted that different results were obtained when the model input was condition-specific compared to the superposition results based on temperature and pressure. The process of benchmarking the 2D model utilized state-of-the-art three-dimensional (3D) finite element capabilities inherent to the ANSYS computer code. Ultimately, a new 3D model of the tubesheet complex was developed and adopted as the reference model for the structural analysis. The 2D axisymmetric model is no longer used in the current tubesheet deflection calculations supporting the analysis of H*.

Prior calculations assumed that contact pressure from the tube would expand the tubesheet bore uniformly without considering the restoring forces from adjacent pressurized tubesheet bores. In the structural model, a tubesheet radius dependent stiffness effect is applied by modifying the representative collar thickness (see Section 6.2.4) of the tubesheet material surrounding a tube based on the position of the tube in the bundle. The basis for the radius dependent tubesheet stiffness effect is similar to the previously mentioned "beta factor" approach. The "beta factor" was a coefficient applied to reduce the crevice pressure to reflect the expected crevice pressure during normal operating conditions in some prior H* calculations and is no longer used in the structural analysis of the tube-to-tubesheet joint. The current structural analysis consistently includes a radius dependent stiffness calculation described in detail in Section 6.2.4. The application of the radius dependent stiffness factor has only a small effect on the ultimate value of H* but rationalizes the sensitivity of H* to uncertainties throughout the tubesheet.

The contact pressure analysis methodology has not changed since 2007 (Reference 1-9). However, the inputs to the contact pressure analysis and how H* is calculated have changed in that period of time. The details describing the inputs to the contact pressure analysis are discussed in Section 6.0.

The calculation for H* includes the summation of axial pull out resistance due to local interactions between the tube bore and the tube. Although tube bending is a direct effect of tubesheet displacement, the calculation for H* conservatively ignores any additional pull out resistance due to tube bending within the tubesheet or Poisson expansion effects acting on the severed tube end. In previous submittals, the force resisting pull out acting on a length of a tube between any two elevations h_1 and h_2 was defined in Equation (1-1):

$$\Big[\qquad\qquad\qquad \Big]^{a,c,e} \qquad (1\text{-}1)$$

where:

F_{HE} = Resistance per length to pull out due to the installation hydraulic expansion,

d = Expanded tube outer diameter,

P = Contact pressure acting over the incremental length segment dh, and,

μ = Coefficient of friction between the tube and tubesheet, conservatively assumed to be 0.2 for the pull out analysis to determine H*.

The current H* analysis generally uses the following equation to determine the axial pull out resistance of a tube between any two elevations h_1 and h_2:

$$\Big[\qquad\qquad\qquad \Big]^{a,c,e} \qquad (1\text{-}2)$$

Where the other parameters in Equation (1-2) are the same as in Equation (1-1) and [

]^{a,c,e}. A detailed explanation of the revised axial pull out equation are included in Section 6.0 of this report. However, the reference basis for the H* analysis is the assumption that residual contact pressure contributes zero additional resistance to tube pull out. Therefore, the equation to calculate the pull out resistance in the H* analysis is:

[ ] a,c,e (1-3)

1.3.2 Leakage Integrity Analysis

Prior submittals of the technical justification of H* (Reference 1-9) argued that K was a function of the contact pressure, P_c, and, therefore, that resistance was a function of the location within the tubesheet. The total resistance was found as the average value of the quantity μK, the resistance per unit length, multiplied by L, or by integrating the incremental resistance, $dR = \mu K\, dL$ over the length L, i.e.,

$$R = \mu\bar{K}(L_2 - L_1) = \mu\int_{L_1}^{L_2} K\, dL \qquad (1\text{-}4)$$

Interpretation of the results from multiple leak rate testing programs suggested that the logarithm of the loss coefficient was a linear function of the contact pressure, i.e.,

$$\ln K = a_0 + a_1 P_c\,, \qquad (1\text{-}5)$$

where the coefficients, a_0 and a_1 of the linear relation were based on a regression analysis of the test data; both coefficients are greater than zero. Simply put, the loss coefficient was determined to be greater than zero at the point where the contact pressure is zero and it was determined that the loss coefficient increases with increasing contact pressure. Thus,

$$K = e^{a_0 + a_1 P_c}\,, \qquad (1\text{-}6)$$

and the loss coefficient was an exponential function of the contact pressure.

The B* distance (L_B) was defined as the depth at which the resistance to leak during SLB was the same as that during normal operating conditions (NOP) (using Equation 1-4, the B* distance was calculated setting $R_{SLB} = R_{NOP}$ and solving for L_B). Therefore, when calculating the ratio of the leak rate during the design basis accident condition to the leak rate during normal operating conditions, the change in magnitude of leakage was solely a function of the ratio of the pressure differential between the design basis accident and normal operating plant conditions.

The NRC Staff raised several concerns relative to the credibility of the existence of the loss coefficient versus contact pressure relationship used in support of the development of the B* criterion:

1. The Model F SG loss coefficient versus contact pressure plot exhibits a higher slope than the case for the Model D5 SG (Reference 1-10).

2. Although the mean of the regression fits for the loss coefficient data for the Model F and the Model D5 SGs are within a factor of three (3) of each other, the slope and intercept properties remain highly divergent (Reference 1-11).

3. The Model D5 loss coefficient data is spread out in range and results in a slightly negative log-linear correlation (Reference 1-11).

The current approach to the leakage analysis shows that there is no significant correlation between loss coefficient and contact pressure based on the available test data. A ratio approach, using the Darcy formulation as noted above and as described in detail in Section 9.0, is the current reference basis for leakage ratio calculations.

1.3.3 Probabilistic Analysis

At a meeting in July 2008, the NRC requested a probabilistic evaluation of H*. Probabilistic evaluations of H* had not been performed. Previously, a limiting worst-case analysis was provided (Reference 1-11) that was based on an H* variability study on individual inputs parameters. The worst-case values of the variables were then combined into an integrated case that resulted in a high probability value of H*. This approach was not accepted as noted in the remaining technical concerns issued in Reference 1-12.

Because of the complexity of the H* calculations (see Section 1.2.1) that involve the combined use of four different models, a pure Monte Carlo approach was not possible. The current analysis of H* is based principally on the semi-statistical approach outlined in the EPRI Integrity Assessment Guidelines (Reference 1-13), in which the uncertainties are combined using a square root of sum of squares (SRSS) approach. Further, to support the SRSS approach, a Monte Carlo approach to the H* calculation was developed that utilized influence factors. For the influence factor approach, a distribution of H* in a single input variable is determined while maintaining all other input variables at their nominal values. This process is completed for each input variable, resulting in H* distributions in every input variable. Monte Carlo sampling is performed from these distributions to develop the integrated variability of H* in all variables. The probabilistic analysis for H* is included in Section 8.0 of this report.

In response to the residual technical issues identified by the Staff, the capability to provide residual contact pressure variability as an input to the H* integration model was developed. The mean value of residual contact pressure is based on test data, and the variability around the mean value is determined for each relevant input variable based on analysis. The individual variability distribution for residual contact pressure is combined in the same manner as discussed above for the probabilistic H* determination. It is noted that the reference H* calculation provided in this report assumes residual contact pressure to be zero. Any positive value of residual contact pressure will decrease the final value of H*.

1.4 CONSERVATISMS IN THE H* ANALYSIS

A conservative approach was taken for the calculation of H*. Notwithstanding that the underlying structural integrity and leakage requirements are inherently conservative, e.g., application of a factor of three (3) on expected normal operating pressure differentials, other conservative assumptions were made that provide significant confidence in the predicted value of H* and the leakage factors. Table 1-1 summarizes the significant conservative assumptions and approaches included in the calculations for H*.

1.5 REPORT OVERVIEW

Section 1.0 provides an introduction to WCAP-17091-P. Section 2.0 provides information on the resolution of all technical issues and NRC requests for additional information on this topic. Section 3.0 addresses the test programs in support of the technical justification of H*. Section 4.0 addresses the structural and leakage analysis acceptance criteria. Section 5.0 discusses the plant operating conditions at the H* plants with Model 44F SGs. Section 6.0 discusses the structural analyses of the tube-to-tubesheet joint. Section 7.0 addresses residual contact pressure and its variability. Section 8.0 uses the results provided in Section 6.0 and Section 7.0 to define the H* values as a function of tubesheet radial location for each of the H* plants for normal operating, postulated steam line break, and feedwater line break conditions to provide a probabilistic assessment of the H* value. Section 9.0 discusses the details of the leakage analysis. Finally, Section 10.0 provides the conclusion of this report.

1.6 REFERENCES

1-1 OE19662 (Restricted & Confidential), "Steam Generators (Catawba Nuclear Power Station)," Institute of Nuclear Power Operations (INPO), Atlanta, GA, USA, December 13, 2004.

1-2 IN 2005-09, "Indications in Thermally-Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," United States Nuclear Regulatory Commission, Washington, DC, April 7, 2005.

1-3 SGMP-IL-05-01, "Catawba Unit 2 Tubesheet Degradation Issues," EPRI, Palo Alto, CA, March 4, 2005.

1-4 OE20339, "Vogtle Unit 1 Steam Generator Tube Crack Indications," Institute of Nuclear Power Operations (INPO), Atlanta, GA, USA, April 4, 2005.

1-5 NEI 97-06, Rev. 2, "Steam Generator Program Guidelines," Nuclear Energy Institute, Washington, DC, May 2005.

1-6 LTR-SST-05-19, Rev. 1, "System State Equivalency Testing Not Required for Windows XP SP-2," June 20, 2005.

1-7 LTR-SST-08-16, "ANSYS 10.0 for HP-UX 11.23i Release Letter," March 28, 2009.

1-8 WCNOC Letter ET-06-0004, "Revision to Technical Specification 5.5.9, Steam Generator Tube Surveillance Program," February 21, 2006.

1-9 LTR-CDME-05-209-P, "Steam Generator Tube Alternate Repair Criteria for the Portion of Tube Within the Tubesheet at the Wolf Creek Generating Station," January 2006.

1-10 LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," April 24, 2007 (Enclosure to WCNOC Letter WO 07-0012, dated May 3, 2007).

1-11 LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," September 24, 2007 (Enclosure to WCNOC Letter ET07-0043, dated September 27, 2007).

1-12 NRC Letter, "Wolf Creek Generating Station – Withdrawal of License Amendment Request on Steam Generator Tube Inspections (TAC No. MD1097)," February 28, 2008.

1-13 Steam Generator Integrity Assessment Guidelines, Revision 2, EPRI, Palo Alto, CA, July 2006, 1012987.

1-14 NRC Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections," August 30, 2004.

a,c,e

Figure 1-1 Analysis Process for H*

Table 1-1 List of Conservatisms in the H* Structural and Leakage Analysis

Assumption/Approach	**Why Conservative?**
The NEI 97-06 performance criteria, which address tube burst, are applied by equating failure to meet the H* distance with tube burst .	Tube burst cannot occur within the tubesheet (see Section 4.1), thus application of the same criteria designed to prevent tube burst in an area where tube burst cannot occur is inherently conservative. Prevention of tube burst is a necessity for preventing excessive leakage, and accident-induced leakage in the tubesheet expansion region is shown to be limited, independent of the H* distance. Therefore, equating failure to meet H* with tube burst, and application of the same criteria to prevent tube burst to H*, is inherently conservative.
H* distances are based on analysis of the worst tube in the bundle.	The distribution of the contact pressure between the tube and the tubesheet varies as a function of radial position in the tubesheet; *the worst-case tube location is used to establish the H* distance* (see Section 6.2.3). All other tubes have lower H* values.
Structural support from the divider plate is ignored.	The H* distances for a severely degraded divider plate (no connection between the tubesheet and the divider plate) bound the H* distances for a non-degraded divider plate (see Section 6.2.6).
Residual Contact Pressure Assumed to be Zero.	All pull out tests to date have shown that there is residual contact pressure from the hydraulic expansion; any non-zero value will decrease H* (see Section 7.0 and Appendix A).
Calculation of Pull out Force.	Assumes mean plus 2 sigma tubesheet bore diameter as basis for tube cross-sectional area (see Section 5.3).
Coefficient of Thermal Expansion.	Use of ASME Code mean is conservative relative to test data for both tubesheet and tubing material (see Section 3.5 and Appendix B).
Coefficient of Friction.	Lower bound value of [ ]a,c,e is used in the determination of the H* distance (see Section 6.2.2.3.3). Standard reference values suggest a reasonable value of coefficient of friction is [ ]a,c,e.
Darcy equation used to model leakage analysis.	The assumed linear relationship between leak rate and differential pressure is conservative relative to alternate models such as Bernoulli or orifice models which assume the leak rate to be proportional to the square root of the pressure differential (see Section 9.1.1 and Reference 9-5 of Section 9.0).

Table 1-1 List of Conservatisms in the H* Structural and Leakage Analysis (Continued)

Assumption/Approach	Why Conservative?
Use of different plant temperature and pressure conditions for structural and leakage calculations.	The conditions that maximize H* are different from those that maximize leakage conditions. Separate maximizing assumptions are made for structural and leakage analysis (see Section 6.4.5 for the structural analysis assumptions and Section 9.4 for the leakage analysis assumptions). Bounding limit values for the most limiting plant operating pressure and temperatures which include maximum licensed steam generator tube plugging levels (i.e., in numbers of tubes plugged) are used to establish the H* distances for the Model 44F SGs (see Section 5.0). A combination of [ ]^{a,c,e} are used for the structural evaluation (see Section 6.2.2.2.2 and Section 6.2.2.2.5). [ ]^{a,c,e} conditions are used for evaluating the overall leakage factors (to maximize the pressure difference ratio between design basis accident conditions and normal operating conditions) (see Section 9.4).
H* distances based on hot leg temperatures and pressure.	The results described in this report conservatively bound the requirements for both the hot leg and the cold leg in any Model 44F SG (see Section 6.2.2.2.3).
Stiffening effect of the presence of tubes ignored in the structural analysis.	Equivalent properties of the tubesheet are calculated without taking credit for the stiffening effect in the tubes, which results in a conservatism in the calculations regarding tubesheet deflection (see Section 6.2.1).
Some local interactions between the tube bore and the tube are ignored.	Additional pull out resistance due to tube bending within the tubesheet or Poisson expansion effects on the severed tube end are ignored (see Section 1.3.1).
Peak reactor coolant system pressures and temperatures are assumed to exist during the entire design basis accidents.	Time varying, or transient pressures and temperatures would reduce the pressure and thermal loads on the tube and the tubesheet (see Section 6.2.2).

Table 1-1 List of Conservatisms in the H* Structural and Leakage Analysis (Continued)

Assumption/Approach	Why Conservative?
A [ ].^{a,c,e}	This is conservative because it reduces the stiffness of the solid and perforated regions of the tubesheet to the lowest level *for each operating condition (see Section 6.2.2.2.2).*
Pressure is not applied to the [ ].^{a,c,e}	Applying pressure to the [ ]^{a,c,e} (see Section 6.2.2.2.4).
The radius dependent stiffness analysis ignores the presence of the [ ].^{a,c,e}	Including these structures in the analysis would reduce the tubesheet displacement and limit the local deformation of the tubesheet hole ID (see Section 6.2.4.4).
The tubesheet bore dilation [ ]^{a,c,e}	Thermal expansions under operating loads were [ ]^{a,c,e} (see Section 6.2.5).

2.0 RESOLUTION OF TECHNICAL ISSUES AND NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) FROM PRIOR H* SUBMITTALS

2.1 CATEGORIZATION OF TECHNICAL ISSUES AND RESOLUTION ROAD MAP

The open technical issues identified by the NRC Staff are included in Reference 2-1. Generally, the significant remaining technical issues are in the following categories:

1. Determination of residual contact pressures and variability of residual contact pressure.

2. Adequacy of the existing tube pull out data to justify residual contact pressure when potentially larger values of H* may be determined.

3. Justification of the mean values and variability of the coefficient of thermal expansion for the tubesheet material (SA508) and the tubing material (A600).

4. Leakage loss coefficient as a function of tube-to-tubesheet contact pressure.

5. Consideration of the potential for incremental tube slippage during pressure and temperature cycles.

Table 2-1 provides a listing of the remaining technical issues related to steam generator (SG) tube inspections based on the H*/B* methodology that were identified in Reference 2-1 and a road map to where these issues are addressed within this report. Since the issuance of Reference 2-1, four additional issues have been identified during NRC/Industry meetings. These issues are labeled as A**, B**, C**, and D** and are also resolved in this report.

2.2 REVIEW OF PRIOR NRC REQUESTS FOR INFORMATION

Wolf Creek Nuclear Operating Corporation (WCNOC) submitted a license amendment request on February 21, 2006 (Reference 2-4) proposing changes to the Technical Specifications for the Wolf Creek Generating Station. The proposed changes were to revise the Technical Specification to exclude portions of the SG tube for a distance from the top of the tubesheet in the SGs from periodic tube inspections based on the application of structural analysis and leak rate evaluation results to re-define the primary-to-secondary pressure boundary. The NRC Staff provided an initial Request for Additional Information (RAI) on June 27, 2006 (Reference 2-5). Subsequently, a second NRC Staff RAI was received by WCNOC via electronic mail on June 22, 2007. The second NRC Staff RAI was documented in Reference 2-6. Responses to these two sets of NRC RAI are included in References 2-2 and 2-3.

All previously issued NRC RAI are identified in Table 2-2 below along with a summary of either the resolution of the issues or identification of where the previous NRC RAI are addressed in this report.

2.3 REFERENCES

2-1 NRC Letter, "Wolf Creek Generating Station – Withdrawal of License Amendment Request on Steam Generator Tube Inspections (TAC No. MD0197)," February 28, 2008.

2-2 LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," Westinghouse Electric Company LLC, Pittsburgh, PA, April 24, 2007 (Enclosure to WCNOC Letter WO 07-0012, dated May 3, 2007).

2-3 LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," Westinghouse Electric Company LLC, Pittsburgh, PA, September 24, 2007 (Enclosure to WCNOC Letter ET 07-0043, dated September 27, 2007).

2-4 WCNOC Letter ET 06-0004, "Revision to Technical Specification 5.5.9, Steam Generator Tube Surveillance Program," February 21, 2006.

2-5 NRC Letter, "Wolf Creek Generating Station – Request for Additional Information (RAI) Related to License Amendment Request (LAR) to Revise the Steam Generator Program (TAC No. MD0197)," June 27, 2007.

2-6 NRC Letter, "Meeting with Representatives of Wolf Creek Nuclear Operating Corporation for Wolf Creek Generating Station (TAC No. MD0197)," August 7, 2007.

Table 2-1 NRC Technical Issue Response Road Map

Technical Issue No.	Issue Description	Report Section Addressing Technical Issue
1	Contact pressure between the tube and the tubesheet (Need to define method for computing residual contact pressure from pull out tests)	Section 7.1(1)
2		
3	Allowed degree of slippage at tube pull out loads	Appendix A(2)
4	Dimensions and yield strength of test specimens	Appendix A(2)
5		
6	Pull out test database adequacy for uncertainties	Section 7.2(3)
7		
8		
9		
10	Thermal expansion coefficient values and variability	Section 3.1 and Appendix B
11	Statistical performance standard for H* adequacy	Section 4.1
12	Propagate input uncertainties to H* uncertainties	Section 7.0 and Section 8.0
13	Accuracy of 2-D Finite Element tubesheet model	Sections 6.1.2
14	Error in the unit load FE analyses for SLB	Section 6.1.2.1.5
15	Input random versus systematic uncertainties	Section 8.1.3 and Section 8.2.2
16	Incremental slippage under normal operation and monitoring	Section 9.8
17	Need to assess accident leakage for feedwater line break (Not Applicable to Model 44F SG)	Section 9.2.3
18	Conservatism of "limiting median crevice pressure approach"	Section 6.4.8 and Section 8.1.1
19	Beta factor adjustment to crevice pressure (tubesheet stiffness)	Section 6.2.4
20	Consider assumptions on divider plate condition	Section 6.2.6
A**	Effects of hole dilation on leakage and contact pressure	Section 6.2.5
B**	Thermal expansion coefficient in the radial direction	Section 3.4 and Appendix B
C**	3D-FEA discrepancies with ANL (gap under DBA)	Section 6.4.6
D**	Accident Leakage Integrity	Section 9.2
** Identified based on Industry activities after February 2008		

(1) Residual contact pressure conservatively assumed to be zero in this report.

(2) Only previous pull out test program results are included in this report. New pull out test results were not available at the time of printing of this report.

(3) Residual contact pressure uncertainties are addressed analytically on this report.

Table 2-2 List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-72 (Reference 2-2)
1	Enclosure 1 of the application, Sections 6.1 and 6.2 - What were the actual yield strengths and wall thicknesses of the tube specimens used for pull out and leakage testing? How do these values compare to minimum values of these parameters at Wolf Creek? Discuss the effect of tube yield strength and wall thickness on contact pressure between the tube and tubesheet after the tube expansion process (i.e., ignoring pressure and temperature loads). Discuss why the test specimen strengths and wall thicknesses were conservative from the standpoint of minimizing the contact pressures between the tube and tubesheet, or discuss what adjustments need to be made to test results to allow for the variability of yield strength and tube wall thickness. Issue Resolution Summary: Additional tube pull and leakage data for the original test specimens as requested by the NRC Staff is provided in Appendix A of this report. Other than to provide specific information about the test specimens used in the pull out test, additional test data, together with a new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants obviate the need to compare the original test data yield strengths and tube wall thicknesses with the tubes at Wolf Creek as requested in this RAI. This RAI has been superseded by Item Numbers 4 and 5 of the list of issues that were outstanding when the Wolf Creek Generating Station amendment was withdrawn. The road map for the resolution of these issues is provided in Section 2.1 of this report.
2	Enclosure 1, Section 6.2.1 - The section states that the leak test program utilized tubesheet simulants (collars) with the nominal tubesheet hole diameter. Was this also the case for the pull out tests? What were the diameters of the tube specimens used in the pull out and leakage tests? Discuss the effect that the field tolerances on these parameters can have on contact pressure between the tube and tubesheet after the tube expansion process (i.e., ignoring pressure and temperature loads). Discuss why the parameter values used for the test specimens were conservative from the standpoint of minimizing the contact pressures between the tube and tubesheet, or discuss what adjustments need to be made to test results to allow for the variability of these parameters. Issue Resolution Summary: In response to the residual technical issues identified by the Staff, the capability to provide residual contact pressure variability as an input to the H* integration model was developed. The mean value of residual contact pressure is based on test data, and the variability around the mean value is determined for each relevant input variable based on analysis (see Section 7.0 of this report). The individual variability distribution for residual contact pressure are combined in the same manner as discussed above for the probabilistic H* determination (see Section 8.0 of this report). It is noted that the reference H* calculation provided in this report assumes residual contact pressure to be zero. Any positive value of residual contact pressure will decrease the final value of H*.
3	Enclosure 1, Section 6.1, page 27 of 127 - Why was the pull out data evaluated at the lower 95th percentile? Discuss how this supports the ability of tubes to sustain pull out loads, versus using an absolute lower bound value? Given the limited number of tests performed (and the many thousands of tubes in the SGs), should not the lower bound value be evaluated to a high confidence value? Issue Resolution Summary: See the response to NRC RAI 2 above.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-72 (Reference 2-2)
4	Enclosure 1, Section 6.2.1.2 - The section states that the hydraulic expansion pressure was approximately [proprietary information]. Was hydraulic expansion pressure a measured parameter during SG fabrication that was used for acceptance of each joint? Was the lower limit of the acceptance standard the same as the lower limit of the assumed [proprietary information]? If the answer to either of these questions is no, what is the basis for the assumed [proprietary information]? Issue Resolution Summary: See the response to NRC RAI 2 above.
5	How does pressure and temperature cycling affect the pull out and leakage resistance of the joints? Cite the available data on this topic, and why it is appropriate that the proposed inspection depths need not specifically account for such cycling. Issue Resolution Summary: This RAI has been superseded by Item Number 16 of the list of issues that were outstanding when the Wolf Creek Generating Station amendment was withdrawn. The road map for the resolution of these issues is provided in Section 2.1 of this report.
6	Pull out resistance per unit length associated with the tube expansion process (residual pull out resistance) was determined on the basis of pull out tests and on the assumption that pull out resistance is uniform along the length of the joint. The axial force in the tube is maximum at the top of the tubesheet and decreases as joint friction incrementally picks up some of the load with increasing distance into the tubesheet. As axial force in the tube declines, with increasing distance in the tubesheet, the Poisson's contraction of the tube diameter decreases causing contact pressure to increase until it reaches a constant value at the location where axial force in the tube has been reduced to zero. At the pull out load, the pull out resistance per unit length near the bottom of the joint will be higher than the average pull out resistance along the entire joint. The pull out resistance over the upper portion of the joint will be less than the average resistance. Referring to Tables 7-6 to 7-10 in Enclosure 1, would not consideration of the actual distribution of the residual pull out resistance as a function of distance below the top of the tubesheet lead to larger H* values than shown on these tables? If not, explain why not. Issue Resolution Summary: See the response to NRC RAI 2 above.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-72 (Reference 2-2)
7	The models used to develop the H* lengths are complex. Describe how these models have been verified to yield conservative H* values. Have these models been verified by test? For example, how well do these models predict the actual residual pull out loads for joint test samples with typical H* lengths (i.e., provide comparative data)? Issue Resolution Summary: This RAI has been superseded by Item Number 12 of the list of issues that were outstanding when the Wolf Creek Generating Station amendment was withdrawn. The road map for the resolution of this issue is provided in Section 2.1 of this report.
8	Enclosure 1, Section 6.2.2 - The section states that room temperature leakage tests were performed on all test specimens at test pressures of 1900, 2650, and 3100 pounds per square inch (psi) (presumably applied on the primary side with nothing more than atmospheric pressure at the top of the joint). However, Table 6-2 only presents room temperature data for a differential pressure of 1000 psi. Where is this latter data discussed? Why aren't the room temperature data for the tests described in Section 6.2.2 included in Table 6-2 and Figure 6-6? Issue Resolution Summary: The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.
9	Enclosure 1, Section 6.2.2-1 - The section states that the elevated temperature tests were performed following the room temperature tests. Section 6.2.2.2 states that the room temperature tests were performed following the elevated temperature tests. Please clarify this discrepancy. Issue Resolution Summary: The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-72 (Reference 2-2)
10	Enclosure 1, Section 6.2.2-2 - The section states that a 1900 psi test pressure was used (simulating normal operating pressure) to keep the pressurizing fluid above saturation pressure. As the Staff understands the report, the pressure at the upper end of the test joint is at atmospheric pressure which is not prototypic for normal operating conditions. As the test leakage goes from the bottom of the joint to the top, pressure at some point drops to less than saturation. Why would the test be expected to show as much leakage through the joint as would be the case under prototypic normal operating conditions? Issue Resolution Summary: The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies. This RAI has been superseded by Item Number 18 of the list of issues that were outstanding when the Wolf Creek Generating Station amendment was withdrawn. The road map for the resolution of this issue is provided in Section 2.1 of this report.
11	The plot of Model F loss coefficient versus contact pressure in Figure 6-6 of Enclosure 1 exhibits a higher slope than is the case for Model D5. The difference appears attributable to lower loss coefficients at lower contact pressures for Model F than for Model D5. Discuss the differences between the Model F and D5 SG designs that explain their different behaviors. If no significant design differences can be identified, discuss the credibility of the loss coefficient data. Issue Resolution Summary: This RAI has been superseded by Item Number D** of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of this issue is provided in Section 2.1 of this report.
12	Enclosure 1, Section 6.2.2.1 - The section states that the leak test results averaged 16 drops per minute (dpm) per joint at 1900 psi compared to 59 dpm at higher pressures. This is a factor of 3.7 difference. Discuss why this difference is so high compared to the factor of 2 which, under the bellwether principle, is assumed to bound the increase in leakage going from normal operating to accident conditions. Issue Resolution Summary: The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-72 (Reference 2-2)
13	Enclosure 1, Section 7.1.2, page 45 of 127: Was the primary pressure unit load applied only to the primary face of the tubesheet, and not to the side of the tubesheet bore holes? Was the secondary pressure unit load applied only to the secondary face of the tubesheet, and not to the side of the tubesheet bore holes? Was the tube end cap pressure load (due to primary and secondary pressures) included in the finite element analyses? Issue Resolution Summary: The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies. This RAI has been superseded by Item Number 19 of the list of issues that were outstanding when the Wolf Creek Generating Station amendment was withdrawn. The road map for the resolution of this issue is provided in Section 2.1 of this report.
14	Enclosure 1, Section 7.1.2, page 45 of 127: The 500 of unit loads represent which of the following; heating up from 70 to 500 °F, or from 70 to 570 °F? If the former, why isn't 70 °F subtracted from 500 °F in the radial deflection scaling factors in Section 7.1.3 (page 46 of 127)? Issue Resolution Summary: The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.
15	Enclosure 1: Regarding the equation for Δ RprTS top of page 48 of 127, should not Pi be Po consistent with the last equation appearing on page 48? If not, why not? Issue Resolution Summary: The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-72 (Reference 2-2)
16	Enclosure 1, Section 7.1.3 - The tube inside and outside radii within the tubesheet after expansion shown on page 49 of 127 appear not to be entirely consistent with the numbers on page 44 of 127. Explain this inconsistency or, alternatively, show that this inconsistency does not significantly affect the outcome of the overall analysis. Issue Resolution Summary: The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.
17	Enclosure 1, Section 7.1.4 - Near the top of page 50 of 127, it is stated that the secondary pressure is conservatively assumed to act on the outside of the tube and the inside of the tubesheet hole. The Staff agrees that this is conservative from the standpoint of maximizing leakage under normal operating conditions, but is concerned that it may be non-conservative from the standpoint of determining conservative ratios of accident leakage to normal operating leakage. Wouldn't the assumption of no secondary pressure yield a lesser value of normal operating leakage, leading to a higher ratio of accident to normal operating leakage? What is the basis for describing the assumption on secondary pressure as conservative? Issue Resolution Summary: This RAI has been superseded by Item Numbers 18, 19 and D** of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of these issues is provided in Section 2.1 of this report.
18	Enclosure 1, Section 8.2 - The ligament tearing discussion in Section 8.2 (starting on page 75 of 127) only addresses circumferential cracks. Please provide corresponding discussion for axial cracks. Issue Resolution Summary: The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies. The original response is also included as Section 9.7.2 of the Final H* Report.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-72 (Reference 2-2)
19	The structural and leakage assessments supporting the proposed technical specification amendment are for tubes with no degradation in the proposed inspection zone. The proposed inspection depths make no allowance for degradation which may occur within this zone prior to the next scheduled inspection. Assess the potential impact of degradation in the inspection zone on (1) contact pressures between the tube and tubesheet, (2) on tube pull out capacity, and (3) on leakage under normal and accident conditions. (Although flaws in this zone will be plugged on detection, this question is relevant to satisfying the tube integrity performance criteria with respect to condition monitoring and operational assessments.) This assessment should address potential axial and circumferential stress corrosion cracks (SCC) and volumetric intergranular attack (IGA) flaws. Issue Resolution Summary: The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.
20	Describe the methodology to be employed for performing condition monitoring and operational assessments for the tubesheet inspection zone (for pull out and accident leakage) assuming that SCC and or IGA mechanisms have started to be active. Issue Resolution Summary: The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.
21	Enclosure 1: The development of the B* distances assumes that crack leakage resistance is not significant relative to the tube-to-tubesheet joint resistance. Discuss the conservatism of the B* distances given the assumption that crack leakage resistance is the dominant resistance to leakage under normal operating conditions. To the extent this discussion relies on assumptions about contact pressure between the tube and tubesheet local to the crack, justify assumptions relative to the influence of the crack on local contact pressure. Issue Resolution Summary: The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-72, "Response to NRC Request for Additional Information Relating to LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-72 (Reference 2-2)
22	Describe the methodology for performing condition monitoring and operational assessments for accident induced leakage stemming from locations below the specified tubesheet inspection depths. <u>Issue Resolution Summary</u>: This RAI has been superseded by Item Number D** of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of this issue is provided in Section 2.1 of this report.
23	By letter dated March 28, 2006, you provided revisions to your proposed technical specifications (TS) in accordance with TSTF-449, Rev. 4, to include the following additional sentence into TS 5.5.9 c.1: "All tubes with degradation identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be removed from service." Describe your plans for revising these words to reflect the February 21, 2006 license amendment and for submitting revisions to this amendment. <u>Issue Resolution Summary</u>: This RAI does not apply to the Model 44F H* plants going forward.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-72 (Reference 2-2)
24	Discuss your plans to revise TS 5.6.10 to include reporting requirements applicable to the implementation of the tubesheet inspection and alternate repair criteria. For example: *A breakout of indications detected within the tubesheet inspection depths with respect to their location, orientation, and measured size. (The only difference here relative to proposed changes associated with Technical Specification Task Force (TSTF) 449, Revision 4, is that the indications in the tubesheet region would be listed separately from those elsewhere.) *The operational primary to secondary leakage rate observed in each steam generator during the cycle preceding the inspection which is the subject of the report, and (2) the calculated accident leakage rate for each steam generator from the portion of tubing below the tubesheet inspection depths for the most limiting accident. If the calculated accident leakage rate for any steam generator is less than 2 times the total observed operational primary to secondary leakage rate, the 12-month report should describe how it was determined. <u>Issue Resolution Summary</u>: Proposed changes to the technical specification for the steam generator tube inspection report are provided by the utility as part of the license amendment request.
25	Enclosure 1, Section 7.1.3, page 46 of 127: The tubesheet bow analysis takes credit for resistance against bow provided by the divider plate. Cracks in the welds connecting the tubesheet and divider plate have been found by inspection at certain foreign steam generators. Describe what actions you are taking to ensure that the divider plates can perform their function, including providing the assumed resistance against tubesheet bow. <u>Issue Resolution Summary</u>: This RAI has been superseded by Item Number 20 of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of this issue is provided in Section 2.1 of this report.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
1	Reference 1, Enclosure I, Table 6-4 - Are the listed F/L, force per length, values correct? If so, please describe in detail how they were calculated. If not correct, please provide all necessary revisions to the H* analysis results. [For Byron 2, Braidwood 2, and Seabrook, F/L is calculated as follows: F/L = (Pull Force/specimen length) x (net contact pressure/total contact pressure) A consistent approach for Wolf Creek (based on allowing 0.25 inch slip) would yield F/L values on the order of 200 pounds per inch (lb/inch) rather than 563 lb/inch as shown in the Table.] Issue Resolution Summary: In response to the residual technical issues identified by the Staff, the capability to provide residual contact pressure variability as an input to the H* integration model was developed. The mean value of residual contact pressure is based on test data, and the variability around the mean value is determined for each relevant input variable based on analysis (see Section 7.0 of this report). The individual variability distribution for residual contact pressure are combined in the same manner as discussed above for the probabilistic H* determination (see Section 8.0 of this report). It is noted that the reference H* calculation provided in this report assumes residual contact pressure to be zero. Any positive value of residual contact pressure will decrease the final value of H*. This RAI has been superseded by Item Numbers 1 and 2 of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of these issues is provided in Section 2.1 of this report.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
2	Reference 2, Enclosure I, Response to RAI questions 1 and 2 - provides the sensitivity of contact pressure to many of the material and geometric parameters used in the analyses. The response provides only a qualitative assessment of these sensitivities to support the conclusion that the values assumed in the H* analyses support a conservative calculation of H*. For example, the sensitivity study showed that contact pressure is sensitive to the yield strength of the tubing. The response states that the yield strength of the tubing used in the pull out test specimens was higher than the documented mean yield strength for prototypical tubing material, but did not indicate to what extent the yield strength of the test material bounds the range of prototypic yield strength variability. Thus, the Staff has no basis to agree or disagree with the conclusion that test specimen contact pressures are conservatively low. The steam generators contain up to 5620 tubes, and it needs to be demonstrated that the computed H* distances are conservative for all the tubes, not simply the average tubes or 95% of the tubes. Please provide a quantitative assessment demonstrating that the assumed values of the material and geometric parameters support a conservative H* analysis for all tubes. This assessment should consider thermal expansion coefficient (TEC) for the tube and tubesheet in addition to the parameters included in the Reference 2 response. Issue Resolution Summary: This RAI has been superseded by Item Number 9 of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of this issue is provided in Section 2.1 of this report.
3	The H* analyses in References 1 and 2 are based, in part, on pull out resistance associated directly with hydraulic expansion process. This pull out resistance was determined by subtracting out the effects of differential thermal expansion between the tube and tubesheet test collar from the measured pull out load. The calculated differential thermal expansion effect was based, in part, on an assumed TEC value of 7.42E-06 in/in/°F for the 1018 steel tubesheet test collar. What is the impact of considering an alternative TEC value of 7E-06 in/in/°F (from Matweb.com for 1018 steel interpolated at 600 degrees Fahrenheit) on the computed pull out force determined from the pull out test and on the computed H* distances? Issue Resolution Summary: This RAI has been superseded by Item Number 12 of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of this issue is provided in Section 2.1 of this report.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
4	Reference 2, Enclosure I, Response to RAI question 7 - The Model D5 steam generator (SG) pull out data in Table 2 indicate that pull out force increases with temperature for the 3-inch long specimens and decreases with temperature for the 6-inch long specimens. For the 4-inch specimens, pull out force increases with temperature to 400°F and decreases with temperature beyond that point. Discuss the reasons for this apparent discrepancy in trends among the data. Discuss whether the reduction in tube yield strength with temperature might be sufficient for some specimens to limit any increase in contact pressure associated with differential thermal expansion between the tube and tubesheet. Issue Resolution Summary: In response to the residual technical issues identified by the Staff, the capability to provide residual contact pressure variability as an input to the H* integration model was developed. The mean value of residual contact pressure is based on test data, and the variability around the mean value is determined for each relevant input variable based on analysis (see Section 7.0 of this report). The individual variability distribution for residual contact pressure are combined in the same manner as discussed above for the probabilistic H* determination (see Section 8.0 of this report). It is noted that the reference H* calculation provided in this report assumes residual contact pressure to be zero. Any positive value of residual contact pressure will decrease the final value of H*.
5	Following up on question 4 above, is there a possibility that any tubes could be stressed beyond the compressive yield strength (at temperature) of the tube material due to differential thermal expansion, internal pressure, and tubesheet hole dilation for the range of yield strengths in the field? Describe the basis for either yes or no to this question. If yes, how has this been factored into the contact pressures, accumulated pull out resistance load as a function of elevation, and H* in Tables 7-6 through 7-10 and 7-6a through 7-10a of Reference 2, Enclosure I? Issue Resolution Summary: The original response to this RAI in LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
6	Reference 2, Enclosure I, Response to RAI question 17 - The response states near the bottom of page 30 of 84 that Case 1 results shown in Table 3.0 are for the limiting cold leg analysis and reflect the following assumption: "Although the pull out test data indicated positive residual mechanical joint strength, the residual joint strength is ignored for SLB [steam line break] accident condition[s] to conservatively account for postulated variability of the coefficient of thermal expansion." The NRC Staff notes, however, that the limiting H* value shown in Table 3.0 for Case 1 is that necessary to resist three times the normal operating pressure end cap load, not that needed to resist 1.4 times SLB. It is the Staff's understanding based on review of Tables 7-6 through 7-10 and 7-6a through 7-10a that the residual mechanical joint strength (522 lb/inch) was reflected in the H* computations for normal operating and accident conditions, including SLB. Discuss and clarify these apparent discrepancies. <u>Issue Resolution Summary</u>: A new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants obviates the need to address the sub-parts of this RAI.
7	Reference 2, Enclosure I, Table 7-6 - This table states that the required pull out force is 1680 lb. Table 7-6 indicates that for a tubesheet radius of 12 inches the needed depth of engagement is less than 10.52 (about 10.2 using linear interpolation). However, the table states that an engagement depth slightly greater than 10.52 (i.e., 10.54) is needed. Discuss and explain this apparent (minor) discrepancy. <u>Issue Resolution Summary</u>: A new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants obviates the need to address this RAI.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
8	Reference 1, Enclosure I, Table 6-4 - The listed F/L values are based on allowing 0.25 inch slippage. Reference 1 does not address the potential for limited, but progressive incremental slippage under heatup/cooldown and other operational load cycles. Nor does Reference 1 address the effects of slippage on normal operating leakage and on accident-induced leakage or the ratio of normal operating and accident induced leakage. The response to RAI question 5 in Reference 2, Enclosure I, does not provide any further insight into this issue. That response specifically addressed test results for tubes with a hard roll expansion, and the Staff believes that the slippage versus axial load characteristics for such an expansion may be entirely different than for a hydraulic expansion. Discuss and address the potential for progressive incremental slippage under heatup/cooldown and other operational load cycles. In addition, address the potential for slippage under operational and accident conditions to affect the ratio of accident-induced leakage to operational leakage. <u>Issue Resolution Summary</u>: This RAI has been superseded by Item Number 16 of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of this issue is provided in Section 2.1 of this report.
9	Discuss your plans for revising the proposed technical specification (TS) amendment to monitor the tube expansion transition locations relative to the top of the tubesheet to ensure that the tubes are not undergoing progressive, incremental slippage between inspections. <u>Issue Resolution Summary</u>: This RAI has been superseded by Item Number 16 of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of this issue is provided in Section 2.1 of this report.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
10	Reference 1, Enclosure I, Section 7.1.4.2 - This section provides a brief discussion of SLB, feed line break (FLB), and loss-of-coolant accident (LOCA) in terms of which is the most limiting accident in terms of tube pull out potential. Expand this discussion to indicate whether SLB and FLB are the most limiting accidents among the universe of design basis accidents (DBA) (or other faulted conditions in the design basis) in terms of both tube pull out, and the margin between the calculated accident-induced tube leakage for each DBA and the assumed accident-induced tube leakage in the safety analysis for that DBA. <u>Issue Resolution Summary:</u> This RAI has been superseded by Item Numbers 17, A** and D** of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of these issues is provided in Section 2.1 of this report.
11	Figure 11 of Reference 2, Enclosure I contains loss coefficient data for Model F SG tubing that was not included in Figure 6-6 of Reference 1, Enclosure 1. This data was for contact pressures ranging from about 1200 psi to about 2000 psi. Why was this data not included in Figure 6-6? Discuss if this is this because of low expansion pressures and if the data that is not included in Figure 6-6 is room temperature data. [If yes, then the NRC Staff observes that the room temperature loss coefficients for the Model F specimens are relatively invariant with contact pressure above a contact pressure threshold of around 700 psi. The 600 degree F data is also invariant with contact pressure. Thus, loss coefficient may not be a direct function of contact pressure once a threshold degree of contact pressure is established. The difference in loss coefficient data between the 600°F data and the room temperature may be due to parameter(s) other than contact pressure. This other parameter(s) may not be directly considered in the B* analysis.] <u>Issue Resolution Summary:</u> This RAI has been superseded by Item Numbers 17, A** and D** of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of these issues is provided in Section 2.1 of this report.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
12	Figure 13 of Reference 2, Enclosure I contains additional loss coefficient data taken from the crevice pressure study in the white paper. Provide a figure showing all individual data points from which Figure 13 was developed. Describe the specific applied pressure differentials from the crevice pressure study used to calculate the contact pressure for each data point. Issue Resolution Summary: This RAI has been superseded by Item Numbers 17, A** and D** of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of these issues is provided in Section 2.1 of this report.
13	Although the means of the regression fits of the loss coefficient data for the Model F and Model D SGs are shown in Figure 13 of Reference 2, Enclosure I, to be within a factor of three of each other, the slope and intercept properties remain highly divergent, seeming to cast further doubt that loss coefficient varies with contact pressure (above some threshold value of contact pressure). Discuss this and describe any statistical tests that have been performed to establish the significance of correlation between loss coefficient and contact pressure. In addition, describe any statistical tests that have been performed to confirm that it is appropriate to combine the data sets to establish the slope and intercept properties of loss coefficient versus contact pressure. Issue Resolution Summary: This RAI has been superseded by Item Numbers 17, A** and D** of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of these issues is provided in Section 2.1 of this report.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
14	Reference 2, Enclosure I, page 25 of 84 - For the case of assumed zero slope of loss coefficient versus contact pressure, two constant loss coefficient values were compared. Does the first assumed value come from Figure 14? If not, provide additional information on where this assumption comes from. If yes, explain the relationship between the assumed value and Figure 14. Does the second assumed value come from Figure 12? If not, provide additional information on where this assumption comes from. If yes, explain the relationship between the assumed value and Figure 12. <u>Issue Resolution Summary</u>: The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.
15	Reference 2, Enclosure I, Figure 15 - clarify the title of Figure 15 in terms of whether it reflects consideration of residual mechanical strength in the joint during an SLB. Is Figure 15 for the hot or cold leg? Explain the following: (1) why the B* values at small tubesheet radii are less than those listed in Reference 1, Enclosure I, Table 11-1 and (2) why the contact pressures shown in Reference 1, Enclosure I, Figures 9-6 and 9-7 are different from those shown in Tables 7-6 and 7-8 of Reference 1, Enclosure I. <u>Issue Resolution Summary</u>: A new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants and a new leakage analysis obviate the need to provide a detailed response to this RAI.
16	Reference 2, Enclosure I - Provide a description of the revised finite element model used to support the revised H* calculations in Tables 6-7 through 6-10 and Tables 6-7a through 6-10a. Compare this revised model to the original model which supported the Reference 1 analysis. Explain why the revised model is more realistic than the original model. <u>Issue Resolution Summary</u>: A new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants obviates the need to provide a detailed response to this RAI.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
17	Reference 2, Enclosure 1, Attachment 1 (The Westinghouse Letter Summary of Changes to B* and H*), page 14 - address the status of the divider plate evaluation being performed under EPRI sponsorship, and the schedule for completion of the various topics being addressed in the evaluation. Describe any inspections that have been performed domestically that provide insight on whether the extent and severity of divider plate cracks is bounded by the foreign experience. Discuss the available options for inspecting the divider plates. Issue Resolution Summary: This RAI has been superseded by Item Number 20 of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of this issue is provided in Section 2.1 of this report.
18	Discuss how the ability of the divider plates at Wolf Creek to resist tubesheet deflection (without failure) under operating and accident loads is assured in the short term, pending completion of the EPRI evaluation. Include in this discussion the actions that are planned in the near term to ensure that the divider plates are capable of resisting tubesheet deflection. Issue Resolution Summary: This RAI has been superseded by Item Number 20 of the list of issues that remain outstanding when the Wolf Creek Generating Station amendment was withdrawn and as a result of industry activities after February 2008. The road map for the resolution of this issue is provided in Section 2.1 of this report.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
19	Reference 2, Enclosure 1, Attachment 1 - Provide a description of the Crevice Pressure Test. This description should address, but not necessarily be limited to the following: a. Description of test specimens, including sketches. b. Description of "pre-treatments" of test specimens (hydraulic expansion pressure, heat relief, etc.). c. Description of test setup, including sketches. d. Description of test procedure. e. What were the secondary side temperatures in Tables 1 and 2 corresponding to the listed secondary side pressures and how were the secondary side pressure and temperatures controlled and monitored? f. How long did each test run and how stable were the pressure readings at each of the pressure taps during the course of each test? g. What was the temperature of (1) the coolant in the crevice and (2) the tube and tubesheet collar as a function of elevation? h. How were the temperature distributions for item g determined? Were direct temperature measurements of the tubesheet collar performed as a function of elevation? Issue Resolution Summary: The original response to this RAI in LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
20	Reference 2, Enclosure 1, Attachment 1 - The pressure tap locations in Figure 2 are different from those shown in Figure 3. Discuss and explain this difference or provide corrected figures. Issue Resolution Summary: The original response to this RAI in LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.
21	Reference 2, Enclosure 1, Attachment 1 - Figures 2 and 3 assume crevice pressure at the top of tubesheet is at the saturation pressure for the primary system. Discuss and explain the basis for this assumption. Why wouldn't the crevice pressure trend to the secondary side pressure near the top of the tubesheet? Issue Resolution Summary: The original response to this RAI in LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.
22	Reference 2, Enclosure 1, Attachment 1 - Figure 3 refers to tests labeled SLB 9 and SLB 10 which are not listed in Table 2. Discuss and explain this, or provide a revised Table 2 and Figure 3 showing all test results. Issue Resolution Summary: The original response to this RAI in LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
23	Reference 2, Enclosure 1, Attachment 1 - Page 6 states in part that the following change should be made to the H*/B* analyses: "The driving head of the leaked fluid has been reduced." Discuss and clarify this sentence. The Staff notes that resistance to leakage occurs from two sources: resistance from the flaw and resistance from the crevice. Because the crevice pressure was assumed to be equal to the secondary pressure, the original analysis assumed the entire pressure drop (the driving head) was across the flaw. The tests described in the white paper eliminate any pressure across the flaw (by using holes rather than cracks) and force the entire pressure drop to occur along the crevice. Thus, there is no net change in the total driving head between the primary and secondary sides. In fact, the driving head from the bottom to the top of the crevice would seem to have been increased. <u>Issue Resolution Summary</u>: A new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants which applies a depth-based crevice pressure obviates the need to provide a detailed response to this RAI.
24	Reference 2, Enclosure 1, Attachment 1 - The top paragraph on page 10 states, in part, "the median value of the crevice pressure ratios provides a conservative value that is an average representation of the behavior at the top of the tubesheet. The median is typically a better statistical representation of the data than the mean because the median is not influenced by a smaller data set but by the total range in values in the sample set." The Staff has the following questions regarding these sentences: a. Discuss and clarify what data set "median value" applies to. For example, does the "median value" for the NOP data set in Table 1 mean the median value of the 15 pressure tap data points obtained during three tests, or does it mean a median value of a subset of these 15 data points? If a subset, what subset and why? Alternatively, does it mean the median value at each pressure tap location? b. Discuss why this median value is a conservative representation of the behavior at the top of the tubesheet. c. Discuss what is meant by "top of the tubesheet." For 17-inch inspection zone amendments, shouldn't this mean the upper 17-inches to ensure a conservative analysis? If not, why not? To ensure a conservative analysis for H* and B*, should not the objective be to establish crevice pressure as a function of elevation that can be directly applied into the H* and B* computations. d. Discuss why the median is not influenced by a smaller data set and how the median is influenced by the total range of values in the sample set. <u>Issue Resolution Summary</u> A new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants which applies a depth based crevice pressure obviates the need to provide a detailed response to this RAI (see Sections 6.0 and 8.0).

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
25	Reference 2, Enclosure 1, Attachment 1 - Provide a copy of Reference 3. The cited web page appears to be no longer available. Also, provide copy of Reference 4. <u>Issue Resolution Summary</u> The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies. A new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants which applies a depth-based crevice pressure obviates the need to provide a detailed response to this RAI.
26	Reference 2, Enclosure 1, Attachment 1 - What were the specific data sets used to compute the Dixon Ratio values at the top of page 11? <u>Issue Resolution Summary</u> The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies. A new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants which applies a depth-based crevice pressure obviates the need to provide a detailed response to this RAI.
27	Reference 2, Enclosure 1, Attachment 1 - In Table 5 under the heading of outliers, rows 1 and 2 refer to "total set," whereas lines 3 and 4 refer to "included." Does "included" mean the same thing as "total set." If not, how does it differ from "total set," and how does it differ from "excluded?" <u>Issue Resolution Summary</u> The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies. A new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants which applies a depth-based crevice pressure obviates the need to provide a detailed response to this RAI.

Table 2-2 (Continued) List of NRC RAI on H* and Resolution Status

RAI No	Source Document for Initial Response: LTR-CDME-07-198 (Reference 2-3)
28	Reference 2, Enclosure 1, Attachment 1 - Provide a step-by-step description (including an example) of how the values in Table 5 were obtained. <u>Issue Resolution Summary</u> The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies. A new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants which applies a depth-based crevice pressure obviates the need to provide a detailed response to this RAI.
29	Reference 2, Enclosure 1, Attachment 1 - Confirm that the "unaltered" case in Table 5 reflects the use of the improved tubesheet/divider plate model with a "divider plate factor" of 0.399. <u>Issue Resolution Summary</u> The response to this NRC RAI is provided for historical purposes only. The original response to this RAI in LTR-CDME-07-198, "Response to NRC Request for Additional Information Relating to LTR-CDME-07-72 P-Attachment and LTR-CDME-05-209-P of the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," still applies. A new structural analysis which involves a fully probabilistic, whole bundle H* depth calculation for each of the Model 44F H* plants which applies a depth-based crevice pressure obviates the need to provide a detailed response to this RAI.

10.0 SUMMARY AND CONCLUSIONS

This report provides a technical justification for re-defining the primary pressure boundary in the steam generators (SG). The original design methods that are reflected in the applicable American Society of Mechanical Engineers (ASME) Code Stress Reports for the Model F, Model D5, Model 44F and Model 51F SGs rely on the tube end welds as the pressure boundary without taking credit for the hydraulic expansion joint between the tubes and the tubesheet. The technical justification provided in this report provides a conservative analysis that shows that the hydraulic expansion joint is capable of functioning as the primary pressure boundary without relying on any aspect of the tube end weld, either as a structural restraint for retaining the tubes in the tubesheet or as a barrier against unacceptable leakage. The title of the technical justification is H* (H-star).

Since 2004, when stress corrosion cracking (SCC) in the tubesheet expansion region was first reported among SGs with hydraulically expanded thermally treated Alloy 600 (A600TT) tubing, multiple plants have inspected throughout the full thickness of the tubesheet with rotating pancake coils (RPC). Since the first report of SCC in the tubesheet expansion region, only one other plant identified SCC in the tubesheet expansion region above the tube-end welds as an isolated incident. However, although not great in number, additional indications of SCC have been reported since the Spring 2008 outages at or within 1 inch of the ends. Unless an alternate repair criterion (ARC) is approved, industry guidelines require that tubes with crack-like indications be plugged and that the inspection scope be expanded, potentially including all SGs in the plants. H* provides a conservative technical justification that a length of undegraded tube, measured from the top of the tubesheet, provides structural and leakage integrity in accordance with the industry requirements without relying on the tube end welds or the lower several inches of the tubes to provide these functions. The value of H* defines the necessary length of undegraded tubing. Application of H* permits keeping acceptable tubes in service, reduces the inspection requirements and requirements for inspection expansions without any impact on public safety.

10.1 RECOMMENDED VALUE OF H*

The recommended value of H* for application to the Model 44F SGs is 13.31 inches. The interpretation of this recommendation is that inspection with a qualified probe to detect stress corrosion cracking in the tubesheet region is required only from the top of the tubesheet (TTS) to 13.31 inches below the TTS. For practical purposes, it is assumed that the tube below the H* value does not exist, or that any degradation below 13.31 inches from the TTS is acceptable, provided that there is no degradation observed in the span from the TTS to 13.31 inches below the TTS.

10.2 H* CONCEPT AND EVOLUTION

In concept, H* is similar to other technical justifications that have been licensed for implementation, such as F* (for hard rolled tubesheet joints), W* (for explosively expanded tubesheet joints using the WEXTEX process), and C* (for explosively expanded tubesheet joints using the Combustion Engineering "Explansion" process). The technical bases for all of these alternate repair criteria (ARC) are similar in that the interaction forces between the tube and tubesheet resulting from the initial expansion process and from the thermal and pressure-induced forces under normal operating conditions (NOP) and design basis accident (DBA) conditions are relied upon to prevent tube pull out and to limit leakage from the primary to the secondary side of the SG to acceptable limits. The difference among the H*, F*, W* and C* ARC lies in the relative tightness of the tubesheet joints achieved by the original manufacturing processes.

This technical justification was preceded by other analyses which had the same objective of replacing the tube end weld as the primary pressure boundary with the hydraulic expansion joint. The prior analyses have undergone extensive reviews, and the lessons learned from these reviews are incorporated into the current analysis. A summary of prior technical review issues is provided in this report to capture the lessons learned and to provide comprehensive documentation of the evolution of H* to its current embodiment.

10.3 DESIGN REQUIREMENTS

The applicable design requirements are those specified in industry performance criteria contained in NEI 97-06, Revision 2 and its sub-tier mandatory and recommended guidelines. The specific design requirements applicable to the H* analyses are:

1. The applicable loads shall be the greater of 3 times the normal operating pressure differential or 1.4 times the accident-induced pressure differential across the tube.

2. DBA (other than a tube rupture) induced primary-to-secondary leakage shall not exceed the leakage assumed in the accident analysis applicable to the specific plants that implement H* in terms of total leakage rate and leakage rate for an individual SG.

3. It is required that the recommended value of H* for every tube in the bundle meets a statistical probability of 95% at 50% confidence. This is known as the "whole bundle" probability.

Because the industry guidelines are designed to address prevention of tube burst, they do not directly apply to the tubesheet expansion region where burst is not possible due to the constraint provided by the tubesheet. Nevertheless, the criteria to prevent burst are conservatively applied by treating "failure to meet H*" as a burst.

10.4 DESIGN CONDITIONS

Several different models of SGs are among the H* candidate population. These include the Model F, Model D5, Model 44F and Model 51F SGs. Except for the Model 51F SGs, multiple plants are represented among each sub-population. The approach utilized in this justification is to consider the design and operating conditions for each plant and to define the plant with the most limiting conditions for H* among each sub-population. Consequently, four reports for H* are provided that have large overlap of contents for methodology, but specific content for each model of SG. This report is specific to the Model 44F SG.

For all models of SGs, the design and operating conditions for normal operation and all design basis accidents that include leakage are evaluated. These include steam line break (SLB), feedwater line break (FLB), control rod ejection (CRE) and locked rotor (LR). The limiting conditions among these are the basis for the applied loads to the tubes and for the end cap loads.

The population of plants with Model 44F SGs includes 2-loop and 3-loop plants. The bounding plant is a 2-loop plant because of the specific operating conditions of that plant that include a significant power uprating. Therefore, the results in this report are significantly more conservative for the other Model 44F plants than the significant conservatism already included for the bounding plant.

10.5 MATERIAL PROPERTIES

The principal factor that enables H* is differential thermal expansion between the tube and the tubesheet. It is assumed that, as a minimum, the as-manufactured condition of the tubes in the tubesheet have zero clearance contact with the tubesheet after the hydraulic expansion process. As the temperature increases to operating conditions, the tube material, thermally treated Alloy 600 (A600TT), expands more than the tubesheet material (SA508), resulting in a significant increase in the contact forces between the tubes and the tubesheet.

Extensive testing summarized in Section 3.0 and Appendix B show that differential thermal expansion between the tube and the tubesheet will always occur, even at significant levels of uncertainty for both materials simultaneously. The tests performed under the H* program significantly increased the available database for coefficient of thermal expansion (CTE) for A600 and SA508 materials. The data also show that the use of the mean values of CTE from the ASME Code (2007 edition) to determine the limiting H* value is conservative. The standard deviations of CTE for A600 and SA508 are 2.33% and 1.44%, respectively, of the at-temperature mean value. (A standard deviation of [ ]^{a,c,e} % is conservatively used for the CTE of SA508 in the structural analysis of the tubesheet.) The data also imply that the bulk of this uncertainty is the result of measurement error, and that the true variance of these properties is actually much smaller than the values noted above. Reductions of the variance of CTE to represent the true values would result in a significant decrease in the recommended value of H*.

10.6 RESIDUAL CONTACT PRESSURE

Prior, and current, test data show that the hydraulic expansion process results in a positive value of contact pressure between the tubes and the tubesheet. However, the technical justification in this report conservatively assumes that the contact pressure due to only the hydraulic expansion process, known as residual contact pressure (RCP), is zero. (Test data provided in Section 7.0 show that positive RCP exists for hydraulic expansions. Negative values of RCP are not possible.) The assumption of zero RCP is conservative because any value of RCP will reduce the recommended value of H*.

10.7 STRUCTURAL ANALYSIS

The structural analysis included in this report is a significant evolution from the prior analyses. The current analysis is based on a three-dimensional (3D) finite element analysis (FEA) of the tubesheet complex, which includes the tubesheet, stub barrel, channelhead and divider plate. Prior analyses utilized a two-dimensional axisymmetric model to calculate the tubesheet radial and axial deflections. The current 3D FEA approach shows that H* results prior to 2008 were significantly conservative. The 3D FEA analysis results were compared against an independently created model of the same geometry and shown to provide essentially the same results when the same inputs were utilized.

The mean value of H* is calculated to be 3.87 inches (including adjustments the location of the bottom of the transition and for the applicable thermal distribution correction) for the limiting operating condition, NOP. This value applies to the limiting radius in the worst sector of the tubesheet, which is the sector of tubes in approximately a 5 degree arc from the perpendicular to the divider plate. All tubes at other locations in the tubesheet have a smaller value of H*. This is a significant conservatism because the probabilistic evaluation is based on this worst-case mean value of H*.

There is much conservatism included in the structural analysis as summarized in Section 1.0. The following are the most significant conservatisms:

1. The mean value of H* is based on the location of the limiting radius of the tubesheet in the limiting sector of the tubesheet for the limiting SG operating condition. All other locations on the tubesheet have a lower value of H*.

2. The analysis assumes that the divider plate has no connection to the tubesheet. It is shown that this assumption results in the largest values of H*.

3. The value of the coefficient of friction utilized is 0.2. A significantly greater coefficient of friction could be justified based on the available literature. A greater coefficient of friction would results in a smaller value of H*.

4. Boundary conditions are applied to yield the most conservative (largest) values of H*.

5. The conditions of the bounding plant in the population of Model 44F SGs are used for all Model 44F plants.

10.8 LEAKAGE ANALYSIS

The leakage analysis is based on application of the Darcy model for flow through a porous medium. The approach used is to determine the ratio of accident-induced leakage to the observed leakage under normal operating conditions. These ratios are termed "Leakage Factors". The applicable maximum leakage ratio for all plants is 2.03 and is based on the peak pressure differential from a postulated FLB event for a Model F SG, which is shown to be a plant cooldown event when taking credit for operator action. The maximum leakage ratio for the Model 44F population is 1.82.

The leakage factor is applied to the normal operating leakage that is associated with the tubesheet expansion region in the condition monitoring and operational assessments. No increase in the accident-induced leakage assumed in the safety analysis results with the implementation of the leakage factor approach; however, an adjustment to the administrative shutdown leakage limit may be required depending on the NOP leakage observed and the possible sources of leakage in the SG. The leakage factor for a postulated FLB bounds the leakage factor required for a postulated SLB event because the pressure differential across the tubesheet is greater during a postulated FLB versus a SLB event; both events result in a plant cooldown event. For most plants, due to the short duration of the transients, no leakage factors are required for a postulated locked rotor or control rod ejection events. The leakage factor for a postulated FLB is conservatively used for those plants with a locked rotor with a stuck open SG power operated relief valve (PORV) as part of the licensing basis.

It is important to note that the implementation of H* does not require that the maximum leakage factor be used by plants that have a lower leakage factor; however, the maximum factor provides a conservative basis for performing the condition monitoring and operational assessments.

10.9 PROBABILISTIC ANALYSIS

Four input parameters directly affect the calculation of H* if it is assumed that RCP is equal to zero. Other variable parameters are introduced if a non-zero value of RCP would be included in the analysis. The applicable variables are the CTE of the tube and tubesheet materials (A600 and SA508) and Young's Moduli of the tube and tubesheet materials. Sensitivity analyses have shown that the variation of Young's Modulus of both the tube and tubesheet materials has insignificant effect on H*. The principal variable affecting H* is the CTE of the tube material, A600.

Sensitivity analyses also indicate that there is an interaction between the CTE of the tube and tubesheet materials when expressed as a variation of H*. The effect of the interaction is that the value of H* increases more when both parameters are varied than the cumulative effect on H* when each parameter is varied separately. However, even for extreme variations of these parameters (3 to 4 standard deviations from the mean in the direction of increasing H*), a value of H* exists; thus, there is no credible event that would invalidate the H* concept. No other interactions among the four effective parameters were identified.

Standard binomial statistical analysis shows a 4.157 standard deviation (σ) variance is required to achieve 95% probability at 50% confidence for a population of 3214 tubes, i.e., the full tube complement of a Model 44F SG. The variations of H* were calculated for a number of assumptions of parameter variations up to and including 5 standard deviations and also specifically including 4.157 standard deviations. These sensitivity analyses showed that the variation of H* was not linear for the parameter input assumptions made.

Both a simplified statistical approach and a fully probabilistic approach using a Monte Carlo simulation technique were applied for the H* analysis as permitted by the industry Integrity Assessment Guidelines for meeting the probabilistic criteria of 95% probability at 50% confidence for every tube in the tube bundle. In the simplified statistical approach, the uncertainties are combined by the square root of the sum of the squares (SRSS) method. In the fully probabilistic approach, simulations are performed by randomly sampling from influence functions of H* for each of the four variables that directly impact the H* calculations, assuming that RCP is equal to zero.

The change in H* due to positive variation of a parameter may be different than the change in H* due to a negative variation of the same parameter. In the probabilistic analysis of H*, only the parameter variations that adversely affect the value of H* (i.e., increase the value of H*) were conservatively used.

The SRSS is the preferred approach to determining the 95/50 whole bundle value of H* because of its simplicity. Using the 4.157 (σ) parameter input variation of H* directly, the 95% probability at 50% confidence value of H* for the whole bundle is 13.31 inches.

Different extreme value analyses were performed based on the Monte Carlo simulation approach under various assumptions of parameter variability and sampling schemes that recognized that the number of tubesheets in the population of H* candidate plants is limited. All of the extreme value cases considered exceed the probabilistic requirement of 95/50 for the entire bundle. Therefore, although some of these extreme value cases yielded H* values slightly greater than the recommended value of H* of 13.31 inches as expected, the results from all of the extreme value cases support the recommended value of 13.31 inches.

10.10 TUBE SLIPPAGE

The technical justification for H* concludes that at a high level of confidence at the value of H* specified, 13.31 inches for the top of the tubesheet, the tubes are fully restrained against motion under very conservative design conditions and very conservative analysis assumptions. Therefore, tube slippage is not a credible event for any tube in the bundle. It is concluded, based on the analyses in this report, that no significant technical case can be made that monitoring for tube slippage is necessary.