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10 CFR 50.90

RS-06-109

August 18, 2006

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: Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification

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 S. J. Campbell (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2 – Request for Additional Information Related to Technical Specification Improvement Regarding Steam Generator Tube Integrity," dated August 4, 2006

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 requested 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."

In the course of their review the NRC determined that additional information is required as documented in Reference 2. The attachments to this letter provide the response to the requested information. The response is subdivided as shown below.

Attachment 1 provides the EGC responses to the Request for Additional Information.

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Attachments 2-A and 2-B include the marked-up TS pages with the proposed changes indicated for Braidwood Station and Byron Station, respectively.
Attachments 3-A and 3-B include the associated typed TS pages with the proposed changes incorporated for Braidwood Station and Byron Station, respectively.

Attachments 4-A and 4-B include the associated revised TS Bases for Braidwood Station and Byron Station, respectively, for information only.

Attachment 5 provides an affidavit for withholding signed by Westinghouse Electric Company, LLC, the owner of proprietary information provided in Attachment 6. Also enclosed are a Westinghouse authorization letter CAW-06-2181, Proprietary Information Notice and Copyright Notice.

Attachment 6 provides a proprietary version of Westinghouse document SG-SGDA-06-20-P-Attachment, Revision 1, "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."

Attachment 7 provides a non-proprietary version of Westinghouse document SG-SGDA-06-20-NP-Attachment, Revision 1.

Attachment 6 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 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 respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-06-2181 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.

The information provided in this letter does not affect the supporting analysis for the original license amendment request as described in Reference 1. No other information contained in the Reference 1 letter is affected by this additional information.

The No Significant Hazards Consideration and the Environmental Consideration provided in Attachment 1 of Reference 1 are not affected by this additional information.

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 attachments to the designated State Official.

If you have any questions about this letter, please contact David Chrzanowski at (630) 657-2816.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 18th day of August 2006.

Respectfully,

Darin M Benyak

Darin M. Benyak
Manager – Licensing

Attachment 1: Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification Change
Attachment 2A: Revised Markup of Technical Specifications Pages for Braidwood Station
Attachment 2B: Revised Markup of Technical Specifications Pages for Byron Station
Attachment 3A: Revised Typed Technical Specifications Pages for Braidwood Station
Attachment 3B: Revised Typed Technical Specifications Pages for Byron Station
Attachment 4A: Revised Typed Technical Specifications Bases Page for Braidwood Station
Attachment 4B: Revised Typed Technical Specifications Bases Page for Byron Station
Attachment 5: Application for Withholding and Affidavit
Attachment 6: Proprietary Version - Westinghouse SG-SGDA-06-20-P-Attachment, Revision 1
Attachment 7: Non-proprietary Version - Westinghouse SG-SGDA-06-20-NP-Attachment, Revision 1

Attachment 1

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457
License Nos. NPF-72 and NPF-77

and

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455
License Nos. NPF-37 and NPF-66

Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification Change

Note that the SG-SGDA-06-20-P-Attachment, Revision 1 (Attachment 6 to this submittal) and SG-SGDA-06-20-NP-Attachment, Revision 1 (Attachment 7 to this submittal) reports provided by Westinghouse Electric Company, LLC (Westinghouse) were intended to provide responses to NRC draft questions posed to Seabrook Station, Braidwood and Byron Stations as well as providing information in support of other utilities proposing similar steam generator tube inspection license amendment requests. The responses to these draft questions in the Westinghouse reports are formatted in the Seabrook Station draft question sequence.

Also, these Westinghouse reports were prepared prior to the formal issuance of the finalized version of the Braidwood Station and Byron Station NRC questions, as a result, the documents make reference to two questions that no longer require clarification. Therefore, the responses provided for "Exelon Byron/Braidwood Draft RAI Questions 5 and 7" contained in the Westinghouse reports are no longer applicable.

Wherever necessary, the specific location in these Westinghouse reports of the relevant Braidwood Station and Byron Station information is identified in the response to the question.

NRC Question 1:

Proposed Specification 5.5.9.f discusses provisions for Unit 2 Steam Generator (SG) tube repair methods. The proposed technical specifications (TS) for Braidwood Units 1 and 2, and Byron Units 1 and 2, including LCO 3.4.19, refer to "plugged or repaired." However, proposed TS 5.5.9.f addresses provisions for SG tube repair methods for Braidwood Station Unit 2 and Byron Station Unit 2 only. The staff is concerned that the proposed TS could be misconstrued to mean that there are no restrictions with respect to repairs for Braidwood Station Unit 1 and Byron Station Unit 1. Discuss your plans for revising specification 5.5.9.f such as to clarify that tube repairs may not be performed for Unit 1.

Exelon Generation Company, LLC (EGC) response to Question 1:

In order to clarify that there are currently no approved SG tube repair methods for Braidwood Station Unit 1 and Byron Station Unit 1, the proposed TS 5.5.9.f wording is being revised. The statement: "There are no approved tube repair methods for the Unit 1 SGs," is being added to Section 5.5.9.f. See Attachment 2-A (for Braidwood Station) and 2-B (for Byron Station) for the revised wording of Insert 5.5-9 and Insert 5.5-10. To aid the review, these inserts are highlighted to indicate the most recent changes. The typed TS pages incorporating these changes are provided in Attachment 3-A (for Braidwood Station) and Attachment 3-B (for Byron Station).

NRC Question 2:

The NRC staff notes that the existing TS reporting requirements do not address issues associated with implementation of the tubesheet inspection and alternate repair criterion. Discuss Exelon's plans to revise proposed TS 5.6.9, "Steam Generator (SG) Inspection Report," to include reporting requirements applicable to the implementation of the tubesheet inspection and alternate repair criterion, similar to what the staff has requested other licensees submitting applications for tubesheet inspection and alternate repair criteria.

EGC response to Question 2:

The proposed TS 5.6.9 wording is being revised to include three additional items to be included in the 180-day SG report for Braidwood Station Unit 2 and Byron Station Unit 2. The additional items address the reporting details regarding indications detected in the upper 17-inches of the tubesheet thickness, the operational primary to secondary leakage rate observed during the previous Unit 2 cycle, and the calculated accident leakage rate from the lowermost 4-inches of tubing.

See Attachment 2-A (for Braidwood Station) and 2-B (for Byron Station) for the revised wording of Insert 5.6-6. To aid the review, this insert is highlighted to indicate the most recent change. The typed TS pages incorporating these TS 5.6.9 changes are provided in Attachment 3-A (for Braidwood Station) and Attachment 3-B (for Byron Station).

NRC Question 3:

On page 7 of 17 of Attachment 1, "Evaluation of Proposed Changes," of the November 18, 2005 submittal, Exelon indicated that the roll transition zone sleeve lower joint is located near the neutral axis of the tubesheet (i.e., within the portion of the tube that will be inspected). Confirm that the Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and Unit 2 Steam Generators Tube Repair Using Leak Tight Sleeves, Final Report," dated April 1995, precludes the establishment of the joint in the lower 4 inches of the tubesheet. If it does not, provide technical justification why a joint in the lower 4 inches of the tubesheet is allowable. Alternatively, rewrite the proposed TSs to preclude the establishment of joints in the lower 4 inches of the tubesheet.

EGC response to Question 3:

As indicated in topical report (CEN-621-P) Figures 4-3A, "Roll Transition Zone Sleeve (Short) Installation," and 4-3B, "Roll Transition Zone Sleeve (Long) Installation," the bottom edge of the ABB-CE sleeve design is 11.25 inches to 17.0 inches from the hot leg tube end, thus installation of the sleeve within the bottom four inches of the tubesheet is prohibited.

NRC Question 4:

In the proposed TS (and TSTF-449), a SG tube is defined as the entire length of the tube, including the tube wall (and any repairs made to it), between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. Given this definition, the proposed repair criteria in TS 5.5.9.c could be misinterpreted. Discuss Exelon's plans to modify the TSs to more clearly define the repair criteria for the sleeved portion of a tube.

EGC response to Question 4:

TS 5.5.9.c, "Provisions for SG tube repair criteria," is being revised. See Attachment 2-A (for Braidwood Station) and 2-B (for Byron Station) for the revised wording of Insert 5.5-8. To aid the review, this insert is highlighted to indicate the most recent change. The typed TS pages incorporating these TS 5.5.9 changes are provided in Attachment 3-A (for Braidwood Station) and Attachment 3-B (for Byron Station).

NRC Question 5:

Regarding the revised BASES,

a. Proposed Page B 3.4.13 - 3 states, "The dose consequences resulting from the Locked Rotor with a Concurrent SG PORV [power-operated relief valve] Failure Accident are well within the limits defined in 10 CFR [Part] 100." This statement replaces the current statement, "The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR [Part] 100." Why has this revision been made? Aren't the consequences of all analyzed events within the limits defined in 10 CFR Part 100? If so, why not simply state that?

b. On page B 3.4.19-4, there is a statement that the accident induced leakage criteria is 1 gpm for all steam generators except for specific types of degradation at specific locations. This wording (i.e., the exception) is not in the proposed TS. Describe Exelon's plans for resolving this discrepancy.

EGC response to Question 5:

The current wording in the Braidwood Station and Byron Station Bases, page 3.4.13-3 states: "The dose consequences resulting from the Locked Rotor with a Concurrent SG PORV Failure accident are well within the limits defined in 10 CFR 100." There were no changes proposed to the current wording. The wording identified in Question 5 that states: "The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100," is the wording proposed in TSTF-449. This generic statement in the TSTF-449 should have been identified as a bracketed change since the limiting design basis accident (DBA) from a dose perspective is site-specific.

For Braidwood and Byron Stations the Locked Rotor with a concurrent SG Power Operated Relief Valve (PORV) Failure presents the design basis challenge for primary to

secondary SG leakage only. As stated earlier in Section B 3.4.13, the Braidwood Station and Byron Station:

> "Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a Locked Rotor with a Concurrent Steam Generator (SG) Power Operated Relief Valve (PORV) Failure accident because such leakage contaminates the secondary fluid. Other accidents or transients involve secondary steam release to the atmosphere, such as a Steam Generator Tube Rupture (SGTR). The SGTR is more limiting than the Locked Rotor with a Concurrent SG PORV Failure for site radiation releases."

While this specific Braidwood Station and Byron Station Bases 3.4.13 wording accurately describes the limiting DBA from a primary to secondary SG tube leakage perspective and no changes are being considered for this TS amendment submittal, the Braidwood Station and Byron Station license amendment request to incorporate Alternative Source Term, currently under NRC review, is proposing to modify the current Base 3.4.13 words to encompass all relevant DBAs by stating:

> "The dose consequences resulting from the MSLB, SGTR, Control Rod Ejection and Locked Rotor accidents are within the limits defined in 10 CFR 50.67."

No additional actions are required to address this part of Question 5.

The original proposed wording on page 4 of new Bases Section B 3.4.19 states:

> "The accident induced leakage requirement of 1 gpm for all SGs, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage, bounds the accident analysis assumptions for primary to secondary LEAKAGE."

This wording was included TSTF-449 to address a licensee's alternate repair criteria allowing for greater than one gpm leakage. Braidwood Station and Byron Station are not invoking this type of alternate repair criteria, therefore, the words will be revised to state:

> "The accident induced leakage requirement of 1 gpm for all SGs bounds the accident analysis assumptions for primary to secondary LEAKAGE."

The typed Bases page incorporating this revised wording is provided, for information only, in Attachment 4-A (for Braidwood Station) and Attachment 4-B (for Byron Station).

<u>NRC Question 6:</u>

Under the proposed 17-inch tubesheet inspection zone, it is the Exelon's contention that the accident leakage integrity of the tubing below the 17-inch inspection zone is ensured by the bell-weather principle. The NRC staff requests that Exelon submit a leakage sensitivity study to support the conservatism of the bellwether approach. That is, leakage during accidents will not exceed two times that observed during normal operating conditions. The NRC staff requests that this study consider axial and circumferential flaws located at the bottom of the tubesheet at three tubesheet radial locations; i.e., at the zero radius, mid-radius, and peripheral locations. For each type of crack at each location, leakage under normal operating and accident conditions should be evaluated considering only the crack leakage resistance, considering only the tube to tubesheet annulus resistance and, lastly, considering the total resistance of the crack and annulus to leakage.

<u>EGC response to Question 6:</u>

The complete response to this question is provided on pages 8 through 16 of the proprietary version of SG-SGDA-06-20-P-Attachment, Revision 1 (i.e., Attachment 6 of this submittal) and is identified as "Response 5." Because the response to this question contains information proprietary to Westinghouse, only partial information is provided on pages 8 through 16 of the non-proprietary version (i.e., Attachment 7).

<u>NRC Question 7:</u>

Section 8.2 of Attachment 7 provides a justification for why ligament tearing of circumferential cracks is not a significant concern. Provide a justification for why ligament tearing of axial cracks at the bottom of the tubesheet at the periphery is similarly not a significant concern.

<u>EGC response to Question 7:</u>

The complete response to this question is provided on pages 20 through 23 of the proprietary version of SG-SGDA-06-20-P-Attachment, Revision 1 (i.e., Attachment 6) and is identified as "Response 9." Because the response to this question contains information proprietary to Westinghouse, only partial information is provided on pages 20 through 23 of the non-proprietary version (i.e., Attachment 7).

NRC Question 8:

Are there any tubes in the Byron and Braidwood, Unit 2 SGs which were not fully expanded (per nominal) within the tubesheet? If so, please describe the extent of this condition and justify why the amendment request is sufficient to ensure the structural and leakage integrity of the affected tube joints.

EGC response to Question 8:

A review of eddy current data for Byron Station Unit 2 was performed during the 12th refueling outage in the Fall of 2005 (B2R12). One SG tube was identified as not being fully expanded through the tubesheet. This tube was rerolled and expanded an appropriate length to accommodate a mechanical plug. The tube was then preventively plugged prior to restart from the B2R12 refueling outage.

As described on page 25 of both the Attachment 6 (proprietary) and the Attachment 7 (non-proprietary) versions of SG-SGDA-06-20-P-Attachment, Revision 1, identified as "Response 10," the eddy current reports for Braidwood Station Unit 2 were reviewed to determine if there were indications of partially or fully unexpanded tubes within the tubesheet. Based on this data all the tubes currently inservice, within the Braidwood Station Unit 2 tubesheet, are fully expanded.

Attachment 2A

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification Change

Revised Markup of Technical Specifications Pages

INSERT 5.5-8

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.

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. 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 be applied as an alternate to the 40% depth-based criteria of Technical Specification 5.5.9.c.1:

   i. For Unit 2 only, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Unit 2, 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

INSERT 5.5-9

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.

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of

INSERT 5.5-10

these Specifications, tube plugging is not a repair.

1. There are no approved tube repair methods for the Unit 1 SGs.

2. All acceptable repair methods for the Unit 2 SGs are listed below.

   i. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, "Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.

Steam Generator Program (Braidwood)

Insert TS 5.6-6

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, 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, 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, and

l. For Unit 2, 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 2B

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455

License Nos. NPF-37 and NPF-66

Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification Change

Revised Markup of Technical Specifications Pages

INSERT 5.5-8

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.

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. 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 be applied as an alternate to the 40% depth-based criteria of Technical Specification 5.5.9.c.1:

   i. For Unit 2 only, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Unit 2, 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

INSERT 5.5-9

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.

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of

INSERT 5.5-10

these Specifications, tube plugging is not a repair.

1. There are no approved tube repair methods for the Unit 1 SGs.

2. All acceptable repair methods for the Unit 2 SGs are listed below.

   i. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, "Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.

Steam Generator Program (Byron)

Insert TS 5.6-6

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, 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, 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, and

l. For Unit 2, 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 3A

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification Change

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5.5-10
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.

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. 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 be applied as an alternate to the 40% depth-based criteria of Technical Specification 5.5.9.c.1:

i. For Unit 2 only, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does 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 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.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair.

1. There are no approved tube repair methods for the Unit 1 SGs.

2. All acceptable repair methods for the Unit 2 SGs are listed below.

i. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, "Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.

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, 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, 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, and

l. For Unit 2, 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 3B

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455

License Nos. NPF-37 and NPF-66

Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification Change

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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.

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i. 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 be applied as an alternate to the 40% depth-based criteria of Technical Specification 5.5.9.c.1:

i. For Unit 2 only, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does 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 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.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair.

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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, 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, 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, and

l. For Unit 2, 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 4A

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification Change

Revised Typed Technical Specifications Bases Page
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BASES

LCO (continued)

significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident induced leakage requirement of 1 gpm for all SGs bounds the accident analysis assumptions for primary to secondary LEAKAGE. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

Attachment 4B

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455

License Nos. NPF-37 and NPF-66

Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification Change

Revised Typed Technical Specifications Bases Page
(Information Only)

B 3.4.19-4

BASES

LCO (continued)

significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident induced leakage requirement of 1 gpm for all SGs bounds the accident analysis assumptions for primary to secondary LEAKAGE. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

Attachment 5

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457
License Nos. NPF-72 and NPF-77

and

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455
License Nos. NPF-37 and NPF-66

Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification Change

Application for Withholding and Affidavit

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
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USA

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

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

Our ref: CAW-06-2181

July 28, 2006

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: SG-SGDA-06-20 P-Attachment, Rev. 1, "Exelon: Byron Unit 2 and Braidwood Unit 2 Response to NRC Request for Additional Information on License Amendment Request 05-08 Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet (TAC No. MC8554) Seabrook Station," dated July 2006 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-06-2181 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-06-2181, 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: G. Shukla/NRC

bcc: J. A. Gresham (ECE 4-7) 1L
R. Bastien, 1L (Nivelles, Belgium)
C. Brinkman, 1L (Westinghouse Electric Co., 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) 1L, 1A (letter and affidavit only)
G. W. Whiteman, Waltz Mill
R. F. Keating, Waltz Mill
H. O. Lagally Waltz Mill
N. R. Brown, Waltz Mill
D. W. Alexander, ECE 561B
J. M. Bunecicky, ECE 560E

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:

PATRICIA L. CROWN COMMONWEALTH OF PENNSYLVANIA NOTARY PUBLIC

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

Sworn to and subscribed before
me this 28th day of July, 2006

Patricia L. Crown
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Patricia L. Crown, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Feb. 7, 2009
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, 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 SG-SGDA-06-20 P-Attachment, Rev. 1, "Exelon: Byron Unit 2 and Braidwood Unit 2 Response to NRC Request for Additional Information on License Amendment Request 05-08 Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet (TAC No. MC8554) Seabrook Station," dated July 2006 (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 joint with a rotating probe within the tubesheet region of the steam generators and is provided for additional information on 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.

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

(a) Obtain NRC approval of LTR-CDME-05-32-P, Rev. 2 "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron Unit 2 and Braidwood Unit 2," August 2005.

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

(c) Provide a primary-to-secondary side leakage evaluation for the Byron Unit 2 and Braidwood Unit 2 steam generators during all plant conditions.

(d) Assist the customer to respond to NRC requests for information.

Further this information has substantial commercial value as follows:

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

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

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

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar licensing support documentation 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).

Attachment 7

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457
License Nos. NPF-72 and NPF-77

and

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455
License Nos. NPF-37 and NPF-66

Response to Request for Additional Information Regarding Application for Steam Generator Tube Integrity Technical Specification Change

Non-proprietary Version - Westinghouse SG-SGDA-06-20-NP-Attachment, Revision 1

"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."

SG-SGDA-06-20-NP-Attachment, Rev. 1

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 (TAC NO. MC8554)

July 27, 2006

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Introduction

This document contains responses to a RAI from the USNRC in response to a license amendment request (LAR) by FP&L Seabrook to permanently limit the extent of rotating coil inspection in the tubesheet expansion zone to the region from the top of the tubesheet to 17 inches below the top of the tubesheet. A number of the questions contained in the RAI are generic to other utilities, including EXELON, who have submitted, or are planning to submit, similar license amendment requests to permanently limit the extent of RPC inspection in the tubesheet expansion zone. To the degree that these responses apply to Byron 2 and Braidwood 2, they are provided to EXELON to support its submittal of a license amendment request based on a B*/H* type of technical justification.

The original NRC questions are reproduced, followed by the responses provided by Westinghouse. Some of the questions are specific to the Seabrook situation and responses to these were provided by Seabrook. The Seabrook-specific responses are not included in this document. However, if the generic questions require a modified response specific to either Byron 2 or Braidwood 2, these modifications have been included in the responses.

EXELON has received a draft of a RAI from the NRC for the EXELON LAR based on the B*/H* type of technical justification. Some of the draft questions received by EXELON are the same as the questions received by Seabrook. Some questions on the Seabrook LAR are not included in the EXELON draft RAI. Some questions received by EXELON are not included in the Seabrook RAI. A correlation of the questions received by Seabrook and the draft questions received by EXELON is provided in Table 1 and is summarized below:

Seabrook questions #1, #2 and #3 are Seabrook-specific; no response is included in this document.

Seabrook question #4 requests that WCAP-16053 be provided. In a draft RAI, question 7, received by EXELON regarding its submittal of a similar license amendment request, the NRC requested a copy of the EXELON-specific WCAP-16152. A response to this EXELON RAI has been prepared and is included following the responses to the Seabrook questions.

Seabrook question #5 is equivalent to EXELON draft question #8. The Seabrook response has been modified to include information that renders the response generic for the different tube diameter at Byron 2 and Braidwood 2.

No questions equivalent to Seabrook questions #6, #7 and #8 were included in the EXELON draft RAI. The Seabrook responses are provided for information and consistency.

Seabrook question #9 is equivalent to EXELON draft question #9. The Seabrook response has been modified to include information that renders the response generic for the different tube diameter at Byron 2 and Braidwood 2.

Seabrook question #10 is the same as EXELON draft question #10. By its nature, the question is plant-specific. Both Byron 2 and Braidwood 2 have performed analyses of their inspection results and, based on this, EXELON will respond to the RAI. In response to a request from Braidwood, Westinghouse has

reviewed the EC results for Braidwood-2 for the presence or absence of unexpanded or partially expanded tubes. A response to the RAI, as it applies to Braidwood 2, is included.

Table 1
Correlation of Seabrook RAI Questions and EXELON Draft RAI Questions

Seabrook		Byron/Braidwood	
RAI Number	**Subject**	**RAI Number**	**Subject**
1	Delete minimum RPC sampling requirements in tubesheet region from TS 4.4.5.2	NA	No equivalent
2	Delete the definitions of bulge and overexpansion since they support only the TS section noted in item 1 which are considered unnecessary	NA	No equivalent
3	Revise TS 4.5.5.5 , 12 month report, to include number, size, ID or OD origin, orientation of flaws in upper 17" of the tubesheet. Also, the corresponding, 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 12-month report should describe how it was determined.	2	Discuss plans to revise TS 5.6.9 , SG Inspection Report to include reporting requirements...indications in the upper 17 inches, location orientation and measured size. Provide observed operational leakage; 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 12-month report should describe how it was determined.
4	Provide copy of WCAP-16053 (the predecessor H* report for the plant)	7	Provide copy of WCAP-16152 (the predecessor H* report for the plant)
5	Submit a leakage sensitivity study to support the conservatism of the bellwether approach.	8	Submit a leakage sensitivity study to support the conservatism of the bellwether approach.
6	Provide the rationale which supports the conservatism of the 600 degree F assumption for MSLB and FWLB	NA	No equivalent

7	Please clarify what is the appropriate comparison to make here; 0.2 gpm (accident leakage adjusted for room temperature conditions) versus the 0.347 gpm allowable limit, or 0.3 gpm (accident leakage for actual accident temperature) versus the 0.347 allowable limit? In other words, does the accident analysis consider the 0.347 gpm allowable limit to be an adjusted value for room temperature conditions or does it treat it as a hot value?	NA	No equivalent
8	Plans for submitting a proposed change to the technical specification operational leakage limit such as to ensure accident leakage will be within the amounts analyzed?	NA	No equivalent
9	Justification for why ligament tearing of axial cracks at the bottom of the tubesheet at the periphery is similarly not a significant concern.	9	Justification for why ligament tearing of axial cracks at the bottom of the tubesheet at the periphery is similarly not a significant concern.
10	Tubes in the Seabrook steam generators which were not fully expanded (per nominal) within the tubesheet?	10	Tubes in the Byron/Braidwood Unit 2 steam generators which were not fully expanded (per nominal) within the tubesheet?

NA	No equivalent	1	Discuss plans for revising specification 5.5.9 such as to clarify that tube repairs may not be performed in Unit 1
NA	No equivalent	3	Confirm that the topical report CEN-621-P precludes the establishment of the (sleeve) joint in the lower 4 in. of the tubesheet.
NA	No equivalent	4	Discuss plans to modify your TS to more clearly define the repair criteria for the sleeved portion of the tube.
NA	No equivalent	5	Leakage criteria in LAR more restrictive than in TSTF-449
NA	No equivalent	6	Describe plans to revise the statement (on dose limits) to say that dose consequences from all analyzed events are within the limits defined in 10CFR100. Describe plans to resolve discrepancy between LAR and TS regarding exception to the 1 gpm criterion.

FPL Energy Seabrook, LLC Responses to
REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST 05-08
LIMITED INSPECTION OF THE STEAM GENERATOR
TUBE PORTION WITHIN THE TUBESHEET (TAC NO. MC8554)
SEABROOK STATION

References:
1. FPL Energy Seabrook, LLC letter SBK-L-05185, "License Amendment Request 05-08, Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet, dated September 29, 2005.
2. FPL Energy Seabrook, LLC letter SBK-L-05186, "Proprietary Information to Support License Amendment Request 05-08, Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet, dated September 29, 2005.

NRC RAI 1: The licensee is proposing to add a new requirement "d". under TS 4.4.5.2, Steam Generator Tube Sample Selection and Inspection. This proposed new requirement would define minimum sampling requirements to be performed with a rotating probe in the tubesheet region. The staff finds this proposed new requirement to be unnecessary and recommends that it be deleted from the proposed amendment. The technical specifications are normally non-specific with respect to the types of eddy current test probes to be used. As stated in NRC Generic Letter 2004-01, once licensees determine that specific degradation mechanisms may be present at various locations along the tube (as part of the degradation assessment), it is the staff's position that they should use probes capable of detecting these forms of degradation. The Generic Letter states that not to do so raises questions about whether the tube inspection practices ensure compliance with the technical specifications in conjunction with 10 CFR 50, Appendix B.

Response provided by FP&L Energy.

NRC RAI 2: The licensee is proposing to add items 10) and 11) under TS 4.4.5.4, "Acceptance Criteria. These new items would define the words bulge and overexpansion, respectively. These new definitions support the use of these terms in the proposed new requirement d. under TS 4.4.5.2. As stated in Question 1 above, the staff finds the proposed item d. under TS 4.4.5.2 to be unnecessary and recommends that it be deleted from the proposed amendment. With deletion of this item, the proposed definitions for bulge and overexpansion are also unnecessary and should be deleted from the proposed amendment.

Response provided by FP&L Energy.

NRC RAI 3: The staff requests a revision to TS 4.4.5.5, Reports, regarding item b. pertaining to the 12-month Special Report. Specifically, the staff requests that the requirements for the 12-month report be expanded to include:

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

- The operational primary to secondary leakage rate observed in each steam generator during the cycle preceding the inspection which is the subject of the report and (2) the corresponding, 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 12-month report should describe how it was determined.

Response provided by FP&L Energy.

NRC RAI 4: Provide copy of WCAP-16053, which is listed as a reference in the technical attachment to Reference 2, above.

Response 4:

(Note that an EXELON-specific response is appended to these responses, titled "Specific Responses for EXELON")

NRC RAI 5: Under the proposed 17-inch tubesheet inspection zone, it is the licensee's contention that the accident leakage integrity of the tubing below the 17-inch inspection zone is ensured by the bell-weather principle. The staff requests that the licensee submit a leakage sensitivity study to support the conservatism of the bell-weather approach. That is, leakage during accidents will not exceed two times that observed during normal operating conditions. We request that this study consider axial and circumferential flaws located at the bottom of the tubesheet at three tubesheet radial locations; i.e., at the zero radius, mid-radius, and peripheral locations. For each type crack at each location, leakage under normal operating and accident conditions should be evaluated considering only the crack leakage resistance, considering only the tube to tubesheet annulus resistance and, lastly, considering the total resistance of the crack and annulus to leakage. [Note, the staff is not so much interested in the absolute values of the leakage predictions as it is in the relative values of the predictions between normal operating and accident conditions. The licensee hasn't requested that the staff review the leakage prediction models, and the staff would not be in a position to approve these models until the accuracy of these models has been validated by test for prototypic situations. This being said, the staff believes that these models, which are based on standard engineering principles, should at least be capable of providing a qualitative demonstration supporting the bell-weather approach.]

Response 5:

The basis for the development of the 17-inch tubesheet inspection zone with regard to leak rate is the ratio of the potential leak rate during a SLB event to that during NOp using the results of data from leak rate tests of the tube-to-tubesheet interface, a.k.a. crevice. Westinghouse had historically developed a computer model for a crevice in series with a crack using the crevice data and independent

data for free-span cracks. The NRC staff expressed concerns regarding the model because of a lack of test data from physical specimens which contained a crack in series with a crevice; hence, the discussion in the last section of the RAI. Westinghouse data obtained from separate testing of the tube-to-tubesheet crevice and axial cracks within a tubesheet with a zero length crevice above the crack demonstrated the resistance of the crack to be comparable to the resistance of the crevice for a larger tube size. The implication from the latter being that an analysis that neglected the effect of the crack would be valid because the effect on the numerator and denominator of the SLB:NOp leak rates ratio would be the same. Other considerations were also made, e.g., for indications within about 56 inches from the center of the tubesheet, the effect of tubesheet bow induced crack closure would be to increase the resistance of the crack.

It is also worth noting the expectations from the analysis based on the crack opening area formulation and the geometries of the model. The opening of circumferential cracks is resisted by the stiffness of the material above and below the crack flanks and by friction on the OD of the tube. For all practical purposes the tube is infinitely long in the axial direction, although, the resistance to opening due to shear increases rapidly. The geometry of the tubesheet does not restrict crack opening in the axial direction. The opening of axial cracks is more restricted owing to the geometry of the problem. For example, there is a line of symmetry 180° from the crack flanks; hence the tube is stiffer in the hoop direction at that location. There is also friction associated with the interface of the tube with the tubesheet. Finally, the confinement provided by the tubesheet means that an axial crack cannot open more than the dilation of the tubesheet hole, which is very small on the periphery. Thus, the effects reported from the DENTFLO analyses would be expected to indicate more effect for circumferential cracks than for axial cracks.

In trying to use the Westinghouse computer model to provide a qualitative demonstration of the veracity of the B* analysis, a significant potential shortcoming associated with the approach must be recognized. The crack flow leak rate model portion of the code was based on a freespan axial crack, not a crack with flanks constrained by the tubesheet hole. Thus, the crack opening area computation could be significantly biased. In order to perform the requested studies the code was modified as follows:

1) The crack opening area model for circumferential cracks of Appendix C of WCAP-15932 (Reference 3) was included in the DENTFLO code.
2) A crack opening area model for axial cracks constrained in the tubesheet was derived accounting for the constraints added to the problem by the presence of the tubesheet and included in the DENTFLO code.

The new model for the axial crack opening that takes into account the guidance and constraint provided by the surrounding tubesheet is described in what follows. See Figure 1, below, for a sketch of the model configuration.

Using these models, a sensitivity study was performed which consisted of a series of analyses to demonstrate the conservatisms of the bellwether approach. These analyses considered the locations and conditions specified in the RAI, i.e., crack only, crevice only and combined crack and crevice.

The models developed for these analyses were for qualitative comparisons only (i.e., not for absolute prediction of leak rates) and were not verified and/or validated beyond that "information only" status.

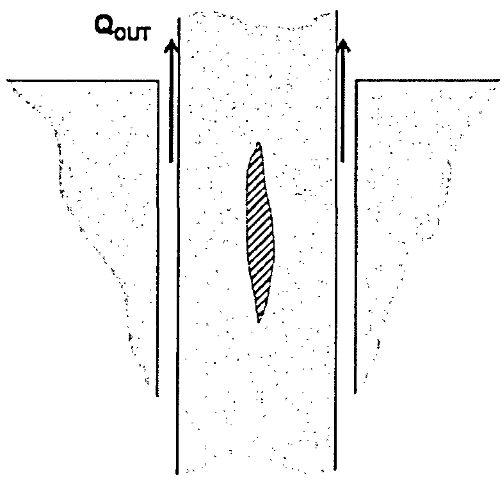

Figure 4: Sketch of Cracked Tube in Tubesheet leading to Leakage Flow-rate, Q_{OUT}.
(Note that the gap shown between the tube and the tubesheet is for illustration purposes only.)

Constrained Axial Crack Opening Area Calculations

A literature search of the significant fracture mechanics texts and journals, e.g. Journal of Engineering Fracture Mechanics, International Journal of Fracture Mechanics, Reference 7, etc., did not yield any previously published models for the crack opening area for a central axial crack that is circumferentially constrained in an internally pressurized tube. A mathematical model was developed to calculate the crack opening area for a constrained SG tube under these conditions. The general form of the equation for the crack opening area in an infinite plate is [2]:

$$\text{COA} = 2\pi a^2 \frac{\sigma}{E}$$

where σ is the far-field stress resulting in the crack opening, E is the Young's Modulus of the tube and a is the half-length of the crack. In the absence of empirical evidence, the general form for the crack opening area can be modified by a functional to deal with the tube configuration. The functional can include the important details regarding the boundary conditions and other effects relative to the constrained and cracked tube. Let H(F(n)) be the modifying functional for the crack opening area, where n represents the influential parameters of the geometry and loading. Then the model for the crack opening area becomes,

$$\text{COA} = H\big(\ F(n)\ \big)2\pi a^2 \frac{\sigma}{E}$$

Given the general constitutive form for the crack opening area it remains to define the modifying functional. The crack opening area for the axial crack will be affected by the interactions of the internal

pressure on the crack flanks and the contact pressure between the tube and the tube sheet, see Reference 3. The largest effect on the crack flanks that can affect the flow rate through the crack will be the radial flexing of the flanks due to the internal pressure in the event that the contact pressure between the tube and the tube sheet decreases, although, as previously noted, the amount of opening can never be greater than the change in the circumference of the hole in the tubesheet.

Comparison Against Established Methods

The resulting model for the crack opening area in an internally pressurized and constrained tube with an axial crack is shown below,

a,c,e

where ζ is a scalar coefficient that describes the local effect of the crack in a tube on an equivalent finite flat plate area and α is a parameter that accounts for the change in resistance to radial flexing of the crack flanks due to the change in tube to tubesheet contact pressure and crack length. This model was compared against several established models for unconstrained internally pressurized tubes with axial cracks. These comparison models include published work by Zahoor [4] and empirical models employed by the American Petroleum Institute (API) and developed by Anderson [6]. Figure 2, below, shows the results that each model predicts for the crack opening area for cracks ranging from 0.02 to 2.00 inch.

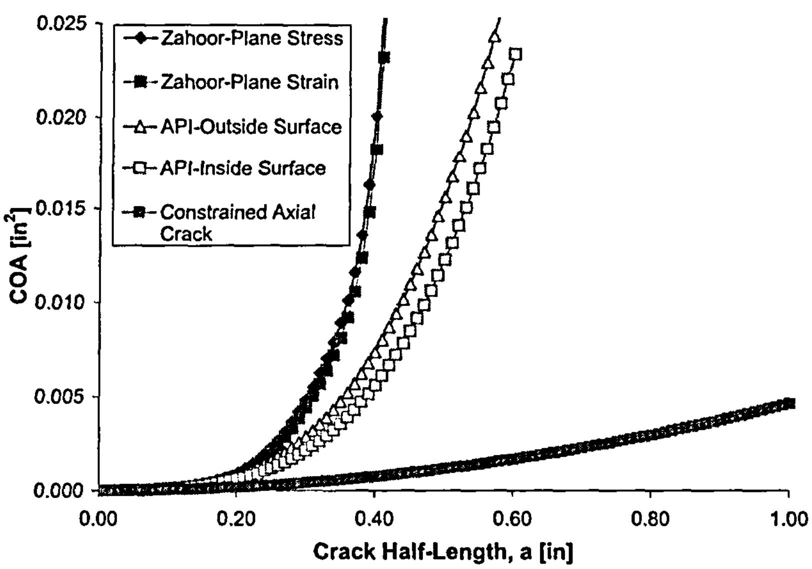

Figure 5: Comparison of Models for Calculating the Crack Opening Area

The comparison above shows that the results for the new model, for a constrained internally pressurized tube, are reasonable in comparison to the other established models for unconstrained tubes. Specifically, the new model predicts a smaller crack opening area as a function of crack length than the unconstrained axial cracks with a smaller rate of increase in the crack opening area. This is an expected result because the other models assume a free span for the tube with no constraining effects. The COA should be come asymptotically linear with increasing length because the constraint from the tubesheet hole will restrict the opening to some maximum amount. Therefore, it is reasonable to use the new model for further calculations to estimate the crack opening area of a constrained and internally pressurized tube with a central axial crack.

Analysis of Circumferential Cracking

A model for a constrained circumferential crack in an internally pressurized tube was developed [3]. This model is appropriate to use for a circumferential crack that occurs in a tube within the tubesheet. A maximum crack half angle of 90°, or a maximum circumferential crack equal to half of the circumference of the tube, was used in this analysis as a simplifying assumption. The model for constrained circumferential cracks was implemented in the code DENTFLO to determine the trend of the leakage rate ratios for normal operating and steam line break conditions.

DENTFLO Analysis Methodology

The program DENTFLO was run to determine the trend of the leak rate ratios of a damaged tube at the bottom of the tubesheet at different radii. There are 36 different cases of interest with respect to the leak rate ratio analysis, i.e., combinations of: 2 thermal-hydraulic conditions (NOp and SLB), 2 crack orientations (axial and circumferential), 3 radial locations (near, mid, and peripheral) and 3 conditions of interest (crack, crevice, and combined crack and crevice). The Normal Operation – Maximum Temperature (NOp-MAX) and the Steam Line Break (SLB) were chosen for the analysis because of the largest change in temperature and pressure between the two cases. The tubesheet radii for each range are: Near (2.0774 In), Mid (33.101 In), Peripheral (60.2475 In).

The contact pressures used in the DENTFLO analysis were taken at each tubesheet radius at several elevations in the tubesheet taken from Section 7 of LTR-CDME-05-32-P, Revision 2. The tubesheet material below 4 inches from the bottom of the tubesheet and any contact pressure generated by that material was conservatively neglected. The loss coefficients for the analysis were taken as a numerically integrated average over the length of the crevice, i.e., from 4 inches above the bottom of the tubesheet to the top of the tubesheet, based on the contact pressure distribution in the tubesheet. These are provided in Section 6 of LTR-CDME-05-32-P, Revision 2. The axial and circumferential crack orientation cases used the models discussed above. The crevice only cases use a model for an unconstrained axial crack based on the work of Paris and Tada [2] because it gives less resistance to flow through a crack than the other models available in DENTFLO. The crack length is also large in the crevice only case in order to best represent a situation where the crack cannot contribute significantly to the flow resistance.

DENTFLO Analysis Results

[

]a,c,e

a,c,e

Figure 6: Leak Rate Ratio as a function of Axial Crack Length for a tubesheet radius of 60.248 in. at the bottom of the tube sheet.

a,c,e

Figure 4: Leak Rate Ratio as a function of Circumferential Crack Length for a tubesheet radius of 60.248 in. at the bottom of the tube sheet.

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]a,c,e

Table 4: Summary of Near Radius Leak Rate Ratio Results

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[

]a,c,e

Table 5: Summary of Mid Radius Leak Rate Ratio Results

[ ]a,c,e

Figure 3 and Figure 4 show the results for the peripheral tube cases. In the case of an axial crack in series with a crevice and the crack only case, the axial crack orientation leak rate ratios plateau at a value [ ]a,c,e This is despite the sharp decrease contact pressure at the periphery between the tubes and the tubesheet. [

]a,c,e

Table 6: Summary of Peripheral Radius Leak Rate Ratio Results

a,c,e

The results of the DENTFLO analysis indicate that the bellwether trend is valid for all tubesheet radii and is bounded by a factor of 2. Therefore, the bellwether principle for the leak rate ratio approach is valid.

Axial Crack Opening Area and Leak Rate as a Function of Tube Diameter

The calculated axial crack opening area can vary with tube size. An increase in the crack opening area is generally proportional to an increase in tube diameter. Comparing crack sizes below 0.65 inch on tubes with diameters of 0.50 and 1.00 inch, the increase in constrained axial crack opening area is predicted to be more than twice the increase in tube size. See Figure 5 for a plot comparing the constrained axial crack opening area (COA) as a function of crack size and tube radius.

The axial crack opening area results shown in the Figure 5 below are not directly indicative of the leak rate behavior during NOp and SLB conditions. However, the axial crack opening area is indicative of other factors that do regulate the leak rate. For instance, an increase in the crack opening area will decrease the potential leak resistance of the crack at NOp and SLB. However, this decrease in leak rate resistance due to the crack is comparable for both conditions; therefore, the leak rate ratio will remain relatively unchanged. The bellwether principle is still reasonable for various tube diameters with respect to the constrained axial crack opening area and the conclusions stated for the Model F steam generator tube size discussed above can be generically applied.

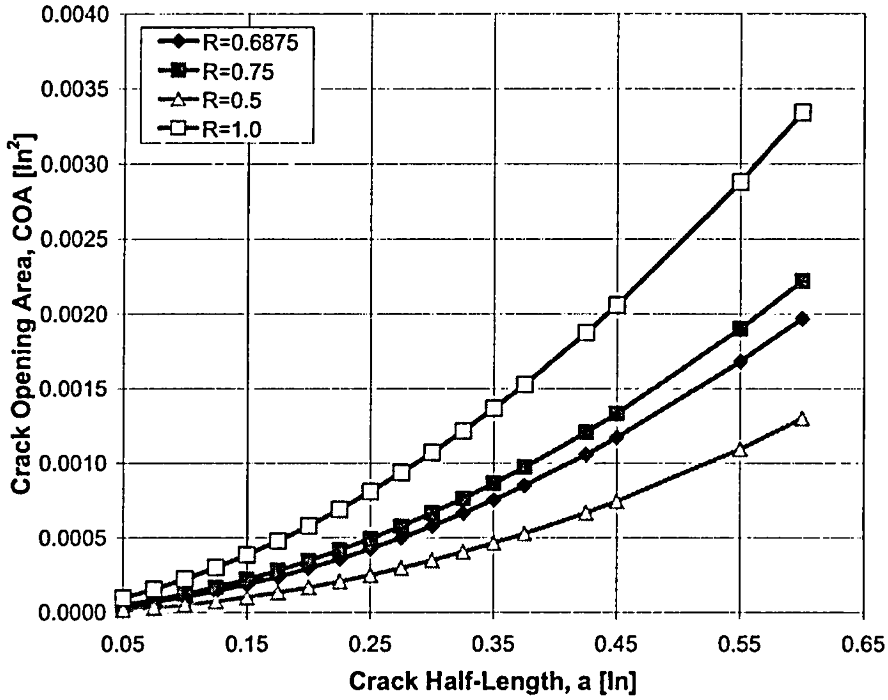

Figure 5: Axial Crack Opening Area as a Function of Crack Length for varying tube diameters at SLB.

References:

8. NSD-RMW-91-026, "An Analytical Model for Flow Through an Axial Crack in Series with a Denting Corrosion Medium," M.J. Sredzienski. 02/05/1991.

9. H. Tada, The Stress Analysis of Cracks Handbook, 2nd Ed., Del Research Corporation, 226 Woodburne Dr., St. Louis, Missouri, USA, 1985.

10. WCAP-15932-P, Revision 1; Improved Justification of Partial-Length RPC Inspection of Tube Joints of Model F Steam Generators of Ameren-UE Callaway Plant; May 2003

11. NP-6301-D, "Ductile Fracture Handbook," Volume 1, EPRI, 3412 Hillview Avenue, Palo Alto, CA, June 1989.

12. EPRI 1001191; Steam Generator Degradation Specific Management Flaw Handbook, EPRI, Palo Alto, CA, USA; 2001.

13. T.L. Anderson, "Stress Intensity and Crack Growth Opening Area Solutions for Through-Wall Cracks in Cylinders and Spheres," WRC Bulletin 430, Welding Research Council, January, 2003.

14. R.J. Sanford, Principles of Fracture Mechanics, Prentice Hall, Upper Saddle River, NJ, 2003.

NRC RAI 6: The bellweather principle maintains that the increase in primary to secondary leakage when going from normal operating to accident conditions is bounded by a factor of 2. This is based, in part, on an assumed main steam line break (MSLB) and feed line break (MFLB) pressure differential of 2560 and 2650 psi, respectively, and a temperature of 600 degrees F. Provide the rationale which supports the conservatism of the 600 degree F assumption. This rationale should consider the time history of primary and secondary pressure and temperature during the accident. The staff's purpose in asking this question is ensure the time integrated leak rate is bounded by the bellweather principle factor of 2 increase relative to normal operating leakage. Provide the primary pressure and temperature curves as a function of time for the MSLB and MFLB accidents under consideration. Also, provide the rationale supporting the conservatism of the bellweather principle for a large break loss of coolant accident.

Response 6:

The calculation of the leak rate during a postulated SLB event is not based on a temperature of 600°F, but on the temperature obtained from an examination of the equipment specification curves for the transient. In addition, the calculation of conditions during normal operation is based on a set of umbrella values resulting in considering primary fluid temperatures of 604.3 and 621.4°F for secondary side pressures of 782 and 947 psig respectively. Both conditions are used.
The SLB transient is assumed to initiate while the plant is at hot standby conditions, i.e., 557°F. The primary temperature drop is 139°F leading to an analysis temperature of 418°F. The secondary temperature drop is 307°F leading to a shell temperature of 260° for the analysis. The design specification pressure changes are illustrated on Figure 1 and the actual pressures for Seabrook are shown on Figure 2. The design specification temperature history during a postulated SLB event is shown on Figure 3. The superposed curves are shown on Figure 4. It is readily seen that the most limiting differential pressure, 2575 psi, occurs when the temperature is at a relatively steady state for both the hot and cold legs. The analyses were performed for the conditions at about 600 seconds into the transient and found to be less limiting than when the pressure difference is at a maximum. The H* engagement length required to resist the SLB differential pressure with a margin of 1.4 is about 0.15 inch greater for the latter condition. The B* length to restrict the SLB leak rate to a factor of 2 relative to that during NOp, based on the method of calculation, is about 7% longer for the conditions at 4200 seconds relative to those at 600 seconds. Examination of Figure 4 leads to the conclusion that the differential pressure is the controlling factor and the value at 4200 seconds will bound any value in between the two.

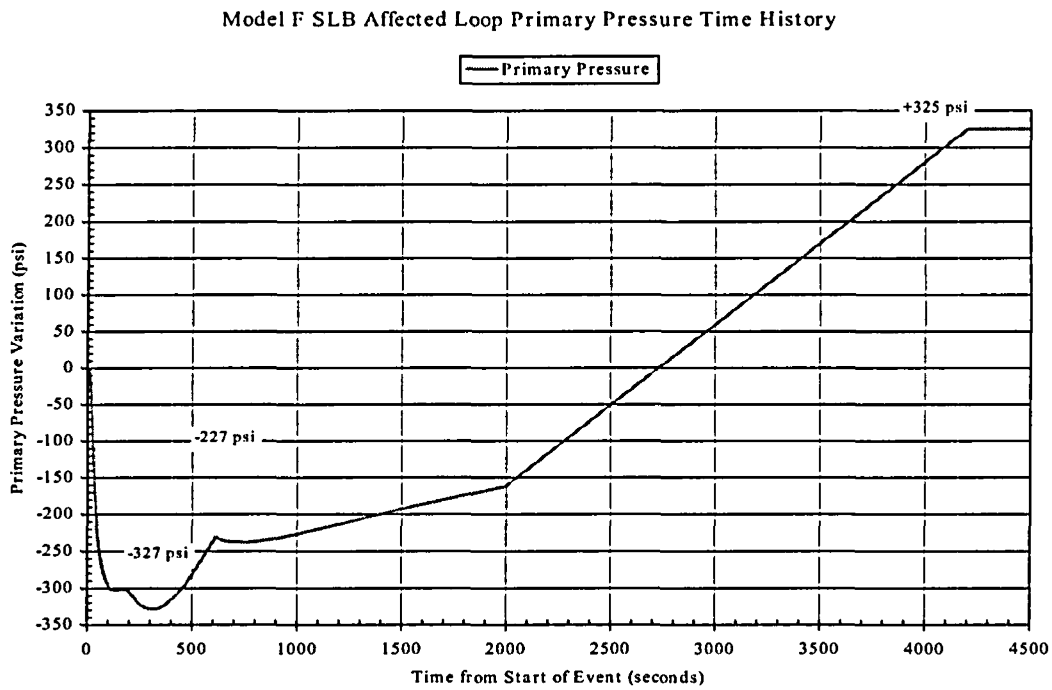

Figure 1: SLB design pressure change history.

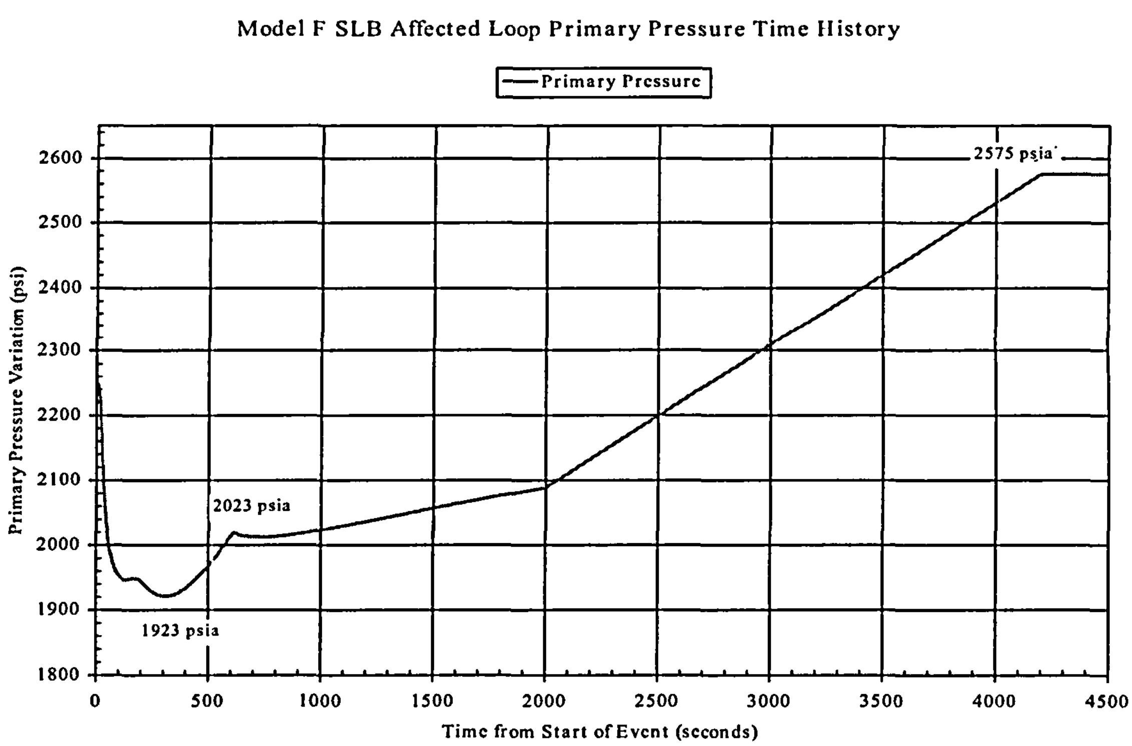

Figure 2: SLB design pressure history.

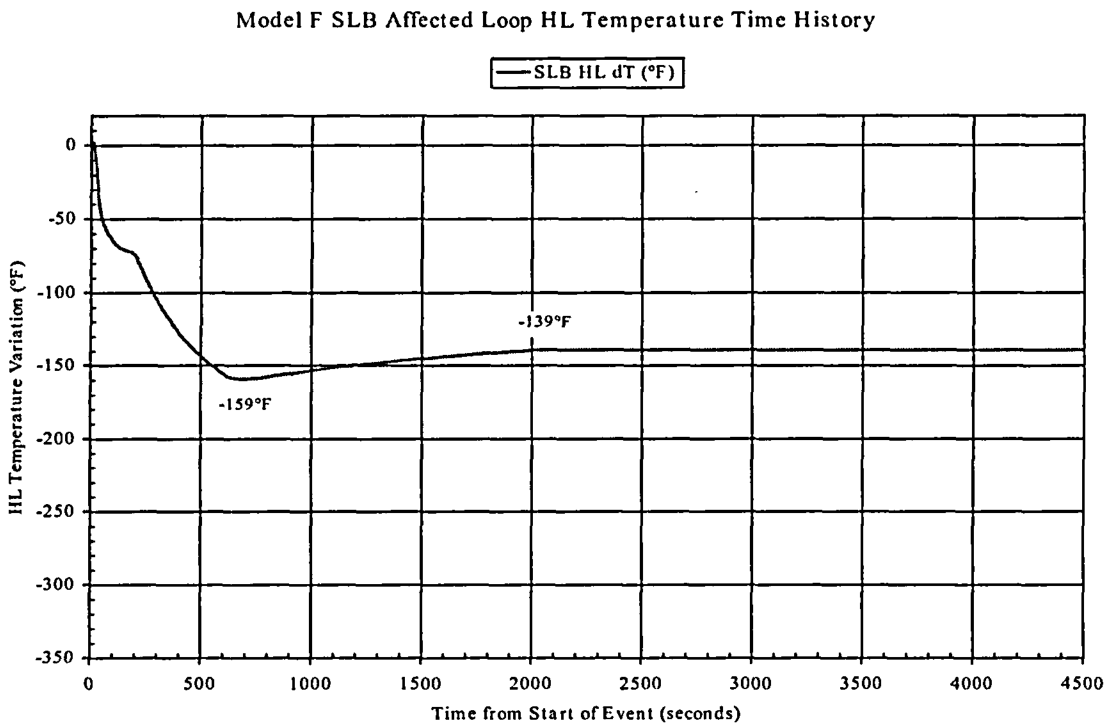

Figure 3: SLB design temperature change history.

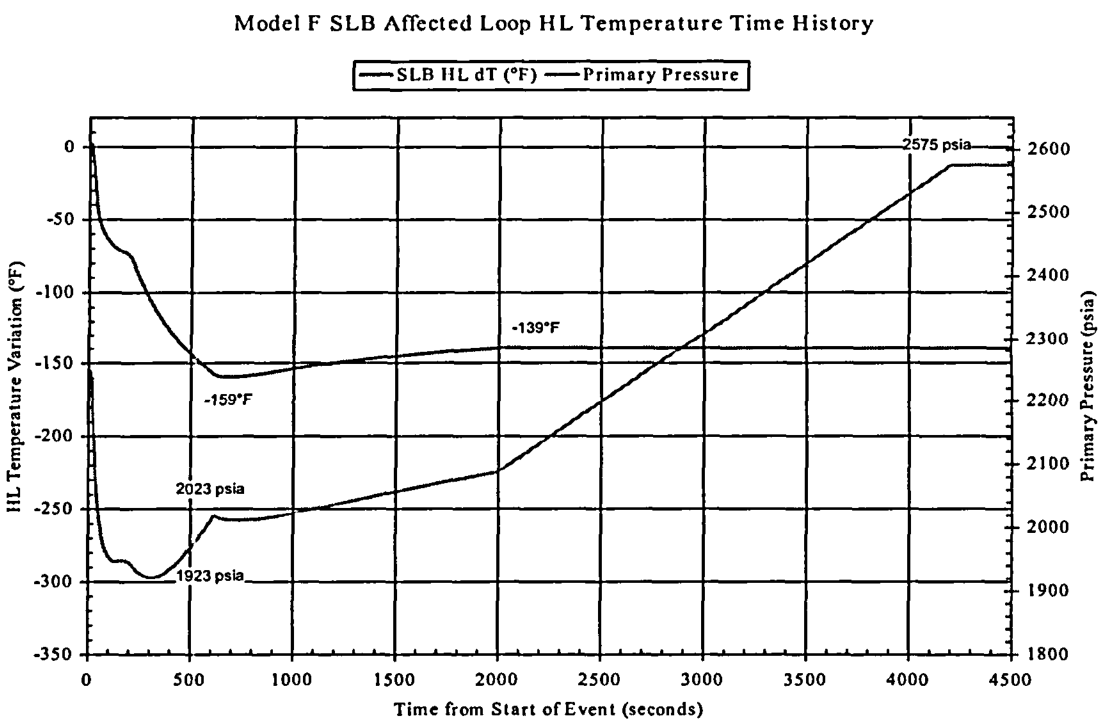

Figure 4: Superimposed SLB design pressure and temperature history.

NRC RAI 7: It is stated on page 20 of the technical attachment to Reference 2 that if the leak rate during normal operation was 0.1 gpm (150 gpd), the postulated accident condition leak rate would be 0.2 gpm (using the Darcy equation) versus the allowable limit of 0.347 gpm when only considering the change in differential pressure. In accordance with the EPRI primary to secondary leak guidelines, operating leakage versus the technical specification limit is evaluated under room temperature conditions. Presumably, this adjustment is based on water density at room temperature versus operating temperature, which means that if the operational leakage measurement is 0.1 gpm, then actual leakage under hot operating conditions is about 0.15 gpm. Assuming a factor 2 increase in leak rate during postulated accidents, the actual leak rate under hot accident conditions is 0.3 gpm which is still less than the allowable limit of 0.347 gpm. Please clarify what is the appropriate comparison to make here; 0.2 gpm (accident leakage adjusted for room temperature conditions) versus the 0.347 gpm allowable limit, or 0.3 gpm (accident leakage for actual accident temperature) versus the 0.347 allowable limit? In other words, does the accident analysis consider the 0.347 gpm allowable limit to be an adjusted value for room temperature conditions or does it treat it as a hot value?

Response 7:

Based on a review by FPL of the basis of the allowable accident leak rate, the allowable limit of 0.347 gpm is at room temperature conditions. The comparison is appropriate as written.

NRC RAI 8: The technical specification primary to secondary leakage limit is 500 gallons per day (gpd) per steam generator and 1 gallon per minute for all generators. These limits appear inappropriate since the 500 gpd limit is equal to the amount assumed in the accident analyses. If operational leakage is just below 500 gpd, the expected leakage during a postulated accident may significantly exceed the amount assumed in the MSLB accident analysis. What are your plans for submitting a proposed change to the technical specification operational leakage limit such as to ensure accident leakage will be within the amounts analyzed?

Response 8:

FPL Energy has submitted LAR 06-02, the Generic Licensing Change Package for TSTF – 449 to the NRC for approval. As part of that submittal, the operational leakage in TS 3.4.6.2 was changed from 500 gallons per day to 150 gallons per day per steam generator.

NRC RAI 9: WCAP-15932-P, Revision 1, Section 6.5 (submitted on the Callaway docket, NRC Accession No. ML022910436) provides a justification for why ligament tearing of circumferential cracks is not a significant concern. Provide a justification for why ligament tearing of axial cracks at the bottom of the tubesheet at the periphery is similarly not a significant concern.

Response 9:

One of the concerns to address when dealing with cracks in SG tubes is the potential for ligament tearing. Ligament tearing may occur during a postulated accident when the differential pressure is significantly greater than during normal operation. The approach to dealing with the question is the same as that for circumferential cracks, that is, what is the ligament that will not tear during NOp

conditions compared to the ligament that will tear during a postulated SLB event. The stress that is applied to the crack flanks during normal operation is the 2250 psi primary pressure. The stress during a SLB event is the 2560 psi pressure associated with the set point of the relief valves. The net difference is only 310 psi, hence the affect is expected to be small. The following evaluation considers the potential for ligament tearing of postulated axial cracks and to what extent such tearing would affect the technical basis for the LAR. For the depth of concern, at 17 inches from the top of the tubesheet, the contact pressure between the tube and the tubesheet, at the periphery, is never less than 800 psi for NOp or 900 psi for SLB.

The tube area required to resist tearing due to an axially oriented crack can be calculated using traditional mechanics. The axial orientation of the damage in the tube means that the required area of the tube cross section to resist tearing and damage should be based on the local strength of the material around the crack. It is conservative, in this case, to neglect the forces in the tubesheet at the periphery that would act to keep a crack closed and compress the flanks in the ligament so that tensile tearing would become unlikely. This includes the far field axial stress on the tube cross section generated by internal pressure end cap loads which would act to close the ligament and any cracks below the H* depth (the depth required to prevent tube pullout). This is in contrast to the typical method used to compare what percent of the area is required to resist ligament tearing in circumferentially damaged tubes based on the amount of force applied to the damaged tube cross section.

The results shown in the table below were obtained using the ASME code minimum material properties [1] and the nominal dimensions of the steam generator tubes in Byron Unit 2 and Braidwood Unit 2. The method of evaluation was to determine the minimum wall thickness necessary to resist tearing due to the internal pressure in each condition and then define what range of pre-crack flaw thickness would result in a tear under the accident conditions but not tear under normal conditions. [

]a,c,e The results for the calculated minimum wall thicknesses to resist tearing under either NOp or SLB at the periphery are summarized in the table below.

[ ]a,c,e

The results of the axial ligament tearing calculations are [

]a,c,e

Another check can be made by looking at the dilation of the TS hole with the change in pressure. The increase in circumference would be equal to π times the diameter change. This is roughly a change in crack opening displacement (COD) and uses a linear approximation to determine the thickness that would lead to ligament tearing, i.e., a ligament of 3 mils can withstand a COD of 3 mils. For example,

the flexibility of the tubesheet is about 6.44E-8 in/psi at 600°F. If 2250 psi acts on the inside of the hole, the circumference increases by 0.345 mils. The corresponding change is 0.392 mils for the SLB pressure, an increase of a little less than 0.05 mils. So, the probability of having cracks that are not torn during NOp, but would tear during a SLB is very small.

Considering the worst-case scenario, the likelihood of ligament tearing from axial cracks at the periphery resulting from an accident pressure increase is [

]^{a,c,e} The potential for axial ligament tearing is considered to be a secondary effect of essentially negligible probability and is not expected to affect the results and conclusions reported for the B* evaluation. The leak rate model does not include provisions for predicting ligament tearing and subsequent leakage. Increasing the complexity of the model to attempt to account for axial or circumferential ligament tearing is not considered necessary.

Axial Ligament Tearing as a Function of Tube Radius

Axial ligament tearing as a function of tube radius is considered in Reference 4. The results of the analysis of axial ligament tearing do not change significantly as a function of tube radius. Table 1, below, summarizes the results for tube diameters ranging from 0.50 to 1.00 inch.

Table 1: Summary of Minimum Ligament Thickness as a Function of Tube Diameter

a,c,e

[ ]

The change in minimum ligament thickness, Δt_m, ranges from 0.34 to 0.35 mils for tube diameters of 0.50 and 1.00 inch, respectively. These results indicate that it is reasonable to assume that for the above axial ligament tearing calculations the bellwether approach applies and is valid for a variety of tube sizes.

5. ASME Boiler and Pressure Vessel Code Section III, Rules for Construction of Nuclear Power Plant Components," 1989 Edition, The American Society of Mechanical Engineers, New York, NY.

6. "Improved Justification of Partial Length RPC Inspection of Tube Joints of Model F Steam Generators of Ameren-UE Callaway Plant", WCAP-15932, Revision 1, May 2003.

7. EPRI 1001191; Steam Generator Degradation Specific Management Flaw Handbook, 2006.

8. LTR-SGDA-06-108; "Data and Analysis Methodology in Support of Axial Ligament Tearing Model," 6/29/06

NRC RAI 10: Are there any tubes in the Seabrook steam generators which were not fully expanded (per nominal) within the tubesheet? If so, please describe the extent of this condition and justify why the amendment request is sufficient to ensure the structural and leakage integrity of the affected tube joints.

Response provided is specific to Seabrook:
(Note: A response is provided for Braidwood 2 in the appended material.)

Specific Responses for EXELON:

Draft RAI #7, received by EXELON on its License Amendment Request is addressed in the following:

NRC Question:

Provide copy of WCAP-16152, which is listed as Reference 5 in Attachment 7.

Response:

WCAP 16152, Reference 5 in Attachment 7 provides no additional technical details than those already included in the technical attachment to Attachment 7. It is referenced for historical perspective only, that is, to document the historical evolution of the H* concept. The requested report is essentially the same as other H* reports that the staff has received and partially reviewed in the past (e.g., WCAP-15932-P). The information from WCAP-16152 required for the technical justification attached to Attachment 7 has been included in the technical justification provided. Indeed, there are very few references to WCAP-16152 in the technical justification provided; these are addressed below with the rationale why WCAP-16152 will not provided additional information.
In the following discussion WCAP-16152 is referred to as "the WCAP" and the Attachment 7 is referred to as the "technical justification".

Section 2.0: Summary Discussion

- A reference to the WCAP is made in the third paragraph from the end of the section in a discussion of flow resistance associated with an increase in the tube-to-tubesheet contact pressure. Figure 6-6 in the technical justification provides the loss coefficient vs. contact pressure data that supports this discussion. No additional data is provided by WCAP-16152.
- A reference to the WCAP is made in the second paragraph from the end of the section in a discussion regarding the minimum inspection distance that would support the structural requirements (i.e., to prevent pullout of the tube. The structural analysis of the tube-to-tubesheet joint is provided in section 7 of the technical justification. Table 7-14 provides a summary table of the tube pullout lengths as the limiting value in each of three zones. The WCAP provides no additional information than that presented in Section 7 of the technical justification provided with the LAR.
- A reference is made to the WCAP in the last paragraph of Section 2.0, regarding the necessary inspection lengths to assure structural integrity (prevent tube pullout). The reference to the WCAP is to the structural analysis; the same analysis is included in section 7 of the technical justification noted above.

Section 6.0 Steam Generator Tube Leakage and Pullout Test Program Discussion

- The WCAP is referenced in the introductory statement of Section 6, noting that a test program for leakage and pullout had been performed and was documented in the WCAP. The remainder of Section 6 of the technical justification then provides a detailed discussion of the test program and its results. Therefore, the WCAP provides no additional information not already provided in the technical justification.

Section 7.2 Determination of Required Engagement Length of the Tube in the Tubesheet

- The WCAP is referenced in the first paragraph of this section, and is an inference to the philosophy of the WCAP, i.e., that exempting any part of the tubesheet expansion zone from inspection depends on the structural analysis of the interfacial contact pressures. Section 7.2 present the details of the same structural analysis contained in the WCAP. Therefore, the WCAP provides no additional information.

Section 10 Conclusions and Inspection Recommendations

- Item 1 under the paragraph which begins with "The recommendations with regard to the inspection of the welds at Braidwood 2 and Byron 2 are based on the following:" references the WCAP. This reference is to the structural analysis contained in the WCAP but also contained in the technical justification. Since the complete structural analysis is included in the technical justification, the WCAP provides no additional information.

Section 11.0 References

- Reference 5 is WCAP 16152. This reference is provided only for historical purposes. As shown by the discussions above, complete information is contained in the technical justification and no additional data is provided in the WCAP. However, the WCAP does contain information not relevant to the current technical justification which the NRC has previously commented on in their review of a like WCAP (WCAP-15932-P) which could mislead review of the current technical justification. The current technical justification is a freestanding report, independent of WCAP-16152.

Draft RAI #10, received by EXELON on its License Amendment Request is addressed in the following:

(Note: The response is limited to Braidwood 2 based on discussions with the steam generator engineers for both Byron 2 and Braidwood 2. The complete response to this question will be provided by EXELON).

NRC Question:

Are there any tubes in the Byron/Braidwood Unit 2 steam generators which were not fully expanded (per nominal) within the tubesheet? If so, please describe the extent of this conditions and justify why the amendment request is sufficient to ensure that the structural and leakage integrity of the affected joints.

Response:

The EC data reports for Braidwood 2 were reviewed for the presence of the codes NTE and PTE. NTE indicates that the tube has not been expanded into the tubesheet. PTE indicates that the tube has only partially (in length) been expanded into the tubesheet. None of these codes was found in the results for any inspection; thus, it is concluded that there are no unexpanded or partially expanded tubes in the Braidwood 2 steam generators.