FPL Energy Point Beach, LLC, 6610 Nuclear Road, Two Rivers, WI 54241

FPL Energy
Point Beach Nuclear Plant

May 28, 2008

NRC 2008-0034
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Point Beach Nuclear Plant, Unit 1
Docket 50-266
Renewed License No. DPR-24

License Amendment Request 257
Technical Specifications 5.5.8 and 5.6.8,
Steam Generator Program & Steam Generator Tube Inspection Report
Interim Alternate Repair Criteria (IARC) for Steam Generator Tube Repair

References

(1) Wolf Creek Nuclear Operating Corporation to NRC Letter dated February 8, 2008, “Docket No. 50-482: Revision to Technical Specification (TS) 5.5.9, ‘Steam Generator (SG) Program’ for Interim Alternate Repair Criteria” (ML080440099)
(2) Vogtle Electric Generating Station to NRC Letter dated February 13,2008, “License Amendment Request to Revise Technical Specification (TS) Sections 5.5.9, “Steam Generator (SG) Program” and TS 5.6.10, “Steam Generator Tube Inspection Report,” for Interim Alternate Repair Criterion” (ML080500223)
(3) Exelon to NRC Letter dated February 25, 2008, “Application for Steam Generator Tube Interim Alternate Repair Criteria Technical Specification Amendment” (ML080560512)
(4) Wolf Creek Nuclear Operating Corporation to NRC Letter dated March 21, 2008, “Docket No. 50-482: Response to Request for Additional Information Related to License Amendment Request for an Interim Alternate Repair Criterion to Technical Specification 5.5.9, Steam Generator (SG) Program” (ML080860248)
(5) Vogtle Electric Generating Plant to NRC Letter dated March 21, 2008, “Response to Request for Additional Information Related to License Amendment Request to Revise Technical Specification (TS) “Steam Generator Tube Inspection Report” for Interim Alternate Repair Criterion” (ML080850256)
(6) Exelon to NRC Letter dated March 27, 2008, “Response to Request for Additional Information Regarding Application for Steam Generator Tube Interim Alternate Repair Criteria Technical Specification” (ML080880057)

In accordance with the provisions of 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit,” FPL Energy Point Beach, LLC is submitting a request for an amendment to the Point Beach Nuclear Plant (PBNP) Technical Specifications (TS) for Facility Operating License DPR-24.

This amendment proposes a one cycle revision to the PBNP TS. Specifically, TS 5.5.8, "Steam Generator (SG) Program," and TS 5.6.8, "Steam Generator Tube Inspection Report," will be revised to incorporate an interim alternate repair criterion into the provisions for SG tube repair for use during the PBNP Unit 1 2008 fall refueling outage (U1R31) and the subsequent operating cycle. The amendment reflects recent industry efforts, including the May 14, 2008, meeting between the Commission and the industry, to resolve technical issues associated with the interim alternate repair criterion.

This license amendment request is based upon similar requests submitted by Wolf Creek, dated February 8, 2008, (Reference 1) Vogtle 1 and 2, dated February 13, 2008 (Reference 2) and Braidwood 1 and 2, dated February 25, 2008 (Reference 3). As part of their review of the three submittals, the NRC issued requests for additional information (RAIs) which included, in aggregate, 17 questions. The plants drafted the responses to Questions 1-5 and Westinghouse Electric Company LLC developed responses to Questions 6-17. These RAI responses were submitted to the NRC by Wolf Creek on March 21, 2008, (Reference 4) Vogtle 1 and 2 on March 21, 2008, (Reference 5) and Braidwood 1 and 2 on March 27, 2008 (Reference 6). These RAI responses have been incorporated in this license amendment request.

Enclosure 1 provides the discussion of the proposed change. Enclosure 2 provides the marked-up versions of the proposed TS pages.

Enclosure 3 contains, "Response to NRC Request for Additional Information relating to LTR-CDME-08-11," dated April 29, 2008 (Proprietary), provided by Westinghouse, This response contains the proprietary version of the response to the NRC RAI 6 through 17 on an interim alternate repair criterion (IARC) that requires full-length inspection of the steam generator tubes within the tubesheet but does not require plugging tubes with a certain arc length of circumferential cracking below 17 inches from the top of the tubesheet. This information is supported by affidavits, signed by Westinghouse, the owner of the information, is contained in Enclosure 5. The affidavits set forth the bases 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 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations. The affidavits are included in Westinghouse authorization letters CAW-08-2423 and CAW-08-2424, "Application for Withholding Proprietary Information from Public Disclosure, " which also includes Proprietary Information Notices and Copyright Notices.

Correspondence with respect to the copyright or proprietary aspects of the Westinghouse information noted above or the supporting Westinghouse affidavits should reference the applicable authorization letter 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. Redacted, non-proprietary versions of the Westinghouse supporting documentation are provided in Enclosure 4.

FPL Energy Point Beach has evaluated the proposed amendment and has determined that it does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for this determination is included in Enclosure 1. FPL Energy Point Beach has also determined that operation with the proposed change will not result in any significant increase in the amount of effluents that may be released offsite and no significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change. The proposed amendment has been reviewed by the Plant Operations Review Committee.

FPL Energy Point Beach requests approval of the proposed license amendment by October 1, 2008, to support the fall PBNP Unit 1 refueling outage, which is currently scheduled to start in October 2008. Once approved, the amendment will be implemented prior to entering MODE 4 during startup of PBNP Unit 1 from the refueling outage.

FPL Energy Point Beach continues to remain engaged in industry activities associated with steam generator tube integrity and alternate repair criteria, both interim and permanent, for plants with thermally treated Alloy 600 tubes.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on May 28, 2008.

Very truly yours,

FPL ENERGY POINT BEACH, LLC

James H. McCarthy
Site Vice President

Enclosures

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

ENCLOSURE 1

FPL ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNIT 1

LICENSE AMENDMENT REQUEST 257
PROPOSED LICENSE AMENDMENT REQUEST

INTERIM ALTERNATE REPAIR CRITERIA (IARC)
FOR STEAM GENERATOR TUBE REPAIR

1.0 SUMMARY DESCRIPTION

This amendment proposes a one cycle revision to the Point Beach Nuclear Plant (PBNP) Technical Specifications (TS) 5.5.8, "Steam Generator (SG) Program," and TS 5.6.8, "Steam Generator Tube Inspection Report," to incorporate an interim alternate repair criterion (IARC) into the provisions for SG tube repair criteria for use during the PBNP 2008 fall refueling outage and the subsequent operating cycle. This amendment application requests approval of an IARC that requires full-length inspection of the tubes within the tubesheet but does not require plugging tubes if any circumferential cracking observed in the region greater than 17 inches from the top of the tubesheet (TTS) is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads. This amendment application is required to preclude unnecessary SG tube plugging while still maintaining tube structural and leakage integrity.

2.0 DETAILED DESCRIPTION

2.1 Proposed Change

The following specific changes to the PBNP TS are proposed:

TS 5.5.8 – Steam Generator (SG) Program

- TS 5.5.8.c currently states:

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

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

  1. For Unit 1 Refueling Outage 30 and the subsequent operating cycle, flaws found in the portion of the tube below 17 inches from the top of the hot leg tubesheet do not require plugging. All tubes with flaws identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged. This alternate tube repair criteria is not applicable to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet.

The criterion would be revised as follows:

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

  1. For Unit 1 Refueling Outage 31 and the subsequent operating cycle, tubes with flaws having a circumferential component less than or equal to 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet do not require plugging. Tubes with flaws having a circumferential component greater than 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet

and above 1 inch from the bottom of the tubesheet shall be removed from service.

Tubes with service-induced flaws located within the region from the top of the tubesheet to 17 inches below the top of the tubesheet shall be removed from service. Tubes with service-induced axial cracks found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging.

When more than one flaw with circumferential components is found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet with the total of the circumferential components greater than 203 degrees and an axial separation distance of less than 1 inch, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet, and the total of the circumferential components found in the tube exceeds 94 degrees, then the tube shall be removed from service. When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet and within 1 inch axial separation distance of a flaw above 1 inch from the bottom of the tubesheet, and the total of the circumferential components found in the tube exceed 94 degrees, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

This alternate tube repair criteria is not applicable to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet.

- TS 5.5.8.d currently states:

  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 1 Refueling Outage 30 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded when the alternate repair criteria in TS 5.5.8.c are implemented. This exclusion does not apply to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet. 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.

The provisions stated in d. would be revised as follows:

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and 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.

<u>TS 5.6.8 – Steam Generator Tube Inspection Report</u>

- TS 5.6.8 currently states:

  A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with TS 5.5.8, Steam Generator (SG) Program. The report shall include:

  a. The scope of inspections performed on each SG,

  b. Active degradation mechanisms found,

  c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

  h. The effective plugging percentage for all plugging in each SG.

TS 5.6.8 would be revised to add the following three additional reporting criteria:

i. Following completion of an inspection performed in Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, whether initiated on primary or secondary side for each service-induced flaw within the thickness of the tubesheet, and the total of the circumferential components and any circumferential overlap below 17 inches from the top of the tubesheet as determined in accordance with TS 5.5.8,

j. Following completion of an inspection performed in Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the primary to secondary LEAKAGE rate observed in each steam generator (if it is not practical to assign leakage to an individual SG, 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

k. Following completion of an inspection performed in Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the portion of the tube below 17 inches from the top of the tubesheet for the most limiting accident in the most limiting steam generator.

2.2 Background

TS 5.5.8 requires that a SG tube program be established and implemented to ensure that SG tube integrity is maintained. SG tube integrity is maintained by meeting specified performance criteria (in TS 5.5.8) for structural and leakage integrity, consistent with the plant design and licensing bases. TS 5.5.8 requires a condition monitoring assessment be performed during each outage during which the SG tubes are inspected to confirm that the performance criteria are being met. TS 5.5.8 also includes provisions regarding the scope, frequency, and methods of SG tube inspections. Of relevance to the amendment application, these provisions require that the number and portions of tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type that may be present along the length of a tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The applicable tube repair criteria, specified in TS 5.5.8, are that tubes found by an inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal tube wall thickness shall be plugged.

Reference 2 provides the technical justification for an IARC that requires full-length inspection of the tubes within the tubesheet, but does not require plugging tubes if the extent of any circumferential cracking observed in the region greater than 17 inches from the top of tubesheet (TTS) is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads [the greater of 3 times the normal operating (NOP) loads or 1.4 times the steam line break (SLB) end cap loads]. Axial cracks below 17 inches from the TTS are not relevant to the tube pullout arguments because axial cracks do not degrade the axial load carrying capability of the tube. Axial cracks do not require plugging if they are below 17 inches from the TTS.

The limiting circumferential ligament has been defined by calculation. The calculation assumes that friction loads between the tube and tubesheet from any source are zero. This assumption avoids potential effects of uncertainties in tube and tubesheet material properties.

Also, based on the same assumption that the contact pressure between the tube and the tubesheet from any source is zero, this evaluation provides a basis for demonstrating that the accident induced leakage will always meet the value assumed in the plant's safety analysis if the observed leakage during normal operating conditions is within its allowable limits. The need to calculate leakage from individual cracks is avoided by the calculation of the ratio of accident induced leakage to normal operating leakage.

3.0 TECHNICAL EVALUATION

An evaluation has been performed in Reference 2 and through responses in Reference 4 to assess the need for removing tubes from service due to the occurrence of circumferentially or axially oriented cracks in a tubesheet. The conclusions of the evaluation are primarily:

1. Axial cracks in tubes below a distance of 17 inches below the TTS can remain in service in the PBNP SGs as they are not a concern relative to tube pullout and leakage capability.

2. Circumferentially oriented cracks in tubes below a distance of 17 inches below the TTS with an azimuthal extent of less than or equal to 203 degrees can remain in service for one cycle of operation (18-month SG tubing eddy current inspection interval).

3. Circumferentially oriented cracks in the bottom 1-inch of the tube or in the tube-to-tubesheet welds with an azimuthal extent of less than or equal to 94 degrees can remain in service for one cycle of operation (18-month SG tubing eddy current inspection interval).

A bounding analysis approach is utilized for both the minimum ligament calculation and the leakage ratio calculation. "Bounding" means that the most challenging conditions from the plants with hydraulically expanded Alloy 600TT tubing are used. Three different tube diameters are represented by the affected plants (11/16" diameter, Model F; 3/4" diameter, Model D5; 7/8" diameter, Model 44F). PBNP Unit 1 SGs are Model 44F. The most limiting conditions for structural evaluation depend on tube geometry and applied normal operating loads; thus the conditions from the plant that result in the highest stress in the tube are used to define the minimum required circumferential ligament. The limiting leak rate ratio depends on the leak rate values assumed in the safety analysis and allowable normal operating leakage that results in the longest length of undegraded tube.

Questions Relating to Interim Alternate Repair Criteria for Steam Generator Tubes

This license amendment request is based upon similar requests submitted by Wolf Creek, dated February 8, 2008 (Reference 1), Vogtle 1 and 2, dated February 13, 2008 (Reference 15) and Braidwood 1 and 2, dated February 25, 2008 (Reference 19). As part of their review of the three submittals, the NRC issued requests for additional information (RAIs) which included, in aggregate, 17 questions. The corporations drafted the responses to Questions 1-5 and Westinghouse developed responses to Questions 6-17. These RAI responses were submitted to the NRC by Wolf Creek on March 21, 2008 (Reference 3), Vogtle 1 and 2 on March 21, 2008 (Reference 14), and Braidwood 1 and 2 on March 27, 2008 (Reference 18). These RAI responses have been incorporated into this FPL Energy Point Beach license amendment request.

Discussion of Performance Criteria

The following performance criteria of NEI 97-06, Revision 2, "Steam Generator Program Guidelines," dated May 2005 (Reference 6), which are included in the PBNP TS, are the basis for these analyses:

The structural integrity performance criterion is:

All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

The structural performance criterion is based on ensuring that there is reasonable assurance that a steam generator tube will not burst during normal operation or postulated accident conditions.

The accident-induced leakage performance criterion as stated in PBNP TS 5.5.8.b.2 is:

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 steam generator. Leakage is not to exceed 500 gallons per day per SG.

Primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a limiting design basis accident. The potential primary-to-secondary leak rate during postulated design basis accidents shall not exceed the offsite radiological dose consequences required by 10 CFR Part 100 guidelines or the radiological consequences to control room personnel required by GDC-19.

The IARC for the tubesheet region are designed to meet these criteria. The structural criterion regarding tube burst is inherently satisfied because the constraint provided by the tubesheet to the tube prohibits burst.

Limiting Structural Ligament Discussion

As defined in Reference 2 and revised through Reference 4, the bounding remaining structural ligament which meets the NEI 97-06, Revision 2, Performance Criterion described above and required for the tube to transmit the operational loads is 126 degrees arc. This assumes that the residual ligament is 100% of the tube wall in depth. A small circumferential initiating crack is predicted to grow to a through-wall condition before it is predicted to reach a limiting residual ligament. A residual ligament in a part-through-wall condition is not a significant concern, because of the assumption that all circumferential cracks detected are 100% through-wall.

Consideration of Non Destructive Examination (NDE) Uncertainty

The NDE uncertainty must be addressed to assure that the as-indicated circumferential arc of the reported crack is a reliable estimate of the actual crack. ETSS 20510.1 (Reference 7) describes the qualified technique used to detect circumferential primary water stress corrosion cracking (PWSCC) in the expansion transitions and in the tubesheet expansion zone (TEZ). The qualification data is provided in the ETSS.

The fundamental assumption for the IARC is that all circumferential cracks detected are 100% through-wall. Thus, even a shallow crack of small length will be considered to be through-wall. Further, tube burst is not an issue for the IARC because of the constraint provided by the tubesheet; rather, it is axial separation of the tube that is the principal concern. Assuming that all circumferential cracks are through-wall reduces the inspection uncertainty to length of the cracks only. Further, the accuracy of the length determination is an issue only when the indicated crack approaches the allowable crack length (the complement of the required residual ligament) and if the indicated crack length is a reasonable estimate of the structural condition of the tube.

Prior investigations have correlated the axial strength of the tube to the percent degraded area (PDA) of the flaw (Reference 8). PDA takes into account the profile of the existing crack, including non-through-wall portions and shallow tails of the crack. Using the data from ETSS 20510.1 for cracks with a 90%, or greater, through-wall condition from both NDE and destructive examination, a comparison of the actual crack lengths and corresponding PDA for the cracks to a theoretical PDA which assumes that cracks are 100% through-wall has been made. All of the points with a PDA of 60% or greater fall below the theoretical PDA line. As the crack lengths increase, the separation of the actual PDA from the theoretical PDA tends to increase.

The conclusion that the as-indicated crack angle is conservative is further supported by considering the characteristics of the eddy current probes. Each probe has a "field of view," that is, a window of finite dimension in which it detects flaws. Therefore, as the probe traverses its path, a flaw will be detected as the leading edge of the field of view first crosses the location of the flaw, continuing until the trailing edge of the field of view passes the opposite end of the flaw. This is known as "lead-in" and "lead-out" of the probe and the effect of these are to render the indicated flaw length greater than the actual flaw length. Therefore, it is concluded that the indicated flaw length will be conservative relative to the actual flaw length, especially when it is assumed that the entire length of the indicated flaw is 100% through-wall.

Based on the above, it is concluded that if the detected circumferential cracks are assumed to be 100% through-wall, the as-indicated crack lengths will be inherently conservative with respect to the structural adequacy of the remaining ligament. Therefore, no additional uncertainty factor is necessary to be applied to the as-measured circumferential extent of the cracks.

Consideration of Crack Growth

The growth of cracks due to PWSCC in this submittal request is dictated by four default growth rates from Reference 2. The distribution of growth rates is assumed to be lognormal. Typical values and conservative values are given, although it is recommended in Reference 9 to use the default values only when the historical information is not available and not to use the typical values unless the degradation is mild. (No significant crack growth data exists for the circumferential cracking in the tubesheet expansion region). Both sets provided in Reference 2 have mean values and 95% upper bound values. For this analysis, the typical 95% upper bound growth rate is used. The circumferential growth rates are expressed as inches per effective full power year (EFPY).

Table 1
Calculation of Required Minimum Ligament for
18-Month Operating Period

	Bounding Structural Ligament	EFPY (1)	Growth (In./EFPY) (2)	Growth (Deg./EFPY) (3)	Growth for Operating Period (degrees)	Minimum Structural Ligament (degrees)	Critical Ligament (degrees)
Tube	18-Calendar Month (CM) Operation	1.5	0.12	20.65	31	115	146
1) It is conservatively assumed that 1 EFPY= 1 Calendar Year 2) 95% upper value of typical growth rates from Reference 2 3) Based on smallest (Model F) mean tubesheet bore dimension							

The residual structural ligament must be adjusted for growth during the anticipated operating period between the current and the next planned inspection. For the PBNP Model 44F SGs, referring to Table 1 above, the maximum allowable through-wall circumferential crack size in a SG tube is 214° (=360° – 146°) for one cycle of operation (18-month SG tubing eddy current inspection interval).

(The maximum allowable through-wall circumferential crack size in a SG tube was reduced to 203° in the response to RAI Question 17 in Reference 4 by increasing the minimum structural ligament to 126°. The total critical ligament was increased to 157°. Thus, the total maximum allowable through-wall circumferential crack size is 203° (=360° - 157) for one cycle of operation.)

Primary-to-Secondary Leakage Discussion

A basis, using the D'Arcy formula for flow through a porous medium, is provided to assure that the accident induced leakage for the limiting accident will not exceed the value assumed in the safety analysis for the plant if the observed leakage during normal operation is within its limits for the bounding plant is discussed in Reference 2. The bounding plant envelopes all plants who are candidates for applying H*/B*. The D'Arcy formulation was previously compared to other potential models such as the Bernoulli equation or orifice flow formulation and was found to provide the most conservative results.

Assuming zero contact pressure in the tube joint, the length of undegraded crevice required to limit the accident induced leakage to less than the value assumed in the safety analysis for the limiting plant is calculated to be 3.78 inches. By definition of the IARC, a tube that can remain in service has an undegraded crevice of 17 inches. Therefore, a safety factor of 4.5 is available (17 inches /3.78 inches). Expressed in length terms, the length margin in the crevice is 13.22 inches.

Significant margin on crevice length is available even if only the distance below the neutral axis of the tubesheet is considered. This distance is approximately 6.5 inches. A factor of safety of 1.72 is available. Expressed in length terms, the length margin in the crevice is 2.72 inches below the neutral axis of the tubesheet. During normal operating conditions, the tubesheet flexes due to differential pressure loads, causing the tubesheet holes above the neutral axis to dilate, and below the neutral axis, to constrict. No mechanical benefit is assumed in the analysis due to tubesheet bore constriction below the neutral axis of the tubesheet; however, first principles dictate that the tubesheet bore and crevice must decrease. Therefore, the leakage analysis provided is conservative.

Based on the above, with a length of undegraded crevice of 17 inches, it is concluded that if the normal operating leakage is within its allowable value, the accident induced leakage will also be within the value assumed in the PBNP safety analysis. The total increase in leakage during a postulated accident condition would be less than a factor of 3.5 (0.35 gpm allowable leakage during a SLB event / 0.1 gpm allowable leakage during normal operating conditions).

For integrity assessments, the ratio of 2.5 will be used in completion of both the condition monitoring (CM) and operational assessment (OA) upon implementation of the IARC. For example, for the CM assessment, the component of leakage from the lower 4 inches for the most limiting steam generator during the prior cycle of operation will be multiplied by a factor of 2.5 and added to the total leakage from any other source and compared to the allowable accident analysis leakage assumption. For the OA, the difference in leakage from the allowable limit during the limiting design basis accident minus the leakage from the other sources will be divided by 2.5 and compared to the observed leakage. An administrative limit will be established to not exceed the calculated value.

Reporting Requirements

FPL Energy Point Beach proposes to report the following additional information associated with the IARC following the Fall 2008 inspections and any additional inspections during the subsequent operating cycle:

- The number of indications and location, size, orientation, whether initiated on primary or secondary side for each service-induced flaw within the thickness of the tubesheet, and the total of the circumferential components and any circumferential overlap below 17 inches from TTS.

- The primary-to-secondary leakage rate observed in each SG (if it is not practical to assign leakage to an individual SG, the entire primary-to-secondary leakage should be conservatively assumed to be from one SG) during this cycle preceding the inspection which is the subject of the report.

- The calculated accident leakage rate from the portion of tube below 17 inches from TTS for the most limiting accident in the most limiting SG. A factor of 2.5 shall be used to relate this accident leakage to the related operational leakage.

The proposed reporting requirements are only required for the applicable period of the IARC.

Inspection and Repair of Tube

The tube below the IARC depth will be examined with a qualified technique, e.g., +Point probe. Axial flaws have no impact on the structural integrity of the tube in this region and may be left in service. Circumferential indications that exceed the maximum acceptable tube flaw size of 203 degrees will be plugged. The detection of flaws will result in sample expansion per EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines" (Reference 20). Stress concentration areas may be used to define the extent of the expansion, e.g., if a repairable indication is located in a bulge/overexpansion (BLG/OXP), the expansion may be limited to the non-inspected BLG/OXPs. The circumferential components of multiple flaws within 1 inch of each other axially will be combined in accordance with TS 5.5.8. Furthermore, the circumferential component of flaws within the bottom 1 inch of the SG tubes is limited to 94 degrees.

4.0 Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

Steam Generator (SG) tube inspection and repair limits are specified in Section 5.5.8, "Steam Generator (SG) Program" of the PBNP Technical Specifications (TS). The current TS require that flawed tubes be repaired if the depths of the flaws are greater than or equal to 40 percent through-wall. The TS repair limits ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with PBNP General Design Criteria (GDC) 9, 33, 34,and 36.

PBNP was licensed prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 50. As such, PBNP is not licensed to the Appendix A GDC. The PBNP Final Safety Analysis Report (FSAR) lists the plant-specific GDC to which the plant was licensed. The PBNP GDC are similar in content to the draft GDC proposed for public comment in 1967. The PBNP GDC addressing the reactor coolant pressure boundary are PBNP GDC 9 (Reactor Coolant Pressure Boundary), GDC 33 (Reactor Coolant Pressure Boundary Capability), GDC 34 (Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention), and GDC 36 (Reactor Coolant Pressure Boundary Surveillance). The applicable criteria for this system are discussed in FSAR Section 4.1, "Reactor Coolant System - Design Basis."

PBNP GDC 9, 33, 34, and 36 require, in part that the reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime; be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component; be designed and operated to reduce to an acceptable level the probability of rapidly propagating. The PBNP GDC are similar to Appendix A GDC 14, 15, 31, and 32.

10 CFR 50.36(c)(5) states that, "Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." The technical analysis performed by FPL Energy Point Beach concludes that the proposed changes to TS 5.5.8 will continue to provide the appropriate procedural and program controls for inservice testing and steam generator tube surveillance.

10 CFR 50.55a specifies that components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). Section 50.55a further requires, in part, that throughout the service life of a pressurized water reactor facility, ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination requirements, in Section XI, "Rules for Inservice Inspection [ISI] of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code.

The tube repair limits in the TSs were developed with the intent of ensuring that degraded tubes (1) maintain factors of safety against gross rupture consistent with the plant design basis (i.e., consistent with the stress limits of the ASME Code, Section III) and (2) maintain leakage integrity consistent with the plant licensing basis while, at the same time, allowing for potential flaw size measurement error and flaw growth between SG inspections.

FPL Energy Point Beach concludes that the proposed changes are in accordance with 10 CFR 50.36(c)(5) with regards to maintaining the necessary procedural and program controls to assure operation of the facility in a safe manner. These changes also continue to meet the applicable requirements of 10 CFR 50.55a. The proposed changes thus continue to be compliant with the above regulatory requirements.

The technical evaluation in Section 3.0 above concludes that the proposed changes to TS 5.5.8 and 5.6.8 will continue to assure that the design requirements of the reactor coolant pressure boundary are met. The proposed changes will not adversely affect the other requirements of these criteria.

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public with the implementation of the IARC discussed above.

4.2 No Significant Hazards Consideration

FPL Energy Point Beach has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

(1) <u>Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?</u>

Response: No

Of the various accidents previously evaluated, the proposed changes only affect the steam generator tube rupture (SGTR) event evaluation and the postulated steam line break (SLB), locked rotor, and control rod ejection accident evaluations. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Model F steam generators has shown that axial loading of the tubes is negligible during an SSE.

At normal operating pressures, leakage from PWSCC below 17 inches from the TTS is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

For the SGTR event, the required structural margins of the steam generator tubes is maintained by limiting the allowable ligament size for a circumferential crack to remain in service to 203 degrees below 17 inches from the TTS for the subsequent operating cycle. Tube rupture is precluded for cracks in the hydraulic expansion region due to the constraint provided by the tubesheet. The potential for tube pullout is mitigated by limiting the allowable crack size to 203 degrees subsequent operating cycle. These allowable crack sizes take into account eddy current uncertainty and crack growth rate. It has been shown that a circumferential crack with an azimuthal extent of 203 degrees for the 18-month SG tubing eddy current inspection interval meets the performance criteria of NEI 97-06, Rev. 2, "Steam Generator Program Guidelines" and Draft Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." Therefore, the margin against tube burst/pullout is maintained during normal and postulated accident conditions and the proposed change does not result in a significant increase in the probability or consequence of a SGTR.

The probability of a SLB is unaffected by the potential failure of a SG tube as the failure of a tube is not an initiator for a SLB event. SLB leakage is limited by leakage flow restrictions resulting from the leakage path above potential cracks through the tube-to-tubesheet crevice. The leak rate during postulated accident conditions (including locked rotor) has been shown to remain within the accident analysis assumptions for all axial or circumferentially oriented cracks occurring 17 inches below the top of the tubesheet. Since normal operating leakage is limited to 150 gpd (approximately 0.10 gpm), the attendant accident condition leak rate, assuming all leakage to be from indications below 17 inches from the top of the tubesheet would be bounded by 500 gpd (approximately 0.35 gpm). This value is within the accident analysis assumptions for the design basis accident for PBNP.

Based on the above, the performance criteria of NEI-97-06, Rev. 2 and Draft Regulatory Guide (RG) 1.121 continue to be met and the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) <u>Does the proposed change create the possibility of a new or different accident from any accident previously evaluated?</u>

Response: No

The proposed change does not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the interim alternate repair criteria. The proposed change does not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

Therefore, based on the above evaluation, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) <u>Does the proposed change involve a significant reduction in a margin of safety?</u>

Response: No

The proposed change maintains the required structural margins of the steam generator tubes for both normal and accident conditions. NEI 97-06, Revision 2 and RG 1.121 are used as the basis in the development of the limited tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting GDC 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. PBNP GDC 9, 33, 31, 34, and 36 are similar to Appendix A GDC 14, 15, 31, and 32. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of a SGTR are reduced.

This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking in a tube or the tube-to-tubesheet weld, References 2 and 4 define a length of remaining tube ligament that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factors applied). Additionally, it is shown that application of the limited tubesheet inspection depth criteria will not result in unacceptable primary-to-secondary leakage during all plant conditions.

Based on the above, it is concluded that the proposed changes do not result in any reduction of margin with respect to plant safety as defined in the Final Safety Analysis Report or Bases of the plant Technical Specifications.

Therefore, FPL Energy Point Beach concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Precedent

Wolf Creek Nuclear Operating Corporation, Vogtle Electric Generating Plant Units 1 and 2, and Braidwood Station, Units 1 and 2 were granted similar TS changes on April 4, April 9, and April 18, 2008, respectively. These changes modified the repair requirements for portions of the SG tubes greater than 17 inches below the top of the tubesheet.

4.4 Conclusion

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public with the implementation of the interim alternate repair criterion discussed above.

5.0 ENVIRONMENTAL CONSIDERATION

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

REFERENCES

1. Letter from T. J. Garrett of Wolf Creek Nuclear Operating Corporation to USNRC dated February 8, 2008 (Serial No. ET 08-0009), "Docket No. 50-482: Revision to Technical Specification (TS) 5.5.9, 'Steam Generator (SG) Program' for Interim Alternate Repair Criteria."

2. Westinghouse Electric Company LLC letter, LTR-CDME-08-11 Revision 1, P-Attachment, "Interim Alternate Repair Criteria (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated April 29, 2008.

3. Letter from T. J. Garrett of Wolf Creek Nuclear Operating Corporation to USNRC dated March 21, 2008 (Serial No. ET 08-0016) "Docket No. 50-482: Response to Request for Additional Information Related to License Amendment Request for an Interim Alternate Repair Criterion to Technical Specification 5.5.9, Steam Generator (SG) Program."

4. Westinghouse Electric Company LLC letter, LTR-CDME-08-43 Revision 1 P-Attachment, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-011 P-Attachment," dated April 29, 2008.

5. TSTF-449, Revision 4,"Steam Generator Tube Integrity," Technical Specifications Task Force Standard Technical Specification Change Traveler, April 14, 2005.

6. NEI 97-06, Revision 2, "Steam Generator Program Guidelines," May 2005.

7. ETSS #20510.1; Technique for Detection of Circumferential PWSCC at Expansion Transitions.

8. EPRI TR-107197; Depth Based Structural Analysis Methods for Steam Generator Circumferential Indications; November 1997.

9. EPRI 1012987; "Steam Generator Integrity Assessment Guidelines," July 2006.

10. NRC Letter, Wolf Creek Generating Station – Issuance of Amendment re: Revision to Technical Specification 5.5.9 on the Steam Generator Program (TAC No. MD8054), April 4, 2008

11. Letter ET 08-0024, Docket No. 50-482: Supplemental Information Related to License Amendment Request for an Interim Alternate Repair Criterion to Technical Specification 5.5.9, "Steam Generator (SG) Program", dated March 30, 2008

12. NRC Letter, Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding Changes to Technical Specification (TS) Sections TS 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" (TAC Nos. MD7450 and MD7451), April 9, 2008.

13. Letter NL-08-0522, Vogtle Electric Generating Plant Supplemental Information Related to License Amendment Request for an Interim Alternate Repair Criterion to Technical Specification 5.5.9, "Steam Generator (SG) Program," dated April 3, 2008.

14. Vogtle Electric Generating Plant Units 1 and 2, Response to Request for Additional Information Related to License Amendment Request to Revise Technical Specification (TS) "Steam Generator Tube Inspection Report" for Interim Alternate Repair Criterion, March 21, 2008.

15. Vogtle Electric Generating Station Plant Units 1 and 2, License Amendment Request to Revise Technical Specification (TS) Sections 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report," for Interim Alternate Repair Criterion, February 13, 2008.

16. NRC Letter, Braidwood Station, Units 1 and 2 – Issuance of Amendments re: Revision to Technical Specifications for the Steam Generator Program (TAC Nos. MD8158 and MD8159), April 18, 2008.

17. Exelon Letter RS-08-046, Supplemental Information Related to Steam Generator Tube Interim Alternate Repair Criteria Technical Specification, April 9, 2008.

18. Exelon Letter RS-08-031, Response to Request for Additional Information Regarding Application for Steam Generator Tube Interim Alternate Repair Criteria Technical Specification, March 27, 2008.

19. Exelon Letter RS-08-016, Application for Steam Generator Tube Interim Alternate Repair Criteria Technical Specification Amendment, February 25, 2008.

20. EPRI 1013706, Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 7, October 2007.

ENCLOSURE 2

FPL ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 257

INTERIM ALTERNATE REPAIR CRITERIA (IARC)
FOR STEAM GENERATOR TUBE REPAIR

TECHNICAL SPECIFICATION MARKUPS

TECHNICAL SPECIFICATION 5.5.8
TECHNICAL SPECIFICATION 5.6.8

4 pages follow

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

for all SGs and leakage rate for an individual SG. Leakage is not to exceed 500 gallons per day per SG.

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

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

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

~~1. For Unit 1 Refueling Outage 30 and the subsequent operating cycle, flaws found in the portion of the tube below 17 inches from the top of the hot leg tubesheet do not require plugging. All tubes with flaws identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged. This alternate tube repair criteria is not applicable to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet.~~

1. For Unit 1 Refueling Outage 31 and the subsequent operating cycle, tubes with flaws having a circumferential component less than or equal to 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet do not require plugging. Tubes with flaws having a circumferential component greater than 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet shall be removed from service.

   Tubes with service-induced flaws located within the region from the top of the tubesheet to 17 inches below the top of the tubesheet shall be removed from service. Tubes with service-induced axial cracks found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging.

5.5 Programs and Manuals

When more than one flaw with circumferential components is found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet with the total of the circumferential components greater than 203 degrees and an axial separation distance of less than 1 inch, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet, and the total of the circumferential components found in the tube exceeds 94 degrees, then the tube shall be removed from service. When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet and within 1 inch axial separation distance of a flaw above 1 inch from the bottom of the tubesheet, and the total of the circumferential components found in the tube exceed 94 degrees, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

This alternate tube repair criteria is not applicable to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet.

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 1 Refueling Outage 30 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded when the alternate repair criteria in TS 5.5.8.c are implemented. This exclusion does not apply to the tube at row 38 column 69 in the A steam generator, which is not expanded the full length of the tubesheet.~~ 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

5.6 Reporting Requirements

5.6.7 Tendon Surveillance Report (continued)

Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

h. The effective plugging percentage for all plugging in each SG.

i. Following completion of an inspection performed in Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, whether initiated on primary or secondary side for each service-induced flaw within the thickness of the tubesheet, and the total of the circumferential components and any circumferential overlap below 17 inches from the top of the tubesheet as determined in accordance with TS 5.5.8, 5.6 Reporting Requirements

j. Following completion of an inspection performed in Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the primary to secondary LEAKAGE rate observed in each steam generator (if it is not practical to assign leakage to an individual SG, 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

k. Following completion of an inspection performed in Refueling Outage 31 (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the portion of the tube below 17 inches from the top of the tubesheet for the most limiting accident in the most limiting steam generator.

Westinghouse

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Our ref: CAW-08-2424

May 19, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-CDME-08-43, Rev. 1 P-Attachment, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11, Rev. 1 P-Attachment," dated April 29, 2008 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-08-2424 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 FPL Energy Point Beach, LLC.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-08-2424, 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,

R. M. Span/FOR

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

Enclosures

cc: Jon Thompson (NRC O-7E1A)

bcc: J. A. Gresham (ECE 4-7A) 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 (letter and affidavit only)
G. W. Whiteman, Waltz Mill
H. O. Lagally, Waltz Mill
C. D. Cassino, Waltz Mill
J. T. Kandra, Waltz Mill
D. E. Peck, ECE 560C

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared B. F. Maurer, 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:

B. F. Maurer, Manager
ABWR Licensing

Sworn to and subscribed before me
this 19th day of May 2008

Sharon L. Markle
Notary Public

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-CDME-08-43, Rev. 1 P-Attachment, "Response to NRC Request for Additional Information (RAI) Relating to LTR-CDME-08-11, Rev.1 P-Attachment," dated April 29, 2008 (Proprietary), for submittal to the Commission, being transmitted by FPL Energy Point Beach, LLC Application for Withholding Proprietary Information from Public Disclosure to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Point Beach Unit 1 is expected to be applicable to other licensee submittals in support of implementing an interim alternate repair criterion (IARC) that requires a full-length inspection of the tubes within the tubesheet but does not require plugging tubes with a certain arc length of circumferential cracking below 17 inches from the top of the tubesheet.

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

(a) Provide documentation of the analyses, methods, and testing for the implementation of an interim alternate repair criterion for the portion if the tubes within the tubesheet of the Point Beach Unit 1 steam generators.

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

Further this information has substantial commercial value as follows:

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

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

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

FPL Energy Point Beach, LLC

Letter for Transmittal to the NRC

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

Enclosed are:

1. 1 copy of LTR-CDME-08-43, Rev. 1 P-Attachment, "Response to NRC Request for Additional Information relating to LTR-CDME-08-11, Rev. 1 P-Attachment," dated April 29, 2008 (Proprietary)

2. 1 copy of LTR-CDME-08-43, Rev. 1 NP-Attachment, "Response to NRC Request for Additional Information relating to LTR-CDME-08-11, Rev. 1 NP-Attachment," dated April 29, 2008 (Non-Proprietary).

Also enclosed is Westinghouse authorization letter CAW-08-2424 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

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

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

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-08-2424 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.

ENCLOSURE 4

FPL ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNIT 1

LICENSE AMENDMENT REQUEST 257
TECHNICAL SPECIFICATION 5.5.8, STEAM GENERATOR PROGRAM
PROPOSED LICENSE AMENDMENT REQUEST,
INTERIM ALTERNATE REPAIR CRITERIA (IARC)
FOR STEAM GENERATOR TUBE REPAIR

WESTINGHOUSE ELECTRIC COMPANY LLC
INTERIM ALTERNATE REPAIR CRITERION (ARC) FOR CRACKS IN THE
LOWER REVION OF THE TUBESHEET EXPANSION ZONE
LTR-CDME-08-11, REVISION 1, DATED APRIL 29, 2008

WESTINGHOUSE ELECTRIC COMPANY LLC
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION RELATING
TO LTR-CDME-08-11, REVISION 1 NP-ATTACHMENT

WESTINGHOUSE ELECTRIC COMPANY LLC
APPLICABILITY OF IARC TECHNICAL JUSTIFICATION TO POINT BEACH 1
LTR-CDME-08-125 DATED MAY 23, 2008

74 pages follow

LTR-CDME-08-11, Rev. 1 NP-Attachment

Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone

April 29, 2008

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

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

An alternate repair criterion (ARC) to limit the inspection depth in the tubesheet expansion zone, known as H*/B*, has been docketed by Wolf Creek Nuclear Operating Corporation since February 2006 and has been undergoing NRC review since that time. The H*/B* ARC seeks to minimize the depth of rotating coil inspection of the SG tubes within the tubesheet. The premise of H*/B* is that the expansion joint provides sufficient structural restraint to prevent the tube from pulling out of the tubesheet under normal operating and accident conditions, and that the accident induced leakage during accident conditions is bounded by a factor of two on the observed normal operating leakage. Because of the technical complexity of H*/B*, review of it cannot be completed in time for the Spring 2008 refueling outages.

This report provides technical justification for an interim alternate repair criterion (IARC) that requires full-length inspection of the tubes within the tubesheet but does not require plugging tubes if any circumferential cracking observed in the region greater than 17 inches from the top of the tubesheet (TTS) is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads (the greater of 3x NOP or 1.4x SLB end cap loads). Axial cracks below 17 inches from the TTS are not relevant to the tube pullout arguments because axial cracks do not degrade the axial load carrying capability of the tube. Axial cracks do not require plugging if they are below 17 inches from the top of the tubesheet.

The calculation of the limiting circumferential ligament is provided in Section 3 of this report. The calculation assumes that friction loads between the tube and tubesheet from any source are zero. This assumption avoids potential effects of uncertainties in tube and tubesheet material properties.

Also, based on the same assumption that the contact pressure between the tube and the tubesheet from any source is zero, this report provides a basis for demonstrating that the accident induced leakage will always meet the value assumed in the plant's safety analysis if the observed leakage during normal operating conditions is within its allowable limits. This analysis is provided in Section 4 of this report. The need to calculate leakage from individual cracks is avoided by the calculation of the ratio of accident induced leakage to normal operating leakage.

The tube-end weld is specifically excluded from the tube by TSTF-449, Rev. 4. Because friction between the tube and the tubesheet is ignored, the weld may become an important component in the transfer of the tube pullout loads to the tubesheet. Therefore, the minimum ligament necessary to transfer the pullout loads is also calculated in Section 3. Because the tube-end weld is not considered a part of the tube, discussion of the inspection methodology is beyond the scope of this technical discussion. Discussion of how the weld will be examined is provided as a separate part of the license amendment request.

A bounding analysis approach is utilized for both the minimum ligament calculation and leakage ratio calculation. "Bounding" means that the most challenging conditions from the plants with hydraulically expanded Alloy 600TT tubing are used. Three different tube diameters are represented by the affected plants (11/16" dia., Model F; ¾" dia. Model D5; 7/8" dia., Model 44F). The most limiting conditions for structural evaluation depend on tube geometry and applied normal operating loads. The conditions from the plant that result in the highest stress in the tube below the top of the tubesheet are used to define the minimum required circumferential ligament. The limiting leak rate ratio depends on the leak values assumed in the safety analysis and allowable normal operating leakage that results in the longest length of undegraded tube/crevice for assuring that acceptable leakage during the limiting design basis accident (i.e., steam line break, locked rotor and control rod ejection) above 17 inches below the tubesheet are used. The limiting cases for structural evaluation and leakage evaluation are not necessarily from the same plant. However, the resulting minimum ligament and required undegraded length of tube below the top of the tubesheet can be safely applied for any of the affected domestic plants identified in Table 4-1.

2.0 PERFORMANCE CRITERIA

The performance criteria of NEI 97-06, Rev. 2 (Reference 2-1) are the basis for these analyses. The performance criteria are:

The structural integrity performance criterion is:

All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

The structural integrity performance criterion is based on ensuring that there is reasonable assurance that a steam generator tube will not burst during normal operation or postulated accident conditions.

The accident induced leakage performance criterion is:

The primary to secondary accident induced leakage rate for any design basis accident, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. Leakage is not to exceed 1 gpm per steam generator, except for specific types of degradation at specific locations when implementing alternate repair criteria as documented in the Steam Generator Program technical specifications.

Primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a limiting design basis accident. The potential primary-to-secondary leak rate during postulated design basis accidents shall not exceed the offsite radiological dose consequences required by 10 CFR Part 100 guidelines or the radiological consequences to control room personnel required by GDC-19, or other NRC-approved licensing basis.

The IARC for the tubesheet region is designed to meet these criteria. The structural criterion regarding tube burst is inherently satisfied because the constraint provided by the tubesheet to the tube prohibits burst. However, the structural integrity criterion is interpreted to mean that tube pullout from the tubesheet is equivalent to a tube burst and must, therefore, be prevented.

The accident induced leakage criterion applies directly. The IARC will demonstrate that the accident induced leakage will not exceed the leakage assumed in the accident analysis for the plant which bounds all of the domestic plants which are anticipated to utilize the IARC.

2.1 REFERENCES

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

3.0 STRUCTURAL EVALUATION FOR MINIMUM CIRCUMFERENTIAL LIGAMENT

3.1 INTRODUCTION

An assessment to determine the remaining ligament in steam generator tubes (relevant to Model D, Model F, and Model 44F) necessary to support the assumed loading conditions in the presence of postulated, partially circumferential and fully circumferential flaws was performed. Two locations were considered, within the steam generator tube wall at a location deep in the tubesheet and within the tube-to-tubesheet weld. In addition, growth of the crack was simulated by using four default primary water stress corrosion crack (PWSCC) growth rates. Failure was determined to occur when the stress in the remaining ligament of tube or weld metal exceeded the flow stress.

3.2 ANALYSIS

3.2.1 Description of the Steam Generator Models

The tube geometries used in three models of steam generator which may utilize the IARC were analyzed. These were Model D, Model F, and Model 44F. The material properties applied in this analysis are LTL properties provided in References 3-1 through 3-4. The tube dimensions, material, and mechanical properties (at 650°F) are listed in Table 3-1.

3.2.2 Flaw Geometries

1. *Partial circumferential flaw in the steam generator tube wall.* This postulated flaw in the steam generator tube wall is assumed to have an initial depth of 0.010 inch and an initial arc length of 0.060 inch on the tube's inner diameter. The flaw extends from the tube's inner diameter to a depth of 0.010 inch such that the side faces of the flaw run parallel to the radii of the tube. Figure 3-1 shows a section of a steam generator tube, its radial and axial axes, and the crack face. Figure 3-2 shows the partial circumferential crack on the crack face. The initial depth and arc length are chosen to represent a typical surface flaw with a semi-elliptic shape and a 3:1 aspect ratio subject to mode I crack opening (Reference 3-5). Thus, the length of the semi-major axis is initially three times that of the semi-minor axis, and the tensile axis of the load which opens the crack is normal to the direction of crack propagation. The initial depth of 0.010 inch is a commonly accepted initial flaw depth upon initiation. The flaw simultaneously grows by PWSCC both radially and circumferentially, and it maintains its initial shape. Upon breaching the outer diameter of the tube, the flaw continues to grow circumferentially until the remaining area of the tube cannot support the applied loading.

2. *Full circumferential flaw in the steam generator tube wall.* The postulated, full circumferential flaw in the steam generator tube wall is assumed to have an initial depth equal to 0.010 inch, consistent with the partial circumferential flaw. The

depth is also measured from the tube's inner diameter. Figure 3-3 shows the geometry for this type of flaw. This type of flaw grows by PWSCC radially only until the remaining ligament can no longer support the applied loading.

3. *Partial circumferential, through-wall flaw in the steam generator tube wall.* This type of geometry was chosen to correspond to the type of flaw that may exist upon detection. The assumed initial arc length of this flaw is 40 degrees, and the flaw grows by PWSCC circumferentially only until the remaining ligament can no longer support the applied loading. The geometry for this flaw is identical to the geometry shown in Figure 3-2 with the exception that the crack depth is through-wall.

4. *Partial circumferential flaw in the weld metal.* This geometry is similar to that described in number 1 above, except that it is in the weld and grows due to PWSCC in the shape of a conical frustum on an angle determined by the plane of maximum principal stress. The initial depth and arc length are 0.010 inch and 0.060 inch, respectively. Figure 3-4 is a schematic of a conical frustum and the surface on which the crack grows, and Figure 3-5 is a schematic of the flaw on that surface. The growth is simultaneously radial and circumferential until the remaining ligament cannot support the applied loading.

5. *Full 360 degree circumferential flaw in the weld metal.* This flaw, of 0.010 inch initial depth grows radially only due to PWSCC. It also grows in the shape of a conical frustum on an angle determined by the maximum principal stress until the remaining ligament cannot support the applied loading. Figure 3-6 is a schematic of this flaw geometry.

3.2.3 Initiation

Implicit in the preceding section is that the flaws are presumed to exist as the initial condition for the crack growth cycle. A crack growth cycle as defined in this analysis is full power operation for the length of time for the crack to grow from initial conditions until the minimum residual ligament is attained. The time variable is important to establish the ultimate required residual ligaments for different planned plant operating periods between inspections.

3.2.4 Pressure Loading for Flaws in the Tube Wall

The requirement for tube integrity is that the tube be able to support loads due to a pressure difference of $3*\Delta P_{NOP}$ or $1.4*\Delta P_{SLB}$, whichever is more limiting. A review of the data available shows that the most limiting condition is due to ΔP_{NOP} of Surry Units 1 and 2 [ ]^{a,c,e} Therefore, the most limiting pressure differential to determine end cap loads is based on $3*\Delta P_{NOP}$ of the Surry Units 1 and 2 and equals [ ]^{a,c,e} This is conservative relative to the actual loads. Once a PWSCC flaw initiates, the faces of that flaw are subject to internal pressure, which in this case is the primary side pressure (2250 psia).

3.2.5 Pressure Loading Effects in the Weld Metal

The plants being addressed for this study all have flush welds. The weld is assumed to have an elliptic shape with a semi-major axis equal to the tube wall thickness, a semi-minor axis equal to 0.014 inch, and a crown extending 0.008 inch below the tubesheet cladding surface. This is a conservative idealization of the actual weld nugget. In-process measurements of the welds have determined that the weld protrusion from the tubesheet surface is between 0.008 inch and 0.013 inch. Also, visual examination of the welds show that the autogenous weld nugget is elliptical and inclined to horizontal with the interface between the weld and the tube approximately 0.035 inch into the tubesheet bore. Therefore, the idealized representation of the weld is conservative to the actual manufacturing condition.

Three main crack paths are most likely to occur due to the applied loading. One is the horizontal surface between the tube bottom and the weld. In the most idealized fashion, the end cap loads result in a tensile stress along this interface. The second crack path is the vertical line from the tube-tubesheet interface to the bottom of the weld metal. In this case, the end cap loads result in a shear stress along this line of crack propagation. The third crack path is in the weld metal, between the previous two paths, and whose loading is a combination of tensile stress and shear stress. Figure 3-7 is a schematic of the weld geometry and the crack paths just discussed. The simplifying assumption used in this study is that the stress tensor of an infinitesimal volume of material in this region is comprised of the stress components calculated for the first two crack paths. This results in the maximum principal stress acting on a line that is approximately 35 degrees counter-clockwise from the tube bottom, where the center of rotation is 0.020 inch above the bottom surface of the tubesheet cladding and along the tube-tubesheet interface. Figure 3-8 is a representation of an infinitesimal volume of material, the applied stress tensor, and the principal stresses. As the crack grows, a decreasing area of the weld metal is subject to the maximum principal stress, however the flaw area is then subject to internal pressure on its faces.

3.2.6 Constraint

The tube region subject to cracking is deep in the tubesheet (>17 inches below the top of the tubesheet). The tubes are assumed to be flush against the tubesheet due to the hydraulic expansion process; however, there is no interference force due to pressure. No motion is possible in the lateral direction. Furthermore, it is also assumed that there is no friction acting on the joint between the tube and the tubesheet. The result of these assumptions is that only vertical displacement is allowed and the stresses in the tube wall are purely tensile; there is no bending stress component because of the lateral restraint of the tubesheet. Similarly, the weld metal is subject only to the tensile loads transmitted by the tube. Therefore, any crack in the weld metal will also open in a purely tensile mode. This is the reason that a weld crack in a direction radiating away from the tube's centerline is not considered here. In this case, the residual weld nugget on the tube results in mechanical interference with the residual weld nugget on the tubesheet, and the tube cannot pull out of the tubesheet.

3.2.7 Force Balance

1. *Partial circumferential flaw in the steam generator tube wall.* The force balance for this scenario is one in which the end cap load plus the force due to the internal pressure acting on the faces of the flaw is balanced by the force reacted over the tube wall's cross-sectional area minus the flaw area. As the flaw grows, the areas of both the tube wall cross-section and the flaw change. The equation used in this part of the study is

a,c,e

[ ]

where

P is the pressure [ ]a,c,e

P_i is the internal pressure (2250 psia),

r_i is the inner radius of the steam generator tube,

d is the crack depth,

$\Delta\theta$ is the arc length of the crack,

σ is the stress reacted by the steam generator tube's cross-section, and

r_o is the outer radius of the steam generator tube.

2. *Fully circumferential flaw in the steam generator tube wall.* The force balance dictated by this case is one in which the end cap load plus the internal pressure acting over the crack faces of a fully circumferential flaw is balanced by the force reacted by the steam generator tube wall's cross-sectional area minus the area of the flaw. Again, the areas of both the flaw and the steam generator tube wall's cross-section change as the flaw grows. The equation used to model this situation is

a,c,e

[ ]

where the variables are the same as previously defined.

3. *Partially circumferential, through-wall flaw in the steam generator tube wall.* This situation is identical to scenario 1 with the exception that the initial flaw is through-wall at the beginning of the crack growth cycle, and the initial arc length of the flaw is 40 degrees. This models a reasonable flaw length that would be detected by +Pt inspection which is assumed to be throughwall. The force balance for this case is

a,c,e

[ ]

where the variables are the same as previously defined.

4. *Partial circumferential flaw in the weld metal.* The welds applicable to the plants under consideration are flush welds. Thus, the weld was modeled as an ellipse. The starting point of the ellipse region is the steam generator tube wall's inner diameter. This case is one in which normal stress and shear stress components are present. The normal stress results from a potential crack propagation path that runs along the interface between the steam generator tube wall and the weld metal. The shear stress component is from a potential crack propagation path that runs vertically from the interface between the steam generator tube and the tubesheet to the crown of the weld. The infinitesimal element of weld metal is assumed to have the normal and shear stress components that result from each of the two crack propagation paths (assuming that only one is active and the other is fixed). Hence, the normal stress component used is

a,c,e

[ ]

and the shear stress component is

a,c,e

[ ]

b is the semi-minor axis (0.014 inch). The three principal stresses that result from calculating the invariants of the stress tensor comprised of the above components are:

a,c,e

[ ]

and the direction of the principal axes is determined by:

a,c,e

[ ]

The crack propagation direction is found to be approximately [ ][a,c,e] extending from the steam generator tube-tubesheet interface toward the centerline of the steam generator tube. This results in a crack propagation surface that is an inverted frustum of a cone. Using the surface of revolution technique (see Reference 3-6), the surface area of this conical frustum is

a,c,e

[ ]

where θ is the approximately [ ][a,c,e] angle defined above, y is the vertical location of the intersection of the crack propagation line and the ellipse, and the rest of the variables are defined for scenario 1 above. The area of a flaw extending a depth d into this surface and over an arc length $\Delta\phi$ extending over this surface is

a,c,e

[ ]

where all of the variables have been previously defined. The resulting force balance for this scenario is

a,c,e

[ ]

where, in this case, σ is the stress reacted by the remaining surface area of the frustum.

5. *Full circumferential flaw in the weld metal.* This number is similar to number 4 with the exception that the flaw is now fully circumferential. The area of the flaw in this case is

a,c,e

[ ]

The resulting force balance is

a,c,e

[ ]

where, again, σ is the stress reacted by the remaining surface of the frustum.

3.3 RESULTS AND DISCUSSION

The required remaining ligaments are shown in Table 3-3. The required remaining circumferential ligaments for initially non-360 degree throughwall circumferential flaws are expressed in terms of degrees of arc. The required remaining radial ligaments for full 360 degree non-throughwall circumferential flaws are expressed in terms of inches.

3.3.1 Steam Generator Tube Wall Cross-Section

The values contained in Table 3-3 indicate that the required remaining ligament for partially circumferential flaws is approximately [ ]^{a,c,e} while the required remaining ligament for fully circumferential flaws is approximately [ ]^{a,c,e} The Model F steam generator tube requires less remaining ligament than do either the Model D or Model 44F steam generator tubes.

3.3.2 Steam Generator Tube Cross-Section with an Initial 40 Degree Arc Length, Through-Wall Flaw

The results contained in Table 3-3 show that a partially circumferential flaw that is initially through-wall requires about the same remaining ligament of material as the case for which the initial flaw was not initially through-wall [ ]^{a,c,e} Since the force balance is based on net tensile force, this result is expected.

3.3.3 Weld Metal

The results for the weld metal calculations are also shown on Table 3-3. The required remaining ligaments for both the partially circumferential and fully circumferential flaws are approximately [ ]^{a,c,e} arc length and approximately [ ]^{a,c,e} for the partially circumferential and fully circumferential flaws, respectively , significantly less than required for the steam generator tube wall.

This situation for the weld is mechanically different than for the steam generator tube wall. In the latter case, the pressure differential that causes the end cap load is based on the internal pressure which acts on the flaw's faces. The end cap loading relieved in the wall during crack growth is replaced by another pressure loading on the crack faces. For the weld, the pressure differential causes an end cap load, which in turn results in a maximum principal stress along an inclined crack propagation path. The maximum principal stress [ ]^{a,c,e} is much greater than the initial stress reacted by the steam generator tube wall [ ]^{a,c,e} However, as the flaw grows in the weld metal, it is the maximum principal stress in the area of the flaw that is relieved and replaced with the primary pressure loading [ ]^{a,c,e} over the crack faces. In addition, the surface area relevant to the weld metal is slightly larger than that contained in the steam generator tube wall due to its incline.

3.4 CONCLUSIONS – STRUCTURAL EVALUATION

- The required arc of ligament for an initial, partially circumferential flaw of 0.010" depth in the steam generator tube is approximately [ ]^{a,c,e} In general, the Model F steam generator tube wall requires the least amount of remaining ligament. However, Model F requires the least amount of time to grow to its critical flaw size. The results of all of the calculations performed are enveloped by an arc length of ligament equal to [ ]^{a,c,e} for this geometry.

- The required arc of ligament for the case when the initial flaw is through-wall over a 40 degree arc is approximately the same as above. This is expected as the critical flaw size is based on net tensile stress. An arc length of ligament equal to [ ]^{a,c,e} is necessary to bound the results for this geometry.

- Initial, fully circumferential flaws in the steam generator tube can grow to approximately [ ]^{a,c,e} through-wall before failure was calculated to occur. The minimum required radial ligament depth is [ ]^{a,c,e} for the bounding case. This is provided for information only since the underlying assumption of the IARC is that circumferential cracks will be considered 100% throughwall.

- Initial, partially circumferential flaws in the weld required a [ ]^{a,c,e} arc of remaining weld material, significantly less than the arc required in the steam generator tube wall. In order to bound the results for this geometry, an arc length of material spanning [ ]^{a,c,e} is required.

- Initial, fully circumferential flaws in the weld metal were able to grow to approximately [ ]^{a,c,e} through-wall before failure was calculated to occur, again significantly less than the ligament required in the steam generator tube wall. A bounding value of [ ]^{a,c,e} of ligament is required for this case. This is provided for information only since the underlying assumption of the IARC is that circumferential cracks will be considered 100% throughwall.

3.5 REFERENCES

3-1 J. A. Begley and J. L. Houtman, WCAP-12522, "Inconel Alloy 600 Tubing-Material Burst and Strength Properties," January 1990.

3-2 G. Whiteman, WCAP-16670-P, "Steam Generator Alternate Repair Criteria for Tube Portion within the Tubesheet at Comanche Peak Unit 2," November 2006.

3-3 WCAP-16124-P, "Justification for the Partial-Length Rotating Pancake Coil (RPC) Inspection of the Tube Joints of the Wolf Creek Model F Steam Generators," September 2003.

3-4 G. Whiteman, WCAP-16506-P, "Steam Generator Alternate Repair Criteria for Tube Portion within the Tubesheet at Turkey Point Units 3 and 4," December 2005.

3-5 T. L. Anderson, Fracture Mechanics Fundamentals and Applications, Second Edition, New York: CRC Press, 1995.

3-6 G. B. Thomas, Jr., Calculus and Analytic Geometry, Alternate Edition, Reading: Addison-Wesley Publishing Company, Inc., 1972.

Table 3-1
Dimensions and Mechanical Properties of the Steam Generator Tubes

Model	D [2]	F [3]	44F [4]
O.D. (in)	0.764	0.703	0.893
Wall Thickness (in)	0.04257	0.0396	0.0495
I.D. (in)	0.664	0.6075	0.775
Material	Alloy 600	Alloy 600	Alloy 600
Heat Treatment	Thermally Treated	Thermally Treated	Thermally Treated
r_o (in)	0.382	0.3515	0.4465
r_i (in)	0.33943	0.3119	0.397

a,c,e

Note [1]: These properties listed are lower tolerance limit (LTL) properties from Reference (3-1).
Note [2]: The expanded tube outer diameter and thinned wall dimensions for the Model D steam generator tubes are from Reference (3-2).
Note [3]: The expanded tube outer diameter and thinned wall dimensions for the Model F steam generator tubes are from Reference (3-3).
Note [4]: The expanded tube outer diameter and thinned wall dimensions for the Model 44F steam generator tubes are from Reference (3-4).

Table 3-2
Interim Alternate Plugging Criterion Pressure Differentials

Plant	Normal Operation ΔP (psi)	Steam Line Break ΔP (psi)	Source Document

a,c,e

Table 3-3
Calculation of Required Minimum Ligament

	Circumferential Extent of Flaw	Minimum Structural Ligament

a,c,e

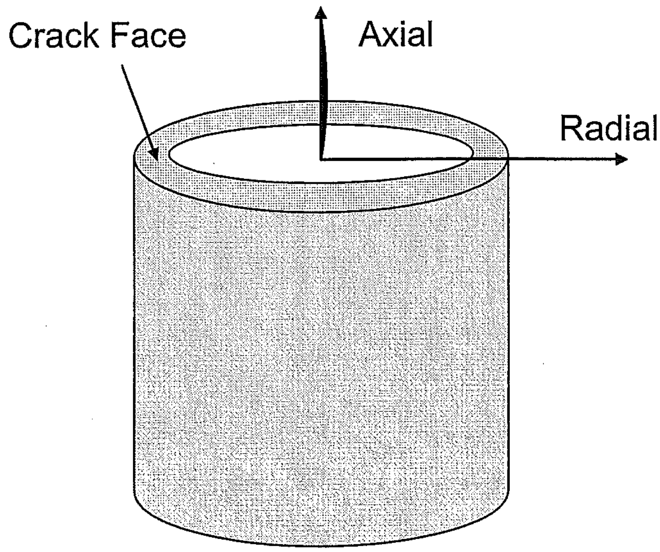

Figure 3-1
A Segment of a Steam Generator Tube Showing the Radial and Axial Axes as Well as the Crack Face

a,c,e

Figure 3-2
The Geometry of a Partially Circumferential Crack on the Crack Surface Shown in Figure 3-1

a,c,e

Figure 3-3
The Geometry of a Fully Circumferential Crack on the Crack Surface Shown in Figure 3-1

a,c,e

Figure 3-4
A Schematic of a Conical Frustum Showing the Surface on Which the Crack Grows

a,c,e

Figure 3-5
Schematic of a Partially Circumferential Flaw in the Weld Metal Along a Conical Frustum

a,c,e

Figure 3-6
Schematic of a Fully Circumferential Flaw in the Weld Metal Along a Conical Frustum

a,c,e

Figure 3-7
The Weld Metal Geometry and the Potential Crack Paths Considered

a,c,e

Figure 3-8
A Schematic Representing an Infinitesimal Volume of Material in the Weld Metal Under the Applied Stress Tensor and Its Transformation to the Principal Stress Tensor.

(This element is in the weld metal to the left of the shear plane vertical line in Figure 3-7.)

4.0 METHOD FOR CALCULATING LEAKAGE

4.1 SUMMARY

The alternate repair criterion (ARC) known as B* (Reference 4-2, 4-3), for "bellwether" approach, specifies the length of sound tubing required for the tube portion within the tubesheet that will assure that a plant's accident induced primary-to-secondary (P/S) leakage limit will not increase greater than a factor of two (2) above the normal operating leakage. The B* criterion relies on the contact pressure between the tube and the tubesheet. Technical issues remain to be resolved in the calculation of contact pressure between the tube and the tubesheet. Therefore, a modified B* approach is presented in this section which demonstrates that a plant with postulated cracks in the tube portion within the lower four inches of the tubesheet will still meet the accident induced leakage limits for safe steam generator operation under the assumption that no contact pressure exists between the tube and the tubesheet.

The modified B* approach shows that for an undegraded 17 inch depth of tube, measured from the secondary side surface of the tubesheet, there is a margin of a factor of 1.7 on the limiting length below the neutral axis of the tubesheet required to meet accident induced leakage limits for the bounding plant among those under consideration. This result means that, for the bounding plant, a 17 inch length of tube in undegraded condition provides more than 1.7 times the length of porous medium (crevice) necessary below the neutral axis of the tubesheet to limit the accident induced leakage to the value assumed in the safety analysis.

Figure 4-1 shows a sketch of the porous medium in the tube-to-tubesheet crevice. The typical machining finish of 125 micro-inches defines the porosity, but is assumed to provide no interlocking or friction.

A summary of the plants that are included in the modified B* analysis is given in Table 4-5. Based on the plant information, the ratio of the allowable accident leak rate to the allowable normal operating leakage limit in the bounding case steam generator is two (2). This value ranges from two (2) to six (6) for the plants under consideration for the IARC. See Table 4-2. This means that the leakage during accident conditions can increase by no more than 2 to 6 times the leak rate during normal operating conditions for the plants under consideration. This section shows that ample margin exists in undegraded crevice length for the bounding plant. The results for the bounding plant envelope all of the plants under consideration.

4.2 MODIFIED B* LEAKAGE ANALYSIS

The approach to the modified B* leakage analysis is similar to that used in the original B* (Reference 4-2). Where B* calculates the length of undegraded tubing, measured from the TTS, required to equilibrate the flow resistance during normal operating and during accident conditions so that the increase in primary to secondary leakage is limited to a function of the ratio of the pressure differential during the limiting design basis accident and normal operating conditions, the Modified B* analysis calculates the ratio of undegraded crevice length determined by eddy current inspection to the length of undegraded crevice required to meet the design basis accident analysis primary to secondary leakage assumption. By definition of the

IARC, 17 inches from the TTS is the available undegraded crevice length because confirmed cracking in this length will require the tube to be plugged. Both the pressure difference ratio and the ratio of the length of crevice during normal operating and the limiting design basis accident are factored into the margin determination as discussed below. By definition, the plant with the smallest allowable accident analysis leakage assumption results in the longest crevice length necessary to assure that accident analysis leakage assumptions are not exceeded. For the plants in question, the Modified B* value ranges from a safety factor of [ ]a,c,e down to [ ]a,c,e at a distance 17 inches below the top of the tubesheet (See the "n" values in Table 4-5). Conservatively using the neutral axis as a reference point, the Modified B* value ranges from [ ]a,c,e down to [ ]a,c,e (See the "n′" values in Table 4-5). Again, these values are the ratio of undegraded tube/crevice length confirmed by eddy current inspection to the length of undegraded crevice calculated using the D'Arcy equation necessary to preclude exceeding the limiting design basis accident analysis leakage assumption.

The D'Arcy formula for axial flow in a porous medium is used to calculate the leakage ratio and to evaluate the potential resistance to leakage in the crevice of the tubesheet. Other available leakage models (Bernoulli, Orifice Flow) are known to be less conservative than the D'Arcy model. Unresolved technical issues regarding the calculation of contact pressure between the tube and the tubesheet in the original B* require that both the bellwether principle and the application of D'Arcy's law do not employ contact pressure equations or relationships in the leakage analysis.

The D'Arcy model for describing axial flow in a porous medium, taken from Reference 4-1 is:

$$Q = \frac{\Delta p}{\mu K l} \tag{1}$$

Where:

Q is the flow rate for the fluid through the medium,

Δp is difference in pressure (or driving head) acting to force the fluid through the medium,

μ is the viscosity of the fluid,

K is the resistance to flow through the medium and

l is the axial length of the medium.

The term $\mu K l$ is the flow resistance, R. In that case, (1) becomes

$$Q = \frac{\Delta p}{R} \tag{2}$$

which produces a relationship between fluid flow, flow resistance and driving potential similar to electrical currents (i.e., $I = V/R$) and allows for similar analogies and assumptions to be made. See Figure 4-1 for a sketch of the system used to describe the porous medium present in the annulus of the tubesheet crevice.

In the following discussion the term R' refers to μK and the axial length of the porous medium is left in the equation as a separate variable as shown in Equation (3).

$$Q = \frac{\Delta p}{\mu K l} = \frac{\Delta p}{R' l} \tag{3}$$

Note that in previous submittals (Reference 4-2, 4-3), the length of the medium was included in the term R (see equation 2), which led to the conclusion that if the resistance of the crack and tubesheet crevice to leakage during normal operating (NOP) conditions was equal to the resistance of the crack and tubesheet crevice during steam line break (SLB), the increase in leakage between NOP and SLB conditions would be governed solely by the pressure differential. The original bellwether ratio of the expected accident leak rate to the required normal operating leak rate of 2 was based on this assumption because the pressure differential at SLB conditions is approximately double that during normal operating conditions. Therefore, the leakage during SLB conditions would be limited to twice that of the leakage during NOP for a length of crevice and a location of the leak that validates the assumption of equal resistance between SLB and NOP conditions.

The purpose of the interim ARC leakage assessment is to calculate the length of porous medium (crevice) required to limit primary-to-secondary (P/S) leakage to an acceptable level during a postulated SLB (or limiting design basis accident) to provide adequate resistance and margin against leakage during accident conditions assuming no contact pressure between the tube and the tubesheet exists. This length is defined as Modified B* and is used to assess the potential for leakage and acceptability of leakage flow rates assuming a full depth inspection of the tube portion with the tubesheet and a 17" length of tube free of all cracking indications. The Modified B* ratio is prescribed as the accident analysis limit divided by the plant Technical Specification limit of 0.1 gpm.

The margin against leakage during an accident event can be defined using equations (1) and (3). An example calculation of the modified B* ratio and the required length of porous medium necessary to accommodate the limiting accident leakage is provided below for the limiting case of zero contact pressure. There is no contact pressure between the tube and the tubesheet ($P_{contact} = 0$ psi) but the tube and the tubesheet are assumed to remain in contact. Assume that a point exists where the viscosity and leakage resistance during normal operating conditions will be equal to that of the viscosity and leakage resistance during accident conditions at some elevation in the tube-to-tubesheet crevice. That is,

$$R'_{NOP} = R'_{DBA} = R' \tag{4}$$

In this case the resistance to flow is calculated assuming that the liquid must flow through a tortuous path that begins at the crack (primary side) and ends at the top of the tubesheet (secondary side). No credit is taken for the increase in contact pressure between the tube and the tubesheet due to tubesheet flexure during accident conditions which would increase the resistance to flow through the crack and crevice.

The following example demonstrates the approach:

If the limiting leakage during NOP is 0.1 gpm and the leakage assumed in the safety analysis for SLB is 0.35 gpm, the ratio between SLB and NOP leakage is:

$$\frac{Q_{SLB}}{Q_{NOP}} = \frac{0.35}{0.10} = 3.5$$

Note that prior knowledge of the shape or orientation of the flaws that contribute to this leakage is not required. The ratio merely reflects the total leakage volume to which the plant is limited during operation. The ratio of the leak rates can be calculated using equations (3) and (4) which gives

$$\frac{Q_{SLB}}{Q_{NOP}} = \frac{\Delta p_{SLB}}{\Delta p_{NOP}} \frac{R'_{NOP}}{R'_{SLB}} \frac{l_{NOP}}{l_{SLB}}$$

$$\frac{Q_{SLB}}{Q_{NOP}} = \frac{\Delta p_{SLB}}{\Delta p_{NOP}} \frac{R'}{R'} \frac{l_{NOP}}{l_{SLB}} = \frac{\Delta p_{SLB}}{\Delta p_{NOP}} \frac{l_{NOP}}{l_{SLB}}$$

$$\frac{Q_{SLB}}{Q_{NOP}} = \frac{\Delta p_{SLB}}{\Delta p_{NOP}} \frac{l_{NOP}}{l_{SLB}} \qquad (5)$$

Substitution of the pressure differentials and the limiting leak rate ratio into equation (5) yields the ratio of the porous medium (crevice) length necessary to maintain the limiting accident analysis leakage assumption. For example, if the limiting primary to secondary pressure differential during normal operations is 1274 psig and the limiting accident pressure differential is 2560 psig the required length ratio for a leak ratio of 3.5 is given by:

$$3.5 = \frac{2560}{1274} \frac{l_{NOP}}{l_{SLB}}$$

$$\frac{l_{NOP}}{l_{SLB}} = 3.5\left(\frac{1274}{2560}\right) = \frac{3.5}{2.009} = 1.74$$

$$\frac{l_{NOP}}{l_{SLB}} = 1.74$$

The length ratio can be used with the data for loss coefficient and viscosity to calculate the required length of tube and crevice necessary to match the limiting leakage flow rate. If the leakage limits for the operating SG are based on "hot" or operational conditions, then the viscosity of the single phase leaked fluid is approximately equal to the viscosity of liquid water at 600°F.[2] The viscosity of liquid phase water at 600°F is approximately 1.76E-6 lbf-s/in^2 (Reference 4-2). The loss coefficient data given in WCAP-16794-P (Reference 4-2) shows that for a contact pressure of approximately 0 psi, the bounding loss coefficient from the 95% confidence interval fit is equal to [ ]^{a,c,e} The value of loss coefficient that approximately bounds all of the test data is [ ]^{a,c,e} (See Figures 4-2 and 4-3).

Note that the primary to secondary leakage at 600°F that corresponds to 0.1 gpm at room temperature conditions is 0.14 gpm. It is necessary to adjust the limiting leak rate for the NOP conditions because the loss coefficient data in WCAP-16794-P (Reference 4-2) is adjusted to represent room temperature conditions. Using the bounding loss coefficient value and the viscosity to calculate the required length of porous medium (crevice) to accommodate the NOP leakage gives

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

$$l_{NOP} = \frac{\Delta p_{NOP}}{\mu_{NOP} K Q_{NOP}}$$

a,c,e

[ ]

$$l_{NOP} = \frac{76440.00}{56918.40} = 1.34 in$$

Recall that:

$$\frac{l_{NOP}}{l_{SLB}} = 1.74$$

Therefore, the length of tube and crevice necessary to maintain the limiting leakage flow rate at accident conditions is

$$\text{Modified B*} = l_{SLB} = 1.34/1.74 = 0.77\text{in}$$

---

[2] : The viscosity and loss coefficient are calculated at normal operating conditions because the normal operating conditions for the set of plants seeking to use the IARC are more closely related. Also, it is conservative to assume that the viscosity of the liquid phase of water during SLB equals the viscosity of the liquid phase of water at NOP condition.

This result shows that the length of porous medium required during the normal operating condition is more limiting compared to the length of porous medium required during an accident condition.

Inspection of the tube to a depth of 17 inches to ensure that the tube is free of cracking indications means that there is at least 17 inches of tube material and crevice to interact and provide leak resistance. Therefore, the available factor of safety against leakage in excess of accident analysis assumptions, *n*, is

$$n = \frac{17}{0.77} \approx 22$$

The result for *n* shows that there is greater than a factor twenty (20) times the length of tube and crevice annulus/porous medium necessary to maintain the maximum allowable leakage limits for plant operation during steam line break conditions in this example.

It is possible for the tubesheet to deflect during operations as the pressure differential from the primary to secondary surface varies so that the tubesheet crevices expand above the tubesheet neutral axis. It is reasonable to expect that the flow resistance of the crevice will decrease as the tubesheet crevice expands. The tubesheet deflection will tend to expand the crevice from the neutral axis of the tubesheet to the secondary side face of the tubesheet in the near and mid-range radii. In the context of this analysis the term near radius refers to the tubesheet radii from the center to a distance of 20 inches, mid range refers to the radius from 20 inches to 40 inches and peripheral refers to tubesheet radii greater than 40 inches from the center. The tubesheet deflection will tend to constrict the tubesheet crevice from the neutral axis to the primary face of the tubesheet in the near and mid-range radii. The effects of the tubesheet deflection are reversed in the peripheral radii so that the crevice tightens above the neutral axis and expands below the neutral axis. In order to accommodate this phenomenon, the available tube-to-tubesheet crevice or available porous medium is only that length within the tubesheet, above or below the neutral axis, which experiences constriction of the tubesheet bore. This will be the reference available crevice length in this analysis. This means that even though there are 17 inches of undegraded crevice available due to the IARC assumptions, only that difference between the neutral axis and 17 inches is assumed to act to provide leakage resistance. In the case of a Model F steam generator the neutral axis is located approximately [ ]a,c,e below the secondary side face of the tubesheet (Reference 4-2). This means that for a Model F steam generator there is a [ ]a,c,e long length of porous medium available to resist leakage that can be assured to not dilate due to tubesheet flexure. Following the example above this means that the actual factor of safety against exceeding the accident induced leakage is:

a,c,e

[ ]

This result for *n′* indicate that if the region of the tubesheet crevice affected by tubesheet bow is removed from consideration there is at least a factor of eight (8) on the available porous medium to resist accident and normal operating leakage in this example.

4.3 CALCULATION OF APPLICABLE DENSITIES AND VISCOSITY

Calculation of the leaked fluid density and the applicable viscosity during NOP conditions is required to determine the required length of porous medium. The density of the leaked fluid is important because different operating plants use different leakage assumptions in their safety analyses. For example, a plant may assume that the leaked fluid is "hot" or at operating temperature, which means that the volume of the fluid is increased relative to a "cold" or room temperature condition. Some of the potential plants under consideration have revised the Plant Technical Specifications to use a mass flow rate for the leakage limit which removes the concern of "hot" or "cold" volumes entirely. The modified B* analysis assumes that all leakage volumes are "cold" leakage volumes even though some plant values for accident analysis leakage are at operating conditions. This results in a lower ratio value for allowable leakage rate during design basis accident conditions to normal operating leakage limit and longer required crevice lengths during the design basis accident.

The modified B* analysis also assumes that the fluid viscosity during NOP bounds the viscosity during any accident at lower temperatures. The viscosity term appears in the denominator of equation (3) so it is conservative to keep it at a lower value which reduces the denominator (viscosity of water increases at lower temperatures) and increases the required length of porous medium.

4.4 CALCULATION OF LIMITING LEAK RATES AND PRESSURE DIFFERENTIALS

The Modified B* IARC leakage analysis represents a bounding approach that describes the limiting leak and length ratios for the potential user plants that are noted on Figure 4-1. These plants meet the definition of an H*/B* plant; that is, steam generators with Alloy 600TT tubing that is hydraulically expanded over the full depth of the tubesheet.

The limiting leak rate ratio, accident induced leakage to normal operating leakage, for the plants on this list is the lowest leak rate ratio for any plant, which is two (2). The bounding analysis for the modified B* must justify a leak rate ratio of two (2). The limiting leak rate ratio is taken from Catawba Unit 2 and is assumed to be a cold volume. No leak rate ratio higher than six has been identified (See Table 4-2).

Table 4-2 through Table 4-5 show the accident and normal operating condition leak rates and the associated pressure differentials for each condition. The pressure differentials are calculated assuming hot leg, low TAVG properties for NOP conditions.

The inputs for the calculation of the limiting length of porous medium (crevice) and the limiting leakage ratio are applied consistently. That is, the pressure differential and leak limit for a single plant is used to calculate the porous medium length and the available margin at 17 inches. The longest required length that bounds all of the other plants under consideration is then taken as the bounding, or limiting length, for all of the plants.

4.5 CALCULATION OF BOUNDING MODIFIED B* FOR INTERIM ARC PLANTS

Applying the limiting leak rate and pressure differential data from Table 4-2 in Equation (5) gives a length ratio of [ ]a,c,e. The calculation of the limiting length ratio is given below

$$\frac{Q_{SLB}}{Q_{NOP}} = \frac{\Delta p_{SLB}}{\Delta p_{NOP}} \frac{l_{NOP}}{l_{SLB}}$$

[ ]a,c,e

[ ]a,c,e

[ ]a,c,e

Calculating the required length of porous medium (crevice) for the limiting plant during NOP conditions yields

$$l_{NOP} = \frac{\Delta p_{NOP}}{\mu_{NOP} k Q_{NOP}}$$

[ ]a,c,e

[ ]a,c,e

[ ]a,c,e

Therefore, the 17 inch length of undegraded crevice within the tubesheet provides more than [ ]^{a,c,e} times the required length required to meet the accident induced leakage limits for the bounding plant. The [ ]^{a,c,e} inch length of undegraded tubing below the neutral axis provides more than [ ]^{a,c,e} times the required length of crevice required to meet the accident induced leakage limits for the bounding plant. The result for the bounding plant envelopes all of the other plants under consideration (see Table 4-5) and the margin for all other plants in Table 4-5 is greater. Therefore, the limiting modified B* result of [ ]^{a,c,e} inches is a bounding result for all of the plants under consideration.

4.6 CONCLUSION

A basis is provided to assure that the accident induced leakage for the limiting accident will not exceed the value assumed in the safety analysis for the plant.

The length of undegraded crevice required to limit the accident induced leakage to less than the value assumed in the safety analysis for the limiting plant is [ ]^{a,c,e} inches. By definition of the IARC, a tube that can remain in service has an undegraded crevice of 17 inches. Therefore, a factor of safety of [ ]^{a,c,e} is available. Expressed in length terms, the length margin in the crevice is [ ]^{a,c,e} inches.

For all IARC candidate plants other than the limiting plant, the margins on length required to limit the accident induced leakage to less than the value assumed in the safety analysis is greater.

In summary, no leakage issue is associated with the IARC unless the normal operating leakage attributable to the tubesheet expansion zone (TEZ) is greater than its limit. Continued operation of the plant with leakage greater than the specified allowable limit is not possible.

4.7 REFERENCES

4-1. NSD-RMW-91-026, M.J. Sredzienski, "An Analytical Model for Flow Through an Axial Crack in Series with a Denting Corrosion Medium." 02/05/1991.

4-2. WCAP-16794-P, G.W. Whiteman, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Vogtle 1 & 2 Electric Generating Plants." 10/2007.

4-3. Wolf Creek ET 07-0043; Docket No. 50-482: "Response to Request for Additional Information Related to License amendment Request to Revise Steam Generator Program"; September 27, 2007.

Table 4-1
List of H*/B* Plants

a,c,e

Site	Alpha	SG Model

Table 4-2
Primary to Secondary Leakage Data and Pressure Differentials for the Domestic Fleet.

SLB = Steam Line Break. LR=Locked Rotor. CRE=Control Rod Ejection. NOP=Normal Operating Condition.

a,c,e

Plant Name	Pressure (psi)		P/S Leakage (GPM)			L_{NOP}/L_{SLB}	L_{NOP}	L_{SLB}
	SLB	NOP	SLB	NOP	Ratio		in	in

Table 4-3
Primary to Secondary Leakage Data and Pressure Differentials for the Domestic Fleet.

SLB = Steam Line Break. LR=Locked Rotor. CRE=Control Rod Ejection. NOP=Normal Operating Condition.

a,c,e

Plant Name	Pressure (psi)		P/S Leakage (GPM)			L_{NOP}/L_{LR}	L_{NOP}	L_{LR}
	LR	NOP	LR	NOP	Ratio		in	in

Table 4-4
Primary to Secondary Leakage Data and
Pressure Differentials for the Domestic Fleet.

SLB = Steam Line Break. LR=Locked Rotor. CRE=Control Rod Ejection.
NOP=Normal Operating Condition.

a,c,e

Plant Name	Pressure (psi)		P/S Leakage (GPM)			L_{NOP}/L_{CRE}	L_{NOP}	L_{CRE}
	CRE	NOP	CRE	NOP	Ratio		in	in

Table 4-5
Summary of Required Accident Length and Available Margin for the Domestic Fleet.

a,c,e

Plant Name	L_{SLB}	L_{LR}	L_{CRE}	MAX	Modified B* Safety Margin Ratio	
					n [(1)]	n' [(2)]
	in	in	in	in	$17/L_{ACCIDENT}$	$6.5/L_{ACCIDENT}$

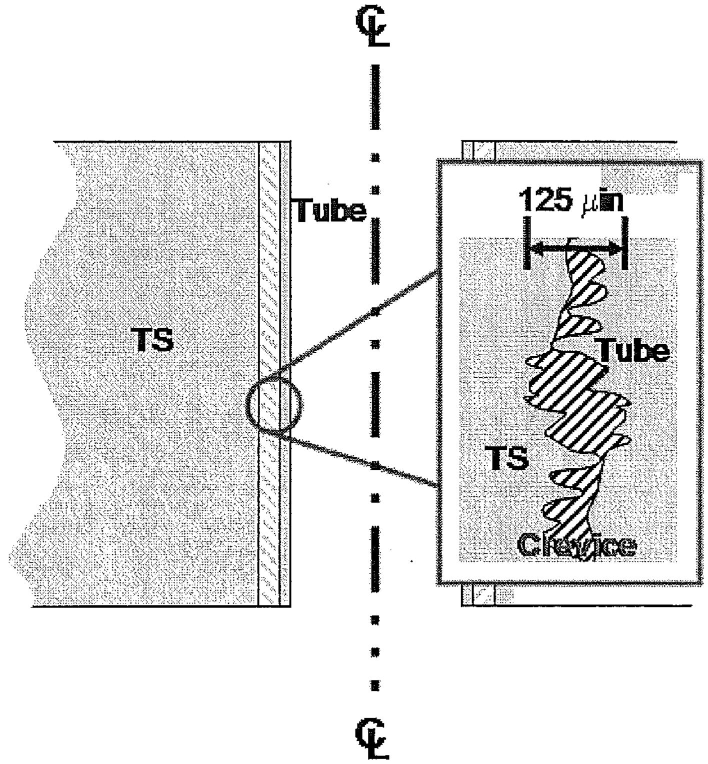

Figure 4-1
Illustration of Tube-to-Tubesheet Crevice and Approximated Porous Medium
Roughness of 125 μin is Typical of Installed Tube and Tubesheet Crevice Surfaces

a,c,e

Figure 4-2
Plot of Loss Coefficient Data as a Function of Contact Pressure for Model F and Model D Steam Generators (Reference 4-2)

a,c,e

Figure 4-3
Plot of Loss Coefficient Data as a Function of Contact Pressure for Model 44F and Model 51F Steam Generators (Reference 4-2)

5.0 IARC CONCLUSIONS

5.1 LIMITING STRUCTURAL LIGAMENT

From Section 3 of this report, the bounding structural ligament required for the tube to transmit the operational loads is 115 degree arc. This assumes that the residual ligament is 100% of the tube wall in depth. For the tube-end weld, the bounding circumferential structural ligament is 35 degrees arc. A small circumferential initiating crack is predicted to grow to a throughwall condition before it is predicted to reach a limiting residual ligament. A residual ligament in a part-throughwall condition is not a significant concern, because of the assumption that all circumferential cracks detected are 100% throughwall.

5.1.1 Consideration of NDE Uncertainty

The NDE uncertainty must be addressed to assure that the as-indicated circumferential arc of the reported crack is a reliable estimate of the actual crack. ETSS #20510.1 (Reference 5-1) describes the qualified technique used to detect circumferential PWSCC in the expansion transitions and in the TEZ. This technique is also considered qualified by the industry, and has been routinely used, for the detection of circumferential indications in the tack expansion region just above the tube-end weld. The qualification data is provided in the ETSS.

The fundamental assumption for the IARC is that all circumferential cracks detected are 100% throughwall. Thus, even a shallow crack of small length will be considered to be throughwall. Further, tube burst is not an issue for the IARC because of the constraint provided by the tubesheet; rather, it is axial separation of the tube that is the principal concern. Assuming that all circumferential cracks are throughwall reduces the inspection uncertainty to length of the cracks only. Further, the accuracy of the length determination is an issue only when the indicated crack approaches the allowable crack length (the complement of the required residual ligament) and if the indicated crack length is a reasonable estimate of the structural condition of the tube.

Prior investigations have correlated the axial strength of the tube to the Percent Degraded Area (PDA) of the flaw (Reference 5-2). PDA takes into account the profile of the existing crack, including non-throughwall portions and shallow tails of the crack. Using the data from ETSS 20510.1 for cracks with a 90%, or greater, throughwall condition from both NDE and destructive examination, Figure 5-1 compares the actual crack length and corresponding PDA for the cracks to a theoretical PDA which assumes that all cracks are 100% throughwall. For all flaws greater than 60 degrees circumferential extent, the theoretical PDA line is bounding. As the crack lengths increase, the separation of the actual PDA from the theoretical PDA tends to increase.

It is concluded that if the detected circumferential cracks are assumed to be 100% throughwall, the as-indicated crack lengths will be inherently conservative with respect to the structural adequacy of the remaining ligament. Therefore, no additional uncertainty factor is necessary to be applied to the as-measured circumferential extent of the cracks.

5.1.2 Consideration of Crack Growth

The growth of cracks due to PWSCC in the present study is dictated by four default PWSCC growth rates from Reference 5-3. The distribution of growth rates is assumed to be lognormal. Typical values and conservative values are given, although it is recommended in Reference 5-3 to use the default values only when the historical information is not available and not to use the typical values unless the degradation is mild. (No significant crack growth data exits for circumferential cracking in the tubesheet expansion region.) Both growth sets provided in Reference 5-3 have mean values and 95% upper bound values. See Table 5-1. For this analysis, the typical 95% upper bound growth rate is used.

The residual structural ligament must be adjusted for growth during the anticipated operating period between the current and the next planned inspection. Typically, the operating periods for the affected plants are 18 calendar months; however, some plants have planned outages in which no primary side inspections will be performed. Therefore, the cycle length adjustments are made to the minimum structural ligament required.

The circumferential growth rates are expressed as inches per EFPY in Table 5-2. Referring to Table 5-2, the maximum allowable throughwall circumferential crack size in a steam generator tube is 214° (=360° – 146° [required minimum ligament]) supporting one cycle of operation. The maximum allowable circumferential crack size in a tube-to-tubesheet weld is 294° (360° – 66° [required minimum ligament]) supporting one cycle of operation.

5.2 LEAKAGE

A basis, using the D'Arcy formula for flow through a porous medium, is provided to assure that the accident induced leakage for the limiting accident will not exceed the value assumed in the safety analysis for the plant if the observed leakage during normal operation is within its limits for the bounding plant. The bounding plant envelopes all other plants who are candidates for applying H*/B*. The D'Arcy formulation was previously compared to other potential models such as the Bernoulli equation or orifice flow formulation and was found to provide the most conservative results.

The length of undegraded crevice required to limit the accident induced leakage to less than the value assumed in the safety analysis for the limiting plant is [ ]^{a,c,e} By definition of the IARC, a tube that can remain in service has an undegraded crevice of 17 inches. Therefore, a factor of safety of [ ]^{a,c,e} is available. Expressed in length terms, the length margin in the crevice is [ ]^{a,c,e}

Significant margin on crevice length is available even if only the distance below the neutral axis of the tubesheet is considered. This distance is approximately [ ]^{a,c,e} During normal operating conditions, the tubesheet flexes due to differential pressure loads, causing the tubesheet holes above the neutral axis to dilate, and below the neutral axis, to constrict. No mechanical benefit is assumed in the analysis due to tubesheet bore constriction below the neutral axis of the tubesheet; however, first principles dictate that the tubesheet bore and crevice must decrease. Therefore, the leakage analysis provided is conservative.

For all IARC candidate plants other than the limiting plant, the margin on length required to limit the accident induced leakage to less than the value assumed in the safety analysis is greater than the values noted above for the bounding plant.

It is also concluded that if the normal operating leakage is within its allowable value, the accident induced leakage will also be within the value assumed in the bounding plants' safety analysis. This conclusion applies for all other plants which would benefit from implementation of the IARC.

5.3 REFERENCES

5-1 ETSS #20510.1; Technique for Detection of Circumferential PWSCC at Expansion Transitions.

5-2 EPRI TR-107197; Depth Based Structural Analysis Methods for Steam Generator Circumferential Indications; November 1997.

5-3 EPRI Document 1012987, "Steam Generator Integrity Assessment Guidelines, Revision 2," July 2006.

Table 5-1
PWSCC Growth Rates (Reference 3-6)

Growth Direction		**Radial (%TW/EFPY)**	**Circumferential (in/EFPY)**
Typical Values	Mean	4.5	0.04
	95% Upper Bound	13.1	0.12
Conservative Values	Mean	7.0	0.08
	95% Upper Bound	20.4	0.24

Table 5-2
Calculation of Required Minimum Ligament for
18 and 36 Months Operating Periods

	Bounding Structural Ligament	EFPY (1)	Growth (In./EFPY) (2)	Growth (Deg./EFPY) (3)	Growth for Operating Period (degrees)	Minimum Structural Ligament (degrees)	Required Minimum Ligament (degrees)
Tube	18 CM Operation	1.5	.12	20.65	31	a,c,e	146
	36 CM Operation	3.0	.12	20.65	62		177
Weld	18 CM Operation	1.5	.12	20.65	31		66
	36 CM Operation	3.0	.12	20.65	62		97

Notes:

4. It is conservatively assumed that 1 EFPY = 1 Calendar Year.
5. 95% upper value of typical growth rates from Reference 5-3.
6. Based on smallest (Model F) mean tubesheet bore dimension.

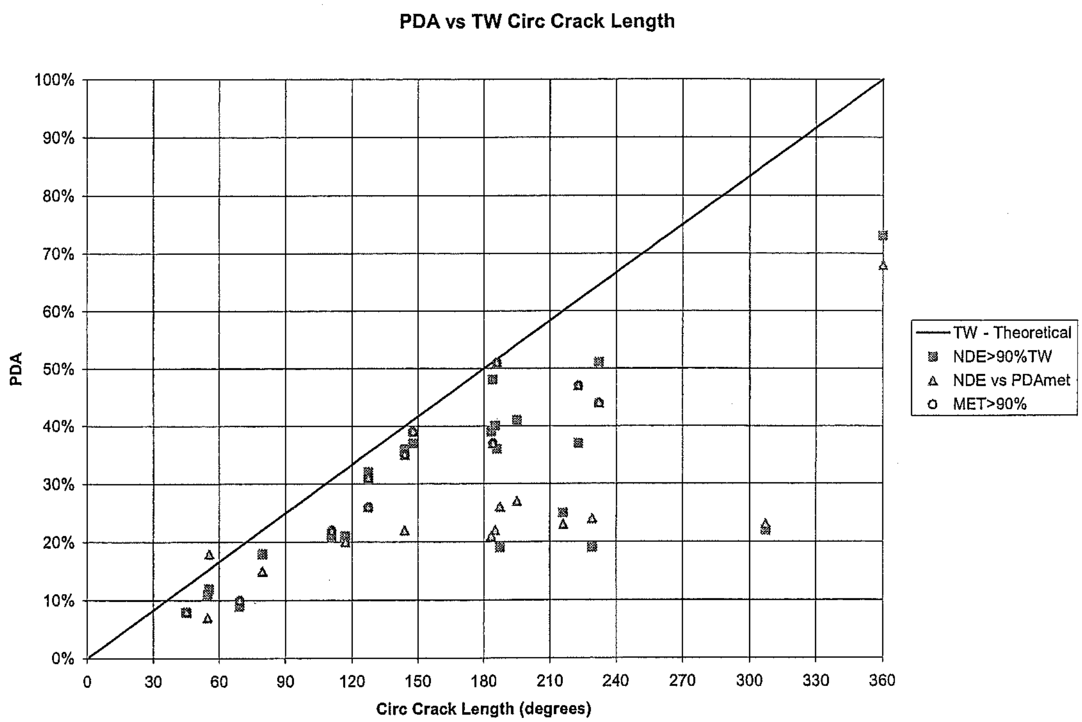

Figure 5-1
Correlation of Circumferential Crack Length and PDA

LTR-CDME-08-43, Rev. 1, NP-Attachment

Response to NRC Request for Additional Information Relating to LTR-CDME-08-11, Rev. 1, NP-Attachment

April 29, 2008

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

QUESTIONS RELATING TO STEAM GENERATOR TUBESHEET AMENDMENT ON INTERIM ALTERNATE REPAIR CRITERIA

The NRC has provided to Wolf Creek Nuclear Operating Corporation (WCNOC) by email dated February 28, 2008 the Request for Additional Information (RAI) relating to an interim alternate repair criterion (IARC) that requires full-length inspection of the steam generator tubes within the tubesheet, but does not require plugging tubes if the extent of any circumferential cracking observed in that region greater than 17 inches from the top of the tubesheet that meets the performance criteria of NEI 97-06, Rev. 2, "Steam Generator Program Guidelines," (Reference 1).

A total of thirteen RAI were provided to WCNOC. Four additional RAI have since been provided to Southern Nuclear Operating Company for Vogtle Units 1 and 2. The same four additional RAI were also provided to Exelon Generation Company for the Braidwood Nuclear Power Station. The responses to RAI 6 through 17 are provided below.

After adjusting for growth as documented in Reference 2, the allowable crack sizes in the tube (203°) and the weld metal (94°) are bounding values and they apply for Model D5, Model F, Model 44F and Model 51F steam generators. The 1.0 inch axial separation criterion discussed herein for multiple circumferential cracks also applies to these same model steam generators. The ASME Code stress report results summarized in response to RAI 9 apply to the Model F steam generator only; however, it has been confirmed that similar results have been obtained for the Model D5 steam generators.

6. *Figure 3-7 (LTR-CDME-08-11-P) needs to provide all geometry details assumed in the weld analysis on pages 7, 9 and 10. (The NRC staff does not understand the assumed weld geometry based on the discussion on pages 7, 9 and 10.) With respect to the equation for S.A. near the top of page 10, what is the parameter whose value is 0.020 and what is the solution for "y"?*

Response: The tube-to-tubesheet weld is modeled in Figure 6-1 below. The tube wall has an inner radius r_i and an outer radius r_o, and it is displaced upward [

]a,c,e

a,c,e

Figure 6-1

The equation of a line, relative to the ellipse is:

y = mx + b, where

the slope = tanθ, and one point is located at (r_o, 0.020). The resulting equation for the line on which the crack grows is:

[ ] a,c,e

Similarly, the equation of the ellipse, as offset from the origin, is:

[ ] a,c,e

where [ ] a,c,e

Simultaneously solving the equations for the line and the ellipse results in the point of their intersection (x, y):

[ ] a,c,e

Setting the points so that they are now relative to the original coordinate system gives the point (x', y').

[ ] a,c,e

The surface area of the frustum, S.A., is calculated by the surfaces of revolution technique and is

[ ] a,c,e

where, the equation for the line can be rewritten as:

[ ] a,c,e

and

Thus,

$$\frac{dx}{dy} = \cot\theta$$

a,c,e

[ ]

and the result is:

a,c,e

[ ]

The previous calculation made use of surfaces of revolution (Φ varies from 0 to 2*π) in order to calculate the surface area of the entire frustum. Now, since the circumferential flaw does not subtend a surface completely around the frustum, the equation must be integrated over an angle of revolution (Φ to Φ+ΔΦ). In addition, as the crack grows along the line of crack propagation, the y-value is integrated from y' to y'+d*sinθ, where d is the crack depth. Thus, in this case, the surface area of the flaw, A_f, is:

a,c,e

[ ]

the final result of which is:

a,c,e

[ ]

The surface area of the circumferential flaw, A_{fc}, is a hybrid of the previous two. The angle of revolution again varies from 0 to 2 π, as in the case of the surface area of the frustum. However, the y-value varies from y' to y'+d*sinθ, just as in the case of the partially circumferential flaw.

Now the integral is:

a,c,e

[ ]

and the result is:

a,c,e

[ ]

7. *On page 10, the assumed flaw is said to extend a distance "d" into this "surface." Does "surface" refer to the outer ellipse or inner ellipse in Figure 3-5? Figure 3-5 suggests it is from the inner ellipse.*

Response: Referring to the frustum pictured in Figure 3-4 on Page 16 of LTR-CDME-08-11, Rev. 1, P-Attachment, viewing the frustum from above (looking down) or viewing the frustum from below (looking up), the view obtained is shown in Figure 3-5. The crack originates in the bottom of the frustum in Figure 3-4 and grows upward along the surface depicted. That is what the crack in Figure 3-5 is attempting to show. The crack originates at the point (x', y') in the first figure provided to answer Question 6.

8. *What was the assumed flow stress for the weld material? What was the basis for selecting this value?*

Response: The weld is an autogenous weld; no filler metal is used. The flow stress assumed for the weld bead is the same as that of the tube (base) metal, which was taken from Westinghouse WCAP-12522 (Reference 3). This is a conservative assumption since the Alloy 182 weld metal used for the tubesheet clad is stronger than the base metal of the tubing. Manufacturer's specifications[1] for Alloy 182 and Alloy 82 weld metal indicate that the yield strength ranges from [ ]^{a,c,e} and the ultimate tensile strength ranges from [ ]^{a,c,e} The flow stress ($0.5*(S_Y+S_{UT})$) then ranges from [ ]^{a,c,e} This range of values is higher than the flow stress used in the tube ligament analysis [ ]^{a,c,e}

9. *LTR-CDME-05-209-P (Reference 5) states that the tube-to-tubesheet welds were designed and analyzed as primary pressure boundary in accordance with the requirements of Section III of the ASME Code. Provide a summary of the Code analysis, including the calculated maximum stress and applicable Code stress limit.*

Response:

General Summary of ASME Code Stress Report Results Relative to the IARC

The existing Model F steam generator tube end weld (TEW) analysis used an axisymmetric finite element model (FEM) to estimate the stress state of the weld material. The assumptions in the weld analysis (Reference 2) closely resemble the assumptions in the IARC (Reference 2). For example, in the Model F FEM analysis there is [

]^{a,c,e}

This result is similar to the [ ]^{a,c,e} plane cited in LTR-CDME-08-11, Rev. 1, P-Attachment, when the different weld surfaces are compared (i.e., the flat plane chosen in the Model F FEM geometry versus the elliptical plane used in LTR-CDME-08-11, Rev. 1, P-Attachment). Therefore, the results described for the limiting weld ligament in LTR-CDME-08-11, Rev. 1, P-Attachment, are reasonable. In addition, the

[1] FAX from Samuel D. Kaiser, P.E., of Inco Alloys Int'l, Inc. Welding Products Co. dated August 31, 1999 to Karan K. Gupta of Westinghouse NEE-Pensacola.

stress results contained in WNET-153, Vol. 6 (Reference 6) for a Model D5 steam generator are bounded by those contained in the Model F steam generator report (Reference 4).

Weld Geometry Model

Figure 9-1 shows the configuration of the weld as modeled in the Code stress analysis. This is a conservative idealization of the actual weld bead, which is approximately an [ ][a,c,e] The interfacing elements to the weld have been added to Figure 9-1 for clarity.

a,c,e

Figure 9-1

The average actual height of the weld bead was determined by destructive examination of 10 factory welds and was found to be [ ][a,c,e] The modeled height of the weld was conservatively set at [ ][a,c,e] To maximize the load applied to the weld, since the dominant loading is tubesheet deflection, a "stiff" tube of [ ][a,c,e] wall thickness was assumed.

Stress Summary

The results of the stress analysis are contained in Table 9-1 for the limiting section of weld [ ][a,c,e]

Table 9-1

Quantity	Design	Emergency	Faulted	Test

a,c,e

Note: P_m is the primary membrane stress intensity

The design primary membrane stress intensity is based on the design pressure differential of [ ]a,c,e and an isothermal temperature of [ ]a,c,e from the Equipment Specification.

Loads and Loading Conditions

There are four sources of applied loads on the weld material:

- Deformation imposed by the tubesheet motion (taken at the center of the tubesheet, assuming no restraint from the divider plate, to maximize the tubesheet deflection). This is the most significant of the loads.
- Primary-to-secondary pressure differences.
- Local temperature gradients. Shown to be "trivial" in the Code stress analysis.
- Isothermal temperature. Local temperature gradients are very small. (Exception: Non-ductile failure evaluation.)

Weld residual stress is not considered because it is stated to be insignificant compared to the operating loads. This is because the ASME Code stress report analysis assumes that there is [

]a,c,e

The end cap loads and fatigue results for the tube end weld were evaluated for several ASME Code defined conditions as specified in the Equipment Specification for the Model F steam generator. The conditions in the analysis included:

- Design Condition
- Normal and Upset Conditions
- Emergency Conditions
- Faulted Conditions
- Test Conditions

Material Properties

The materials used in the FEA model are:

- Tubesheet Ligament: SA-508 Cl 2a
- Tube: SB-163 (Code Case 1484)
- Tubesheet Cladding: Inconel Weld

See the tables below for a detailed description of the appropriate data from the applicable Code year.

TABLE 4-1

MATERIAL PROPERTIES VS. TEMPERATURE FOR SA-508-CL. 2a

Temperature	TC (Btu/hr-ft-°F)	TD (ft^2/hr)	$\alpha \times 10^6$ (in/in-°F)	$E \times 10^{-6}$ (psi)	S_m (ksi)	S_y (ksi)	S_u (ksi)

a,c,e

TC = Thermal Conductivity

TD = Thermal Diffusivity

α = Mean Coefficient of Expansion going from 70°F to indicated temperature.

E = Modulus of Elasticity

S_m = Design Stress Intensity

S_y = Yield Strength

S_u = Ultimate Strength

TABLE 4-2

MATERIAL PROPERTIES VS. TEMPERATURE FOR SB-163 (Code Case 1484)

a,c,e

Temperature	TC (Btu/hr-ft-°F)	TD (ft^2/hr)	$\alpha \times 10^6$ (in/in-°F)	$E \times 10^{-6}$ (psi)	S_m (ksi)	S_y (ksi)	S_u (ksi)

TC = Thermal Conductivity

TD = Thermal Diffusivity

α = Mean Coefficient of Expansion going from 70°F to indicated temperature.

E = Modulus of Elasticity

S_m = Design Stress Intensity

S_y = Yield Strength

S_u = Ultimate Strength

The thermal properties and the elastic modulus of the cladding are assumed to be the same as those for the tube.

Thermal Analysis

The thermal analysis considered a bounding transient for Normal and Upset conditions, Inadvertent RCS Depressurization. For this transient, the maximum calculated temperature difference between the nodes represented in the FEA model is [ ]^{a,c,e} It was concluded that the [

]^{a,c,e}

Method of Analysis

The analysis was performed with an axisymmetric finite element analysis in the WECAN computer program with a very fine nodal mesh in the weld area and its interfaces with the tube and the tubesheet clad. The elements consisted of [

]^{a,c,e} Applied loads were due to deformation imposed by the tubesheet motion, primary-to-secondary pressure differences, local temperature gradients, and isothermal temperature.

Calculated Stresses

The following tables are reproductions of the tables included in the code stress analysis for the tube end weld.

Table 7-5 shows that the [

]^{a,c,e} The section numbers in Table 7-5 correspond to the section numbers in the model description figure above. In order to demonstrate acceptability, [

]^{a,c,e}

TABLE 7-1

DESIGN CONDITION STRESSES

Section	Stress Category	Stress Components, ksi σ_Z σ_R τ_{RZ} σ_H	Principal Stresses, ksi S_1 S_2 S_3	Maximum Stress Intensity, ksi	Allowable Stress Limit, (ksi)

a,c,e

TABLE 7-2

EMERGENCY CONDITION STRESSES

Section	Stress Category	Stress Components, ksi σ_Z σ_R τ_{RZ} σ_H	Principal Stresses, ksi S_1 S_2 S_3	Maximum Stress Intensity, ksi	Allowable Stress Limit, (ksi)

a,c,e

TABLE 7-3

FAULTED CONDITION STRESSES

Section	Stress Category	Stress Components, ksi				Principal Stresses, ksi			Maximum Stress Intensity, ksi	Allowable Stress Limit, (ksi)
		σ_Z	σ_R	τ_{RZ}	σ_H	S_1	S_2	S_3		

a,c,e

TABLE 7-4

TEST CONDITION STRESSES

Section	Stress Category	Stress Components, ksi				Principal Stresses, ksi			Maximum Stress Intensity, ksi	Allowable Stress Limit, (ksi)
		σ_Z	σ_R	τ_{RZ}	σ_H	S_1	S_2	S_3		

a,c,e

TABLE 7-5

NORMAL AND UPSET CONDITION
PRIMARY PLUS SECONDARY STRESS INTENSITY RANGE

Location	$(P_L + P_b + Q)_{max}^{**}$ (ksi)	Allowable Limit $3S_m$ (ksi)

a,c,e

[redacted]

* Section numbers are identified in the figure included with the Weld Geometry Description, above.

** All transients creating primary-plus-secondary stress intensity ranges greater than $3S_m$ are evaluated inelastically.

Summary of Fatigue Usage from Code Stress Analysis of the Tube End Weld:

a,c,e

[ ]

The point of maximum usage factor, where [$]^{a,c,e}$ is the most likely fatigue crack initiation point, although the usage is still less than 1.0.

<u>Non-Ductile Failure Evaluation</u>

The methods of evaluating non-ductile failure are [$]^{a,c,e}$

10. Regarding the weld repair criterion:

a. A detailed stress analysis (e.g., finite element) would be expected to reveal a much more complex stress state than that assumed in the licensee's analysis, which may impact the likely locations for crack initiation and direction of crack propagation. In addition, the dominant stresses for crack initiation and crack growth may involve residual stresses in addition to operational stresses. Also, flaws may have been introduced during weld fabrication. Thus, the 35-degree conical "plane" is not the only plane within which cracks may initiate and grow.

b. One hypothetical crack plane, which appears more limiting than the one assumed by the licensee, is the cylindrical "plane" defined by the expanded tube outer diameter where the weld is in a state of shear. Assuming a flow stress of 63.7 ksi and an effective weld depth of 0.035 inches (as shown in LTR-CDME-05-209-P, Figure 2-1), the NRC staff estimates that the required circumferential ligament to resist an end cap load of 1657 lb is greater than 180 degrees (without allowances).

Address these concerns and provide a detailed justification for why the submitted analysis is conservative.

Response: Weld residual stress (WRS) was not considered since there is no definitive basis for any value used. Both the original Wolf Creek code stress analysis and a more recent code stress analysis for different models of steam generators dismiss residual stresses in the weld as negligible.

Development of credible residual stresses using FEA methods is extremely difficult, particularly for small welds like the tube-end weld. A comprehensive test program involving deep/shallow hole drilling, or finite element analyses which include the birthing of elements under very high temperatures to simulate the welding process would be required in order to develop a value for use. Verification of finite element WRS analysis results by deep/shallow hole drilling can only be accomplished for larger volumes of weld metal as removal of cores of trepanned material is required. For small volumes of weld metal, verification of the finite element analysis is much more difficult and thus, the WRS values assumed are more uncertain.

In the ASME Code stress analyses, the operating loads on the weld are characterized as overshadowing any effects of WRS. Current development of residual stress models (unpublished) for consideration as a Code Case indicate that the stress on the inner diameter of the tube is compressive, and not conducive to crack opening. The WRS values used as the basis of the modeling were taken from the heat affected zone (HAZ) of stainless steel welds; therefore, the actual WRS profile may be different. The profile is tensile in some areas and compressive in others (only tensile components of WRS have a deleterious effect). Consideration of WRS further complicates the analysis, but does not necessarily add any conservatism.

The weld region is not in a state of pure shear. There are tensile loads as well as the pressure acting on the face of the weld exposed to primary coolant. Therefore, the limits for pure shear (ASME B&PV Code Section III, NB-3227.2) are not considered to apply. Thus, the ASME code is satisfied with respect to pure shear. The shear plane used in the IARC weld ligament calculation was only used to calculate the shear component of the stress state. This is consistent with the original Wolf Creek code stress analysis in which shear was not explicitly considered, and the shear plane identified was not found to be the limiting plane. The most likely crack initiation point, due to fatigue usage, was on a plane extending from the weld root almost normal to the face of the weld. A recent code stress analysis for another plant did consider pure shear explicitly and determined that the weld region is not in a state of pure shear, thus supporting the WCNOC stress analysis. This report definitively stated that the pure shear limit of NB-3227.2 ($0.6S_m$) does not apply.

The crack opening performed in the weld region for the Wolf Creek IARC was assumed to open due to maximum principal stress, which is tensile, and flow stress was chosen as the limiting strength parameter. While reviewing the Wolf Creek IARC report, it was found that the component stresses, which generate the principal stresses, were not being recalculated as the flaw grew. The correction to this problem (see below), which is documented in Reference 7, changed the bounding required remaining ligament for partially circumferential flaws in the weld region to [ ]^{a,c,e} (not adjusting for growth) from the approximately [ ]^{a,c,e} originally reported in LTR-CDME-08-11, Rev. 1, P-Attachment, (reference Table 3-3). The value of [ ]^{a,c,e} supersedes the old value of [ ]^{a,c,e} Westinghouse believes that these corrections make the consideration of the flaw area in the left hand side of the force balance equations correct.

The normal stress component was:

a,c,e

[ ]

The normal stress component now is:

a,c,e

[ ]

The shear stress reported in the Wolf Creek IARC was:

a,c,e

[ ]

The shear stress component, until the flaw breaches the weld root is now:

a,c,e

[ ]

b is the semi-minor axis (0.014 inch). This is due to the shear path being uninterrupted until that point. After breaching the weld root, there is a lack of a stress path. The shear stress at that point, is:

a,c,e

[ ]

11. *The proposed tube and weld repair criteria do not address interaction effects of multiple circumferential flaws which may be in close proximity (e.g., axial separation of one or two tube diameters). Address this concern and identify any revisions which may be needed to the alternate tube repair criteria and the maximum acceptable weld flaw size.*

Response: In order to ascertain how far apart cracks must be in order to be considered to respond independently to an applied far field stress, a fracture mechanics approach was undertaken. The assumed case was [

]a,c,e

[

][a,c,e] Therefore, a conservative estimate of the distance necessary to prevent the interaction between cracks is [

][a,c,e] and is equal to 1.0 inch. It is also worthy to note that 1.0 inch, which is between 1 and 2 tube diameters, bounds the 0.5 inch result contained in the ASME Boiler and Pressure Vessel Code, Section XI, Article IWA-3000.

a,c,e

[

]

Figure 11-1. Individual Steam Generator Results for the Distance Necessary for σ_{yy} to Equal σ

a,c,e

Figure 11-2. Combined Steam Generator Results for the Distance Necessary for σ_{yy} to Equal σ

The impact of the crack separation analysis is summarized below. Refer to Figures 11-3 through 11-5 for explanations of the crack geometries and combinations of crack-like indications considered in the analysis. Table 11-1 is a summary of the text description of the crack separation analysis impacts. The details described in Table 11-1 apply only to the portion of the tube within the tubesheet 17 inches below the top of the tubesheet (TTS-17 inches).

An Industry Peer Review was conducted on March 12, 2008 at the Westinghouse Waltz Mill Site with the purpose of reviewing the Fall 2007 Catawba Unit 2 cold leg tube end indications to establish whether the reported indications are in the tube material or the weld material. A consensus was reached that the 2007 Catawba Unit 2 cold leg indications most likely exist within the tube material. However, some of the indications extend close enough to the tube end that the possibility that the flaws do extend into the weld could not be ruled out. Therefore, in order to address the potential for cracking in the tube weld in parallel to crack-like indications in the tube, the more limiting ligament size of [ ][a,c,e] (including the adjustment for growth) for the weld is used to establish the allowable crack size in the tube for cracks less than 1.0 from the tube end.

Crack-like indications in a tube:

1. If any circumferential crack-like indication in the tube exceeds 203°, plug the tube.

2. If there is more than one circumferential crack-like indication in a tube, and no single crack angle exceeds 203°, and the minimum axial distance of separation between the crack-like indications is

greater than or equal to 1.00 inch, then the maximum crack angle is used to describe the flaw and the tube remains in service.

3. If there is more than one circumferential crack-like indication in a tube, and no single crack angle exceeds 203°, and the minimum axial distance of separation between the crack-like indications is less than 1.00 inch, and the non-overlapping sum of the crack angles plus the overlapped crack angle is less than or equal to 203°, the tube may remain in service.

4. If there is more than one circumferential crack-like indication in a tube, and no single crack angle exceeds 203°, and the minimum axial distance of separation between the crack-like indications is less than 1.00 inch, and the non-overlapping sum of the crack angles plus the overlapped crack angle is greater than 203°, plug the tube.

Crack-like indications in a tube less than 1.0 inch from the tube end:

5. If there are one or more cracks in the tube that are each less than or equal to 94°, and there is a minimum axial separation distance between the tube end and the tube cracks of less than 1.00 inch, and the non-overlapping sum of the tube crack angles plus the overlapped crack angle is less than or equal to 94°, the tube may remain in service.

6. If there is a crack-like indication in the weld less than or equal to 94° and there are one or more cracks in the tube that are each less than or equal to 94°, and there is a minimum axial separation distance between the tube end and the tube cracks of less than 1.00 inch, and the non-overlapping sum of the tube crack angles plus the overlapped crack angle is greater than 94°, plug the tube.

Table 11-1: Summary of Crack Separation Analysis and Interactions					
	Multiple Cracks?	Max. Crack Angle in Tube, $\theta^{1,2}$	Max. Crack Angle in Weld, α^{1}	Min. Axial Separation Distance, L	Required Action
Case	-	Degrees (°)	Degrees (°)	inch	-
1	No	> 203	No Crack	N/A	Plug Tube
2	Yes	$\theta_1, \theta_2, \theta_n \leq 203$	No Crack	≥ 1.00	Cracks do not interact. Report max. crack angle less than 203°. Leave in Service.
3	Yes	$\theta_1+\theta_2+\theta_n \leq 203$	No Crack	< 1.00	Sum of total non-overlapping crack angle plus overlap angle less than 203°. Leave in Service.
4	Yes	$\theta_1+\theta_2+\theta_n > 203$	No Crack	< 1.00	Sum of total non-overlapping crack angle plus overlap angle greater than 203°. Plug Tube.
5	Yes	$\theta_1+\theta_2+\theta_n+\alpha \leq 94$	*Possible Crack in Weld*	< 1.00[3]	Sum of total non-overlapping crack angle plus overlap angle less than 94°. Cracks in weld and tube do interact. Leave in Service.
6	Yes	$\theta_1+\theta_2+\theta_n+\alpha > 94$	*Possible Crack in Weld*	< 1.00[3]	Sum of total non-overlapping crack angle plus overlap angle greater than 94°. Cracks in weld and tube do interact. Plug Tube.

1. See Figures 11-3, 11-4 and 11-5 for tube crack angle and weld crack angle definition.
2. θ_n is the sum of any remaining crack angles after the first two crack-like indications. For example, the statement: $\theta_1+\theta_2+\theta_n \leq 203°$ is equivalent to writing: $\theta_1 + \theta_2 + \theta_3 + \ldots \leq 203°$.
3. Separation distance, L, is measured from the tube end.

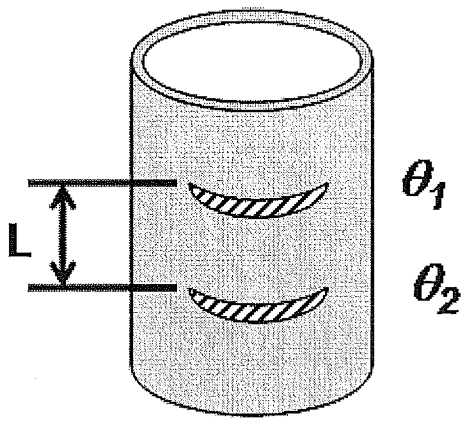

Figure 11-3: Tube Crack Geometry

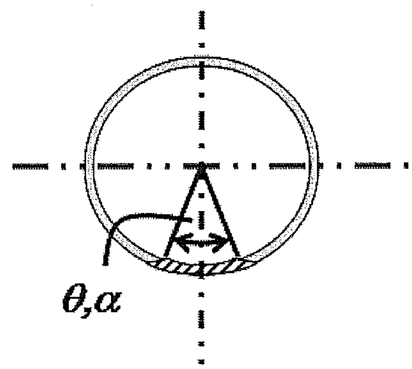

Figure 11-4: Tube and Weld Crack Angle Measurement

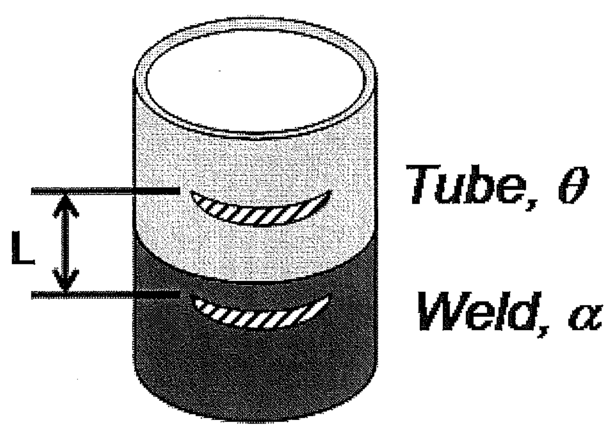

Figure 11-5: Axial Separation Distance Between Weld and Tube Crack-Like Indications

12. *The technical support document for the interim ARC amendment does not make it clear how licensees will ensure they satisfy the accident induced leakage performance criteria. Describe the methodology to be used to ensure the accident induced leakage performance criteria is met. Include in this response (a) how leakage from sources other than the lower 4-inches of the tube will be addressed (in the context of ensuring the performance criteria is met), and (b) how leakage from flaws (if any) in the lower 4-inches of the tube will be determined (e.g., determining the leakage from each flaw; multiplying the normal operating leak rate by a specific factor).*

Response:

The Modified B* leakage analysis in the IARC report calculates the ratio of undegraded crevice length determined by eddy current inspection to the length of undegraded crevice required to meet the design basis accident analysis primary-to-secondary leakage analysis assumption for the limiting design basis accident. By definition of the IARC, 17 inches from the top of the tubesheet is the available undegraded crevice length because confirmed cracking in this length will require the tube to be plugged. Both the pressure difference ratio and the length of crevice during normal operating and design basis accident are factored in the margin determination.

Referring to Table 4-5 of the IARC report, the limiting design basis accident for WCGS is a postulated steam line break (SLB) event. Referring to Table 4-2 of the IARC report, it is calculated that [ ]^{a,c,e} of undegraded crevice length is required to preclude exceeding the SLB accident analysis leak rate assumption of 0.25 gpm. This corresponds to a safety factor of approximately [ ]^{a,c,e} in terms of the ratio of non-degraded crevice as confirmed by eddy current inspection (17 inches) to the crevice length calculated using the D'Arcy equation necessary to preclude exceeding the SLB accident analysis leakage assumption [ ]^{a,c,e} Therefore, the maximum leakage rate that would occur during a postulated SLB event from cracks occurring 17 inches below the top of the tubesheet is calculated to be [ ]^{a,c,e} from the faulted SG. This provides a margin of [ ]^{a,c,e} on leakage rate for other sources of accident-induced leakage.

The table below shows the available margin for leakage sources other than the tubesheet based on the IARC method for calculating the estimated leakage for which a bounding zero-contact-pressure value of loss coefficient, based on the available test data, is used.

Table 12-1: Calculation of Available Margin for Leakage Sources Other Than in the Tubesheet During the Limiting Plant Design Basis Accident (DBA)

a,c,e

Plant	NOP Leak Limit	Limiting Plant DBA	DBA Leak Limit	L Required for DBA	Safety Margin	DBA Leak Margin Available

The response to Question 13 (following) further clarifies the methodology for satisfying the accident-induced leakage performance criteria. For the underlying assumptions of the IARC – no contact pressure between the tube and the tubesheet in the hydraulic expansion region – the discussion above shows that significant margins exist over the length of the crevice required in the 17 inch span below the top of the tubesheet. However, a conservative factor of 2.5 will be applied to that part of the observed normal operating leakage that cannot be associated with degradation mechanisms outside the tubesheet expansion region to calculate the accident-induced leakage from the tubesheet region. The resulting calculated accident-induced leakage will be added to the predicted leakage from other degradation mechanisms that have been detected in the SGs that have the potential to result in accident-induced leakage for evaluation against the accident-induced leakage performance criteria.

13. *The proposed "modified B*" approach relies to some extent on an assumed, constant value of loss coefficient, based on a lower bound of the data. This contrasts with the "nominal B*" approach which, in its latest form (as we understand it) is not directly impacted by the assumed value of loss coefficient since this value is assumed to be constant with increasing contact pressure between the tube and tubesheet. Given the amount of time for the NRC staff to review the interim ARC, the NRC staff will not be able to make a conclusion as to whether the assumed value of loss coefficient in the "modified B*" approach is conservative. However, the NRC staff has performed some evaluations regarding the potential for the normal operating leak rate to increase under steam line break conditions using various values of (l_{NOP}/ l_{SLB}) determined from the "nominal B*" approach (which does not rely on an assumed value of loss coefficient). With*

these analyses and recognizing the issues associated with some of these previous H/B* analyses, it would appear that a factor of 2.5 reasonably bounds the potential increase in leakage that would be realized in going from normal operating to steam line break conditions. Discuss your plans to modify your proposal to indicate that the leak rate during normal operation (for flaws in the lower 4-inches of tube) will increase by a factor of 2.5 under steam line break conditions.*

[The NRC staff makes two observations here in response to possible industry concerns regarding Item 11. First, the NRC staff acknowledges that the ratio of the allowed accident leakage and the operational leakage is only 2.5 for Wolf Creek, which is equal to the factor of 2.5 above. (This ratio is 3.5 for Vogtle and 5 for Byron/Braidwood). This is not an atypical situation as is discussed in NRC RIS 2007-20. The operational leakage limit in the technical specifications can never be assumed to ensure that accident leakage will be within what is assumed in the accident analysis, even if the technical specification limit is zero. For example, part through wall flaws in the free span which are not leaking under normal operating conditions may pop through wall and leak under accident conditions. For cracks in the free span which are leaking under normal operating conditions, the ratio of SLB leakage to normal operating leakage can be substantially greater than 2.5 depending on the length of the crack. It is the licensee's responsibility to ensure that the accident leakage limits are met through implementation of an effective SG program, including an engineering assessment of any operational leakage that may occur in terms of its implications for leakage under accident conditions (based on considerations such as past inspection results and operational assessments, experience at similar plants, etc.).

Second, the NRC staff is not aware of any operational leakage to date from the tubesheet region for the subject class of plants, and there seems little reason to expect that this situation will change significantly in the next 18 months. Thus, the NRC staff's approach discussed above is not expected to have any significant impact for the licensees requesting relief from the tube repair criteria in the lower 4-inches of the tube.]

Response:

The proposed ratio of 2.5 of the SLB to NOP leakage is conservative from the perspective of predicted SLB leak rate from a postulated flaw below TTS-17 inches based on the analysis below. Based on the D'Arcy Model for flow in an axial porous medium, if no value for loss coefficient is assumed, the increase in predicted leakage from the tubesheet region would be lower than that determined by using a factor of 2.5 and also than that provided in the IARC justification.

For example, assume that both the loss coefficient and the length of porous medium surrounding a tube above a postulated crack are constant during both normal operating (NOP) and steam line break (SLB) conditions. The crevice below the neutral axis of the tubesheet will be tighter during accident conditions even if no credit is taken for thermal lockup between the tube and the tubesheet due to increased pressure differential across the tube. If the pressure differential across the tube at SLB conditions is discounted, the resulting condition is still an increase in contact pressure due to structural deflections and rotations. Thus, there is no basis to assume a lower loss coefficient at SLB condition than at NOP condition. Further, the viscosity during a SLB accident would be higher, due to the reduced temperatures in the crevice. Therefore, the assumption of a constant value for loss coefficient is, in fact, the worst case, and is reasonable and conservative for the IARC because the flow resistance is expected to increase during a postulated SLB event below 17 inches from the top of the tubesheet.

Following the assumptions described in Question 13 (above), the D'Arcy Model becomes:

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

$K_{NOP} = K_{SLB} = K$

$l_{NOP} = l_{SLB} = 17 \text{ in} = l$

This assumption forces the estimated increase in leakage to be a factor based on the ratio of differential pressures and the ratio of the applicable viscosities only. For the Wolf Creek steam generators, the viscosity of the fluid during NOP conditions is approximately 1.75×10^{-6} lbf-sec/in^2 and during SLB is approximately 2.66×10^{-6} lbf-sec/in^2. The pressure differential ($\Delta p = P_{PRI} - P_{SEC}$) for Wolf Creek during NOP is 1443 psig and the pressure differential during SLB is 2560 psig. Substitution of these values into the D'Arcy Model gives,

$$Q_{SLB} = \frac{2560}{2.66e-6(Kl)} = 9.624e8 / Kl$$

$$Q_{NOP} = \frac{1443}{1.75e-6(Kl)} = 8.245e8 / Kl$$

$$\frac{Q_{SLB}}{Q_{NOP}} = \frac{9.624e8}{8.245e8}\frac{Kl}{Kl} = \frac{9.624e8}{8.245e8}\frac{1}{1} = \frac{9.624e8}{8.245e8} = 1.167$$

Using the D'Arcy Model to calculate the estimated increase in leakage during SLB yields a result of approximately 1.17. This is less than the conservative ratios which range from 2 to 6 as reported in the IARC description and the 2.5 factor proposed by the NRC staff.

For integrity assessments, the ratio of 2.5 will be used in the completion of both the condition monitoring (CM) and operational assessment (OA) upon implementation of the IARC. For example, for the CM assessment, the component of leakage from the lower 4 inches for the most limiting steam generator during the prior cycle of operation will be multiplied by a factor of 2.5 and added to the total leakage from any other source and compared to the allowable accident analysis leakage assumption. For the OA, the difference in leakage from the allowable limit during the limiting design basis accident minus the leakage from the other sources will be divided by 2.5 and compared to the observed leakage. An administrative limit will be established to not exceed the calculated value.

It is not planned to modify the existing IARC report, but, as noted above, a constant multiplier of 2.5 will be used in CM and OA evaluations to calculate SLB leakage from the lower 4 inches.

14. *The mathematical constant π has been omitted from the first term of the equation near the top of page 8 and the equation at the bottom of page 8. It is not clear if this is a typographical error, or if π has been purposefully omitted. If the omission is intentional, please explain.*

Response:

Two typographical errors have been identified in the left hand side of the equations for force balance for the partial circumferential flaw in the steam generator tube wall and the partially circumferential, through-wall flaw in the steam generator tube wall on Page 8 of LTR-CDME-08-11, Rev. 1, P-Attachment. A factor of π was omitted in each equation in the report but not in the actual calculations. The calculation results are not affected by the typographical errors.

15. *The last term of the equation at the bottom of page 8 includes the parenthetical (r_o^2+r_i^2). The staff believes that this should be (r_o^2-r_i^2). It is not clear if this is a typographical error, or if the radii are intentionally being summed. If intentional, please explain why the squared radii should be summed and not subtracted.*

Response:

Westinghouse agrees that the plus sign (+) should indeed be a minus sign (-). The error is typographical and did not affect the calculations. The last term in the force balance equation for the partially circumferential, through-wall flaw in the steam generator tube contains a $\sigma \times (1/2) \times (r_o^2 + r_i^2) \times \Delta\theta$ term on the right hand side of the equation. That should read $\sigma \times (1/2) \times (r_o^2 - r_i^2) \times \Delta\theta$.

16. *Explain why it is necessary to subtract A_f (area of the flaw) from S.A. (surface area of the frustum) in the first term of the force balance on page 10. (The staff believes that this term should be deleted.)*

Response:

The area of the flaw must be subtracted from the surface area of the frustum when calculating the force balance because that area is no longer contiguous and cannot react to the applied stress. In other words, the flaw area is no longer available to the principal stress, but, is instead loaded by the internal pressure.

17. *Explain the use of the mathematical constant P_i (internal pressure) rather than P (3ΔP or 4800 psi) on the equations on pages 8 and 10. The explanation on page 11 is not sufficient and appears to the staff to be incorrect.*

Response:

It remains Westinghouse's position that it is conservative and correct to use an internal pressure of 2250 psi on the crack flank to calculate an acceptable remaining ligament for crack-like indications that may be present in the tube and weld. However, at the NRC staff's request, the allowable ligament sizes for the tube and the weld were recalculated assuming a 4800 psi differential pressure on the crack flank. The revised values for remaining ligament for the tube and the weld are [ ]^{a,c,e} (including an adjustment for growth) respectively.

For completeness, a summary of the Westinghouse position on the justification for the use of an internal pressure of 2250 psi is provided below.

A SG tube is a thick-wall cylinder. This is consistent with the ASME Code stress analysis of the steam generator tubing. Roark (Reference 8) defines a thin-wall cylinder as a cylinder with an inside radius to thickness ratio (R/t) greater than 10. For the Model F tube, R/t = 8.8, therefore, the tube is considered a thick-wall cylinder.

Reference 9 provides the equation of axial stress in the thick wall cylinder as:

$$\sigma_{ZZ} = \frac{p_1 a^2 - p_2 b^2}{b^2 - a^2} + \frac{P}{\pi\left(b^2 - a^2\right)}$$

Where P is an active external load (for this case = 0)

p_1 is the internal pressure

p_2 is the external pressure

a is the inside radius

b is the outside radius

The second term in the equation, $\frac{P}{\pi\left(b^2 - a^2\right)}$, goes to zero because the applied external load in this case is zero.

The equation is conservatively simplified by assuming the $p_2 b^2$term is negligible. Making this assumption conservative since retaining the term would reduce the axial calculated stress σ_{zz}.

The equation is reduced to let p_1 equal the pressure differential Δp. This is consistent with the equation in example 11.2 of Reference 9. This equation, and the following limitations, are echoed in Roark (Reference 8) Table 13.5, Case 1.b. The final equation for the calculation of stress due to the end cap load becomes

$$\sigma_{ZZ} = \frac{\Delta p a^2}{b^2 - a^2}$$

Calculation of the end cap load using this form of the equation is inherently conservative.

The limitation of the equation for axial stress in the thick-wall cylinder due to end cap load, and for the stress equations in the cylinder, is that the section of interest is far removed from the end caps (Reference 9). Consequently, the stress in the degraded section of the cylinder is increased by the reduced wall, but the end cap load remains constant. Calculating the end cap load for the thick-wall cylinder using the degraded wall thickness is equivalent to assuming that the wall thickness for the entire tube is the same as for the degraded local section.

It is the Westinghouse position that the load on the crack flank should be calculated separately from the end cap load. This is based on the fact that the end cap load already takes into account any variation in the cross section of the tube.

The underlying assumption for the IARC is that all circumferential cracks detected are 100% through wall over the entire indicated length. The Westinghouse crevice pressure test data (Reference 10) shows that the pressure in the crevice external to the tube in the immediate area of the penetration is the same as the internal pressure; therefore, there is no differential pressure at that location and $3\Delta p$ equals zero. The existing analysis conservatively applies the entire primary side pressure to the crack face. There is no operating condition that justifies using triple the primary pressure differential on the crack face and the required safety by the ASME Code for this situation (classification as secondary stress) would imply a safety factor of 1.0 on any primary side pressure.

Finally, the stresses calculated on the degraded section are compared to the flow stress which is very conservative for this situation. The condition of interest is one of pure axial separation under the assumption of the IARC, i.e., no axial friction forces between the tube and the tubesheet, but the tubesheet is present in close contact to prevent bending forces. For pure axial separation, it is appropriate to use the ultimate strength of the material, since no bending can occur and burst is not possible due to the constraint provided by the tubesheet.

References:

1. NEI-97-06, Rev. 2, "Steam Generator Program Guidelines," May 2005.

2. LTR-CDME-08-11, Rev. 1, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," April 29, 2008.

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

4. WNET-180 (Proprietary), Volume 11, Rev. 0, "Model F Steam Generator Stress Report," Westinghouse Electric, Pittsburgh, PA, September 1980.

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

6. WNET-153 (Proprietary), Volume 6, Rev. 0, "Model D5 Steam Generator Stress Report," Westinghouse Electric, Pittsburgh, PA, December 1981.

7. CN-CDME-08-4, Rev. 1, "Structural Evaluation of the Minimum Circumferential Ligament Required as Part of the WCNOC IARC," March 2008.

8. Roark's Formulas for Stress and Strain, Warren C. Young and Richard G. Budynas, Seventh Edition, McGraw-Hill, 2002.

9. Advanced Mechanics of Materials, Arthur P. Boresi and Richard J. Schmidt, Sixth Edition, John Wiley and Sons, 2003.

10. STD-MC-06-11-P, Rev. 1, "Pressure Profile Measurements During Tube-to-Tubesheet Leakage Tests of Hydraulically Expanded Steam Generator Tubing," August 30, 2007.

To: D. Peck
G. Turley

cc: G. W. Whiteman
J. A. Gresham
E. P. Morgan
C. D. Cassino
J. T. Kandra

Date: May 23, 2008

From: H.O Lagally

Your ref:

Ext: (724) 722-5082

Our ref: LTR-CDME-08-125

Fax: (724) 722-5909

Subject: **Applicability of the IARC Technical Justification to Point Beach 1**

References:

The technical justification for the tube-end IARC, LTR-CDME-08-11,Rev 1 "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone" and LTR-CDME-08-43, Rev 1, "Response to NRC Request for Additional Information (RAI) Relating to LTR-CDME-08-11 P-Attachment", together with the Affidavits of Withholding for them were transmitted to Point Beach under separate cover. Each of these letters includes the proprietary and non-proprietary version of the respective document. Together, these documents provide the technical basis for justification of an Interim Alternate Repair Criterion for the lower 4 inches of the tubesheet expansion region.

As a product of a jointly-funded effort among a number of utilities for the development of the IARC, the technical justification was developed as a bounding case for the affected plants with hydraulically expanded Alloy 600TT tubing, including Point Beach Unit 1. Therefore, the technical justification contained in these documents applies directly to Point Beach Unit 1.

Please contact the undersigned should you have any questions or concerns.

Author:
* H. O. Lagally, Fellow Engineer
Chemistry, Diagnostics and
Materials Engineering

Verifier:
* J. T. Kandra
Chemistry, Diagnostics and
Materials Engineering

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

ENCLOSURE 5

FPL ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNIT 1

LICENSE AMENDMENT REQUEST 257
TECHNICAL SPECIFICATION 5.5.8, STEAM GENERATOR PROGRAM
PROPOSED LICENSE AMENDMENT REQUEST,
INTERIM ALTERNATE REPAIR CRITERIA (IARC)
FOR STEAM GENERATOR TUBE REPAIR

WESTINGHOUSE ELECTRIC COMPANY LLC AUTHORIZATION LETTERS:

CAW-08-2423,
"APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE,"
DATED MAY 19, 2008.

CAW-08-2424,
"APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE,"
DATED MAY 19, 2008.

10 pages follow

Westinghouse

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Our ref: CAW-08-2423

May 19, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-CDME-08-11, Rev. 1 P-Attachment, "Interim Alternate Repair Criterion (IARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated April 29, 2008 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-08-2423 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 FPL Energy Point Beach, LLC.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-08-2423, 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,

R. M. [signature] FOR

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

Enclosures

cc: Jon Thompson (NRC O-7E1A)

bcc: J. A. Gresham (ECE 4-7A) 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 (letter and affidavit only)
G. W. Whiteman, Waltz Mill
H. O. Lagally, Waltz Mill
C. D. Cassino, Waltz Mill
J. T. Kandra, Waltz Mill
D. E. Peck, ECE 560C

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared B. F. Maurer, 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:

B. F. Maurer, Manager
ABWR Licensing

Sworn to and subscribed before me
this 19th day of May, 2008

Sharon L. Markle
Notary Public

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-CDME-08-11, Rev. 1 P-Attachment, "Interim Alternate Repair Criterion (IARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated April 29, 2008 (Proprietary), for submittal to the Commission, being transmitted by FPL Energy Point Beach, LLC Application for Withholding Proprietary Information from Public Disclosure to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Point Beach Unit 1 is expected to be applicable to other licensee submittals in support of implementing an interim alternate repair criterion (IARC) that requires a full-length inspection of the tubes within the tubesheet but does not require plugging tubes with a certain arc length of circumferential cracking below 17 inches from the top of the tubesheet.

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

(a) Provide documentation of the analyses, methods, and testing for the implementation of an interim alternate repair criterion for the portion if the tubes within the tubesheet of the Point Beach Unit 1 steam generators.

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

Further this information has substantial commercial value as follows:

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

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

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

FPL Energy Point Beach, LLC

Letter for Transmittal to the NRC

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

Enclosed are:

1. 1 copy of LTR-CDME-08-11, Rev. 1 P-Attachment, "Interim Alternate Repair Criterion (IARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated April 29, 2008 (Proprietary)

2. 1 copy of LTR-CDME-08-11, Rev. 1 NP-Attachment, "Interim Alternate Repair Criterion (IARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated April 29, 2008 (Non-Proprietary).

Also enclosed is Westinghouse authorization letter CAW-08-2423 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

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

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

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-08-2423 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.