Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon.
Nuclear

10 CFR 50.90

RS-07-132

November 29, 2007

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. 50-456 and 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

Subject: Application for Steam Generator Tube Alternate Repair Criteria Technical Specification Amendment

References:
(1) Letter from J. A. Bauer (Exelon Generation Company, LLC) to U. S. NRC, "Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity," dated November 18, 2005

(2) Letter from R. F. Kuntz (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Braidwood Station, Unit 2 – Issuance of Amendments Re: Steam Generator Inspection Criteria (TAC No. MC8969)," dated October 24, 2006

(3) Letter from R. F. Kuntz (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station Unit Nos. 1 and 2 and Braidwood Station, Unit Nos. 1 and 2 – Issuance of Amendments Re: Steam Generator Tube Surveillance Program (TAC Nos. MC8966, MC8967, MC8968, and MC8969)," dated March 30, 2007

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Appendix A Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed changes were to revise the TS requirements related to steam generator tube integrity. The change was consistent with NRC approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity."

A001
NRR

Along with theTSTF-449 Revision 4 changes, EGC included, in the Reference 1 application, a proposal to revise TS 5.5.9, "Steam Generator Program, " which excluded the portion of the tube below 17 inches from the top of the hot leg tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators. This proposed alternate repair criteria (ARC) redefined the Braidwood Station, Unit 2, and Byron Station, Unit 2, primary pressure boundary from the hot leg tube end weld to 17 inches below the top of the hot leg tubesheet. This change was supported by Westinghouse Electric Company, LLC, LTR-CDME-05-32, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2," Revision 2, dated August 2005. The nonproprietary and proprietary versions of this document were provided as Attachments 6 and 7, respectively, to the Reference 1 submittal.

Reference 2 approved the 17-inch ARC for Braidwood Station Unit 2 for Refueling Outage 12 and the subsequent operating cycle. Reference 3 approved the portion of the Reference 1 request that modified the TS requirements related to steam generator tube integrity in accordance with TSTF-449 for Braidwood Station Units 1 and 2 and Byron Station Units 1 and 2. Reference 3 also approved the Byron Station Unit 2 request for the above-described ARC for Refueling Outage 13 and the subsequent operating cycle.

In the Safety Evaluation that accompanied Reference 3, the NRC stated that:

> "*Although the revised analyses continue to support the conservatism of the requested exclusions, the NRC staff could not complete its review of the revised analyses in time to support issuance of an amendment, without a limited applicability of the exclusion, prior to the spring 2007 outage at Byron Unit No. 2. Accordingly, the NRC staff is partially approving the proposed amendment request.* "

Subsequent to the Reference 1 submittal and the Reference 2 and 3 ARC partial approvals, additional technical issues were identified in another licensee's steam generator ARC license amendment request. These issues, and their impact on other similar ARC proposals, were discussed in a July 11, 2007, public meeting (ADAMS accession number ML071650400). The responses to the questions applicable to this Braidwood Station Unit 2 - Byron Station Unit 2 application are provided in attachments to this letter (i.e., Westinghouse Electric Company, LLC, (Westinghouse) Letter LTR-CDME-07-212, "Supplemental Information Relating to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2," dated November 1, 2007).

After the Reference 3 partial approval, the NRC ceased review of the Reference 1 permanent ARC request and the Technical Assignment Control (TAC) numbers associated with the submittal were closed-out. EGC is providing this license amendment request to reestablish NRC review of the Braidwood Unit 2 and Byron Unit 2 permanent ARC proposal. The information previously provided and supplemented by that provided in Attachment 1 and Attachment 4 to this letter provide the necessary justification for the acceptability of the Braidwood Station Unit 2 and Byron Station Unit 2 uniform 17-inch inspection depth ARC proposal and this conservative approach provides the necessary safety margin to justify a permanent implementation.

Letter LTR-CDME-07-212 (Attachment 4) contains information proprietary to Westinghouse and is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, it is requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390.

The attached request is subdivided as shown below.

Attachment 1 provides an evaluation of the proposed changes.

Attachments 2-A and 2-B include the marked-up TS pages with the proposed changes indicated for Braidwood Station and Byron Station, respectively.

Attachment 3 provides an affidavit for withholding the proprietary information provided in Attachment 4 (Including Appendices A, B, and C). Also provided is the Westinghouse authorization letter, CAW-07-2352, "Application for Withholding Proprietary Information from Public Disclosure."

Attachment 4 provides Westinghouse document LTR-CDME-07-212 P, "Supplemental Information Relating to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2, " (Proprietary). This Attachment includes the following proprietary Appendices:

- Appendix A, STD-MC-06-11, Revision 1, "Pressure Profile Measurements During Tube-to-Tubesheet Leakage Tests of Hydraulically Expanded Steam Generator Tubing," dated August 30, 2007
- Appendix B, LTR-SGDA-07-4-P, Revision 3, "Letter Summary of Changes to B* and H* Analysis Due to New Crevice Pressure and Divider Plate Data," dated September 24, 2007
- Appendix C, "Hot and Cold Leg H* and B* Results," October 2007

Attachment 5 provides Westinghouse document LTR-CDME-07-212 NP, "Supplemental Information Relating to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2," (Nonproprietary). This Attachment includes the nonproprietary versions of the following Appendices:

- Appendix A, STD-MC-06-11, Revision 1, "Pressure Profile Measurements During Tube-to-Tubesheet Leakage Tests of Hydraulically Expanded Steam Generator Tubing," dated August 30, 2007
- Appendix B, LTR-SGDA-07-4-NP, Revision 3, "Letter Summary of Changes to B* and H* Analysis Due to New Crevice Pressure and Divider Plate Data," dated September 24, 2007
- Appendix C, "Hot and Cold Leg H* and B* Results," October 2007

Attachment 6 provides a Regulatory Commitment that would require EGC to remove a Byron Station Unit 2 steam generator tube from service by preventative plugging. The tube is in the 2A steam generator with a row-column designation of R34-C46. The reason for the preventative plugging of this tube is provided in Attachment 1 to this submittal.

The following responses further supplement the information previously provided to the NRC.

LTR-CDME-05-32-P, Rev. 2, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2"	The proprietary version was provided as Attachment 7 to the letter from J. A. Bauer (Exelon Generation Company, LLC) to U S. NRC, "Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity," dated November 18, 2005.
LTR-CDME-07-13, "Response to NRC Request for Additional Information Related to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2."	The proprietary version was provided as Attachment 4 to the letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification," dated February 15, 2007.
LTR-CDME-07-31, "Response to NRC Draft Request for Additional Information Related to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2"	The proprietary version was provided as Attachment 1 to the letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Supplement to Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification," dated February 23, 2007.

Although the proposed changes only affect Braidwood Station Unit 2 and Byron Station Unit 2, this submittal is being docketed for Braidwood Station Units 1 and 2 as well as Byron Station Units 1 and 2 since the TS are common to Units 1 and 2 for each station.

The proposed amendment has been reviewed by the Braidwood Station and the Byron Station Plant Operations Review Committees and approved by their respective Nuclear Safety Review Boards in accordance with the requirements of the EGC Quality Assurance Program.

In accordance with 10 CFR 50.91(b), "State consultation," EGC is providing the State of Illinois with a copy of this letter and its non-proprietary attachments to the designated State Official.

EGC requests that this proposed license amendment change be approved by April 4, 2008, to support the inspection activities for Braidwood Station Unit 2, Refueling Outage 13. If you have any questions about this letter, please contact Mr. David Chrzanowski at (630) 657-2816. This letter contains one Regulatory Commitment described in Attachment 1 and specifically identified in Attachment 6 to this submittal.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 29th day of November 2007.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Attachment 1
Evaluation of Proposed Changes

Application for Steam Generator Tube Alternate Repair Criteria Technical Specification Amendment

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. 50-456 and 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

INDEX

1.0 DESCRIPTION

The proposed amendment revises Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," to permanently exclude the portion of the tube below 17 inches from the top of the tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators. This Alternate Repair Criteria (ARC) request was previously submitted (Reference 3) and one-cycle approvals were granted for Braidwood Station Unit 2 (References 1 and 6) and Byron Station Unit 2 (References 2 and 10). These previous requests limited the ARC to the hot leg side tubesheet area. The proposed license amendment request redefines the Braidwood Station, Unit 2, and Byron Station, Unit 2, primary pressure boundary from the tube end weld to 17 inches below the top of the tubesheet on both the hot leg and cold leg sides of the steam generators. These changes are supported by information previously provided in References 3, 4, 7, 8, and 9 as well as by the information contained in Westinghouse Electric Company, LLC, (Westinghouse) LTR-CDME-07-212, "Supplemental Information Relating to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2," provided as Attachment 4 (proprietary) and Attachment 5 (non-proprietary) to this submittal.

The proposed amendment also adds three additional reporting requirements to Braidwood Station TS 5.6.9, "Steam Generator (SG) Tube Inspection Report," that were originally requested by the NRC and agreed to in Reference 4; however, during the Braidwood Station implementation of the Reference 10 amendment, it was noted that the addition of the three additional reporting requirements at that time could present a conflict with TS 5.6.9. That is, the Reference 6 license amendment provided Braidwood Station Unit 2 with requirements that specified report content and a submittal timeframe applicable to their post-refueling outage 12 and subsequent Operating Cycle 13. The implementation of the three new 5.6.9 requirements in the middle of the Braidwood Station Unit 2 operating cycle and after the required SG report due date could have presented a TS non-compliance and therefore the NRC agreed to withhold these additional requirements until the permanent approval of the ARC.

2.0 PROPOSED CHANGE

EGC proposes to revise TS 5.5.9, "Steam Generator (SG) Program," by removing the one-cycle restriction contained in the current wording in the Braidwood Station and Byron Station TS, as well as in TS 5.6.9, "Steam Generator (SG) Tube Inspection Report," for Byron Station. Also, as discussed above, EGC proposes to add three reporting requirements (i.e., TS 5.6.9 paragraphs j, k and l) to Braidwood Station TS 5.6.9. The evaluation of these additional requirements was documented in the Reference 4 submittal. Also editorial changes are proposed for Braidwood Station and Byron Station TS 5.5.9, paragraph c.2.i, adding "For Unit 2 only," and paragraph c.4, to change "may" to "shall." These changes provide consistency with the wording throughout TS 5.5.9.

The marked-up TS pages provided in Attachments 2-A and 2-B indicate the specific wording changes for Braidwood Station and Byron Station, respectively.

3.0 BACKGROUND

Braidwood Station, Unit 2, and Byron Station, Unit 2, each contain four Westinghouse Model D5 recirculating, pre-heater type SGs. Each SG contains 4,570 thermally treated Alloy-600 U-tubes that have an outer diameter of 0.750 inch with a 0.043 inch nominal wall thickness. The support plates are 1.12 inch thick stainless steel and have quatrefoil broached holes. The tubing within the tubesheet is hydraulically expanded throughout the full thickness of the tubesheet. The tubesheet is approximately 21 inches thick. The low row U-bend region, up through row nine, received additional thermal stress relief following tube bending. The units operate on approximately 18-month fuel cycles.

The SG inspection scope is governed by: Braidwood Station TS 5.5.9; Byron Station TS 5.5.9; the Electric Power Research Institute (EPRI) Pressurized Water Reactor (PWR) SG Examination Guidelines; regulatory documents and commitments; Exelon ER-AP-420 procedure series (Steam Generator Management Program Activities); and the results of Braidwood Station, Unit 2, and Byron Station, Unit 2, degradation assessments.

The inspection techniques and equipment are capable of reliably detecting the known and potential specific degradation mechanisms applicable to the Braidwood Station, Unit 2, and Byron Station, Unit 2, SGs. The inspection techniques, essential variables and equipment are qualified to Appendix H, "Performance Demonstration for Eddy Current Examination," of the EPRI PWR SG Examination Guidelines.

In order to preclude unnecessarily plugging tubes in the Braidwood Station, Unit 2, and Byron Station, Unit 2, SGs, analyses were performed to identify the portion of the tube within the tubesheet necessary to maintain structural and leakage integrity for both normal operating and accident conditions. Tube inspections will be limited to identifying and repairing flaws in this portion of the tubes. The technical justification for the limited inspection depth was originally provided in Reference 3 and supplemented by information provided in References 4, 7, 8, and 9 including additional information provided in Attachment 4 to this submittal. The limited tubesheet inspection criteria were developed for the tubesheet region of Model D5 SGs considering the most stringent loads associated with plant operation, including transients and postulated accident conditions.

The limited tubesheet inspection criteria were selected to prevent tube burst and axial separation due to axial pullout forces acting on the tube and to ensure that the steam line break (SLB) leakage limits are not exceeded. Reference 3 provided technical justification for allowing tubes with indications that are below 17 inches from the top of the hot leg side of the tubesheet (i.e., within approximately four inches of the tube end) to remain in-service. This original information was supplemented by that provided in References 4, 7, 8, and 9 including additional information provided in Attachment 4 to this submittal. This supplemental information provides the technical justification for allowing tubes with indications that are below 17 inches from the top of the hot and cold leg sides of the tubesheet to remain in-service.

Constraint provided by the tubesheet precludes tube burst for cracks within the tubesheet. The criteria for tube burst described in Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," Revision 2 dated May 2005, and NRC Draft Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976, are satisfied due to the constraint provided by the tubesheet. Through application of the limited tubesheet inspection scope proposed in this submittal, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur during a postulated SLB event. Sensitivity studies to support the leakage criteria were originally provided in Reference 4 and were updated by information provided in References 7 and 8 and Attachment 4 to this submittal.

Implementation of this proposed ARC involves limited inspection of the tubes within the tubesheet to depths of 17 inches from the top of the tubesheet using specialized rotating eddy current probes. The limited tubesheet inspection length of tubing must be demonstrated to be non-degraded below the top of the tubesheet interface. If cracks are found within the top of tubesheet to 17 inches below the top of tubesheet, the tube must be repaired or removed from service.

To further support implementation of this proposed ARC, a review of fabrication records identified that during SG fabrication, the hole for tube location R34-C46 on the cold leg side of the Byron Station 2A SG tubesheet was drilled slightly larger than specified and although this was determined to be acceptable with no impact on the integrity of the SG tube, the oversized hole diameter is outside the bounding dimensions of the analyses discussed in Reference 4 for application of the 17-inch ARC on the cold leg side of the tubesheet. Therefore the tube at this location will be removed from service by preventative plugging. EGC is providing a Regulatory Commitment (discussed in Attachment 6 of this submittal) to plug this tube at the next Byron Station Unit 2 outage if the proposed ARC is approved.

The NRC has previously granted amendments for a hot leg side ARC, on a one-cycle basis, for Braidwood Station Unit 2 (References 1 and 6), and Byron Station Unit 2 (References 2 and 10). In addition, the NRC approved a 17-inch ARC for both hot and cold leg side application in a letter from J. Stang (U. S. NRC) to J. R. Morris (Duke Power Company, LLC), "Catawba Nuclear Station, Unit 2, Issuance of Amendment Regarding Technical Specification 5.5.9, 'Steam Generator Tube Surveillance Program,'" dated October 31, 2007.

4.0 TECHNICAL ANALYSIS

The following technical justifications have been developed to identify the safety significant portion of the tube within the tubesheet. The technical justifications provided in these submittals were reviewed and approved in accordance with the requirements of Exelon Generation Company, LLC (EGC) procedures.

Westinghouse LTR-CDME-05-32, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2," Revision 2, dated August 2005 (provided in Reference 3)

SG-SGDA-06-20, "Exelon: Byron Unit 2 and Braidwood Unit 2, Response to Request for Additional Information on FP&L Seabrook License Amendment Request 05-08 – Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet" (provided in Reference 4).

LTR-CDME-07-13, "Response to NRC Request for Additional Information Related to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2 (TAC Nos. MC8966, MC8967, MC8968 and MC8969)" (provided in Reference 7).

LTR-CDME-07-31, Rev. 1, "Response to NRC Draft Request for Additional Information Related to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2 (TAC NOS. MC8966, MC8967, MC8968, and MC8969)" (provided in Reference 8).

LTR-CDME-07-212, "Supplemental Information Relating to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2 (Attachment 4).

The safety significant portion of the tube is the length of tube that is engaged in the tubesheet from the secondary face that is required to maintain structural and leakage integrity over the full range of steam generator operating conditions, including the most limiting accident conditions. The approach utilized in demonstrating SLB as the limiting transient for tube pullout and accident-induced leakage (i.e., response to Question 10 in Attachment 4) is based on the SG design specifications; however, the conclusions are supported by the Braidwood Station – Byron Station Design Basis transient analyses for SLB, feedwater line break, locked rotor and control rod ejection. The evaluations determined that flaws in tubing below the safety significant portion of the tube do not require repair or plugging and serves as the basis for the tubesheet inspection program.

Structural Integrity

The bases for determining the safety significant portion of the tube within the tubesheet is based upon analyses and testing programs that quantified the tube-to-tubesheet radial contact pressure for bounding plant conditions was originally provided in Reference 3. Based on questions from the NRC regarding the phase-state of the assumed leakage, a re-evaluation of contact pressure and the tube engagement distance for the required margin against pullout was performed. The results of this re-evaluation, provided in References 7 and 8, concluded the necessary engagement distance varied from 6.2 to 11.5 inches along the radius of the hot leg tubesheet. Although the revised engagement distances are well within the proposed 17-inch inspection zone, additional evaluations were performed to determine the effect of variations in the tubesheet thermal expansion coefficient (TEC) as well as the impact of a degraded divider plate on the engagement distance. As provided in Reference 9, with worse case assumptions on TEC values and a completely degraded divider plate, the required engagement distance varied from 11.5 to 12.6 inches on the hot leg side – still well within the proposed 17-inch inspection depth.

At the request of the NRC, Westinghouse performed a sensitivity study (provided in Attachment 4) evaluating the sources of potential variability on engagement distance. Westinghouse used a "stacked" model of the uncertainties to determine a limiting combination of engagement distances. The model is considered to be a "stacked" model because all of the uncertainties are superimposed simultaneously in a conservative direction (i.e., a direction that results in the greatest required engagement distance). This assessment also assumes a degraded stub runner to divider plate weld. The results of this very conservative assessment, provided in response to Question 2 in Attachment 4 to this submittal, results in a maximum required engagement depth of 12.34 inches on the hot leg side and 13.76 inches on the cold leg side – still bounded by the proposed 17 inch inspection depth.

Divider Plate

Conservative assumptions regarding the ability of the divider plate to resist tubesheet deflection during accident conditions and the impact of this on contact pressure along the tube engagement length were discussed in References 7 and 8. In response to generic issues regarding divider plate integrity, additional information is provided in the responses to Questions 16 through 18 in Attachment 4 to this submittal. The analyses provided in the above listed documents conclude that the proposed 17-inch inspection depth provides significant tube structural and leakage margin even if the divider plate is not credited.

Leakage Integrity

Since the proposed 17-inch tube inspection depth traverses below the mid-plane of the tubesheet, the tube-to-tubesheet contact pressure significantly aids in restricting primary-to-secondary leakage as differential pressure increases. Based on the analyses provided in References 7 and 8 and Attachment 4, given that there is no significant primary-to-secondary leakage during normal operation, there will be no significant leakage during postulated accident conditions from indications located below the mid-plane of the tubesheet (i.e., greater than approximately 10.5 inches below the top of the tubesheet).

The rationale for this conclusion, supported by the analysis, is the interaction of temperature and tubesheet bending effects that increase the contact pressure between the tube and the tubesheet, thereby increasing the resistance to primary-to-secondary leakage during normal operating or accident conditions.

Primary-to-secondary leakage from tube flaws in the tubesheet area during the limiting accident (i.e., SLB) is limited by flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications. The primary-to-secondary leak rate during postulated SLB accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet (i.e., including indications in the tube end welds). This conclusion is based on the observation that while the driving pressure causing leakage increases by approximately a factor of two, the flow resistance associated with an increase in the tube-to-tubesheet contact pressure, during a SLB, increases by up to approximately a factor of three. While such a leakage decrease is expected, the postulated accident leak rate could be conservatively bounded by twice the normal operating leak rate if the increase in contact pressure is ignored. Since normal operating leakage is limited to less than 0.104 gpm (i.e., 150 gpd) per TS 3.4.13, "RCS Operational Leakage," the associated accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by approximately 0.2 gpm. This value is well within the assumed faulted SG accident leakage rate of 0.5 gpm discussed in Byron/Braidwood Updated Final Safety Analysis Report, Table 15.1-3, "Parameters Used in Steam Line Break Analyses." Therefore, it is reasonable to omit any consideration of inspection of the tube, tube end weld, bulges/overexpansions or other anomalies below 17 inches from the top of the tubesheet.

The proposed inspection sampling length of 17 inches from the top of the tubesheet provides a high level of assurance that the leakage criteria are maintained during normal operating and accident conditions. The response to Question 2 of Attachment 4, provides a maximum required tube engagement depth of 8.17 inches on the hot leg side and 7.49 inches on the cold leg side – still bounded by the proposed 17 inch inspection depth.

Flaws found in the portion of the tube below 17 inches from the top of the tubesheet do not require repair or plugging as described in LTR-CDME-05-32, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2," Revision 2, dated August 2005, (i.e., Attachment 7 to the Reference 3 submittal) as supplemented by information provided in References 4, 7, 8, 9, and Attachment 4 to this submittal.

<u>Summary</u>

- The information provided in the original Reference 3 submittal and supplemented by that in References 4, 7, 8, and 9, and Attachment 4 to this submittal confirms that the structural integrity requirements of NEI 97-06 and Draft Regulatory Guide 1.121 are met with the proposed 17-inch inspection depth. The region of the tube below 17 inches from the top of the tubesheet, including the tube-to-tubesheet weld, is not needed for structural integrity during normal operation or accident conditions. Inspections will be performed to a depth of 17 inches from the top of the tubesheet as determined by the degradation assessment.
- Current Braidwood Station and Byron Station TS define the tube as extending from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, but specifically excludes the tube-to-tubesheet weld from the definition of the tube.
- The welds were originally designed and analyzed as the primary pressure boundary in accordance with the requirements of Section III of the 1971 edition of the American Society of Mechanical Engineers (ASME) Code, Summer 1972 Addenda and selected paragraphs of the Winter 1974 Addenda for the Braidwood Station, Unit 2 and Byron Station, Unit 2, SGs. This proposed license amendment request redefines the primary pressure boundary from the tube end to 17 inches below the top of the tubesheet.
- Section XI of the ASME Code deals with the in-service inspection of nuclear power plant components. The ASME Code (i.e., Editions 1971 through 2004) specifically recognizes that the SG tubes are under the purview of the NRC through the implementation of the requirements of the TS as part of the plant operating license.

5.0 REGULATORY ANALYSIS

A description of this proposed change, as originally provided in Reference 3, and its relationship to applicable regulatory requirements and guidance was provided in the NRC Biweekly Notice of Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations published on May 23, 2006 (71 FR 29676). An evaluation of the current request is as follows.

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed amendment excludes the portion of the tube below 17 inches from the top of the tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators from TS 5.5.9.d, "Provisions for SG tube inspections." This proposed license amendment request redefines the Braidwood Station, Unit 2, and Byron Station, Unit 2, primary pressure boundary from the tube end to 17 inches below the top of the tubesheet. In addition, the proposed amendment deletes the cycle specific references in the Byron Station TS 5.6.9, "Steam Generator (SG) Tube Inspection Report," and adds three reporting requirements to Braidwood Station TS 5.6.9. These three TS 5.6.9 report items were previously reviewed and approved but not implemented in the current Braidwood Station operating cycle.

EGC has evaluated whether or not a significant hazards consideration is involved with the proposed TS change by focusing on the three criteria set forth in 10 CFR 50.92 as discussed below:

Criteria

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

   Response: No.

   The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed changes that alter the SG inspection criteria do not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed changes will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   Of the applicable accidents previously evaluated, the limiting transients with consideration to the proposed changes to the SG tube inspection criteria, are the SG tube rupture (SGTR) event and the steam line break (SLB) accident.

   During the SGTR event, the required structural integrity margins of the SG tubes will be maintained by the presence of the SG tubesheet. SG tubes are hydraulically expanded in the tubesheet area. Tube rupture in tubes with cracks in the tubesheet is precluded by the constraint provided by the tubesheet. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. Based on this design, the structural margins against burst, discussed in Draft Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR SG Tubes," are maintained for both normal and postulated accident conditions.

The proposed changes do not affect other systems, structures, components or operational features. Therefore, the proposed changes result in no significant increase in the probability of the occurrence of a SGTR accident.

At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below the proposed limited inspection depth is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the hydraulic expansion by precluding tube deformation beyond its initial hydraulically expanded outside diameter.

The probability of a SLB is unaffected by the potential failure of a SG tube as this failure is not an initiator for a SLB.

The consequences of a SLB are also not significantly affected by the proposed changes. During a SLB accident, the reduction in pressure above the tubesheet on the shell side of the SG creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet thereby restricting primary-to-secondary leakage below the midplane.

Primary-to-secondary leakage from tube flaws in the tubesheet area during the limiting accident (i.e., SLB) is limited by flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications. The primary-to-secondary leak rate during postulated SLB accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet (i.e., including indications in the tube end welds). This conclusion is based on the observation that while the driving pressure causing leakage increases by approximately a factor of two, the flow resistance associated with an increase in the tube-to-tubesheet contact pressure, during a SLB, increases by up to approximately a factor of three. While such a leakage decrease is logically expected, the postulated accident leak rate could be conservatively bounded by twice the normal operating leak rate if the increase in contact pressure is ignored. Since normal operating leakage is limited to less than 0.104 gpm (150 gpd) per TS 3.4.13, "RCS Operational Leakage," the associated accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by approximately 0.2 gpm. This value is well within the assumed accident leakage rate of 0.5 gpm discussed in Updated Final Safety Analysis Table 15.1-3, "Parameters Used in Steam Line Break Analyses." Hence it is reasonable to omit any consideration of inspection of the tube, tube end weld, bulges/overexpansions or other anomalies below 17 inches from the top of the tubesheet. Therefore, the consequences of a SLB accident remain unaffected.

Based on the above discussion, the proposed changes do not involve an increase in the consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes do not involve the use or installation of new equipment and the currently installed equipment will not be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed changes will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases.

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

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

   Response: No.

   The proposed changes maintain the required structural margins of the SG tubes for both normal and accident conditions. Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," Revision 2 and Draft Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, "Reactor coolant pressure boundary," GDC 15, "Reactor coolant system design," GDC 31, "Fracture prevention of reactor coolant pressure boundary," and GDC 32, "Inspection of reactor coolant pressure boundary," by reducing the probability and consequences of a SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall flaws the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

   For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, Westinghouse document LTR-CDME-07-212, "Supplemental Information Relating to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2," defines a length of degradation-free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited tubesheet inspection depth criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for margins between calculated and actual leakage values in the proposed limited tubesheet inspection depth criteria.

Therefore, the proposed changes do not involve a significant hazards consideration under the criteria set forth in 10 CFR 50.92(c).

5.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include technical specifications (TS) as part of the license. The Commission's regulatory requirements related to the content of the TS are contained in Title 10, Code of Federal Regulations (10 CFR), Section 50.36, "Technical specifications. The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation (LCO), (3) surveillance requirements, (4) design features, and (5) administrative controls. The SG tube inspection requirements are included in the TS in accordance with 10 CFR 50.36(c)(5), "Limiting Conditions for Operation."

As stated in 10 CFR 50.59, "Changes, tests, and experiments," paragraph (c)(1)(i), a licensee is required to submit a license amendment pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," if a change to the TS is required. Furthermore, the requirements of 10 CFR 50.59 necessitate that the NRC approve the TS changes before the TS changes are implemented. EGC's submittal revising the requirements of TS 5.5.9, "Steam Generator Program" to exclude the portion of the tube below 17 inches from the top of the tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators from TS 5.5.9.d, "Provisions for SG tube inspections," as well as the changes to the Braidwood and Byron Station TS 5.6.9, "Steam Generator (SG) Tube Inspection Report," meet the requirements of 10 CFR 50.59(c)(1)(i) and 10 CFR 50.90.

Draft RG 1.121 margins against burst are maintained for both normal and postulated accident conditions due to the constraint provided by the tubesheet.

NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," dated April 7, 2005, provides additional regulatory insight regarding SG tube degradation.

6.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment revises TS 5.5.9, "Steam Generator Program", and TS 5.6.9, "Steam Generator (SG) Tube Inspection Report," to exclude the portion of the tube below 17 inches from the top of the tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators from TS 5.5.9.d, "Provisions for SG tube inspections," and adds three reporting requirements to Braidwood Station TS 5.6.9. EGC has determined that these proposals change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for protection against radiation," or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in paragraph (c)(9) of 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review." Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 PRECEDENT

The approval to exclude the portion of the tube below 17 inches from the top of the hot leg tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators from inspection has been previously granted, on a one-cycle basis, for Braidwood Station, Unit 2, (References 1 and 6) and Byron Station, Unit 2, (References 2 and 10). In addition, the NRC approved a 17-inch ARC for both hot and cold leg side application in a letter from J. Stang (U. S. NRC) to J. R. Morris (Duke Power Company, LLC), "Catawba Nuclear Station, Unit 2, Issuance of Amendment Regarding Technical Specification 5.5.9, 'Steam Generator Tube Surveillance Program,'" dated October 31, 2007.

8.0 REFERENCES

(1) Letter from G. F. Dick (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Braidwood Station, Units 1 and 2 – Issuance of Exigent Amendments Re: Revision of Scope of Steam Generator Inspections for Unit 2 Refueling Outage 11 – (TAC Nos. MC6686 and MC6687)," dated April 25, 2005

(2) Letter from J. B. Hopkins (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station Unit 2 – Issuance of Amendment (TAC No. MC7219)," dated September 19, 2005

(3) Letter from J. A. Bauer (Exelon Generation Company, LLC) to U. S. NRC, "Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity," dated November 18, 2005

(4) Letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification," dated August 18, 2006

(5) Letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Response to Request for Additional Information to Support NRC Approval of a License Amendment Related to Technical Specification 5.5.9, "Steam Generator (SG) Tube Surveillance Program," for Use During the Unit 2 Refueling Outage 12 and Subequent Operating Cycle," dated September 28, 2006

(6) Letter from R. F. Kuntz (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Braidwood Station, Unit 2 – Issuance of Amendments Re: Steam Generator Inspection Criteria (TAC No. MC8969)," dated October 24, 2006

(7) Letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification," dated February 15, 2007

(8) Letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Supplement to Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification," dated February 23, 2007

(9) Letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting Application for Steam Generator Tube Integrity Technical Specification Improvement," dated March 7, 2007

(10) Letter from R. F. Kuntz (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station Unit Nos. 1 and 2 and Braidwood Station, Unit Nos. 1 and 2 – Issuance of Amendments Re: Steam Generator Tube Surveillance Program (TAC Nos. MC8966, MC8967, MC8968, and MC8969)," dated March 30, 2007

Attachment 2-A

Application for Steam Generator Tube Alternate Repair Criteria Technical Specification Amendment

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. 50-456 and 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

Braidwood Station

Marked-up Technical Specifications Pages

5.5-8
5.5-9
5.6-6

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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

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

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws in a non-sleeved region with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate repair criteria discussed in TS 5.5.9.c.4. For Unit 2 only, ~~during Refueling Outage 12 and the subsequent operating cycle,~~ flaws identified in the portion of the tube from the top of the ~~hot leg~~ tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection.

2. Sleeves found by inservice inspection to contain flaws with a depth equal to or exceeding the following percentages of the nominal sleeve wall thickness shall be plugged:

   i. For Unit 2 only, TIG welded sleeves (per TS 5.5.9.f.2.i): 32%

3. Tubes with a flaw in a sleeve to tube joint that occurs in the sleeve or in the original tube wall of the joint shall be plugged.

4. The following tube repair criteria ~~may~~ shall be applied as an alternate to the 40% depth-based criteria of Technical Specification 5.5.9.c.1:

   i. For Unit 2 only, ~~during Refueling Outage 12 and the subsequent operating cycle,~~ flaws found in the portion of the tube below 17 inches from the top of the ~~hot leg~~ tubesheet do not require plugging or repair.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Unit 2 only, ~~during Refueling Outage 12 and the subsequent operating cycle,~~ the portion of the tube below 17 inches from the top of the ~~hot leg~~ tubesheet is excluded. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

2. Inspect 100% of the Unit 1 tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

   Inspect 100% of the Unit 2 tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI, 1992 Edition with the 1992 Addenda.

5.6.9 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged or repaired to date,

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

h. The effective plugging percentage for all plugging and tube repairs in each SG, and

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

Insert 1

Insert 1

j. For Unit 2, the number of indications and location, size, orientation, and whether initiated on primary or secondary side for each indication detected in the upper 17-inches of the tubesheet thickness.

k. For Unit 2, following completion of an inspection, the operational primary to secondary leakage rate observed (greater than three gallons per day) in each steam generator (if it is not practical to assign the leakage to an individual steam generator, the entire primary to secondary leakage should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report.

l. For Unit 2, following completion of an inspection, the calculated accident leakage rate from the lowermost 4-inches of tubing for the most limiting accident in the most limiting steam generator. In addition, if the calculated accident leakage rate from the most limiting accident is less than 2 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined.

Attachment 2-B

Application for Steam Generator Tube Alternate Repair Criteria Technical Specification Amendment

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. 50-456 and 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

Byron Station

Marked-up Technical Specifications Page

5.5-8
5.5-9
5.6-6
5.6-7

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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

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

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws in a non-sleeved region with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate repair criteria discussed in TS 5.5.9.c.4. For Unit 2 only, ~~during Refueling Outage 13 and the subsequent operating cycle,~~ flaws identified in the portion of the tube from the top of the ~~hot leg~~ tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection.

2. Sleeves found by inservice inspection to contain flaws with a depth equal to or exceeding the following percentages of the nominal sleeve wall thickness shall be plugged:

   i. For Unit 2 only, TIG welded sleeves (per TS 5.5.9.f.2.i): 32%

3. Tubes with a flaw in a sleeve to tube joint that occurs in the sleeve or in the original tube wall of the joint shall be plugged.

4. The following tube repair criteria ~~may~~ shall be applied as an alternate to the 40% depth-based criteria of Technical Specification 5.5.9.c.1:

   i. For Unit 2 only, ~~during Refueling Outage 13 and the subsequent operating cycle,~~ flaws found in the portion of the tube below 17 inches from the top of the ~~hot leg~~ tubesheet do not require plugging or repair.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Unit 2 only, ~~during Refueling Outage 13 and the subsequent operating cycle,~~ the portion of the tube below 17 inches from the top of the ~~hot leg~~ tubesheet is excluded. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

2. Inspect 100% of the Unit 1 tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

Inspect 100% of the Unit 2 tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI, 1992 Edition with the 1992 Addenda.

5.6.9 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged or repaired to date,

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

h. The effective plugging percentage for all plugging and tube repairs in each SG, and

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

j. For Unit 2, ~~following completion of an inspection performed in Refueling Outage 13 (and any inspections performed in the subsequent operating cycle),~~ the number of indications and location, size, orientation, and whether initiated on primary or secondary side for each indication detected in the upper 17-inches of the tubesheet thickness.

5.6 Reporting Requirements

5.6.9 Steam Generator (SG) Tube Inspection Report (continued)

k. For Unit 2, ~~following completion of an inspection performed in Refueling Outage 13 (and any inspections performed in the subsequent operating cycle),~~ the operational primary to secondary leakage rate observed (greater than three gallons per day) in each steam generator (if it is not practical to assign the leakage to an individual steam generator, the entire primary to secondary leakage should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report.

l. For Unit 2, ~~following completion of an inspection performed in Refueling Outage 13 (and any inspections performed in the subsequent operating cycle),~~ the calculated accident leakage rate from the lowermost 4-inches of tubing for the most limiting accident in the most limiting steam generator. In addition, if the calculated accident leakage rate from the most limiting accident is less than 2 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined.

Attachment 3

Application for Steam Generator Tube Alternate Repair Criteria Technical Specification Amendment

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. 50-456 and 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

Westinghouse Authorization Letter

CAW-07-2352

Application for Withholding Proprietary Information from Public Disclosure

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-4643
Direct fax: (412) 374-4011
e-mail: greshaja@westinghouse.com

Our ref: CAW-07-2352

November 1, 2007

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-CDME-07-212 P-Attachment, "Supplemental Information Relating to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2 (Proprietary)"

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-07-2352 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Exelon Generation Company, LLC.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-07-2352, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: Jon Thompson (NRC O-7E1A)

CAW-07-2352

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 1st day of November, 2007

Sharon L. Markle

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

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

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

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

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

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

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

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

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-CDME-07-212 P-Attachment, "Supplemental Information Relating to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2," dated November 1, 2007 (Proprietary), for submittal to the Commission, being transmitted by Exelon Generation Company, LLC Application for Withholding Proprietary Information from Public Disclosure to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Byron Unit 2 and Braidwood Unit 2 is expected to be applicable to other licensee submittals in support of implementing a limited inspection of the tube hydraulic expansion joints within the tubesheet region of the steam generators.

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

(a) Provide documentation of the analyses, methods, and testing for the implementation of an alternate repair criteria for the portion if the tubes within the tubesheet of the Byron Unit 2 and Braidwood Unit 2 steam generators.

(b) Assist the customer in obtaining NRC approval of the Technical Specification changes associated with the alternate repair criteria.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for the purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

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

COPYRIGHT NOTICE

Exelon Generation Company, LLC

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC:

Enclosed are:

1. 1 copy of LTR-CDME-07-212 P-Attachment, "Supplemental Information Relating to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2," dated November 1, 2007 (Proprietary)

2. 1 copy of LTR-CDME-07-212 NP-Attachment, "Supplemental Information Relating to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2," dated November 1, 2007 (Non-Proprietary)

Also enclosed is Westinghouse authorization letter CAW-07-2352 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's' regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-07-2352 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Attachment 5

Application for Steam Generator Tube Alternate Repair Criteria Technical Specification Amendment

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
<u>NRC Docket Nos. 50-456 and 50-457</u>

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
<u>NRC Docket Nos. 50-454 and 50-455</u>

Westinghouse LTR-CDME-07-212

Non-Proprietary Version

LTR-CDME-07-212 NP-Attachment

Exelon Generation Company, LLC

Supplemental Information Relating to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2

November 1, 2007

Westinghouse Electric Company LLC
P.O. Box 158
Madison, PA 15663

Byron 2 and Braidwood Unit 2 Responses to the NRC Request for Additional Information for the Wolf Creek Steam Generators

Steam Generator Tube Alternate Repair Criteria
for the Portion of the Tube Within the Tubesheet

(LTR-CDME-05-32-P, Rev. 2, LTR-CDME-07-13 and LTR-CDME-07-31 P-Attachment)

The purpose of this letter report is to provide responses to the NRC staff's second RAI for the permanent application of the H*/B* criteria at the Wolf Creek Generating Station (WGCS) as they apply to the Byron and Braidwood Unit 2 Model D5 steam generators. The RAI responses supplement the information previously provided to the NRC staff in LTR-CDME-05-32-P, Rev. 2, "Limited Insepction of the Steam Generator Portion Wihin the Tubesheet at Byron 2 and Braidwood 2," LTR-CDME-07-13, "Response to NRC Request for Additional Information Related to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2 (TAC Nos. MC8966, MC8967, MC8968, and MC8969)," and LTR-CDME-07-31, "Response to NRC draft request for Additional Information Related to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2 (TAC Nos. MC8966, MC8967, MC 8968 and MC 8969."

Provided below are responses to the second NRC staff RAI. Reference 1 and Reference 2 identified in the NRC questions are:

1. Wolf Creek Nuclear Operating Corporation (WCNOC) letter dated February 21, 2006 (ET 06-0004)

2. WCNOC letter dated May 3, 2007 (WO 07-0012)

REFERENCES

1. Wolf Creek Nuclear Operating Corporation (WCNOC) letter ET 06-0004 dated February 21, 2006.

2. LTR-CDME-07-72 P-Attachment, "Response to NRC Request for Additional Information on the Wolf Creek Generating Station (WCGS) Permanent B* License Amendment Request," (Enclosure I to WCNOC letter WO 07-0012).

3. LTR-CDME-05-209-P, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Wolf Creek Generating Station," (Enclosure I to WCNOC letter ET 06-0004).

4. EPRI Report 1012987, "Steam Generator Integrity Assessment Guidelines," Rev. 2.

5. USNRC, Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," December 1998.

6. NEI 97-06, "Steam Generator Program Guidelines," Rev. 2.

7. NCE-88-271, "Assessment of Tube-to-Tubesheet Joint Manufacturing Processes for Sizewell B Steam Generators Using Alloy 690 Tubing," November 1988.

8. WCAP-12522, "Inconel Alloy 600 Tubing-Material Burst and Strength Properties," January 1990.

9. WNEP-9725, "The Westinghouse Tube to Tubesheet Joint Hydraulic Expansion Process," July 1997.

10. Westinghouse Drawing 6525D47.

11. PR27988-52336, Anter Laboratories, Inc.

12. EPRI Report 10104982, "Analysis of Primary Water Stress Corrosion Cracking and Mechanical Fatigue in the Alloy 600 Stub Runner to Divider Plate Weld Material."

13. LTR-SGDA-07-242, "Byron Unit 2/Braidwood Unit 2 Model D-5," October 2007.

14. CN-SGDA-02-33, Rev. 0, "Steam Generator Normal/Upset Delta-P Evaluation for Byron/Braidwood Unit 2, Feedwater Temperature Reduction," May 2006.

15. "Statistics, Probability and Reliability for Civil and Environmental Engineers," McGraw-Hill, © 1997.

16. LTR-CDME-07-201; Technical Support Letter and Spreadsheet Summary for LTR-SGDA-07-4, Rev. 1; September 14, 2007.

17. OP-SGDA-03-1, "Model D5 Tube-To-Tubesheet Joint Determination."

18. LTR-CDME-05-32-P, Rev. 2, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2," August 2005.

19. LTR-CDME-07-31, "Response to NRC Draft Request for Additional Information Related to the Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity for Byron Units 1 and 2 and Braidwood Units 1 and 2 (TAC Nos. MC8966, MC8967, MC8968, and MC8969)," February 2007.

20. LTR-CDME-07-223, "Technical Support for the Byron 2/Braidwood Unit 2 H*/B* Report," October 2007.

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.)

<u>Response to RAI Number 1:</u>

There have been two approaches to calculating the F/L value using the empirical test data. The first involves a basic calculation using first principles and the second involves a model based on the theory of elasticity. Although the equation for calculating F/L is not the same as noted in the question, the first principles method yields the same value.

The following response details the two methods that have been used for calculating the ratio F/L. Following the explanation of the methods, F/L is recalculated for the Byron and Braidwood Unit 2 steam generators using the first principles method and room temperature data only. The values provided in Reference 3 are believed to be correct based on application of the theory of elasticity; however, for consistency with prior submittals of H*/B*, the recalculation based on first principles was performed, and is also now the basis for the technical justification of H*/B* for the Wolf Creek steam generators.

The pullout force for the Model D5 and Model 44F SGs used the following formula to calculate the contact pressure (P_c), based on the first principles approach:

a,c,e

(1)

Next, an average value minus one standard deviation value (-1σ) is determined for contact pressure based on the measured pullout force data. The F/L values, in units of lbf/in are calculated using the formula:

a,c,e

[ ] (2)

Whereas, the theory of elasticity model was used to calculate the residual contact pressure and the pullout force for the Model F steam generators (at Wolf Creek and other Model F SGs), reflecting the advance in the state of this analysis compared to prior application such as Seabrook, Byron and Braidwood.

Using the theory of elasticity model:

- The listed value of 522.3 lb_f/in. is correct.

- The values are based on 0.25 inch slip pullout test data and contact pressures that were calculated using the theory of elasticity:

a,c,e

[ ]

Where:

a,c,e

[ ]

- The value for *Po* is calculated for each test using a coefficient of friction of 0.3.

a,c,e

[ ]

- The average *F/L* value is then calculated using the equation:

a,c,e

[ ] (3)

Where: a,c,e

[ ]

The theory of elasticity model was chosen because there is a decrease in the contact pressure between the tube and the tubesheet due to Poisson's contraction of the tube in response to the application of the axial end cap load. Conversely, an increase in the contact pressure would result from the application of an axial load tending to push on the tube; a push load arises from pressure on the crack flanks in operation. (Assuming a 360° throughwall crack.) The decrease in contact pressure results in a radial inward springback of the tubesheet and a radial outward springback of the tube outside radius. Resistance to pullout of the tube is manifested as a shear stress from the contact pressure between the tube and the tubesheet. The pullout resistance incrementally decreases along the interface as the axial load is applied. Not all of the axial load is transmitted downward into the tubesheet because of the resistance provided by the shear stress, thus the Poisson contraction is progressively less with distance into the tubesheet. The complex theory of elasticity model is required to properly describe the complicated relationship between the deformations of the tube and the tubesheet.

The load carrying capability of a joint, F, was calculated by considering a force equilibrium for a short cylindrical element of the tube to establish a differential equation involving axial stress and net contact pressure. Compatibility and force deformation relations were then used to express the contact pressure as a function of depth in the tubesheet resulting in a first order differential equation (DE). When the contact pressure is constant the DE is homogeneous. When the contact pressure is a function of depth into the tubesheet the DE is not homogeneous. By considering the initial contact pressure to be linear (i.e., a constant slope) through the tubesheet, a linear first order DE results.

The theory of elasticity method used to calculate the residual contact pressure and pullout force for the Wolf Creek Model F SGs is different from the calculation of the residual contact pressure and pullout forces in the Model D5 (and Model 44F H*/B* submittals).

In order to be consistent with all H*/B* submittals going forward, F/L is recalculated for the Byron and Braidwood Unit 2 steam generators as discussed in Reference 18 using the first principles method and room temperature data only for the Model D5 steam generator (See Table 1-1 below). Note that only room temperature data will be used to determine tube resistance to pullout so that temperature effects do not have to be subtracted out from the test data. This is conservative because the value of mean – 1 σ bounds essentially all of the data points as shown in Figures 1-1, 1-2 and 1-3, below.

Table 1-1
0.25 Inch Pullout Forces for Model D5 Tube Tests

Test	Specimen Number	Test Temp. (°F)	Length (in)	Pullout Force (Lbf)	Residual Contact Pressure (psi)[1]	Contact Pressure from Temp.	Net F/L (Lbf/in)
1	D5H-R3-1	70	2.95	878.0	413.6	0	297.63
2	D5H-R5-1	70	4.95	2478.0	695.2	0	500.61
3	D5H-R7-1	70	6.90	6211.0	1250.9	0	900.14
						Mean	566.13
						Mean - 1σ	259.57

[1] The mean residual contact pressure is calculated to be 786.23 psi. The mean – 1 sigma value is 360.49 psi.

The resulting force per length number for the Byron Unit 2 and Braidwood Unit 2 Model D5 steam generators is the mean minus one standard deviation value for force per unit length and is calculated to be [ ]a,c,e This number is conservative and remains below all of the room temperature pullout data. See Figure 1-1. The results for other Model SGs are shown in Figures 1-2 (Model F) and 1-3 (Model 44F).

a,c,e

Figure 1-1 Tube Pullout Force Plot – Model D5 SGs

a,c,e

Figure 1-2 Tube Pullout Force Plot – Model F SGs

a,c,e

Figure 1-3 Tube Pullout Force Plot – Model 44F Data

A comparison of the pullout forces per length values for all Model D5, 44F, and F steam generators using this same methodology is provided in the Table 1-2 below:

Table 1-2
Room Temperature Tube Pullout Force (F/L) Values

Model Steam Generator	Original Pullout Force Value (lbf/in)	Revised Pullout Force value (Lbf/in)

a,c,e

1. Based on Theory of Elasticity model.
2. Based on first principles approach
3. Based on first principles approach and room temperature test data only

The revised joint pullout force value of [ ]a,c,e for the Byron Unit 2 and Braidwood Unit 2 steam generators will be factored into the response to NRC RAI question number 2.

It is important to note that the use of room temperature only data does not significantly change the previous values for pullout force used for the Model D5 (326.7 lbf/in versus 259.57 lbf/in) for which both room temperature and elevated temperature data were considered.

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 pullout 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.*

<u>Response to RAI No. 2:</u>

As was the case for the Model F tubing discussed in Reference 2 (LTR-CDME-07-72), the yield strength of the tubing used in the pullout test specimens was higher than the documented mean yield strength for prototypical tubing material in the Byron 2 and Braidwood 2 SGs. The yield strength of the tubing for the pullout test data for specimen numbers 1 through 3 of Table 1-1, was [ ][a,c,e] (Reference 17). The average yield strength of the tubing for the Byron 2 and Braidwood 2 SG is [ ][a,c,e] (Reference 8).

The following sources of potential variability were identified and the impact on H*/B* distances of varying these parameters has been quantified (References 16 and 20):

a,c,e

[ ]

Only six (6) of these must be addressed numerically in the uncertainty study for H*/B*, and only five (5) of these affect the residual contact pressure for hydraulic expansion. The remaining three variables (Hydraulic Expansion Pressure, Strain Hardening, Tube Outer Diameter) are inherently modeled in the analysis process and do not require independent variation and are discussed below. The greatest potential quantitative impacts on the H*/B* distances are the result of tolerances in [

]^{a,c,e}

Analysis Approach

Three approaches were considered for combining the uncertainties associated with determining the H*/B* distances. The three approaches considered are defined in Reference 4, EPRI Report 1012987, Steam Generator Integrity Assessment Guidelines, Rev. 2 and they are:

- Arithmetic strategy
- Simplified statistical strategy
- Monte Carlo strategy

Only the arithmetic strategy and simplified statistical methods were used to combine the uncertainties for the sensitivity study discussed below. This is the case because the arithmetic approach to combining uncertainties is expected to provide conservative results relative to both the simplified statistical and the Monte Carlo approaches. Moreover, the simplified statistical and Monte Carlo results typically provide similar results when it is assumed that the independent variables are normally distributed (as is the case with the parameters discussed above).

The method discussed below is a modified arithmetic strategy were the parameters were combined using at least a mean plus/minus one standard deviation value, whichever results in the greatest increase in H* distance. The method for combining the uncertainties is defined as a "stacked model" because all of the identified uncertainties are superimposed in a conservative direction.

The simplified statistical method result for combining uncertainties is provided in the discussion below but the explanation of the method for combining the uncertainties is not because the method is entirely consistent with that described in Section 7.3.5.2 of Reference 4, EPRI Report 1012987, Steam Generator Integrity Assessment Guidelines, Rev. 2.

The sensitivity study is divided into two parts:

21. An evaluation of those variables which affect the <u>residual contact pressure due to tube expansion</u>. Only five (5) of the nine (9) listed parameters directly affect the residual contact pressure used in the analysis of the Model D5 tubesheet joint. The applicable parameters do not include the uncertainty in the tube TEC and tubesheet TEC, tube outer diameter and tubesheet hole size.

22. An evaluation of, those variables which affect the contact pressure developed between the tube and the tubesheet during all plant operating conditions.

The uncertainty in all of the parameters above is considered in a sensitivity study. The maximum contribution from hydraulic expansion to tube-to-tubesheet contact pressure is calculated to be [ ]a,c,e This is the mean minus 1σ value based on the equation in the response to RAI No. 1. The contribution of residual contact pressure to tube pullout resistance is small compared to the contribution from differential thermal expansion between the tube and the tubesheet, internal pressure, and tubesheet bow (depending on tube radial location and elevation in the tubesheet).

The steam generator tube integrity acceptance criterion for the sensitivity study is cited on page 20 of the Draft Regulatory Guide DG-1074 (Ref. 5). The conditional probability noted in DG-1074 states that "burst of one or more tubes under postulated accident conditions shall be limited to 1.0E-2." The 1.0E-2 value is 40% of the limiting 0.025 value described in Draft Regulatory Guide DG-1074. This value is conservatively used as an acceptable probability for combining uncertainties for the critical parameters identified above to calculate H* and B* distances. For the calculation of the H*/B* distances, this value represents the probability that the performance criteria of NEI-97-06, Rev. 2 (Ref. 6) will <u>not</u> be met and, in this application, does <u>not</u> represent the probability of a steam generator tube rupture event. Tube burst is prevented by the constraint provided by the tubesheet, and the probability of tube pullout or excessive leakage during accident conditions will be considerably lower because the analysis for H*/B* already assumes a factor of 3 on the normal operating pressure differential. The calculation of a probability smaller than the acceptance criterion of 1.0E-2 supports the technical specification requirement that all tubes have adequate margin against burst (or tube pullout).

The empirical rule is used to assign a probability to each individual parameter, assuming a normally distributed population, given the mean and standard deviation of the data set. The empirical rule states that for a normal distribution:

- 68% of the data will fall within 1 standard deviation of the mean.
- 95% of the data will fall within 2 standard deviations of the mean.
- Almost all (99.7%) of the data will fall within 3 standard deviations of the mean.

This translates into a probability of 0.16 of varying from the mean value by +/- one standard deviation. The probability of varying by +/- two standard deviations is 0.025. And, likewise, the probability of varying by 3 standard deviations is 0.0015.

A "stacked" model of the uncertainties is used to determine a limiting combination of H*/B* distances. The model is considered to be a "stacked" model because all of the uncertainties are superimposed simultaneously in a conservative direction (i.e., a direction that results in the greatest H*/B* distances). The combination of uncertainties on the variables that result in an acceptably low probability closest to the acceptance criterion of 0.01 is used. Each of the parameters evaluated are assumed to be independent of each other. This is a valid assumption because no functional relationship exists between component dimensions and material properties, nor among the individual material properties noted above. The probability (Φ) that the "stacked" model H*/B* distances will not meet the performance criteria of NEI 97-06, Rev. 2 is calculated using the equation:

a,c,e

The parameters considered in the analysis that affect residual contact pressure are addressed below. Figure 2 of Reference 2 presents a normalized sensitivity analysis for various parameters based on an FEA study for hydraulic expansions performed at approximately 40 ksi. This sensitivity study is considered to also apply to the Byron 2 and Braidwood Unit 2 SGs whose tube expansions were performed at pressures between 30 ksi and 33 ksi.

In the context of this sensitivity analysis, an increase in the H* or B* distance means that the required inspection depth into the tubesheet, relative to the top of the tubesheet, increases. A decrease in the H* or B* distance means that the required inspection depth into the tubesheet, relative to the top of the tubesheet, decreases. The discussion below describes the effect on the H* and B* analysis from a variation of each parameter. Table 2-1 summarizes the discussion and provides a view of relative sensitivities.

These results are based on more than 100 individual sensitivity studies to determine the most limiting parameters and trends in the sensitivity of the H* and B* criteria to variations in the input parameters.

The following lists the parameters in order of greatest to smallest effect on the H* distance.

a,c,e

Similarly, the following lists the parameters in order of greatest to smallest effect on the B* distance.

a,c,e

[

]

The magnitude of the variability of each parameter was altered based on its importance to the H* and B* criteria. The term "greatest effect" in this context means that a variation in that parameter caused a larger difference in the final value of H* or B* than the next lower ordered parameter. For example, varying the [

]a,c,e

Variables Not Addressed Independently

As shown in Figure 2 in LTR-CDME-07-72 P-Attachment (Reference 2), three other variables can affect residual contact pressure. These are [

]a,c,e

[

]a,c,e

Impact of the Variability of [ ]a,c,e

The uncertainty on [

Impact of the Variation of [ ]a,c,e

Similarly, the variation in [

]a,c,e

[

]a,c,e

Impact of [ ]a,c,e

Based on NCE-88-271, Reference 7, dimensional data for [

]a,c,e

Impact of [ ]a,c,e

In solid mechanics, [

]a,c,e

Impact of [ ]a,c,e

A calculation was performed using the [

]a,c,e

[

]a,c,e

The values for the various parameters for the stacked model are included in the table below. The stacked model includes the variability of each parameter and the results of a series of sensitivity studies to determine the worst case, or maximum effect, on the H* and B* analysis. For example, the effect of varying the [

]a,c,c

The probability (Φ) that the stacked model H*/B* distances will not meet the performance criteria of NEI 97-06, Rev. 2 is determined to be:

a,c,e

= 2.6E-6.

0.01 >> 2.6E-6

This probability compares favorably with the acceptance criterion of 1.0E-2 which is *conservatively* based on the conditional probability of tube burst for one or more tubes under accident conditions.

Table 2-1
Stacked Model Parameters and Corresponding Uncertainties

Parameter Varied	Definition of Uncertainty Used	Input Parameter Value	Probability of Occurrence

a,c,e

The revised values for H*/B* using the parameters from the stacked model analysis are provided in Table 2-2 below. Table 2-2 includes the hot leg results. Table 2-3 includes the cold leg results. The combined probability term included in Table 2-2 includes a probability of:

a,c,e

Table 2-2
Hot Leg Result Summary for B* and H*
Using Stacked Worst Case Property Input Values

Case #	TEC Source	Combined Probability	Divider Plate Factor	Max. H* (in.)	Max. B* (in.)

a,c,e

Table 2-3
Cold Leg Result Summary for B* and H*
Using Stacked Worst Case Property Input Values

Case #	TEC Source	Combined Probability	Divider Plate Factor	Max. H* (in.)	Max. B* (in.)

a,c,e

a,c,e

HOT LEG RESULTS

[

]

Figure 2-1 Case #89, B* and H* Analysis Results for Model D5 Hot Leg Stacked Input Case.
[

]a,c,e

a,c,e

Figure 2-2 Case #92, B* and H* Analysis Results for Model D5 Hot Leg Stacked Input Case.
[

]a,c,e

COLD LEG RESULTS

a,c,e

Figure 2-3 Case #89, B* and H* Analysis Results for Model D5 Cold Leg Stacked Input Case.
[

]a,c,e

a,c,e

Figure 2-4 Case #92, B* and H* Analysis Results for Model D5 Cold Leg Stacked Input Case.
[

]a,c,e

The results of the sensitivity study show that the H* distance remains the limiting criterion for establishing the inspection depths in the Byron 2/Braidwood 2 steam generators at every radial location.

- For the hot leg, the required H* distance is 12.34 inches from the top of the tubesheet. (Case #89)
- For the cold leg, the required H* distance is 13.53 inches from the top of the tubesheet. (Case #89)

This value includes the uncertainties associated with the [ ]^{a,c,e} identified in Table 1.0 above when stacked in a manner analogous to the arithmetic process outlined in Reference 4, (EPRI 1012987; Steam Generator Integrity Assessment Guidelines: Revision 2, 2006.). As an alternative method for checking the validity of the "stacked model" results, a simplified statistical process analogous to that described in the EPRI Integrity Assessment Guidelines was applied and a similar result was obtained for the combined uncertainties. This alternative check included the limiting and more conservative definitions of variability in each parameter defined above, and the limiting H* distance of 13.70 inches that was calculated using the simplified statistical method bounds the length calculated using the "stacked model" discovered above (13.53 inches) for the cold leg factor 1.645σ recommended in the EPRI Steam Generator Integrity Guidelines.

The combined uncertainties, all biased in a direction to increase the H* distance, with conservative assignment of standard deviation from their means, results in a whole bundle probability of 2.6E-6 of not meeting the structural and leakage performance criteria specified in NEI 97-06, Revision 2.

The maximum H* distances are based on using a divider plate factor of 0.64 versus a divider plate factor of 1.00. It has been shown that, without the divider plate to stub runner weld present, the tube plate vertical deflection is still limited to 36% less than if the divider plate was not present at all, Reference 12 (EPRI 1014982; Analysis of Primary Water Stress Corrosion Cracking and Mechanical Fatigue in the Alloy 600 Stub Runner to Divider Plate Weld Material.). In other words, most of the structural benefit of the divider plate is derived from the welds to the channelhead, and not from the weld to the tubesheet (stub runner). No degradation has been reported in the welds between the divider plate and the bowl, in contrast to the reports of degradation in the weld between the divider plate and the stub runner.

The results discussed in this response are not expected to change in the highly unlikely case that an input parameter takes on an "extreme" value due to some combination of installation artifacts or deviation from mean material properties. This is for two reasons:

1. The variability of the parameters that are most important to the analysis is known and the extreme values have already been considered. For example, the thermal expansion coefficient of the tube and the tubesheet is very important to the results of an H* analysis. However, the ANTER labs data shows that the variability of those parameters is limited to +/- 1.5% with 95% confidence interval. The conclusion, then, from the most recent data means that there will always be a difference between the TEC of the tube and the tubesheet and therefore there will always be differential thermal growth between the tube and the tubesheet. The sensitivity study results presented above include the effects of varying both the ANTER labs properties and the ASME code properties to the extreme limits of the available test data and still show adequate margin with respect to the desired inspection distances into the tubesheet. Note that significant margin exists despite the fact that the other input parameters are also being varied, many to the extreme

range of the known data for that property (e.g., Young's Modulus of the tube and the tubesheet). Therefore, extreme values of most of the parameters critical to the H* and B* analysis (i.e., Young's Modulus of the tube, thermal expansion coefficient of the tube and the thermal expansion of the tubesheet) have already been evaluated in the analysis.

The only critical parameter for which an extreme value has not been evaluated is the yield strength of the tube. The results of varying the tube yield strength to either extremely negative limits or extremely positive limits still provide adequate margin with respect to the desired inspection limits. For example, if the yield strength in the tube approaches a value of zero, the contact between the tube material and the tubesheet material will be nearly perfect. This would dramatically increase the residual contact pressure between the tube and the tubesheet and make it much more difficult for the tube to pull out under the limiting loads. On the other hand, assuming that the yield strength of the tube is triple the highest value (approaching a value of 6σ) the residual contact pressure from installation would tend toward zero. The results of a B*/H* analysis for the Model D5 SG, with no residual contact pressure from installation, but with mean ASME code material properties and design input values for all other parameters, is a B* of 8.36 inches and an H* of 13.74 inches for the limiting cold leg. Therefore, an extreme variation in tube yield strength is nearly bounded by the analysis results from the "stacked" model cases discussed above.

2. The bellwether concept inherently addresses any postulated effect of an extreme value result. If a tube in the steam generator is postulated to exhibit the worst case value for every parameter that is used in the B*/H* analysis, and those parameter values were beyond those considered in any of the sensitivity studies described here, and the tube is assumed to be still in service, the expected conditions would be such that the tube would leak during normal operating conditions. If the leak during normal operating conditions is significant, it will be located and plugged. A significant leak would require a gross crack that occurs in a short time (i.e., one operating cycle) or have a prior history of leakage. There is no known mechanism that would cause a complete circumferential throughwall crack in one operating cycle to cause a significant leak, which, in any case, would cause a plant shutdown if it were to occur. Even for the case of a severe circumferential crack, not of 360 degree extent, or a large axial crack, structural integrity of the tube would be maintained by the remaining tube ligaments. Therefore, the postulate of an extreme condition tube is extremely unlikely due to the normal plant leakage monitoring requirements.

It is concluded that the H* and B* inspection criteria conservatively provides significant margin in the tubesheet for all of the tubes in the bundle even if:

- The extremely unlikely case that a specifically stacked set of material properties variations and dimensional conditions, all biased in a direction to result in greater H* values, exist in the steam generator;
- The divider plate weld to the tubesheet is 100% degraded;
- Lower bound residual pullout strength and contact pressures are used.

3. *The H* analyses in References 1 and 2 are based, in part, on pullout resistance associated directly with hydraulic expansion process. This pullout resistance was determined by subtracting out the effects of differential thermal expansion between the tube and tubesheet test collar from the measured pullout 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 pullout force determined from the pullout test and on the computed H* distances?*

Response to RAI No. 3:

The impact of varying values for thermal expansion coefficient for both the tube and the tubesheet is assessed in the response to RAI No. 2.

The effect of considering the alternate value of TEC for the tubesheet noted in the question for the analysis of the pullout test is bounded by the effect of using the same TEC in the analysis for H* and B*. It is not proper to selectively use different values of the same parameter in different phases of a single analysis of the same conditions. The effect of using a reduced TEC for both the pullout analysis and the tubesheet displacement analysis is non-conservative. The lower residual pullout strength resulting from analysis of the test data has a much smaller effect (increase) on H* and B* than the use of the same value for TEC in the H*/B* analysis. A lower value of TEC applied to the tubesheet analysis significantly reduces the tubesheet displacement, resulting in increased contact pressure between the tube and the tubesheet. As a consequence, the smaller value of tubesheet TEC significantly decreases the H*/B* values.

The use of the ANTER data plus one standard deviation value for the thermal expansion coefficient for the tubesheet material of 7.2E-6 in/in/°F is judged to be extremely conservative when used in concert with a reduced thermal expansion coefficient of 7.5 E-6 in/in/°F for the steam generator tubing for the reasons discussed in the response to NRC RAI Number 2. The value of 7.2E-6 in/in/°F is the at the extreme limit of the ANTER test data. The use of this value is in the most conservative possible with respect to the H*/B* analysis.

The impact of reducing the value for TEC to 7E-06 in/in/°F for the tubesheet without a corresponding change in the thermal expansion coefficient of 7.82E-6 in/in/°F for the tube would result in a less conservative (shorter) H*/B* distance. Therefore, the impact of the MATWEB.com value for thermal expansion coefficient for the tubesheet region on the computed pullout force has not been calculated.

4. *Reference 2, Enclosure I, Response to RAI question 7 - The Model D5 steam generator (SG) pullout data in Table 2 indicate that pullout 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, pullout 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.*

<u>Response to RAI No. 4:</u>

(Note: The specimen number for test number 1 in Table 2.0 of Reference 2, LTR-CDME-07-72 P-Attachment, response to RAI question 7, is D5H-R3-1 not D5H-R5-1 which may have contributed to the misinterpretation of the data.)

The decrease in tube yield strength with increasing temperature is not likely to reduce or limit the contact pressure between the tube and tubesheet. First, the loads and conditions on the limiting section in the tube are not sufficient to cause plastic yielding in the tube. See the response to RAI No. 5. A decrease in the tube yield strength is likely to increase the contact pressure between the tube and tubesheet up to the point of plastic yielding of the tube material. For example, consider a load state on the tube that would generate a state of plastic yielding in the tube material. In that case, the plastically yielded tube material would flow with no resistance in an isochoric manner (i.e., volume preserving deformation). This kind of deformation would have two effects on the interface between the tube and the tubesheet.

1. The tube material would be able to plastically flow into the crevices and surface peaks and valleys of the tubesheet local to the plastic tube material. This would create a state of near perfect intimate contact between the tube and the tubesheet; thereby preventing any leakage through the crevice and allowing for the dispersal of large amounts of strain energy which would further reduce the possibility of tube pullout.

2. Although the surface of the tubesheet could push back on the plastically yielded tube material, the internal pressure in the tube (the primary pressure of the SG) would keep the tube material in contact with the tubesheet regardless of the tubesheet deformation because there would be no stiffness in the tube material to resist the plastic flow.

In conclusion, it is highly unlikely the tube material in the tubesheet will plastically yield due to the loads and constraints applied to the tube (See RAI No. 5). However, in the event that the tube did plastically yield, the effect would be to increase both the leakage resistance and the pullout resistance of the tube. Therefore, it is more conservative to consider both the tube and the tubesheet material as elastic structures that retain their respective stiffnesses in the event that contact pressures and conditions allow the two material surfaces to deflect away from each other. This more conservative case is the condition that is considered in the H*/B* analysis.

The pullout strength for the Model D5 tubing and the Model 44F H*/B* submittals is a lower bounding value based on pullout test results from both room temperature and elevated temperature testing. The data plotted in both Figures 4-1 and 4-2 have had the effects of thermal expansion due to elevated temperature testing subtracted out, thus only the mechanical contribution of the joint are reflected in the figures. As can be seen from Figure 4-1 and 4-2 provided below, the average minus one standard deviation value for pullout strength essentially bounds the data cloud for both the Model 44F and Model D5 test data. Regardless of the tube size, it can be seen that using the empirically determined mean minus one standard deviation resulting value for pullout strength is a conservative value for the bulk of the data for both the Model D5 and 44F steam generators.

The data plotted in both Figures 4-1a and 4-2a have had the effects of thermal expansion due to elevated temperature testing subtracted out. It can be seen by Figures 4-1 and 4-2 that, for the different expansion lengths, the room temperature data does not always result in the lowest pullout strength for the Model D5 and 44F test specimens as well. For the specific example cited by the NRC staff, i.e., the Model F result for Wolf Creek, it can be seen that the Model D5 7-inch expansion length room temperature test specimen results in a higher tube pullout strength than the elevated temperature test results. For the Model 44F steam generator, however, the approximate 7-inch expansion length room temperature test specimen results in a lower tube pullout strength than that for the elevated 400°F test specimen and a comparable pullout strength to the elevated 600°F test result. Therefore, it is concluded that what is interpreted in the question as a data trend reflecting temperature effects is, in reality, normal variability among the different test specimens.

a,c,e

a) elevated temperature data adjusted to remove thermal expansion component

a,c,e

b) Room Temperature Data

Figure 4-1 Model D5 Pullout Test Results for Force/inch at 0.25 inch Displacement

a,c,e

a) (elevated temperature data adjusted to remove thermal expansion component)

a,c,e

b) Room Temperature Data

Figure 4-2 Model 44F Pullout Test Results for Force/inch at 0.25 inch Displacement

As discussed in the response to NRC RAI Number 1 above, it is noted that the analysis of the residual contact pressure and the pullout force for the Wolf Creek Model F steam generators used a theory of elasticity model. In order to obtain a consistent approach for defining the tube pullout strength, it was decided to only use room temperature results to determine tube resistance to pullout so that the temperature effects do not have to be subtracted out of the data. The mean minus one standard deviation room temperature values for tube pullout strength for the Model F, Model D5, and 44F steam generators are repeated here for convenience:

a,c,e

Room Temperature Tube Pullout Force (F/L) Values		
Model Steam Generator	**Original Pullout Force Value (lbf/in)**	**Revised Pullout Force value (lbf/in)**

1. Based on Theory of Elasticity model.
2. Based on first principles approach
3. Based on first principles approach and room temperature test data only

As discussed in the response to NRC RAI Number 1, the use of room temperature data only does not significantly change the previous values for pullout force used for the Model D5 and 44F steam generators when both room temperature and elevated temperature data are considered.

The NRC staff's concern whether a reduction in tube yield strength with increasing temperature might be sufficient for some specimens to limit any increase in contact pressure associated with differential thermal expansion between the tube and tubesheet was considered. As stated in the response to NRC RAI No. 7 in LTR-CDME-07-72, Reference 2, the data in Table 2.0 is intended to show that for an expansion length as short as 2.95 inches, a significant increase in pullout force occurs at elevated temperatures (e.g., pullout force increases from [ ]^{a,c,e} when temperature increases from 70°F to 600°F, respectively). Therefore, since the test results inherently include the variation of yield strength with temperature, the results indicate that yield strength reduction with temperature does not limit the increase in contact pressure due to thermal expansion.

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 pullout 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 1?*

Response to RAI No. 5:

It is not possible for a SG tube with an intact cross section to compressively fail due to the applied pressures calculated in the H* and B* analysis. The basis for this statement is a stress analysis of the tube material using the Tresca yield criteria. The analysis approach and results follow:

The maximum applied contact pressure from Tables 7-6 through 7-11 and Tables 7-6a through 7-11a in Appendix C of this report is 3629.42 psi (3989.91 psi – 360.49 psi) or approximately 3.6 ksi. The maximum contact pressure occurs during FLB at the bottom of the tubesheet (21.03 inches below the top of the tubesheet). For 3/4 inch Alloy 600 tubing, the minimum yield strength is [ ][a,c,e] at room temperature, Reference 8 (WCAP-12522). At a temperature of 650°F the tensile yield stress [ ][a,c,e] Reference 8 (WCAP-12522). For a material element at the outside surface of the tube wall (the tube material in contact with the tubesheet wall), assuming the full axial load is applied to the tube despite its location in the tubesheet, the state of stress is equivalent to that of an element in a thin-walled pressure vessel with the third principal stress not equal to zero.

For the FLB condition, the principal stresses on the tube wall element are:

$$\sigma_1 = \sigma_{XX} = \frac{p_{PRI} r}{t} = \frac{2835 psi \cdot 0.3607 in}{0.0426 in} = 24004.3 psi = 24.0 ksi$$

$$\sigma_2 = \sigma_{YY} = \frac{p_{PRI} r}{2t} = \frac{2835 psi \cdot 0.3607 in}{2 \cdot 0.0426 in} = 12002.2 psi = 12.0 ksi$$

$$\sigma_3 = \sigma_{ZZ} = P_{CONTACT} = -3629.42 psi = -3.6 ksi$$

Applying the Tresca theory, or maximum shear stress yield criteria, gives:

$$\frac{Y}{2} = \tau_{MAX} = \left| \frac{\sigma_1 - \sigma_3}{2} \right|$$

$$Y = 35.5 ksi > \left| 24.0 ksi - (-3.6 ksi) \right| = 27.6 ksi$$

This shows that the tube has a safety factor of at least 1.29 with respect to yielding using the minimum yield strength for Alloy 600 at elevated temperature. At room temperature conditions, the safety factor increases to nearly 1.83. If the material properties and conditions in the steam generator are varied for a worst case analysis then the maximum contact pressures in the tubesheet will be reduced. Any reduction in contact pressure between the tube and the tubesheet make it even less likely that a tube will yield in compressive failure.

In conclusion, it is not possible for an intact tube to yield in compression under the state of stress created by the transient event with the maximum applied contact pressure.

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 pullout 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>Response to RAI No. 6:</u>

The tables in Reference 2, Enclosure I, Response to RAI question 17 print residual pullout value for the analysis using a cell reference from the calculation spreadsheet. This cell reference was not deleted to make it clear that the residual mechanical joint strength was excluded from some portions of the analysis. The residual mechanical joint strength was not included in the H* computations for accident conditions but was included for normal conditions. The result of the analysis shows that the NOP case is limiting despite the elimination of the residual joint strength for the SLB case. The residual mechanical joint strength was included in the H* computations for both accident conditions and for normal operating condition for the Byron 2 and Braidwood 2 steam generators.

7. *Reference 2, Enclosure I, Table 7-6 - This table states that the required pullout 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>Response to RAI No. 7:</u>

The H* distance is determined using an automated MS-EXCEL spreadsheet routine written in VBA code. Beginning at the bottom of the tubesheet, the code integrates the resistance forces over the length of the tubesheet and compares the results to the external load applied, i.e., 3 times the end cap load. The stability of the calculation routine depends on the increment size: A too-small increment results in code instability; a too-large increment size can result in hunting for a solution. If the integrated resisting force is greater than the applied force, the prior reference location, i.e., the bottom of the tubesheet, initially, the code performs the integration again based on the shorter distance. As the net force approaches zero, the increment is automatically reduced. This iteration and incrementation is continued until the resisting force is less than the applied force. The final net-zero distance is established by interpolating between the closest negative and positive net forces.

It is possible that a minor difference, in this case 0.020 inch, can result from the variation of the calculation increment.

After a solution has been obtained with the iteration code, a constant factor of 0.3 inch (taken from the top of the tubesheet down) is added to the result to account for the potential variation in the location of the bottom of the expansion transition (BET). Compared to this constant factor applied for potential variation of the BET from the top of the tubesheet, a 0.02 inch variation of the calculated value of H* is negligible.

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.*

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.*

Response to RAI No. 8 and 9:

No revisions to the Technical Specifications are proposed.

Cyclic loading testing was conducted on five tubes (11/16 outer diameter), including A600TT tubes, which were hydraulically expanded into a test block (Reference 7 – NCE-88-271, "Assessment of Tube-to-Tubesheet Joint Manufacturing Processes for Sizewell B Steam Generators Using Alloy 690 Tubing," November 1988). The tubes were loaded to a given load between [ ]a,c,e The axial position of the primary side end of the tube was measured before loading and after unloading. The direct measurements of the displacement of the primary end of the tubes were in reasonable agreement with displacements recorded from the secondary side by the displacement gage. Reference 7 concludes that the tubes underwent rigid body displacement (breakaway) at a [ ]a,c,e relative to the maximum pullout load. Based on the much steeper unloading and reloading curves relative to the initial loading curves (See R12C4 and R12C5), it can be concluded that the tubes did slip at small loads.

Incremental slippage is not expected to be an issue for the Byron /Braidwood Unit 2 steam generators for the following reasons:

- Incremental motion, like that observed in manufacturing testing, is not expected because of the constraints in an operating SG. The tests that exhibited slight incremental motion did not simulate the operating conditions (temperature and pressure) that would lead to additional constraint provided by tubesheet bow and the thermal effects.

- For all tubes, a circumferential sever is required for any motion to occur. The development of a 360° circumferential crack from crack initiation requires significant time. Normal inspection to H* will identify a circumferential crack within the H* distance if incremental motion is postulated.

- In the extremely unlikely event of a complete separation of a tube, the minimum difference between H* and B* is approximately 2.3 inches. Therefore, motion of up to 2.3 inches (See Figure 8-2 below) will not affect leakage, and provides a definitive, observable degree of displacement. This extent of motion is not possible in the interior of the bundle. Because of axial constraints provided by the tube bundle, and the lateral restraint provided by the AVBs, the maximum axial displacement of a tube is limited to less than one tube pitch (1.0625 inch), even if the u-bend moves toward an adjacnet column. Therefore, any concern about displacement should focus principally on the peripheral tubes. For the peripheral tubes, normal bobbin inspection will detect significant displacement since the expansion transition will be reported above the top of the tubesheet.

- Since the Bellwether Principle applies, normal leakage monitoring will determine the need to shut down the plant when primary to secondary leakage is detected. Therefore, excessive accident induced leakage within the tubesheet expansion region can never occur. If a tube is found to be leaking, normal inspections to H* will determine if a flaw exists above H*. In that event, the tube must be plugged. All tubes except the peripheral tubes can be monitored during normal inspections for potential tube to tube contact. If tube to tube contact is observed, the source of the contact will be determined and appropriate action taken to repair the condition.

a,c,e

Figure 8-2 Difference Between H* and B*

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 pullout 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 pullout, 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.*

Response to RAI No. 10:

The following accidents model primary to secondary leakage in the Byron Unit 2 and Braidwood Unit 2 FSAR:

- Section 15.1.5, Steam Generator Piping Failure at Zero Power
- Section 15.3.3, Reactor Coolant Pump Shaft Seizure (Locked Rotor)
- Section 15.3.3.6, Reactor Coolant Pump Shaft Seizure (Locked Rotor) with Failed-Open PORV
- Section 15.4.8, Spectrum of Rod Cluster Control Assembly Rejection Accidents

An evaluation of these transients has been conducted to determine the duration of time that the primary-to-secondary pressure differential exceeds the normal operating condition primary-to-secondary pressure differential (References 13 and 14. It has been determined that the duration of time that the primary-to-secondary pressure differential exceeds the normal operating pressure differential for a control rod ejection event is [ ]^{a,c,e} The locked rotor transient by itself results in an increase in primary to secondary pressure differential above normal operating pressure differential for less than 30 seconds. The locked rotor event with a failed open PORV is defined to occur for a 10 second time interval and follow the locked rotor transient primary-side pressure curve. However, no information is given for the secondary-side steam pressure/temperature transient in the steam generator with the failed-open PORV or the intact steam generators. Therefore, this transient is assumed to continue for 300 seconds with the primary side returning to 2235 psig after 10 seconds. The secondary side pressure is conservatively represented by the curve for the large steam line break which shows the steam pressure drops to 14.7 psi after approximately 250 seconds. From this point forward, the pressure differential in the loop with the failed-open PORV is assumed to remain at 2250 psi. Although the secondary side steam pressure transient remains undefined for the locked rotor with failed- open PORV transient for the intact steam generators, it is concluded that the driving pressure differential remains less than a factor of two above the normal operating pressure differential (2235 psi/1423 psi = 1.6) in the intact steam generators for the duration of the transient. The SLB event remains the limiting accident. The steam generator tube leakage rate assumed during the SLB accident is 0.218 gpm (each steam generator) in the intact steam generators and 0.5 gpm in the faulted SG. The leakage rate that is assumed for the locked rotor event (without a failed open PORV) and the control rod ejection event is 1.0 gpm total for all SGs. The leak rate for the locked rotor event (without a failed open PORV) and the control rod ejection event can be integrated over one minute to compare to the limit. Since the time above the normal operating pressure differential is less than [ ]^{a,c,e} the time integration should permit an increase in acceptable peak leakage at the peak pressure differential by at least a factor of 2. As the allowable primary to secondary leak rate is limited to 150 gpd (0.1 gpm) in the Byron Unit 2/Braidwood Unit 2 SGs during normal operating conditions, accident induced leakage is expected to remain less than [ ]^{a,c,e}

[ ][a,c,e] This leakage rate (0.2 gpm) is bounded by the current accident analysis assumptions. This is also determined to be the case for the locked rotor event with the failed-open PORV. The primary-to-secondary leakage assumed in this transient is 0.218 gpm in the intact steam generators and 0.5 gpm in the steam generator with the failed-open PORV. The maximum primary-to-secondary leakage that may be experienced in the intact steam generators or the steam generator with the failed-open PORV for the duration of the transient is also less than 0.2 gpm with the implementation of the H*/B* criteria. See Tables 10-1 and 10-2 and Figures 10-1 and 10-2 below

For tube pullout considerations, the limiting load applied to the tube is 3 times the normal operating pressure differential. This value bounds 1.4 times both the SLB and FLB loads.

Table 10-1
Primary-to-Secondary Side Pressure Differential Time-Histories for RCP Locked Rotor and Control Rod Ejection Transients (Byron Unit 2/Braidwood Unit 2 Steam Generators)

Time sec	Normal Operation P psi	RCP Locked Rotor (Dead Loop) P psi	RCP Locked Rotor (Active Loop) P psi	Control Rod Ejection P psi
0	1423	1423	1423	1423
1	1423	1671	1677	2057
1.5	1423	1794	1804	2094
2	1423	1805	1818	2130
3	1423	1826	1845	1807
4	1423	1766	1771	1485
5	1423	1641	1632	1162
8	1423	1268	1216	1093
10	1423	1229	1147	1048
11	1423	1207	1126	1024
15	1423	1121	1039	977
20	1423	1063	1019	917
24	1423	1044	1028	869
40	1423	969	953	836
50	1423	964	902	814
60	1423	959	898	792
70	1423	950	905	770
80	1423	940	912	770
120	1423	1005	959	770
300	1423	1005	959	770

Time sec	RCP Locked Rotor w/ PORV Failure P psi
0	1533
4	2071
5	1967
9	1795
10	1774
25	2072
250	2235
300	2235

Table 10-2
Primary-to-Secondary Side Pressure Differential Time-History for Main Steam Line and Feedwater Line Breaks (Byron Unit 2/Braidwood Unit 2 Steam Generators)

Time sec	Normal Operation ΔP psi	Feedwater Line Beak ΔP psi
0	1423	1423
20	1423	1707
35	1423	2023
40	1423	2322
100	1423	2560
600	1423	2560
4000	1423	2235

Time sec	Steam Line Break ΔP psi
0	1423
150	1277
260	1432
400	1475
800	1695
900	1765
1200	1975
1600	2265
4000	2560

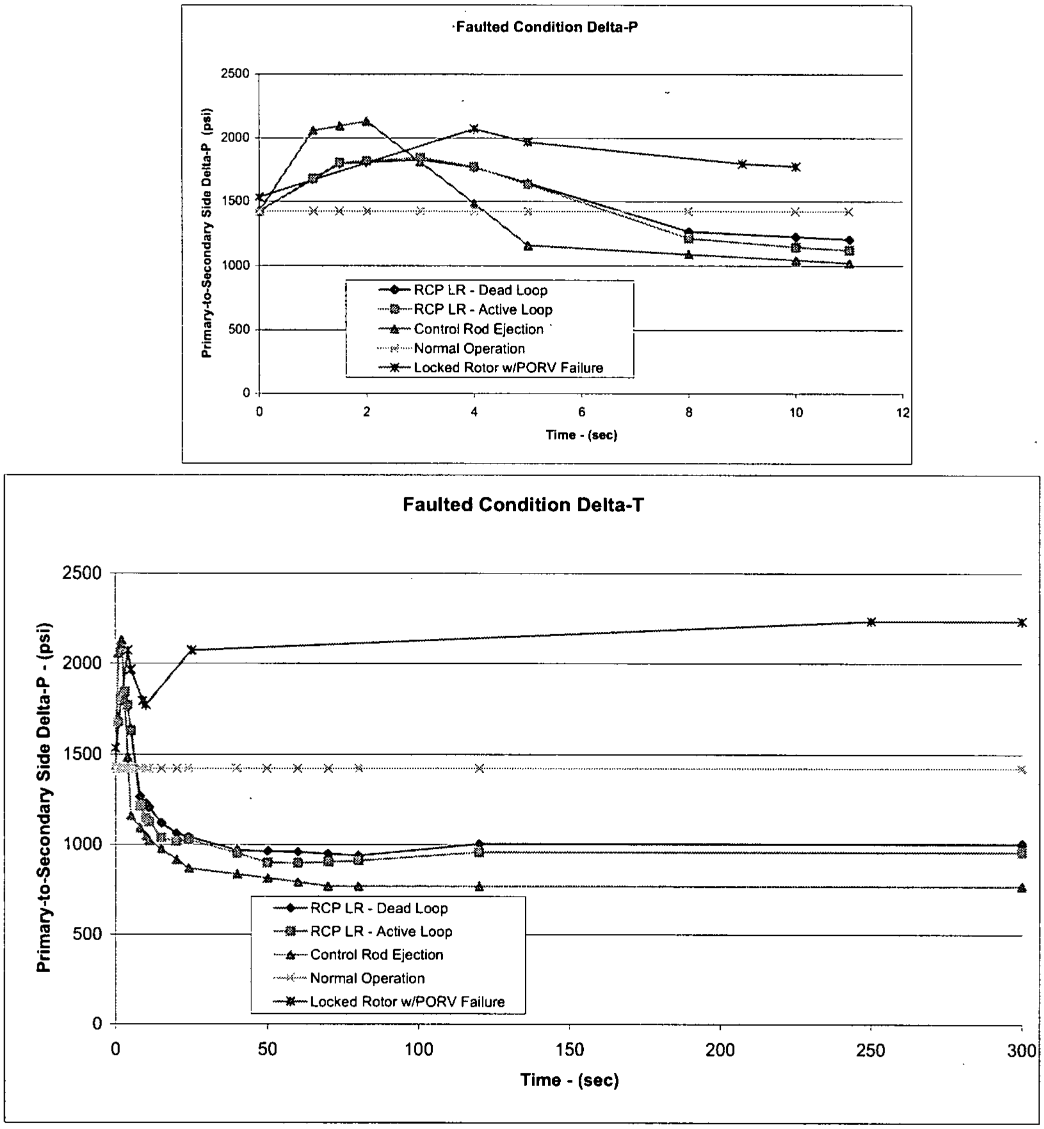

Figure 10-1 Plots of the Primary-to-Secondary Side Pressure Differential Time-Histories for RCP Locked Rotor and Control Rod Ejection Transients (Byron Unit 2/Braidwood Unit 2 Steam Generators)

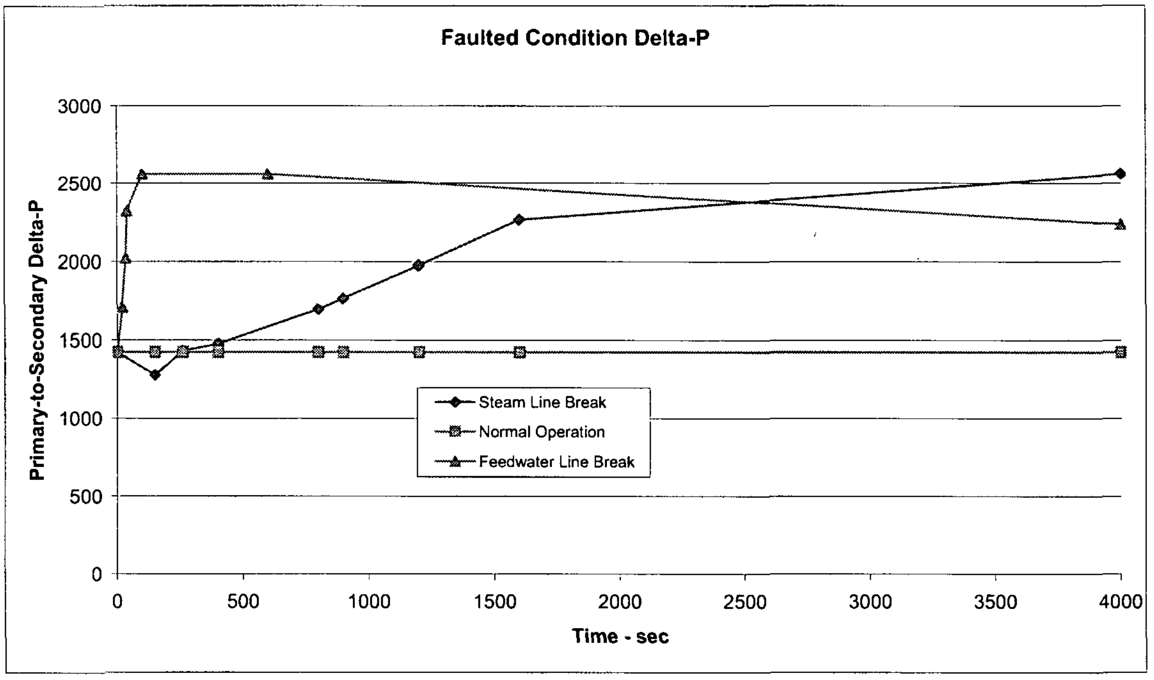

Figure 10-2 **Plot of the Primary-to-Secondary Side Pressure Differential Time-History for Main Steam Line and Feedwater Line Breaks (Byron Unit 2/Braidwood Unit 2 Steam Generators**

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.)*

Response to RAI No. 11:

The reasons for not including the data are discussed in the response to NRC RAI No. 11 from Enclosure I to WCNOC letter WO 07-0012.

The additional data include:

1. Test results from the Model F specimens that were not prepared in accordance with criteria of the test specifications (i.e., the test specification expansion pressure was not achieved).

2. Test results from Model D5 specimens that resulted in no leakage. The Model D5 testing was specifically oriented toward leakage; therefore, since a zero-leak test does not provide usable data for leakage, these tests were not previously reported.

The data that were added includes both room temperature data and elevated temperature data.

The Staff's observations about the leak rate data and the relationship between the loss coefficient and contact pressure for the Model D and Model F tests are noted. However, Westinghouse believes, based on the data, that there is a definable relationship between the loss coefficient of a TS crevice and the contact pressure between the tube and tubesheet. The data available provides the information necessary to define a conservative log-linear relationship between contact pressure and loss coefficient. See Figure 11-1 for a plot of the total Model F and Model D data set with a curve of the 95% confidence fit obtained by a log-linear regression of the combined Model D and Model F data sets.

a,c,e

Figure 11-2 Plot of Loss Coefficient, k, as a Function of Contact Pressure, P_c.

The results plotted in Figure 11-1 illustrate four important characteristics in the Model F and Model D data and the resulting 95% confidence interval fit of the combined data set.

a,c,e

These results indicate that the Model D loss coefficient data is [

]a,c,e

[

]a,c,e

If only the Model D data were used, [

]a,c,e

Therefore, the available test data is considered valid and is applied in a very conservative manner via the Darcy model for axial flow through a porous medium. The Staff has noted in the past that the Darcy model is a more conservative approach than either the Bernoulli model or a converging orifice model. A different model is likely to give a less conservative result compared to the current bounding approach using the fit of the combined data set. In conclusion, using the 95% confidence fit of the combined Model F and Model D data set is an extremely conservative approach which results in a curve that bounds most of the loss coefficient data and maximizes the required B* distance.

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.*

Response to RAI No. 12:

The data supporting Figure 13 is provided below in Figure 12-1.

The pressure differentials used to calculate the contact pressures for each data point are listed in Table 12 -1. The crevice pressure differentials were calculated using bounding crevice pressure ratio values. The crevice pressure ratio (CPR) values were determined by taking the limiting median value as described in the White Paper (Appendix B) and are summarized in Table 12-2.

The crevice pressure ratios were applied based on the pressure differential in the leak test, for example, the data from the White Paper tests that had a ΔP of 1000 psi was applied to cases that also had a Δ P of 1000 psi to determine the appropriate contact pressure.

a,c,e

Figure 12-1 Loss Coefficient versus Contact Pressure (Combined Model F and Model D5)

Table 12-1
Data Supporting Figure 13 of the White Paper (Appendix B)

SG MODEL	PRIMARY SIDE TEST PRESSURE (PSI)	TEST TEMPERATURE (°F)	CONTACT PRESSURE (PSI)	CREVICE PRESSURE DIFFERENTIAL (PSI)

a,c,e

Table 12-2
Summary of Crevice Pressure Ratios from White Paper (Appendix B)

a,c,e

(Primary pressure units are psi)

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.*

Response to RAI No. 13:

Figure 2 from Rev 0 of the White Paper (Rev. 3 is Appendix B of this document) are presented below as Figure 13-2 for ease of reference. The constant loss coefficient values noted in the White Paper are highlighted on the plot as well. Also please refer to the responses to RAI No. 11 and RAI No. 12.

No statistical tests were performed to evaluate combining the two data sets. A conservative approach was used that essentially bounds all of the available data, thus a statistical evaluation was not required.

While the data for the Model F tests results in a reasonable correlation; the Model D results in a slightly negative correlation. When the two data sets are combined, the linear regression results for the combined dataset is more conservative than interpolating between the regression fits of each of the Model D or Model F data. The high variability of the combined data set produces a 95% confidence interval fit that bounds the lower limit of almost the entire data population. See the Response to RAI No. 11.

If no correlation is assumed to exist, the intercept for the log linear equation is the average value of the population and the slope of the log linear correlation is zero. A lower bounding value for loss coefficient is also shown.

The maximum B* inspection distance for the case of the lower bounding limit of the loss coefficient, assuming mean ASME code material properties and mean design inputs, is 11.15 inches. In comparison, the maximum B* inspection distance for the case of the mean loss coefficient value, assuming mean ASME code material properties and mean design inputs, is approximately 6 inches. Since the lower 95% confidence fit for the combined data is well below the mean value and close to the lower bounding value, and essentially bounds the entire available data, it is conservative to use the lower 95% confidence fit for the B* calculation.

In conclusion, the 95% confidence fit of the entire data set results in a loss coefficient versus contact pressure relationship that bounds almost the entire set of data and produces highly conservative B* inspection distances. Even if the lower bounding limit of the loss coefficient values is used the B* inspection distance is less limiting than the H* inspection criteria.

a,c,e

[ ]

Figure 13-3 **Comparison of 95% Confidence Limit Scaled Model F and Model D Loss Coefficient Data Fits for the Case of No Flashing in the Crevice, Pc Calculated using TE and Varied Crevice Pressure with No Data Excluded**

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.*

Response to RAI No. 14:

See Figure 13-2 in the Response to RAI No. 13 for a plot that illustrates the two constant loss coefficient values. The first assumed value for zero slope loss coefficient comes from the lowest value from the Model D curve fit from Figure 14. The second assumed value for zero slope loss coefficient does not come from Figure 12. The second assumed value for zero slope represents the mean of all of the data points included in Figure 11 of Enclosure I to WCNOC letter WO 07-0012.

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.*

Response to RAI No. 15:

- The results shown in Figure 15 of Enclosure I to WCNOC letter WO 07-0012 do include the consideration of mechanical strength in the joint during SLB.
- The results shown in Figure 16 of Enclosure I to WCNOC letter WO 07-0012 do not include the residual strength of the joint during SLB.
- The results shown in both Figure 15 and Figure 16 of Enclosure I to WCNOC letter WO 07-0012 are for the cold leg.
- The results shown in Figure 15 of Enclosure I to WCNOC letter WO 07-0012 are not used in the current determination of B* and H* inspection distances and were provided only as a means of comparison to the results shown in Figure 16 of Enclosure I to WCNOC letter WO 07-0012.
- The results shown in Figure 15 of Enclosure I to WCNOC letter WO 07-0012 and Figure 11-2 do not reflect the current level of technology nor the changes required to address recent RAI *(circa 2006-2007).*
- The B* and H* values at small tubesheet radii in Figure 15 of Enclosure I to WCNOC letter WO 07-0012 are slightly less than those listed in Table 11-1 of Enclosure I to WCNOC letter ET 06-0004 for the cold leg. This is due to the correction of minor errors in the spreadsheet used to calculate the B* values.

- The contact pressures shown in Tables 7-6 and 7-8 of Enclosure I to WCNOC letter WO 07-0012 do not include the residual contact pressure due to hydraulic expansion.

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.*

Response to RAI No. 16:

Three finite element models were used to justify the contact pressure calculations for H* and B*.

- The first, original, finite element analysis is an axisymmetric model, as shown in Enclosure I to ET 06-0004, that calculates the radial deflection of the tubesheet under unit load conditions as a function of the tubesheet elevation and radius. For convenience, this model will be called Model O-1.

- The second finite element model is a three-dimensional solid model that calculates the vertical deflection of the tubesheet as a function of tubesheet radius for various load cases assuming different stiffnesses and/or cracking in the divider plate. The output from the second finite element model defines the divider plate factor and was used to check the results from the first. For convenience, this model will be called Model O-2.

- The third finite element model was an improved version of Model O-2, which included changes further discussed below.

The original finite element model (Model O-1), that was used to calculate the tubesheet deflection, was not revised as it does not model the divider plate in the lower SG complex.

The second finite element model (Model O-2), originally used to determine the divider plate factor used in the H*/B* analysis, was revised to become Model O-3. The modifications to the O-2 model include a solid tubelane, the addition of the weldments and connections between the divider plate and the channelhead and tubesheet, non-axisymmetric boundary conditions and variable stiffness in the divider plate material. The revised model, O-3, more accurately reflects the true geometry, loading and material conditions of the tubesheet. Therefore, the revised model, O-3, more completely represents the actual SG configuration than the original FEA analytical model (O-2) did.

The divider plate factor was conservatively chosen as 1.00 for the development of Tables 7-6 through 7-11 and 7-6a through 7-11a in Reference 19 for Byron Unit 2 and Braidwood Unit 2. A divider plate factor of 1.00 means that no structural credit is taken for the presence of the divider plate and the divider plate is assumed to be entirely absent; no support is derived for the DP either in the bowl or at the tubesheet. This condition is not possible because the residual weld beads would retain the divider plate even if it were postulated that the welds had completely cracked. Further, there has been no evidence in the industry of cracking in the weld between the divider plate and the channel head, nor is such degradation expected because these welds are thermally stress relieved. The results of a study funded by EPRI (Reference 12) and data from original design documents from Westinghouse show that the majority of the structural benefit of the divider plate is derived from the divider plate connection to the channelhead. The divider plate factor for the case where the structural credit for the divider plate to tubesheet connection is negated is DP = 0.64. This means that even in the event that the upper 5.00 inches of the divider plate, stub runner and weld material somehow disappears the tubesheet displacements are still reduced by a minimum of 36%.

The results shown in Tables 7-6 through 7-11 and Tables 7-6a through 7-11a for Byron Unit 2 and Braidwood Unit 2 (for LTR-CDME-05-32-P, Rev. 2) are highly conservative because the DP value that should be used based on the available data on cracking indications in the divider plate is DP = 0.64.

Replacement figures and tables, based on the final models using the stacked worst case and a divider plate factor of 0.64 are provided in Appendix C of this letter report.

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.*

<u>Response to RAI No. 17:</u>

The status of the EPRI program is not related to the B*/H* analysis presented in Reference 19 and any results from the EPRI program will have no impact on the Byron 2/Braidwood Unit 2 B*/H* analysis , which does not take credit for the presence of a divider plate to stub runner weld or the region of the divider plate that has been observed to degrade. The analysis presented in Reference 19 to utilizes a divider plate factor of 1 which assumes that the divider plate is non-existent. Since this is not a credible condition as discussed in the response to RAI No. 16, a divider plate factor DP=0.64 is used that assumes that the divider plate to stub runner weld is completely degraded, but that the divider plate to channel heads welds are intact. This is a valid assumption because these welds are thermally stress relieved, and no degradation has been reported in these welds.

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.*

<u>Response to RAI No. 18:</u>

Please refer also to the response to RAI No. 16.

The B* and H* inspection depths reported in LTR-CDME-07-31 P-Attachment do not take any credit for the presence of the divider plate, either at the tubesheet or in the channelhead. This is reflected in the analysis via the divider plate factor which is set to 1.00 such that no displacements are scaled in the calculation of the contact pressure. The divider plate to stub runner weld is not required to restrict any deflections of the tubesheet in the Byron Unit 2 and Braidwood Unit 2 steam generators to support the B*/H* alternate repair criteria.

As noted in the responses to several RAIs above, the current analysis is based on the use of a divider plate factor of 0.64. This factor reflects the structural effect of a divider plate which is welded to the channelhead but assumes that the weld to the stub runner is entirely missing. Justification for the use of this divider plate factor (DP=0.64) was provided previously.

Therefore, divider plate inspections are not necessary to support the H*/B* criteria for the Byron 2 and Braidwood Unit 2 steam generators because:

- the H*/B* values do not depend on a direct connection between the divider plate and the tubesheet, and the weld between the divider plate, and

- the channelhead to divider plate welds were stress relieved, have shown no evidence of degradation and are not anticipated to crack.

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:*

   e. *Description of test specimens, including sketches.*

   f. *Description of "pre-treatments" of test specimens (hydraulic expansion pressure, heat relief, etc.).*

   g. *Description of test setup, including sketches.*

   h. *Description of test procedure.*

   i. *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?*

   j. *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?*

   k. *What was the temperature of (1) the coolant in the crevice and (2) the tube and tubesheet collar as a function of elevation?*

   l. *How were the temperature distributions for item g determined? Were direct temperature measurements of the tubesheet collar performed as a function of elevation?*

Response to RAI No. 19:

The results of an experimental test program to determine the pressure profile within a steam generator tube-to-tubesheet crevice under leakage conditions are described in *STD-MC-06-11, Rev. 1 "Pressure Profile Measurements During Tube-to-Tubesheet Leakage Tests of Hydraulically Expanded Steam Generator Tubing," August 2007.* This report is provided as Appendix A of this letter report.

For convenience, a "road map" to where subparts a) through h) of RAI No. 19 are located in Appendix A of this report is provided below:

a. Description of test specimens, including sketches.

A description of the test specimens along with sketches and pictures are included on Pages 4-1 through 4-3 of Section 4, "Experimental Description" of Appendix A of this letter report.

b. Description of "pre-treatments" of test specimens (hydraulic expansion pressure, heat relief, etc.).

A description of a portion of the "pre-treatments" of the test are included in Figures 19-1 through 19-6 below. Additional description is provided on Pages A-27 and A-28 in Section 5, "Outline of Test Procedure" of Appendix A of this letter report.

c. Description of test setup, including sketches.

A description of the test facility is provided on Pages A-17 through A-27 in Section 4.2, "Test Facility" of Appendix A of this letter report.

d. Description of test procedure.

A description of the test procedure is provided on Pages A-28 through A-32 in Section 5.4, "Summary of Different Types of Leak Test Procedures" of Appendix A of this letter report. The test matrix is shown in Table 5-3 on Page A-33 in Section 5.5, "Test Matrix" of Appendix A of this letter report.

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?

A summary of the leak rate test procedures is provided on Pages A-28 through A-32 in Section 5.4, "Summary of Different Types of Leak Rate Test Procedures" of Appendix A of this letter report. Only secondary side pressure was monitored. The corresponding temperature can be obtained from the steam tables.

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?

The crevice pressure profile test results are provided on Pages A-34 through A-64 in Section 6, "Results" of Appendix A of this report. A discussion of the leak rate test results is provided on Pages A-73 through A-85 in Section 7, "Discussion" of Appendix A of this letter report.

g. What was the temperature of (1) the coolant in the crevice and (2) the tube and tubesheet collar as a function of elevation?

The temperature of the coolant in the crevice was not measured. Only the inlet temperature was measured. See Section 5.4 of Appendix A of this letter report.

h. How were the temperature distributions for item g determined? Were direct temperature measurements of the tubesheet collar performed as a function of elevation?

No measurements were taken of the temperature of the tubesheet collar. See Section 5.4 of Appendix A of this letter report.

a,c,e

Figure 19-1

a,c,e

Figure 19-2

a.c.e

Figure 19-3

a.c.e

Figure 19-4

a,c,e

Figure 19-5

a,c,e

Figure 19-6

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.*

<u>Response to RAI No. 20:</u>

The abscissa values, the horizontal axis, in Figure 2 of the White Paper are correct. The abscissa values used in the original Figure 3 were incorrect. The corrected Figure 3 is provided below as Figure 20-3.

a,c,e

[ ]

Figure 20-3 Corrected Figure 3 from the White Paper

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?*

Response to RAI No. 21:

Figures 2 and 3 in the White Paper show only the average, or mean, saturation pressure for all of the test results plotted.

The saturation pressures come from measured test data and are not assumptions. The results are presented in Tables 1 and 2 in the White Paper.

The pressure does approach the secondary side pressure near the very top of the specimen (within less than 1 inch from the top) in both the SLB and NOP cases, as shown in Tables 1 and 2 in the White Paper.

The ability of the fluid pressure in the crevice to reduce to the level of the secondary side pressure is limited by the physical constraints of the crevice. The results of the tests suggest that the crevice is too tight for the water to expand in volume and thereby reduce in pressure. This is because the contact pressure between the tube and the simulated tubesheet collar is too great during the simulated operational conditions to allow a path through the crevice that is large enough to allow for the fluid to expand. The contact pressure near the top of the crevice, between the tube and the tubesheet, in the simulated tubesheet collars appears to decrease which allows for a more open crevice. As the crevice opens up, the fluid velocity and volume can increase which allows the fluid to approach the conditions on the simulated secondary side of the tubesheet

An updated version of the White Paper is provided as Appendix B to this letter report.

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.*

Response to RAI No. 22:

The attached White Paper (Appendix B of this letter report) has been revised to correct any typographical errors in the text and the figures.

The data in Table 2 of the White Paper is correct.

A revised Figure 3 is provided below as Figure 22-3.

a,c,e

Figure 22-3 Corrected Figure 3 from the White Paper

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>Response to RAI No. 23:</u>

The objective for incorporating the crevice pressure results into the B* and H* models was to obtain the largest penalty for both the leakage resistance (maximize the B* depth) and the resistance to pullout (maximize the H* depth) assuming a 360° sever in a tube and 100% degradation of the tube material below the severed portion of the tube. The limiting crevice pressure ratio determined using the approach described in the White Paper conservatively accounts for any changes in driving potential and limits the leakage resistance in the crevice too. Further, the limiting crevice pressure ratio approach more realistically models the driving head on the leaked fluid from the tube to the crevice and the resistance of the crevice to leaked fluids at higher tubesheet elevations. See Figure 23-1 and Figure 23-2 for a plot of the difference in pressure between the primary side pressure in the tube and the measured pressure in the simulated tubesheet crevice during NOP and SLB. The zero slope curves in each plot are the result of the limiting median crevice pressure approach described in the White Paper. The varied slope curves in each graph are representative data plotted from the crevice pressure test results. The definition of the crevice pressure ratio is the pressure in the crevice divided by the primary pressure in the system. The definition of the depth ratio is the distance from the bottom of the tubesheet divided by the total tubesheet depth.

a,c,e

Figure 23-1 Plot of the Pressure Difference Between the Primary Pressure in the Crevice Pressure Test Compared to the Measured Pressure in the Crevice During NOP as a Function of Depth Ratio.

a,c,e

Figure 23-2 Plot of the Pressure Difference Between the Primary Pressure in the Crevice Pressure Test Compared to the Measured Pressure in the Crevice During SLB as a Function of Depth Ratio.

Referring to Figure 23-1, the use of the constant, limiting median crevice pressure value (crevice pressure net) during normal operating conditions results in a conservatively lower pressure differential between the primary pressure and the pressure in the crevice (less contact pressure) for the top half of the recommended H* distance (12.34 inches), the limiting hot leg distance. This effect would be expected to cancel out the effect of the depth dependent pressure differential being less than the limiting crevice pressure value (which results in a non-conservative contact pressure) for the lower half of the H* distance. No change in the recommended H* distance is required.

Addressing SLB conditions at the top of the tubesheet, referring to Figure 23-2, because the pressure differential is greater across the tube for most of the H* distance, use of the limiting median crevice pressure value results in a shorter H* distance than would be calculated if a depth dependent pressure differential is used for the SLB condition. However, the most limiting H* distance calculated for the Byron Unit 2 and Braidwood Unit 2 steam generators is based on the NEI 97-06, Rev. 2 structural performance criterion of 3 times normal operating pressure differential. A 0.6 inch margin exists between the H* distance calculated for the normal operating versus SLB condition (12.34 inches versus 11.71[3] inches – see Appendix C of this letter report). The increase in H* distance that results during a postulated SLB if a depth dependent pressure differential across the tubes is used is calculated to be 0.69 inches (12.4 inches). Therefore, no change to the limiting H* distance is required. As no change in H* distance is required and the H* distance bounds the limiting B* distance by 2.3 inches at all radial locations in the Byron Unit 2 and Braidwood Unit 2 steam generators, no change in the B* distances to address the effect of depth dependent crevice pressure is necessary.

Based on the above, the limiting crevice pressure ratio approach results in a conservative H* and B* distance for all plant conditions.

[3] This value is the calculated H* value for SLB conditions, excluding the 0.3 inch uncertainty to cover variability in the location of the BET.

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.

Response to RAI No. 24:

Please refer to the figures and text in the response to RAI No. 23 in addition to the responses below.

The objective for incorporating the crevice pressure results into the B* and H* models was to obtain the largest penalty for both the leakage resistance (maximize the B* depth) and the resistance to pullout (maximize the H* depth) assuming a 360° sever in a tube and 100% degradation of the tube material below the severed portion of the tube. The choice of crevice pressure model affects both the structural calculations for H* and the leakage calculations for B*. The reasons why the limiting crevice pressure ratio approach is conservative are discussed below.

a. The "median value" with respect to the limiting crevice pressure ratio reported in the White Paper is the median value of the sorted data set of all test specimens including only the last two pressure readings after the outliers identified by the Dixon ratio test have been removed from the set. See the response to RAI No. 26 for a sample calculation of the Dixon ratio test and the resulting limiting crevice pressure ratio.

b. The median value approach described in the White Paper is a conservative representation of the behavior at the top of the tubesheet because it captures the significant difference between the NOP and SLB conditions at the top of the tubesheet and it provides the maximum penalty on both the H* and B* distances.

c. The term top of the tubesheet refers to the uppermost elevation of the tubesheet, relative to the primary face. This analysis does not target a specific inspection depth but rather, it defines a very conservative inspection depth that meets all structural and leakage performance requirements.

d. Both the mean and the median values are used to evaluate which application of the constant limiting crevice pressure ratio is the most conservative with respect to B* and H*. The most conservative result occurs when the median crevice pressure value is used.

There are many sources available for detailed discussions of the application of the mean and the median in statistics. The discussion in the paragraph below is paraphrased from a discussion board hosted by Purdue University (http://www.cyto.purdue.edu/hmarchiv/1998/ 0824.htm) and several text books. Similar comments can be found in reliability engineering text books (e.g., Statistics, Probability and Reliability for Civil and Environmental Engineers, McGraw-Hill, © 1997).

The median, or 50th centile, is the value that corresponds to the middle item in a ranked list (e.g, sorted by magnitude) of all recorded measurements in a data set. The median is a robust statistical measure in that it doesn't necessarily change in response to small numbers of outliers, or to skewing of the tails of a distribution, whereas the mean is tugged by both. This is why the median is typically described as a "resistant" measure. One situation where the median is probably the only valid measure is when data congregate at one or both extremes. However, as long as more than 50% of the data are clear of the extremes you get a valid median, but any type of mean (geometric or arithmetic) will be less accurate.

In a smaller data set, it is less likely to obtain a significant portion of outliers, but the presence of outliers can make a drastic change to statistical interpretations of a small data set. The Dixon Ratio test is used to assess the character (i.e., mostly average values, a small number of outliers, entirely composed of outlier values, etc.) of the data set and remove potential outliers that could influence the limiting crevice pressure ratio result. The use of the mean and the median to check the results of the crevice pressure ratio approach is conservative and ensures that the limiting value can be determined regardless of the character of the data set.

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.*

Response to RAI No. 25:

The URL given in the White Paper (Reference 3 in the White Paper) was moved by the academic institution responsible for the website.

References 3 and 4 in the White Paper have been provided by Wolf Creek Nuclear Operating Corporation in its latest submittal to the NRC staff.

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?*

Response to RAI No. 26:

Note: The following response is based on the original White Paper. An updated White Paper is included as Appendix B to this letter report.

The specific data used to calculate the Dixon Ratio values, and the crevice pressure ratio values come from the data presented in Table 1 and Table 2 of the White Paper. The data in Table 1 and Table 2 were then adjusted using one of the crevice pressure model approaches described in the White Paper on page 8 and page 9 of the originally submitted White Paper. The different model approaches are:

a,c,e

See Figure 4 and Figure 5 in the White Paper for plots of the different model results during NOP and SLB conditions (provided below for convenience).

Table 1 a,c,e

Crevice Pressure Specimen Data from Steady State NOP Conditions

[ ]

Table 2 a,c,e

Crevice Pressure Specimen Data from Steady State SLB Conditions

[ ]

a,c,e

[ ]

Figure 4 Plot of Crevice Pressure Model Comparisons Using Average Test Data Results for the Normal Operating Condition. (Ref. original White Paper)

a,c,e

Figure 5 Plot of Crevice Pressure Model Comparisons Using Average Test Data Results for the SLB Accident Condition. (Ref. original White Paper)

Model 1 was determined to be the limiting approach to using the crevice pressure data. The results shown in Table 3 and Table 4 of the original White Paper, were obtained from the following data sets.

Table 26-1
Data Set for Calculating the Dixon Ratio Test NOP Results Using Model 1

a,c,e

Table 26-2
Data Set for Calculating the Dixon Ratio Test NOP Results Using Model 2

a,c,e

Table 26-3
Data Set for Calculating the Dixon Ratio Test NOP Results Using Model 3

a,c,e

Table 26-4
Data Set for Calculating the Dixon Ratio Test SLB Results Using Model 1

a,c,e

Table 26-5
Data Set for Calculating the Dixon Ratio Test SLB Results Using Model 2

a,c,e

Table 26-6
Data Set for Calculating the Dixon Ratio Test SLB Results Using Model 3

a,c,e

Note that the values of the depth ratio for each data set will change based on the assumptions of how the crevice pressure test data is used in the model (See Figure 4 and Figure 5, from the White Paper, above). No confidence limit or scaling on the data are used prior to the Dixon ratio test analysis because lowering the data would produce a less conservative result. Taking the values from the top of the tubesheet, the values associated with a depth ratio of 0.94 and 1.00 in Model 1 and 0.40 and 1.00 in Model 2 and Model 3, and putting them in rank order gives the results in the Table 26-7 and Table 26-8.

Table 26-7
Rank Ordered Data Set for NOP Condition

a,c,e

Table 26-8
Rank Ordered Data Set for SLB Condition

a,c,e

The equation used to calculate the Dixon Ratio test value for a data set changes based on the size of the data population and whether the higher or lower values are suspect. In this context it is desirable to remove the lowest crevice pressure ratio values because the lower the crevice pressure ratio the larger the difference in pressure between the primary and secondary side pressures. The equation for determining the Dixon ratio test value for the NOP case is:

$$\tau = \frac{x_2 - x_1}{x_n - x_1}$$

Where τ is the Dixon ratio test value, x_n refers to the largest value in the data set, x_2 refers to the second lowest value in the data set and x_1 refers to the lowest value in the data set. The equation for determining the Dixon ratio test value for the SLB case is:

$$\tau = \frac{x_3 - x_1}{x_{n-1} - x_1}$$

Where x_3 refers to the third lowest ranked value in the data set and x_{n-1} refers to the second largest value in the data set. Calculating the Dixon ratio test values yield the results shown in Table 26-9 and Table 26-10 below (also shown in the White Paper as Table 3 and Table 4).

Table 26-9
Comparison of Dixon Ratio Test Values for NOP

a,c,e

[ ]

Table 26-10
Comparison of Dixon Ratio Test Values for SLB

a,c,e

[ ]

The results in Table 26-9 and Table 26-10 were used to determine which crevice pressure model was the most limiting with respect to the NOP and SLB conditions. The NOP results indicated that data values in the neighborhood of x_1 ($x \sim 0.35$) were potential outliers. These values were removed from further consideration in all of the models for the NOP data sets as shown in Table 26-7 in the bold type font. The SLB results were more varied than the NOP results, so a conservative limit of twice τ (2τ) was used to determine which values in the data sets could be potential outliers. Applying these criteria to the data in

Model 1 removed the lowest six data points from further consideration (as shown in the bold type font in Table 26-8. The data in Model 2 was determined to be entirely non-conservative because a limit of twice τ suggested that all of the data was a potential low value outlier. Further consideration of Model 2 was performed for the sake of consistency only. Applying the criteria to Model 3 showed that none of the data in the set was a potential outlier. The remaining data sets for Model 1, Model 2 and Model 3 were then used to calculate the average and median values using different assumptions. The summary of the calculations is provided in Table 5 of the White Paper, reprinted below for convenience.

Table 5
Limiting Crevice Pressure Ratios from 3 Models (Ref. White Paper)

a,c,e

[ ]

The "Total Set" results refer to using the entire data set (e.g., all of the data in the column for each model) in Table 26-1 through Table 26-6. The "Included" results refer to using all of the available data (shown in bold and plain type font) in the columns in Table 26-7 and Table 26-8. The "Excluded" results refer to using only the non-bold data in Table 26-7 and Table 26-8. The conclusion of the limiting crevice pressure analysis shown in Table 6 of the White Paper proves that Model 1, Median Outlier Excluded, gives the most conservative and appropriate result for both the NOP and SLB conditions, i.e., the values of both H* and B* are maximized.

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>Response to RAI No. 27:</u>

Please refer to the Response to RAI No. 26 as well.

The terms in Table 5 of the White Paper refer to:

- 1) The mean of the entire data set and the median of the entire data set (Total Set), 2) a skewed mean and median (Included), and 3) a skewed mean and median with potential data outliers removed (Excluded).
- The description "total set" refers to calculating the limiting crevice pressure ratio using the entire set of crevice pressure ratio data (from the bottom of the tubesheet to the top of the tubesheet). This title refers to items 1 and 2 above.

- The description "included" refers to calculating the limiting crevice pressure ratio using the entire set of data from the last two data points nearest the top of the tubesheet. None of the outliers that could potentially skew the limiting crevice pressure result to a lower, and less conservative value, are eliminated from consideration. This title refers to item 2 above.

- The description "excluded" refers to calculating the limiting crevice pressure ratio using the set of data from the top of the tubesheet, in the same fashion as the "included" data set, but with any potential outliers removed to obtain the most conservative result. The outliers are determined using Dixon's ratio test. This title refers to item 3 above.

- The term "skewed" in this context refers to the bias in the data selection, that is, for the Excluded and Included analysis taking only the data from the upper portion of the samples. The term skewed should not be confused with the statistical implication of a skewed distribution.

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>Response to RAI No. 28:</u>

A detailed example of how the values in Table 5 were calculated is provided in the response to RAI No. 26.

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>Response to RAI No. 29:</u>

- The unaltered case is listed in Table 6 of the White Paper, which compares the various B* and H* depths

- The results in Table 6 reflect the crevice pressure results in Table 5 of the White Paper.

- The divider plate factor, and the divider plate, does not influence the results given in Table 5.

- As stated on Page 16 of the White Paper (Appendix B of this report) the "Unaltered" case listed in Table 6 uses a divider plate factor of 0.399. Note that the current recommendation is to use a divider plate factor of 0.64, which conservatively removes the portion of the divider plate and welds that has been shown to exhibit cracking in some foreign steam generators.

APPENDIX A

STD-MC-06-11, Rev. 1

Pressure Profile Measurements During Tube-to-Tubesheet Leakage Tests of

Hydraulically Expanded Steam Generator Tubing

August 30, 2007

Westinghouse

Westinghouse Electric Company
Science and Technology Department
401 Building
1340 Beulah Road
Pittsburgh, Pennsylvania 15235-5082
USA

Nicole Brown
Gary Whiteman

Direct tel: (412) 256-1268
Direct fax: (412) 256-1221
e-mail: jackorj@westinghouse.com

Your ref:
Our ref: STD-MC-06-11-NP, Rev. 1

cc: Herm Legally

August 30, 2007

Nicole and Gary,

Attached is the report summarizing the efforts directed towards leak rate testing of hydraulically expanded steam generator tubes in 2005. The report is titled, "Pressure Profile Measurements during Tube-to-Tubesheet Leakage Tests of Hydraulically Expanded Steam Generator Tubing." This report was revised to identify Westinghouse Electric Company proprietary information.

Call me if you have additional questions.

Regards,

Richard J. Jacko, Manager, Materials Technologies
Materials Center of Excellence
Science and Technology Department

* Electronically Approved Records Are Authenticated in the Electronic Document Management System

Pressure Profile Measurements During Tube-to-Tubesheet Leakage Tests of Hydraulically Expanded Steam Generator Tubing

August 2007

Richard J. Jacko

Westinghouse Electric Co. LLC
Science and Technology Department
Pittsburgh, PA

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

Table of Contents

List of Tables

List of Figures

List of Figures (Cont.)

List of Figures (Cont.)

List of Figures (Cont.)

List of Figures (Cont.)

1 PURPOSE

This report describes the results of an experimental test program to determine the pressure profile within a steam generator tube-to-tubesheet crevice under leakage conditions. For this program, the tube-to-tubesheet joint of tubes were hydraulically expanded into simulated tubesheet collars that model the tubesheets of Westinghouse-designed steam generators. This test program was designed to collect empirical pressure profile and leakage data. The testing was performed in accordance with the test prospectus described in TP-SGDA-04-6 entitled, "Steam Generator Tubesheet Crevice Pressure Profile Testing" [10].

2 REFERENCES

1 WP-4.22, Revision 7, Field Service – Qualification and Functional Testing.
2 WP-4.18, Revision 3, Test Control.
3 WP-11.1, Revision 3, Control of Inspection, Measuring and Test Equipment.
4 WEC 23.11, Revision 0, Science and Technology Department/Nuclear Services, SQS Interface Agreement.
5 ASME Steam Tables, Properties of Steam and Water using the 1967 IPC formulation for Industrial Use, ASME, 1993.
6 Westinghouse Drawing 1B81396, latest revision, "D5 Tube Leakage Test Specimens, Final Assembly, 3", 6", 9" and 12" Hydraulically Expanded Samples".
7 Westinghouse Drawing 1B81387, latest revision, "D5 Tube Leakage Test Specimens 9 inch and 12 inch Hydraulic Expansions and Part Details".
8 Westinghouse Data Package, STD-DP-1997-8015.
9 Westinghouse STD Letter Report STD-MCE-03-49, A. Pearce and M. Nagle, Determination of Model D5 Tube-to-Tubesheet Leakage Resistance for H-Star Program for CBE/CDE/DDP/TCX.
10 TP-SGDA-04-6, "Steam Generator Tubesheet Crevice Pressure Profile Testing", 2/9/2005.
11 "PROPERTIES OF STEAM & WATER using the 1967 IFC FORMULATION FOR INDUSTRIAL USE and other IAPWS releases", 1992, AMERICAN SOCIETY OF MECHANICAL ENGINEERS

3 BACKGROUND

Relatively little work has been done to determine leak rates past a simulated tube-to-tubesheet joint made by hydraulic expansion. Due to the excellent stress corrosion cracking performance of hydraulically expanded steam generator tubing, there has been little driving force to fully characterize the result of through-wall defects. However, changes in mandated inspection practices have required utilities to perform time-intensive inspections to the lower portions of steam generator tubes. Defects in this lower region will not leak substantially, because leakage will be constrained by flow through the hydraulic expansion crevice. One reason for performing leak rate tests on this configuration is to be able to understand and model the processes involved so that realistic predictions of leak rates during normal operation and accident conditions can be calculated. If the calculated rates are appropriately small, the hope is that some of the time-intensive inspections can be reduced.

In 1997, leak rate testing of hydraulically expanded joints was reported in STD-DP-1997-8015 [8]. Another program was performed in 2003 and reported in STD-MCE-03-49 [9] by Pearce. Both tests appeared to raise some questions regarding corrosion in the gap between the expanded steam generator tubing and the simulated steel tubesheet material. The 1997 tests were performed in a small dedicated rig using deaerated water. The 2003 tests were done in an autoclave and used hydrogenated primary water. This autoclave testing applied appropriate primary and secondary side conditions to determine the leak rates across a simulated through-wall crack. Additional efforts were spent to obtain and maintain prototypic oxides on the test specimen surfaces in the testing performed in 2003 and beyond.

Still basic questions remained from these tests regarding the nature of the flow in the hydraulically expanded, tube-to-tubesheet crevice. It was not known whether most of the flow in the crevice was mostly water, mostly steam or a two-phase, water/steam mixture. The goal of this work was to instrument test samples with pressure taps along the steam generator crevice to determine the phase of the fluid at various points along the expansion. This information should assist in the ability to model the phenomena.

4 EXPERIMENTAL DESCRIPTION

4.1 TEST CONCEPT

The goal of this testing was to determine the pressure distribution in a hydraulically expanded, steam generator tube-to-tubesheet crevice during situations where there could be primary-to-secondary leakage through cracks present within the tubesheet crevice. During the testing, primary water flowed through artificial through-wall defects machined into the tube and entered the tight, hydraulically-expanded crevice and eventually flowed to the top of the crevice where it mixed with the simulated secondary-side environment. During this period of time, pressure was monitored at five different locations within the tube-to-tubesheet crevice. Pressure was also monitored at primary-side and secondary-side positions. The main goal of the experiments was to determine where in the crevice the leaking primary water flashed to steam. The volume expansion and flow characteristics of the two-phase water and steam mixtures are much different than those of single phase water. Therefore, an accurate model of steam generator tube leakage can only be determined if the position where the leakage changes from single phase water to two phase (water + steam) is known. This information is needed to model leak rates at these locations during both normal operation and accident conditions.

The test specimens were designed to meet plant conditions for Westinghouse-designed Model D-5 steam generators for Alloy 600 tubes that have simulated through-wall defects per Reference 7. The specimens are simulated steam generator tube-to-tubesheet joints where the tube is hydraulically expanded into a low alloy steel collar that simulates the constraint offered by the steam generator tubesheet. The collar is thick enough to simulate both the thermal and the mechanical properties of the tubesheet in a steam generator. One-eighth inch holes were drilled through-wall at the bottom of the tube to simulate through wall SCC cracks. This was done so that the crack geometry would not limit the pressure drop in the steam generator crevice. These through-wall defects allow pressurized primary water to flow into the annulus between the tube and the simulated tubesheet collar.

Figure 4-1 shows the drawing of the modifications made to the existing leak rate specimens to accommodate the pressure taps. Four pressure taps were located along a given circumferential orientation on the specimen. The fifth tap was located 180° away from the other four pressure taps. Small stainless steel pressure lines, 0.125 inch OD were brazed into the holes leading to the pressure taps at the tube-to-tubesheet joint as shown in Figure 4-2 for specimens numbered 7 and 8 used for these tests. Note that these two specimens had been tested in previous leak rate programs and background data exists.

a,c,e

Figure 4-1 Drawing showing modifications to leak rate specimens 7 and 8 to install pressure taps in the tube-to-tubesheet crevice

a,c,e

Figure 4-2 Photographs of specimens 7 and 8 with pressure tap lines brazed in place

4.2 TEST FACILITY

Figure 4-3 shows a schematic of the details of the testing facility. The facility operated in a slightly different manner depending on what type of leak rate test was being performed. Four different types of leak rate tests were performed during the course of this testing:

- Room Temperature Leak Rate Testing
- Elevated Temperature, Normal Operation Leak Rate Testing – primary water to superheated steam
- Elevated Temperature, Normal Operation Leak Rate Testing – primary water to pressurized water (single phase)
- Elevated Temperature, Accident Condition Leak Rate Testing – primary water to low pressure steam

For the room temperature tests, the test equipment consisted of [

]a,c,e

The equipment used for high temperature tests is shown in Figure 4-4. For these tests, the specimen was contained in a [

]a,c,e

For the normal operation tests, the leakage was calculated based on changes in [

] a,c,e

[

] a,c,e

Figure 4-5 shows the final specimen arrangement for the room temperature tests. [

] a,c,e

a,c,e

Figure 4-3 System used for room temperature tests

a,c,e

Figure 4-4 System used for high temperature tests

a,c,e

Figure 4-5 Final test specimen arrangement for room temperature leak rate tests

a,c,e

Figure 4-6 Initial, unsuccessful test specimen arrangement for room temperature leak rate tests

Figure 4-7 shows an autoclave used for the high temperature tests. [

]a,c,e

Figure 4-8 shows two images of the arrangement for the high temperature tests. [

]a,c,e

a,c,e

Figure 4-7 Image of the test autoclave used for the elevated temperature leak rate tests at STD. [

]a,c,e

a,c,e

Figure 4-8 Test specimen is placed in a carbon steel cylinder that is attached to the autoclave head. [

]a,c,e

5 OUTLINE OF TEST PROCEDURE

The detailed test procedure used for these leak rate tests is listed in Westinghouse Science and Technology Department procedure MCT-147. Some additional details are listed below.

5.1 Specimen Conditioning

The test plan required that the oxides that form on the steel and Alloy 600 surfaces are typical of those that form under reducing PWR steam generator conditions. The test specimens were conditioned in a prototypic PWR environment prior to testing. The test specimens were fabricated using low alloy steels, nickel-based alloy tubing and stainless steel fittings. The corrosion resistances of these materials are quite different in high temperature water.
[

]a,c,e

5.2 Primary Water with Dissolved H_2

Simulated primary-side water was used during the leak rate testing. [

]a,c,e

5.3 Provisions to Minimize Corrosion

Because of the low alloy steel materials used in the specimen fabrication and the PWR steam generator, the specimens will be subject to corrosion when wet at room temperature. Any corrosion films that form on the outside of the specimen probably have no consequence on leak rates measured; however, corrosion products that form in the annulus between the inside of the low alloy steel cylinder and the external tube surface will directly influence the leak rates measured. For this reason, it is necessary to take efforts to minimize the presence of water and moisture

- prior to testing
- between test sequences and
- after testing.

To help minimize the presence of water or water vapor in the annulus between the low alloy steel cylinder and the nickel alloy tube, the tube specimens were [

]a,c,e

5.4 Summary of Different Types of Leak Rate Test Procedures

5.4.1 Room Temperature Leak Rate Testing

For room temperature leak rate testing, the flow configuration was previously described. The specimen is connected to the primary water source and the pressure accumulator. The specimen is [

]a,c,e

5.4.2 Elevated Temperature Normal Operation Leak Rate Testing – Primary Water to Superheated Steam

Previous testing indicated that using the [

]a,c,e

[

]^{a,c,e}

The first step in this calculation process was to [

]^{a,c,e}

For normal operation tests, the leak rate was calculated based on [

]^{a,c,e}

Table 5-1 Density of Superheated Steam at Typical Test Conditions

P	P			Steam Density as f(Temperature, Pressure)							
psig	psia				Density in (grams/cm^3)				Temp. (deg F)		
		575	580	585	590	595	600	605	610	615	620	625
780	***795***	0.02457	0.02434	0.02412	0.02390	0.02369	0.02349	0.02329	0.02310	0.02292	0.02273	0.02256
790	***805***	0.02496	0.02472	0.02449	0.02427	0.02405	0.02385	0.02364	0.02345	0.02326	0.02307	0.02289
800	***815***	0.02534	0.02510	0.02486	0.02464	0.02442	0.02420	0.02400	0.02379	0.02360	0.02341	0.02322
810	***825***	0.02573	0.02548	0.02524	0.02500	0.02478	0.02456	0.02435	0.02414	0.02394	0.02375	0.02356
820	***835***	0.02612	0.02586	0.02562	0.02538	0.02514	0.02492	0.02470	0.02449	0.02429	0.02409	0.02390
830	***845***	0.02651	0.02625	0.02600	0.02575	0.02551	0.02528	0.02506	0.02485	0.02464	0.02443	0.02423
840	***855***	0.02691	0.02664	0.02638	0.02613	0.02588	0.02565	0.02542	0.02520	0.02499	0.02478	0.02457
850	***865***	0.02731	0.02703	0.02676	0.02651	0.02626	0.02602	0.02578	0.02556	0.02534	0.02512	0.02492
860	***875***	0.02771	0.02743	0.02715	0.02689	0.02663	0.02638	0.02615	0.02591	0.02569	0.02547	0.02526
870	***885***	0.02812	0.02782	0.02754	0.02727	0.02701	0.02676	0.02651	0.02627	0.02604	0.02582	0.02561
880	***895***	0.02852	0.02822	0.02794	0.02766	0.02739	0.02713	0.02688	0.02664	0.02640	0.02617	0.02595
890	***905***	0.02894	0.02863	0.02833	0.02805	0.02777	0.02751	0.02725	0.02700	0.02676	0.02653	0.02630
900	***915***	0.02935	0.02904	0.02873	0.02844	0.02816	0.02789	0.02762	0.02737	0.02712	0.02688	0.02665
910	***925***	0.02977	0.02945	0.02913	0.02883	0.02854	0.02827	0.02800	0.02774	0.02749	0.02724	0.02701
920	***935***	0.03019	0.02986	0.02954	0.02923	0.02894	0.02865	0.02837	0.02811	0.02785	0.02760	0.02736
930	***945***	0.03062	0.03028	0.02995	0.02963	0.02933	0.02904	0.02875	0.02848	0.02822	0.02796	0.02772
940	***955***	0.03105	0.03070	0.03036	0.03004	0.02972	0.02943	0.02914	0.02886	0.02859	0.02833	0.02808
950	***965***	0.03148	0.03112	0.03077	0.03044	0.03012	0.02982	0.02952	0.02924	0.02896	0.02869	0.02844
960	***975***	0.03192	0.03155	0.03119	0.03085	0.03053	0.03021	0.02991	0.02962	0.02933	0.02906	0.02880
970	***985***	0.03236	0.03198	0.03161	0.03126	0.03093	0.03061	0.03030	0.03000	0.02971	0.02943	0.02916
980	***995***	0.03280	0.03241	0.03204	0.03168	0.03134	0.03101	0.03069	0.03038	0.03009	0.02980	0.02953
990	***1005***	0.03325	0.03285	0.03247	0.03210	0.03175	0.03141	0.03109	0.03077	0.03047	0.03018	0.02990
1000	***1015***	0.03371	0.03329	0.03290	0.03252	0.03216	0.03182	0.03148	0.03116	0.03085	0.03056	0.03027
1010	***1025***	0.03417	0.03374	0.03334	0.03295	0.03258	0.03222	0.03188	0.03156	0.03124	0.03094	0.03064
1020	***1035***	0.03463	0.03419	0.03378	0.03338	0.03300	0.03264	0.03229	0.03195	0.03163	0.03132	0.03102
1030	***1045***	0.03510	0.03465	0.03422	0.03381	0.03342	0.03305	0.03269	0.03235	0.03202	0.03170	0.03139
1040	***1055***	0.03557	0.03511	0.03467	0.03425	0.03385	0.03347	0.03310	0.03275	0.03241	0.03209	0.03177
1050	***1065***	0.03604	0.03557	0.03512	0.03469	0.03428	0.03389	0.03351	0.03315	0.03281	0.03248	0.03216
1060	***1075***	0.03652	0.03604	0.03557	0.03513	0.03471	0.03431	0.03393	0.03356	0.03321	0.03287	0.03254
1070	***1085***	0.03701	0.03651	0.03603	0.03558	0.03515	0.03474	0.03435	0.03397	0.03361	0.03326	0.03293
1080	***1095***	0.03750	0.03699	0.03650	0.03603	0.03559	0.03517	0.03477	0.03438	0.03401	0.03366	0.03332
1090	***1105***	0.03800	0.03747	0.03697	0.03649	0.03604	0.03561	0.03519	0.03480	0.03442	0.03406	0.03371
1100	***1115***	0.03850	0.03796	0.03744	0.03695	0.03649	0.03604	0.03562	0.03522	0.03483	0.03446	0.03410
1110	***1125***	0.03901	0.03845	0.03792	0.03742	0.03694	0.03649	0.03605	0.03564	0.03524	0.03486	0.03450
1120	***1135***	0.03953	0.03895	0.03840	0.03789	0.03740	0.03693	0.03649	0.03606	0.03566	0.03527	0.03490
1130	***1145***	0.04005	0.03945	0.03889	0.03836	0.03786	0.03738	0.03693	0.03649	0.03608	0.03568	0.03530
1140	***1155***	0.04057	0.03996	0.03938	0.03884	0.03832	0.03783	0.03737	0.03692	0.03650	0.03609	0.03570
1150	***1165***	0.04111	0.04048	0.03988	0.03932	0.03879	0.03829	0.03781	0.03736	0.03693	0.03651	0.03611
1160	***1175***	0.04165	0.04100	0.04039	0.03981	0.03927	0.03875	0.03826	0.03780	0.03735	0.03693	0.03652
1170	***1185***	0.04220	0.04152	0.04090	0.04030	0.03975	0.03922	0.03872	0.03824	0.03779	0.03735	0.03694
1180	***1195***	0.04275	0.04206	0.04141	0.04080	0.04023	0.03969	0.03917	0.03869	0.03822	0.03778	0.03735
1190	***1205***	0.04331	0.04260	0.04193	0.04131	0.04072	0.04016	0.03964	0.03914	0.03866	0.03820	0.03777
1200	***1215***	0.04388	0.04315	0.04246	0.04182	0.04121	0.04064	0.04010	0.03959	0.03910	0.03864	0.03819
1210	***1225***	0.04446	0.04370	0.04299	0.04233	0.04171	0.04112	0.04057	0.04005	0.03955	0.03907	0.03862
1220	***1235***	0.04505	0.04426	0.04353	0.04285	0.04221	0.04161	0.04104	0.04051	0.04000	0.03951	0.03904
1230	***1245***	0.04564	0.04483	0.04408	0.04338	0.04272	0.04211	0.04152	0.04097	0.04045	0.03995	0.03948
1240	***1255***	0.04624	0.04541	0.04463	0.04391	0.04324	0.04260	0.04201	0.04144	0.04090	0.04039	0.03991

Table 5-2 Autoclave Inventory of Superheated Steam as a function of Temperature and Pressure [ ]a,c,e

a,b,c

5.4.3 Elevated Temperature Normal Operation Leak Rate Testing Primary Water to Pressurized Water (single phase)

Leak rates were determined with [

]a,c,e

5.4.4 Elevated Temperature Accident Condition Leak Rate Testing – Primary Water to Low Pressure Steam

A similar procedure was used for analyzing leak rate tests conducted under steam line break conditions. [

]a,c,e

5.5 Test Matrix

The test matrix is shown in Table 5-3 below. NODP refers to normal operation differential pressure. SLB refers to Steam Line Break Accident Conditions. [

]a,c,e

Table 5-3 The Test Matrix

a,c,e

6 RESULTS

Conditioning treatment for specimens 7 and 8 was conducted from 2/17/05 to 2/20/05 in STD autoclave Run A21R263. [

]a,c,e

6.1 SPECIMEN #7

The test results for specimen 7 are presented in the order described in the test prospectus. The results are summarized in Appendix D-1. A starting and ending date and time are listed in Appendix D-1 along with information as specified in the test prospectus. An overall summary of the test conditions and leak rates are shown in Table 6-1 below.

Table 6-1 Summary of Test Conditions and Leak Rates for Specimen #7

a,b,c

[ ]

Note: * denotes that leak rate determined during the last half of the final step.

Graphs showing the pressure, temperatures and leak rates are shown in Figures 6-1 through 6-14.

Specimen 7
Test Condition 1, Room Temperature, Normal Operation and Accident delta P Tests

[

]a,c,e

Specimen 7
Test Condition 2, 600°F, Normal Operation Tests

This test was performed on 11-March-2005 and leak rate was determined by the change in autoclave pressure resulting from the leakage. The test records are shown in Figure 6-3. [

]a,c,e

[

]a,c,e

Specimen 7
Test Condition 3, 600°F, Steam Line Break Test with a Primary Pressure of 2850 psia
This test was performed on 17-March-2005 and the leak rate measured directly by collecting the condensed steam. The target conditions were 600°F, a primary pressure of 2850 psia with atmospheric secondary pressure. [

]a,b,c

Specimen 7
Test Condition 4, 600°F, Steam Line Break Test with a Primary Pressure of 2575 psia
This test was performed on 17-March-2005 and the leak rate measured directly by collecting the condensed steam. The target conditions were 600°F, a primary pressure of 2575 psia with atmospheric secondary pressure. [

]a,b,c

Specimen 7
Test Condition 5, 420°F, Steam Line Break Test with a Primary Pressure of 2850 psia
This test was performed on 18-March-2005 and the leak rate was measured directly by collecting the condensed steam. The target conditions were 420°F and a primary pressure of 2850 psia with atmospheric secondary pressure. The data are presented in Figures 6-6 and 6-7. [

]a,b,c

This test was repeated to try to get conditions to be within the specified temperature range. The second attempt at this test is shown in Figure 6-7. The specimen temperature for this second attempt increased throughout the test. [ ]a,b,c

[

]a,b,c

Specimen 7
Test Condition 6, 420°F, Steam Line Break Test with a Primary Pressure of 2575 psia
This test was also performed on 18-March-2005 and the leak rate was measured directly by collecting the condensed steam. The target conditions were 420°F, a primary pressure of 2575 psia with atmospheric secondary pressure. The data are presented in Figures 6-8. [

]a,b,c

Specimen 7
Test Condition 7, 590°F, Steam Line Break Test with Primary Pressures of 2575 psia (7a), 2850 psia (7b) and 2900 (7c)
These tests were performed on 19-March-2005 and consisted of multiple steps each. The test records are shown in Figures 6-9, 6-10 and 6-11. [

]a,b,c

[

]^{a,b,c}

Specimen 7
Test Condition 8, 420°F, Steam Line Break Test with Primary Pressures of 2575 psia (8a), 2850 psia (8b) and 1800 (8c)

[

]^{a,b,c}

[

]a,b,c

a,b,c

Figure 6-1 Took approximately 30 min for all pressure tap lines to fill with fluid and pressurize. Leakage was observed after ~33 min. Leakage was steady from 33 to 59 minutes. Significant pressure drop within crevice.

a,b,c

Figure 6-2 Pressure taps lines were full from the previous test. Lines were fully pressurized within 4 min. Leakage was steady from 5 to 25 minutes. Significant pressure drop within crevice.

a,b,c

[

]

Figure 6-3 Leak rates were calculated from the change in secondary side water/steam inventory over the pressure range from 695 to 895 psia for this normal operation condition leak rate test. [

]a,b,c

a,b,c

Figure 6-4 Specimen temperature exceeded the specified range after ~ 25 min. Leak rate was steady from 3 to 45 min. [

]a,b,c

a,b,c

Figure 6-5 Leak rate was steady from 5 to 40 min. [ ][a,b,c]

a,b,c

[

]

Figure 6-6 Specimen temperature averaged 466°F, P pri = 2800 psia. Leak rate was steady from 2 to 50 min. [
]a,b,c

a,b,c

Figure 6-7 Specimen temperature changing throughout the experiment. Temperature was in the desired range from [ ]a,b,c

a,b,c

Figure 6-8 Specimen temperature lower than specified for most of the test. [

]a,b,c

a,b,c

Figure 6-9 Specimen temperature decreased during the test. [

]a,b,c

a,b,c

Figure 6-10 Primary pressure was increased each 10 minute period. [

]a,b,c

a,b,c

[

]

Figure 6-11 Primary pressure was increased each 10 minute period and the secondary side was filled with water.
[

]a,b,c

a,b,c

Figure 6-12 Primary pressure was increased each 10 minute period up to 2564 psia. [

]a,b,c

a,b,c

Figure 6-13 Primary pressure was increased each 10 minute period up to 2834 psia. [

]a,b,c

a,b,c

Figure 6-14 This is a water-to-water leak rate test. [

]a,b,c

6.2 SPECIMEN #8

The test results for specimen 8 are presented in the order described in the test prospectus. The results are summarized in Appendix D-2. A starting and ending date and time are listed in Appendix D-2 along with information as specified in the test prospectus. An overall summary of the test conditions and leak rates are shown in Table 6-2 below.

Table 6-2 – Summary of Test Conditions and Leak Rates for Specimen #8

a,b,c

[ ]

Note: * denotes that leak rate determined during the last half of the final step.

Graphs showing the pressure, temperatures and leak rates are shown in Figures 6-15 through 6-28.

Specimen 8
Test Condition 1, Room Temperature, Normal Operation and Accident delta P Tests
The room temperature tests were performed on 30-March-2005, and the results are shown in Figures 6-15 and 6-16. [

]a,b,c

[

]a,b,c

Specimen 8
Test Condition 2, 600°F, Normal Operation Tests
This test was performed on 24-March-2005 and the leak rate was determined by the change in autoclave pressure resulting from the leakage. The test records are shown in Figure 6-17. [

]a,b,c

Specimen 8
Test Condition 3, 600°F, Steam Line Break Test with a Primary Pressure of 2850 psia
Two tests were performed for this condition on 24-March-2005 and the leak rate was measured directly by collecting the condensed steam. The test records are shown in Figure 6-18 and 6-19.
[

]a,b,c

Specimen 8

Test Condition 4, 600°F, Steam Line Break Test with a Primary Pressure of 2575 psia

This test was performed on 24-March-2005 and the leak rate was measured directly by collecting the condensed steam. The target conditions were 600°F and a primary pressure of 2575 psia with atmospheric secondary pressure. The data are presented in Figure 6-20. [

]a,b,c

Specimen 8

Test Condition 5, 420°F, Steam Line Break Test with a Primary Pressure of 2850 psia

This test was performed on 24-March-2005, and the leak rate was measured directly by collecting the condensed steam. The target conditions were 420°F, a primary pressure of 2850 psia with atmospheric secondary pressure. The data are presented in Figures 6-21. [

]a,b,c

Specimen 8

Test Condition 6, 420°F, Steam Line Break Test with a Primary Pressure of 2575 psia

This test was performed on 24-March-2005, and the leak rate was measured directly by collecting the condensed steam. The target conditions were 420°F and a primary pressure of 2575 psia with atmospheric secondary pressure. The data are presented in Figure 6-22. [

]a,b,c

Specimen 8

Test Condition 7, 590°F, Steam Line Break Test with Primary Pressures of 2575 psia (7a), 2850 psia (7b) and 2900 (7c)

These tests were performed on 29-March-2005, and consisted of multiple steps each. The test records are shown in Figures 6-23, 6-24 and 6-25. The secondary pressure was atmospheric for 7a and 7b. [ ]a,b,c

[

]a,b,c

Specimen 8
Test Condition 8, 420°F, Steam Line Break Test with Primary Pressures of 2575 psia (8a), 2850 psia (8b) and 1800 (8c)

These tests were performed on 28-March-2005 and consisted of multiple steps each. The test records are shown in Figures 6-26, 6-27 and 6-28. [

]a,b,c

[

]a,b,c

The final condition (8c) for specimen 8 was a [

]a,b,c

a,b,c

Figure 6-15 Took approximately 9 min for all pressure tap lines to fill with fluid and pressurize. Leakage was observed after ~12 min. Leakage was steady from 12 to 59 min. Significant pressure drop observed within crevice.

a,b,c

Figure 6-16 Pressure taps lines were full from the previous test. Lines were fully pressurized within 2 min. Leakage was steady from 4 to 28 min. Significant pressure drop within crevice.

a,b,c

Figure 6-17 Leak rate was computed by the change in autoclave inventory [

]a,b,c

a,b,c

[

]

Figure 6-18 Specimen temperature below the specified range right after the test started. [

]a,b,c

a,b,c

Figure 6-19 Specimen temperature below the specified range right after the test started. [

]a,b,c

a,b,c

Figure 6-20 Temperature stability problems complicated this test. [

]a,b,c

a,b,c

Figure 6-21 Specimen temperature was below the specified range 382°F. [ ]a,b,c

a,b,c

Figure 6-22 Specimen temperature higher than specified for most of the test. [

]a,b,c

a,b,c

Figure 6-23 Specimen temperature decreased during the test. [

]a,b,c

a,b,c

[

]

Figure 6-24 Primary pressure was increased each 10 min period. [

]a,b,c

a,b,c

Figure 6-25 The secondary side was filled with water. [

]a,b,c

a,b,c

Figure 6-26: Primary pressure was increased each 10 minute period up to 2567 psia. [

][a,b,c]

a,b,c

Figure 6-27 Primary pressure was increased each 10 minute period up to 2834 psia. [

]a,b,c

a,b,c

Figure 6-28 This is a water-to-water leak rate test. [

][a,b,c]

7 DISCUSSION

7.1 Comparison of Leak Test Results on Specimens 7 and 8

A comparison of the leak rates observed on specimens 7 and 8 are shown in Figure 7-1 and 7-2. [

]a,b,c

a,b,c

Figure 7-1 Comparison of Best-Estimate Leak Rates observed on specimens 7 and 8 for test conditions 1 through 6

a,b,c

[

]

Figure 7-2 Comparison of Best-Estimate Leak Rates observed on specimens 7 and 8 for test conditions 7 and 8

Evidence of these subtle differences can also be seen by comparing how the pressure drop was distributed along the crevice. Comparisons were made examining each specimen responded during comparable tests. A simple comparison can be made for the room temperature test 1d for each specimen. The table below shows how the pressure drop was distributed during one of the single phase, room temperature water tests.

Table 7-1 Comparison of the distribution of the total pressure drop observed during room temperature tests

a,b,c

[

]

[

]a,b,c

[

]a,b,c

7.2 Comparison of Leak Test Results 2005 vs. 2003

These same two specimen were previously tested in 2003 and the results were reported by Pearce in Westinghouse Report STD-MCE-03-49. Comparisons were made for specimen 7and 8 in Figures 7-3 and 7-4, respectively. Average temperatures and pressures are reported on the figures to describe small differences in experimental conditions.

a,b,c

Figure 7-3 Comparison of Leak Rates observed on Specimen 7 under similar test conditions

Note: for 2005 test 2a, [

]a,b,c

[

]a,b,c

a,b,c

Figure 7-4 Comparison of Leak Rates observed on specimen 8 under similar test conditions

[

]a,b,c

7.3 Pressure Profiles and Flow in the Tube/Tubesheet Crevice

The main reason for conducting this test program was to determine the pressure profile along the tubesheet crevice of a tube-to-tubesheet joint that was hydraulically expanded into tubesheets of Westinghouse designed steam generators. In addition to collecting this empirical, pressure profile data, an important factor was to determine for elevated temperature tests, where in the crevice did the flow change from primary water to steam.

A complete collection of pressure profiles is shown in Appendix E. [

]a,b,c

a,b,c

Figure 7-5 Final Pressure Profile for Specimen 7, test condition 1-d; single phase test at room temperature

Figure 7-6 shows the final pressure profile room temperature test 7c. [

]a,b,c

a,b,c

[

]

Figure 7-6 Final Pressure Profile for Specimen 7, test condition 7-c; single phase test at 587°F

Figures 7-7 and 7-8 show the pressure profiles for two similar tests on Specimen 8. [

]a,b,c

a,b,c

Figure 7-7 Final Pressure Profile for Specimen 8, test condition 1-d; single phase test at 68°F

a,b,c

Figure 7-8 Final Pressure Profile for Specimen 8, test condition 7-c; [ ]^{a,b,c}

The pressure profiles observed during accident condition tests were [

]a,b,c

For all the examples shown, most of the crevice remains [

]a,b,c

a,b,c

Figure 7-9 Final Pressure Profile for Specimen 7, test condition 3-a; accident test at 624°F

a,b,c

Figure 7-10 Final Pressure Profile for Specimen 7, test condition 5-a; accident test at 467°F

a,b,c

[ ]

Figure 7-11 Final Pressure Profile for Specimen 8, test condition 3-b; accident test at 547°F

a,b,c

[ ]

Figure 7-12 Final Pressure Profile for Specimen 8, test condition 5-a; accident test at 382°F

a,b,c

Figure 7-13 Location where water flashed to steam in the specimen 7 tube-to-tubesheet crevice

a,b,c

Figure 7-14 Location where water flashed to steam in the specimen 8 tube-to-tubesheet crevice

a,b,c

Figure 7-15 This graph shows a comparison of the density of water and steam over temperature range of concern for steam generator operation. Once water flashes to steam in a crevice, the fluid velocity needs to increase to allow for the density change between water and saturated steam.

8 SUMMARY

These data obtained in this test series provided a unique insight into the phenomena associated with leakage through a crevice between a hydraulically expanded steam generator tube and the tubesheet. These results indicate that the fluid in the crevice is [

]a,b,c

a,b.c

Figure 8-1 Distribution of leak rates measured in this test series.

9 CONCLUSIONS

1 The ability to monitor pressure at various locations along the steam generator tube to tubesheet crevice provided key information to understand the factors that affect leak rates under normal operation and accident conditions. A key finding of this test series was that for the majority of the cases considered, the crevice between the steam generator tube and the tubesheet material is [

]a,b,c

2 The flow rates measured in these tests appeared to be modest. Review of the entire hydraulic expansion database may support arguments for obtaining inspection relief for interested utilities. The point would be that even if a through-wall leak existed near the bottom of a hydraulically expanded steam generator tubesheet joint, the potential for measuring significant leakage is not high.

3 Subtle differences in the pressure profiles were observed between the two specimens tested in the resistance to flow along the hydraulic expansion region. [

]a,b,c

4 For the cases where the average test temperatures were less than 500°F which may be typical of the times following a loss-of-coolant accident, [

]a,b,c

10 APPENDICES

APPENDIX A	–	CALIBRATION RECORDS
APPENDIX B	–	TEST CONDITIONS MATRIX
APPENDIX C	–	SPECIMEN FABRICATION RECORDS
APPENDIX D	–	SUMMARY OF LEAK TEST RESULTS
APPENDIX E	–	FINAL PRESSURE PROFILES
APPENDIX F	–	DEVIATIONS AND/OR UNSATISFACTORY RESULTS
APPENDIX G	–	CERTIFICATION OF TEST RESULTS

10.1 APPENDIX A - CALIBRATION RECORDS

a,c,e

10.2 APPENDIX B - TEST CONDITIONS MATRIX

a,c,e

10.3 APPENDIX C - SPECIMEN FABRICATION RECORDS

Welding Sign-off Sheet – Project SAP 113806 Specimen 7

[

]a,b,c

Welding Sign-off Sheet – Project SAP 113806 Specimen 8

[

]a,b,c

10.4 Appendix D – Specimen Results

APPENDIX D-1 SPECIMEN 7 RESULTS

a,b,c

a,b,c

APPENDIX D-1 SPECIMEN 7 RESULTS (Continued)

[redacted]

a,b,c

APPENDIX D-1 SPECIMEN 7 RESULTS (Continued)

a,b,c

APPENDIX D-1 SPECIMEN 7 RESULTS (Continued)

a,b,c

APPENDIX D-2 SPECIMEN 8 RESULTS

a,b,c

APPENDIX D-2 SPECIMEN 8 RESULTS (Continued)

a,b,c

APPENDIX D-2 SPECIMEN 8 RESULTS (Continued)

a,b,c

APPENDIX D-2 SPECIMEN 8 RESULTS (Continued)

10.5 APPENDIX E - FINAL PRESSURE PROFILES

a,b,c

[ ]

a,b,c

[ ]

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

a,b,c

10.6 APPENDIX F - DEVIATIONS AND/OR UNSATISFACTORY RESULTS

DEVIATIONS AND/OR UNSATISFACTORY RESULTS

Sheet 1 of 2

Deviation or Unsatisfactory Results or Comments: total number - 9 a,b,c

[redacted]

Resolutions/Necessary Actions: number of actions 9 a,b,c

[redacted]

DEVIATIONS AND/OR UNSATISFACTORY RESULTS

Sheet 2 of 2

Deviation or Unsatisfactory Results or Comments: total number - 9 a,b,c

Resolutions/Necessary Actions: number of actions 9 a,b,c

10.6 APPENDIX G - CERTIFICATION OF TEST RESULTS

1.0 This test has been performed in accordance with test procedure ***RJ Jacko***

2.0 Computer software application and version Not applicable

Equipment drawing number and revision SK1B81517

3.0 Test data and results have been reviewed against the requirements and acceptance criteria of the test procedure and are accepted.

4.0 Any deviations to this procedure and/or any unsatisfactory results are resolved and documented.

5.0 List any actions not affecting equipment/process acceptability:

N/A

6.0 This test completes all testing required by the following Design Review Action Items:

Test Data Package Reviewers:

Project Manager	Richard J. Jacko	Date	5-Jul-2006
System Eng.		Date	
Quality		Date	

7.0 The results of this test, including any deviations and unsatisfactory results with their resolutions, have been reviewed and are acceptable. The equipment/process/computer software are hereby deemed acceptable for field service application.

* Cognizant Managers:

Product		Date	
Engineering		Date	
Quality		Date	

APPENDIX G - CERTIFICATE OF TEST COMPLETION (Continued)

EQUIPMENT/PROCESS: ____

8.0 The test acceptance criteria have been partially satisfied. Limited field use is certified as follows:

____

____

____

The functional specification has been revised consistent with the above limitations.

Westinghouse

STD Final Report Verification Checklist

Note: If completing form electronically, highlight the checkbox and type an "x" to fill in the checkbox.

Document Title: Pressure Profile Measurements During Tube-to-Tubesheet Leakage Tests of Hydraulically Expanded Steam Generator Tubing		
STD Letter Number: STD-MC-06-11	**Date 7/05/06**	
Test Prospectus Verifier: L. A Nelson (NS)	**Revision 0**	
Checklist	**Yes**	**No**
1.0 Is the approved test prospectus provided?	☒	☐
2.0 Are the STD procedures identified?	☒	☐
3.0 Were the calculations correctly performed?	☒	☐
4.0 Were data properly transcribed?	☒	☐
5.0 Were test samples correctly identified with respect to lot number, fabrication history, etc.?	☒	☐
6.0 Were test deviations identified?	☒	☐
7.0 If yes to 6.0, were there any possible effects to the test results? If so, please list	☒	☐
Verifier Statement – List steps and techniques used in verification		
Reviewed final report for accuracy and completeness		
Checked calculations using steam tables		
Verified conclusions		
Verification Signature Rachel L DeVito	**Date 7/05/06**	

APPENDIX B

LTR-SGDA-07-4-P, Rev. 3

Letter Summary of Changes to B* and H* Analysis Due to

New Crevice Pressure and Divider Plate Data

September 24, 2007

Westinghouse

To: H. O. Lagally

Date: September 24, 2007

cc: P. R. Nelson, E. P. Morgan, G. W. Whiteman, B. A. Bell, J. G. Thakkar, W. K. Cullen

From: C. D. Cassino
Ext: 724 722-6018
Fax: 724 722-5889

Your ref:
Our ref: LTR-SGDA-07-4-NP, Rev. 3

Subject: **Letter Summary of Changes to B* and H* Analysis due to New Crevice Pressure and Divider Plate Data**

The technical basis for H* and B* as documented in the Alternate Repair Criteria (ARC) WCAPs and Calc Notes (see Reference 1 for an example) is based, in part, on of the fundamental assumption that leakage through a postulated crack below H* flashes to steam in the crevice. This establishes the pressure in the crevice as the saturation pressure. Test data show that leakage through a crack below H* does not flash to steam and remains a single-phase fluid; therefore, the original assumption is not justified and changes must be made to the B* and H* analysis inputs to reflect the new test results.

The purpose of the test was to determine the pressure in the crevice between the tube and the tubesheet. The tests show that there is a distribution of pressure in the tubesheet crevice that is [ ]^{a,c,e} The results showed that the fluid in the crevice remained single phase to very near the top (secondary side face) of the [long test specimens. Therefore, the crevice pressure is [ ]^{a,c,e} An increased pressure in the crevice will result in:

1. The driving potential on the leaked fluid from the primary side to the crevice has been reduced.

2. The driving potential on the leaked fluid from the crevice to the secondary side is increased at the bottom of the tubesheet and decreased at the top of the tubesheet.

3. The resistance to flow from viscous effects has increased.

4. The tube expansion component of the contact pressure analysis has been reduced.

5. The tube expansion component of the leakage resistance analysis has been reduced.

A discussion of the impact of the test results documented in Reference 1 on H*/B* analyses from a generic perspective is provided. The effects of varying the divider plate factor on a generic H*/B* analysis are also discussed. Note that the flaw in the test specimens discussed in this document was specifically [ ]a,c,e. It is possible to maintain a large pressure drop across the tube wall in smaller crack geometries. [ ]a,c,e Therefore, the results described in this letter are bounding for the worst case scenario [ ]a,c,e exists in the tube portion within the tubesheet.

If there are any questions regarding the contents of this letter please contact either Chris Cassino or Herm Lagally.

Author:

C.D. Cassino

Chemistry, Diagnostics and Materials Engineering

Reviewer:

H.O. Lagally

Chemistry, Diagnostics and Materials Engineering

References

1. STD-MC-06-11, Rev.1.

2. LTR-CDME-05-32-P, Rev. 2.

3. http://www.cee.vt.edu/ewr/environmental/teach/smprimer/outlier/outlier.html, 07/01/2007, 11:34:07 AM EST.

4. M.R. Chernick, "A Note on the Robustness of Dixon's Ratio Test in Small Samples", *American Statistician*, Vol. 36, No. 2 (May, 1982), p. 140.

5. W.B. Middlebrooks, D.L. Harrod, R.E. Gold, "Residual Stresses Associated with the Hydraulic Expansion of Steam Generator Tubing into Tubesheets," Nuclear Engineering and Design 143 (1993) 159-169 North-Holland.

6. LTR-SGDA-06-156.

7. LTR-SGDA-06-157.

8. CN-SGDA-07-6.

9. Terakawa, T., Imai, A., Yagi, Kazushige, Fukada, Y., Okada, K., "Stiffening Effects of Tubes in Heat Exchanger Tube Sheet", Journal of Pressure Vessel Technology Transactions, ASME Vol. 106, No.3, August 1984.
10. LTR-SGDA-06-160.

11. LTR-SGDA-07-3.

12. TP-SGDA-03-2, Rev.1.

13. *Divider Plate Cracking in Steam Generators: Results of Phase 1: Analysis of Primary Water Stress Corrosion Cracking and Mechanical Fatigue in the Alloy 600 Stub Runner to Divider Plate Weld Material.* EPRI, Palo Alto, CA: 2007. 1014982.

14. Stress Report: 51 Series Steam Generator Calculated and Measured Strains and Deflections for Steam Generator Tubesheet Channel Head Model Under Limit Conditions, Volume 1, MPR Associates, November 1969.

15. WCAP-16820-P.

16. WCAP-16053-P.

17. LTR-SGDA-07-201.

1.0 Discussion of Crevice Pressure Test Results

The tests documented in Reference 1 were performed to determine the pressure distribution in the crevice of a hydraulically expanded tubesheet region with a postulated through wall flaw near the bottom of the expansion. [

]a,c,e The data from the NOP and SLB tests from both specimens [1], taken after the pressure in the crevice reached steady state conditions, are shown in Table 1and Table 2 below.

Table 1 Crevice Pressure Specimen Data from Steady State NOP Conditions

a,c,e

Table 2 Crevice Pressure Specimen Data from Steady State SLB Conditions

a,c,e

a,c,e

Figure 1 Picture of Typical Test Specimens Used in Crevice Pressure Experiments.

[

]a,c,e

a,c,e

[redacted]

Figure 2 Plot of Crevice Pressure Ratio as a Function of Depth Ratio into the Test Specimen for Simulated NOP conditions.

a,c,e

Figure 3 Plot of Crevice Pressure Ratio as a Function of Depth Ratio into the Test Specimen for Simulated SLB conditions.

[

]a,c,e

2.0 Discussion of Increased Crevice Pressures relative to an H*/B* Analysis

[

]a,c,e

1. [
   ]a,c,e
2. [
   ]a,c,e
3. [ ]a,c,e
4. [
   ]a,c,e
5. [
   ]a,c,e

[

]a,c,e

The contact pressure between the tube wall and the tubesheet hole is calculated in the H* and B* analysis for two reasons:

1. [ ]a,c,e
2. [
   ]a,c,e

The components that contribute to the contact pressure between the tube material and the tubesheet crevice are:

- [
  ]a,c,e

- [ ]a,c,e
- [ ]a,c,e
- [ ]a,c,e

Of these, only the contribution of the [

]a,c,e The contribution of the thermal growth in each material will not be affected by an [ ]a,c,e Similarly, [

]a,c,e

The unrestrained radial expansion of a tube OD due to a pressure differential across the tube wall is

a,c,e

[ ]

In the original analysis, P_o was assumed to be equal to the secondary side pressure. If the value of P_o is increased, the value will ΔR_{to}^{pr} decrease. For example, [

]a,c,e

An interesting observation from Figures 2 and 3 is [

]a,c,e

[

]a,c,e

3.0 Calculation of the Limiting Crevice Pressure Ratio

[

]a,c,e

[

]

Figure 4 Plot of Crevice Pressure Model Comparisons using average test data results for the normal operating condition.

[ ]a,c,e

Figure 5 Plot of Crevice Pressure Model Comparisons using average test data results for the SLB accident condition.

[

]a,c,e

There are many sources available for detailed discussions of the application of the mean and the median in statistics. The discussion in the paragraph below is paraphrased from a discussion board hosted by Purdue University (http://www.cyto.purdue.edu/hmarchiv/1998/0824.htm) and several text books. Similar comments can be found in reliability engineering text books (e.g. Statistics, Probability and Reliability for Civil and Environmental Engineers, McGraw-Hill, © 1997).

The median, or 50th centile, is the value that corresponds to the middle item in a ranked list (e.g., sorted by magnitude) of all recorded measurements in a data set. The median is a robust statistical measure in that it doesn't necessarily change in response to small numbers of outliers, or to skewing of the tails of a distribution, whereas the mean is tugged by both. This is why the median is typically described as a "resistant" measure. One situation where the median is perhaps the only valid measure is when data congregate at one or both extremes. However, as long as more than 50% of the data are clear of the extremes a valid median is obtained, but any type of mean (geometric or arithmetic) will be less accurate.

A commonly used statistical tool to determine outliers in a limited population of data is the Dixon Ratio test. The Dixon Ratio test is used to assess the character (i.e., mostly average values, a small number of outliers, entirely composed of outlier values, etc.) of the data set and limit the inlfuence of potential outliers that could affect the limiting crevice pressure ratio result. The following text is adapted from the tutorial on the detection and accommodation of outliers from the web library of Virginia Polytechnic Institute and State University department of Civil and Environmental Engineering [3]. Dixon's test is generally used for detecting a small number of outliers. This test can be used when the sample size is between 3 and 25 observations [4], but is typically employed whenever a sample set is less than an ideal population to apply standard statistical tools. In a smaller data set, it is less likely to obtain a significant portion of outliers, but the presence of outliers can make a drastic change to statistical interpretations of a small data set. The data is ranked in ascending order and then sorted on the sample size. The τ statistic for the highest value or lowest value is computed. [

]a,c,e The chart below gives a list of how to calculate the appropriate Dixon Ratio values.

Observations	Highest value suspect	Lowest value suspect
3 to 7	$\tau = \frac{x_n - x_{n-1}}{x_n - x_1}$	$\tau = \frac{x_2 - x_1}{x_n - x_1}$
8 to 10	$\tau = \frac{x_n - x_{n-1}}{x_n - x_2}$	$\tau = \frac{x_2 - x_1}{x_{n-1} - x_1}$
11 to 13	$\tau = \frac{x_n - x_{n-2}}{x_n - x_2}$	$\tau = \frac{x_3 - x_1}{x_{n-1} - x_1}$
14 to 20-30	$\tau = \frac{x_n - x_{n-2}}{x_n - x_3}$	$\tau = \frac{x_3 - x_1}{x_{n-2} - x_1}$

The τ statistic is compared to a critical value at a chosen value of α. If the τ statistic is less than the critical value, the null hypothesis is not rejected, and the conclusion is that no outliers are present. If the τ statistic is greater than the critical value, the null hypothesis is rejected, and the conclusion is that the most extreme value is an outlier. To check for other outliers, the Dixon test can be repeated, however, the power of this test decreases as the number of repetitions increases. .]a,c,e

[

]a,c,e

Table 3 Data Set for Calculating the Dixon Ratio Test NOP results using Model 1 a,c,e

[ ]

Table 4 Data Set for Calculating the Dixon Ratio Test NOP results using Model 2 a,c,e

[ ]

Table 5 Data Set for Calculating the Dixon Ratio Test NOP results using Model 3 a,c,e

[ ]

Table 6 Data Set for Calculating the Dixon Ratio Test SLB results using Model 1 a,c,e

Table 7 Data Set for Calculating the Dixon Ratio Test SLB results using Model 2 a,c,e

Table 8 Data Set for Calculating the Dixon Ratio Test SLB results using Model 3 a,c,e

[

]a,c,e

Table 9 Rank Ordered Data Set for NOP Condition a,c,e

[ ]

Table 10 Rank Ordered Data Set for SLB Condition a,c,e

[ ]

The equation used to calculate the Dixon Ratio test value for a data set changes based on the size of the data population and whether the higher or lower values are suspect. [

]a,c,e The equation for determining the Dixon ratio test value for the NOP case is: a,c,e

[ ]

Where τ is the Dixon ratio test value, x_n refers to the largest value in the data set, x_2 refers to the second lowest value in the data set and x_1 refers to the lowest value in the data set. The equation for determining the Dixon ratio test value for the SLB case is: a,c,e

[ ]

Where x_3 refers to the third lowest ranked value in the data set and x_{n-1} refers to the second largest value in the data set. Calculating the Dixon ratio test values yield the results shown in Table 11 and Table 12 below.

Table 11 Comparison of Dixon Ratio test values for NOP

a,c,e

[ ]

Table 12 Comparison of Dixon Ratio test values for SLB

a,c,e

[ ]

[

]a,c,e The summary of the calculations is provided in Table 13. The effect of the limiting crevice pressure ratio on the H* and B* inspection distances are evaluated in section 6.0.

4.0 The Crevice Pressure Effects on the Loss Coefficient Data

The effective contact pressure between the tube and the tubesheet is a function of four phenomena:

1. thermal growth/mismatch between the tube and the tubesheet,
2. tubesheet displacement resulting in hole dilation,
3. tube expansion due to the pressure differential, and
4. residual mechanical joint strength due to the tube expansion process during installation.

[

]a,c,e

See Appendix C, response to RAI No. 11, in Reference 15 for a discussion of the results of incorporating different crevice pressure assumptions into the loss coefficient versus contact pressure regression analysis. See Reference 16 for the development of the theory of elasticity model used to calculate the contact pressure associated with a primary to secondary pressure differential during a leak rate experiment. See Reference 1 for the data used to calculate the crevice pressure ratios used in the contact pressure analysis.

[

]a,c,e

The crevice pressure ratios used with each applied ΔP are summarized in Table 13. The results of the varied crevice pressure with applied ΔP are summarized in Table 14. The spreadsheets and calculation tools used in the analysis are captured in the attachments to Reference 17.

Table 13 Crevice Pressure Ratio Summary for Leak Rate Analysis

a,c,e

Table 14 Summary of Contact Pressure Results for Loss Coefficient Analysis

SG Type	Primary Side Test Pressure (PSI)	Test Temperature (°F)	Contact Pressure (PSI)	Crevice Pressure Differential (PSI)

a,c,e

5.0 The Effect of the Divider Plate Factor on B* and H* Analysis

Indications of cracks in the divider plates have been reported in several steam generators located in France. These indications have been observed in steam generators located at the Chinon, Saint-Laurent, Blayais, Dampierre and Gravelines nuclear power stations. The cracks were observed on the hot leg side of the divider plate in the stub runner divider plate weld, stub runner base metal and also at or in the divider plate itself. See Figure 6 for a sketch of the region where cracking has been observed to occur.

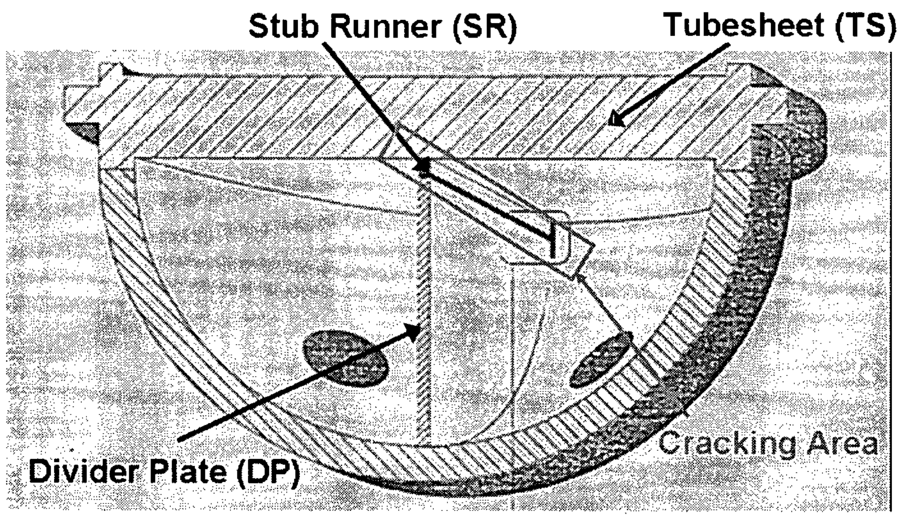

Figure 6 Sketch of Divider Plate, Channel Head and Tubesheet with potential cracking areas highlighted.

The network of cracks has been reported to extend along most of the divider plate (~6 feet) and have also been reported to be relatively shallow with depth, typically less than 2 mm (~75 mils deep).

The French utilities inspected this location to determine if any indications of cracking could be found during a visual inspection because these steam generators used an Alloy 600 material in the divider plate to stub runner weld. During the initial visual inspection it was reported that indications of cracks were observed but that they appeared to be shallow in depth. Various other methods were used in subsequent refueling outages to determine the extent of cracking and to determine the crack growth rate. Available information indicates that these inspections have been performed since 1993 using a combination of liquid penetrant examination (PT) and visual examination (VT) methods with indications of cracking observed in some of these plants. Through the winter of 2005, a total of thirty five inspections using VT and PT were performed in the French 900 megawatt (MW) and 1300 MW units with indications of cracking being found in at least four of the plants as noted above.

Primary water stress corrosion cracking (PWSCC) is a know mechanism of cracking in Alloy 600 and it is likely this is the primary contributor to cracking at this location. However, other potential contributors to cracking have been reported to be defects in the weld or base material, along with deformations associated with loose part impingement and these may be contributing factors. See Reference 13.

The maximum depth of the majority of the cracks observed in the French units has been reported to be about 2 mm (~75 mils). The maximum crack depth indication that has been observed is 7 mm (~0.28 inch) however this indication is the likely result of loose part damage on the hot leg side of the divider plate in the affected generator. Various inspection methods (VT, PT, and then UT) have been used in plants with indications of divider plate cracking. It has been reported that consecutive inspections using identical methods have not been performed to date; therefore, it is not possible to develop an accurate growth rate from the French inspection data. From the available information it can be inferred that the cycle-to-cycle growth rate of the cracks is small based on the following: The difficulty in obtaining an accurate measure of the depth of the crack due to the shallowness of the crack (smaller cracks are harder to detect than larger cracks), the continued reports of finding only shallow depth cracks, and the relatively long period of time that these cracks have been known to exist.

The majority of the cracks included by the French experience are small with a relatively small cycle-to-cycle growth rate; therefore, the effect on the divider plate function is also expected to be small. It would be expected that cracks of the size reported would not affect the general displacement response of the tubesheet since only a very small change in divider plate stiffness would be expected. In addition, it would not be expected that cracks of the size reported would rapidly grow due to mechanically induced loadings resulting from normal/upset events or during a faulted event. However, there may be a potential for long term growth of these cracks which could eventually affect tubesheet displacements and result in an increased rate of crack propagation. See Reference 13 for a conservative analysis estimate of crack growth in the divider plate.

Tubesheet displacements can directly affect multiple regions in the SG that include such areas as:

a. Stress in the tubesheet/shell and tubesheet/channelhead connections

b. Tube stresses and field repairs

c. Plug retention/acceptability issues.

The divider plate is accounted for in B* and H* analyses via a divider plate factor, which is the ratio of the maximum vertical tubesheet displacements with an intact divider plate compared to the maximum vertical displacements of a tubesheet with no divider plate present. The factor is based on the ASME stress report provided for the SGs, which considered both to conservatively calculate stresses in the tubesheet and in the components attached to the tubesheet. The ratio of the maximum tubesheet displacement with and without the benefit of the divider plate is [ ]^{a,c,e}, which means that the maximum vertical displacement of the tubesheet with an intact divider plate is [ ]^{a,c,e} less than the maximum vertical displacement of a tubesheet without a divider plate based on the ASME Code Stress Report for the SGs. This value [ ]^{a,c,e} is used for the divider plate factor in the B* and H* analyses prior to 2007. A value of [ ]^{a,c,e} for the divider plate factor is used in the H* and B* analyses to evaluate the condition where the divider plate does not restrain the vertical tubesheet displacements of the tubesheet.

[

]^{a,c,e}

[

]^{a,c,e}

by an order of magnitude. There have been no cracking indications observed in the divider plate to channelhead welds. This is likely because the divider plate to channelhead connections are heat treated prior to the divider plate to stub runner weld being made. Therefore, it is possible to conclude that the majority of the stiffening effect of the divider plate comes from the divider plate to channelhead connection, not the divider plate to tubesheet connection, and that this connection is not subject to the same cracking risks as the stub runner weld.

The change in divider plate factor for the case of only the channelhead to divider plate connections being intact is equal to [

]a,c,e

The effect of a reduced divider plate factor with a non-degraded divider plate will [

]a,c,e

To evaluate the effect of a degraded divider plate, a bounding analysis was performed which assumed that the divider plate provides [

]a,c,e

Evaluation of divider plate degradation is continuing under EPRI sponsorship. The effects of long term operation with postulated larger cracks in the divider plate must be evaluated to determine if the cracks could grow to a point where either rapid crack growth could occur during operation of the SG or if increased tubesheet displacements could affect other aspects of the steam generator, such as tubesheet stress, secondary side shell stress, channel head stress, tube stress, plug retention/acceptability issues and the ARCs [6, 13].

The following conclusions are reached based on the current evaluation of divider plate degradation:

1. The original divider plate factor from the ASME Code stress report, the ratio of the maximum tubesheet displacement assuming a fully effective divider plate to that assuming no contribution from the divider plate, is [ ]a,c,e

2. Based on a more detailed finite element model of the tubesheet/divider plate assembly, the revised divider plate factor is [ ]^{a,c,e}

3. The divider plate factor obtained by comparing the displacements of the tubesheet with and without the contribution of the potentially cracked region of the divider plate is [ ]^{a,c,e}

4. The preliminary conservative estimate of H* and B* assuming no structural contribution from the cracked region of the divider plate is bounded by [ ]^{a,c,e}

5. The presence or absence of the cracked region of the divider plate does not impact a 17 inch inspection depth, since sufficient margin exists between [ ]^{a,c,e} The structural model used for this assessment is the refined finite element model of the tubesheet/divider plate assembly.

6.0 Results from Implementing Changes in H* and B* Analysis

Table 15 below summarizes the limiting crevice pressure ratios [

]a,c,e

Table 15 Limiting Crevice Pressure Ratios from 3 Models

a,c,e

From Section 1, the pressure ratio [

]a,c,e

Therefore, the smallest pressure drop [

]a,c,e

Table 16 H* and B* Prediction for Different Models of Crevice Pressure

(Data based on improved tubesheet/divider plate structural model)

a,c,e

[ ]

(H* and B* are referenced to the bottom of the expansion transition)

The results [

]a,c,e

The following figures show [

]a,c,e

Figure 7 shows [

]a,c,e

[ ]a,c,e

Figure 7 Unaltered Data and Methods for B* and H*. Crevice Pressure = $P_{Pri} - P_{Sec}$, DP = 0.76.

[ ]a,c,e

a,c,e

[ ]

Figure 8 Updated Input Data and Methods for B* and H*. Crevice Pressure = $CP*P_{Pri}$, DP = 0.399.

The results for the updated analysis input with a divider plate factor of [ ][a,c,e] (i.e., no structural restraint provided by the potentially cracked region of the divider plate) are shown in Figure 9

a,c,e

Figure 9 Updated Input Data and Methods for B* and H*. Crevice Pressure = CP^*P_{Pri}, DP = 0.64.

Comparing the results shown in Figure 7 and Figure 9 proves that the changes in the B* and H* inputs due to the increased crevice pressure and divider plate effects are reasonable and follow similar trends compared to the prior results. The results shown in Figure 9 prove that the generic bounding analysis conditions (using mean ASME material properties and design inputs) in the event that the divider plate to stub runner is fully degraded are still below the previously reported bounding value of 12.50 inches. [

][a,c,e]

The choice of crevice pressure model maximizes the B* and H* depths by minimizing the structural and leakage resistance of the tube to tubesheet crevice joint. An alternative method to using a limiting constant crevice pressure ratio is to use a depth based approach. That is, to vary the crevice pressure ratio at each depth based on the available test data so that the pressure difference across the tube varies as a function of tubesheet elevation. The depth based crevice pressure approach [

]a,c,e

1. [

]a,c,e

2. [

]a,c,e

3. [

]a,c,e

The constant crevice pressure approach does yield different results from the depth based approach during accident conditions. [

]a,c,e

In conclusion, [

]a,c,e

7.0 Summary and Conclusions

The following summarizes this "White Paper" regarding the effects of new test data and updated analysis methods on the H*/B* technical justifications:

1. Recently obtained test data indicate that postulated leakage through a tube crack in the tubesheet expansion region [ ]^{a,c,e}
2. Updated finite element analysis of the tubesheet/divider plate assembly shows that the ratio of the maximum deflection of the tubesheet with an un-degraded divider plate to the maximum deflection with no structural restraint from the divider plate is much [ ]^{a,c,e} than the factor derived from the original ASME Code Stress Report.
3. Analysis using the updated divider plate factor shows that the bounding value for H*/B* is about [ ]^{a,c,e} (using mean ASME code material properties and design inputs). Only the "true" B* value will be affected if the divider plate is assumed to be non-functional. Significant margin exists for [ ]^{a,c,e} inch inspection depth.
4. Several models were developed to represent the new crevice pressure test data. The most conservative model, that minimizes the pressure drop from the primary side to the crevice, was identified.
5. Integrated analysis accounting for both the divider plate degradation and revised crevice pressure show that the justification for H* and B* is still valid when the most conservative crevice pressure model and the refined structural model for the tubesheet/divider plate assembly are used.

APPENDIX C

Hot and Cold Leg H* and B* Results

November 2007

"The required revisions to the tables and figures previously provided to the NRC staff in Reference 19 follow. The H* and B* results included in Appendix C are based on Reference 20."

HOT LEG RESULTS

Table 7-6. Tube/Tubesheet Maximum & Minimum Contact Pressures & H* for Byron/Braidwood Unit 2 Steam Generators

a,c,e

Table 7-7. Cumulative Forces Resisting Pull Out from the Top of the Tubesheet

Byron/Braidwood 2 – Hot Leg Normal Conditions

Low T_{ave}, High T_{sec}

a,c,e

Table 7-8. Cumulative Forces Resisting Pull Out from the TTS

Byron/Braidwood 2 – Hot Leg Normal Conditions

High T_{ave}, Low T_{sec}

a,c,e

Table 7-9. Cumulative Forces Resisting Pull Out from the TTS Byron/Braidwood 2–Faulted (SLB) Conditions

a,c,e

Table 7-10. Cumulative Forces Resisting Pull Out from the TTS Byron/Braidwood 2–FLB Conditions Low T_{ave}, High T_{sec}

a,c,e

Table 7-11. Cumulative Forces Resisting Pull Out for FLB Conditions High T_{ave}, Low T_{sec}

a,c,e

Table 7-13. Summary of H* Calculations for Byron/Braidwood Unit 2

a,c,e

Table 7-14. H* Summary Table

Zone	Limiting Loading Condition	Engagement from TTS (inches)
A	1.4 SLB ΔP	6.51
B	1.4 SLB ΔP	11.21
C	3 Times NO ΔP	12.34

a,c,e

Figure 7-3. Contact Pressures for Normal Condition (T_{min}) at Byron/Braidwood 2
DP Factor = 0.64

a,c,e

Figure 7-4. Contact Pressures for Normal Condition (T_{max}) at Byron and Braidwood Unit 2
DP Factor = 0.64

a,c,e

Figure 7-5. Contact Pressures for SLB Faulted Condition at Byron and Braidwood 2
DP Factor = 0.64

a,c,e

Figure 7-6. Contact Pressures for FLB Faulted Condition at Byron and Braidwood 2 (Tmin), DP Factor = 0.64

a,c,e

Figure 7-7. Contact Pressures for FLB Faulted Condition at Byron and Braidwood 2 (Tmax), DP Factor = 0.64

a,c,e

Figure 8-1. Change in Contact Pressure at 10.5 Inches Below the TTS, DP Factor = 0.64

a,c,e

Figure 8-2. Change in Contact Pressure at 12.6 Inches Below the TTS, DP Factor = 0.64

a,c,e

Figure 8-3. Change in Contact Pressure at 16.9 Inches Below the TTS, DP Factor = 0.64

a,c,e

Figure 8-4. Change in Contact Pressure at the Bottom of the Tubesheet, DP Factor = 0.64

a,c,e

Figure 8-5. Change in Contact Pressure at 8.25 Inches Below the TTS, DP Factor = 0.64

COLD LEG RESULTS

Table 7-6a. Tube/Tubesheet Maximum & Minimum Contact Pressures & H* for Byron/Braidwood Unit 2 Steam Generators (Cold Leg)

a,c,e

Table 7-7a. Cumulative Forces Resisting Pull Out from the Top of the Tubesheet

Byron/Braidwood 2 – Cold Leg Normal Conditions

Low T_{ave}, High T_{sec}

a,c,e

Table 7-8a. Cumulative Forces Resisting Pull Out from the TTS

Byron/Braidwood 2 – Cold Leg Normal Conditions

High T_{ave}, Low T_{sec}

a,c,e

Table 7-9a. Cumulative Forces Resisting Pull Out from the TTS Byron/Braidwood 2–Faulted (SLB) Conditions

a,c,e

Table 7-10a. Cumulative Forces Resisting Pull Out from the TTS Byron/Braidwood 2–FLB Conditions Low T_{ave}, High T_{sec} (Cold Leg)

a,c,e

Table 7-11a. Cumulative Forces Resisting Pull Out for FLB Conditions

High T_{ave}, Low T_{sec} (Cold Leg)

a,c,e

Table 7-13a. Summary of H* Calculations for Byron/Braidwood Unit 2 (Cold Leg)

a,c,e

Table 7-14a. H* Summary Table (Cold Leg)

Zone	Limiting Loading Condition	Engagement from TTS (inches)
A	3 Times NO ΔP	6.97
B	3 Times NO ΔP	13.22
C	3 Times NO ΔP	13.53

a,c,e

Figure 7-3a. Contact Pressures for Normal Condition (T_{min}) at Byron/Braidwood 2 DP Factor = 0.64

a,c,e

Figure 7-4a. Contact Pressures for Normal Condition (T_{max}) at Byron and Braidwood Unit 2 DP Factor = 0.64

a,c,e

Figure 7-5a. Contact Pressures for SLB Faulted Condition at Byron and Braidwood 2
DP Factor = 0.64

a,c,e

Figure 7-6a. Contact Pressures for FLB Faulted Condition at Byron and Braidwood 2 (Tmin), DP Factor = 0.64

a,c,e

Figure 7-7a. Contact Pressures for FLB Faulted Condition at Byron and Braidwood 2 (Tmax), DP Factor = 0.64

a,c,e

Figure 8-1a. Change in Contact Pressure at 10.5 Inches Below the TTS, DP Factor = 0.64

a,c,e

Figure 8-2a. Change in Contact Pressure at 12.6 Inches Below the TTS, DP Factor = 0.64

a,c,e

Figure 8-3a. Change in Contact Pressure at 16.9 Inches Below the TTS, DP Factor = 0.64

a,c,e

Figure 8-4a. Change in Contact Pressure at the Bottom of the Tubesheet, DP Factor = 0.64

a,c,e

**Figure 8-5a. Change in Contact Pressure at 8.25 Inches Below the TTS,
DP Factor = 0.64**

Attachment 6

Regulatory Commitment
Supporting the Application for Steam Generator Tube Alternate Repair Criteria
Technical Specification Amendment

Byron Station Unit 2

COMMITMENT	COMMITTED DATE OR "OUTAGE"	COMMITMENT TYPE	
		ONE-TIME ACTION (Yes/No)	PROGRAMMATIC ACTION (Yes/No)
Byron Station will remove Steam Generator Tube R34-C46, located in the Unit 2, 2A Steam Generator, from service by plugging. This commitment is contingent on approval of the alternate repair criteria license amendment.	Byron Station Unit 2 Refueling Outage 14 (B2R14)	Yes	No