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10 CFR 50.90

RS-08-075

June 17, 2008

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

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

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

References:

(1) Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Application for Steam Generator Tube Interim Alternate Repair Criteria Technical Specification Amendment," dated February 25, 2008

(2) Electronic mail from M. David (U. S. Nuclear Regulatory Commission) to K. Nicely (Exelon Generation Company, LLC), "RAIs for Braidwood Interim SG Tube ARC LAR," dated March 11, 2008

(3) Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Application for Steam Generator Tube Interim Alternate Repair Criteria Technical Specification," dated March 27, 2008

(4) Letter from M. J. David (U. S. Nuclear Regulatory Commission) to C. G. Pardee (Exelon Generation Company), "Braidwood Station, Units 1 and 2, Issuance of Amendments Re: Revision of Technical Specifications for the Steam Generator Program (TAC Nos. MD8158 and MD 8159)," dated April 18, 2008

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC (EGC) is requesting an amendment to Appendix A Technical Specifications (TS), of Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. This amendment application proposes a one cycle revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," to incorporate an interim alternate repair criteria in the provisions for SG tube repair criteria during Byron Station Unit 2 refueling outage 14 and the subsequent operating cycle.

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In addition, reporting requirement changes are proposed to TS 5.6.9, "Steam Generator (SG) Tube Inspection Report."

While the industry, Westinghouse Electric Company LLC, (Westinghouse) and the NRC work on resolution of issues to obtain approval of the permanent alternate repair criteria (ARC) (i.e., the H*/B* methodology), the NRC recently indicated that they would not be able to approve a permanent ARC in time to support the Fall 2008 Byron Station Unit 2 refueling outage. In response to this, EGC is submitting this one-cycle amendment request. This request is consistent to that approved in the Spring of 2008 for Wolf Creek Nuclear Operating Corporation (WCNOC), Southern Nuclear Operating Company (SNC), and EGC's Braidwood Station (Reference 4) and therefore contains the same technical justifications used to support the Spring 2008 submittals.

These changes are supported by Westinghouse documents, LTR-CDME-08-11, Revision 3, P-Attachment, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," and LTR-CDME-08-43 P-Attachment, Revision 3, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11, Revision 3, P-Attachment."

An IARC amendment request was previously submitted for Braidwood Station (Reference 1). In the course of their review of the Reference 1 submittal as well as similar submittals provided by Wolf Creek Nuclear Operating Corporation (WCNOC) and the Southern Nuclear Operating Company (SNC), the NRC determined that additional information was required as identified in Reference 2. The response to this request for additional information was provided in the Reference 3 submittal and is being provided with this request as Attachment 7. The NRC subsequently approved the use of the Interim Alternate Repair Criteria (IARC) for Braidwood Station Unit 2 in the Reference 4 transmittal.

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

The attached request is subdivided as shown below.

Attachment 1 provides an evaluation of the proposed changes.

Attachment 2 includes the marked-up TS pages with the proposed changes indicated for Byron Station.

Attachment 3 provides affidavit CAW-08-2435 for withholding the proprietary information provided in Attachment 4. Also provided is the Westinghouse authorization letter, CAW-08-2435, "Application for Withholding Proprietary Information from Public Disclosure."

Attachment 4 provides Westinghouse LTR-CDME-08-11, Revision 3, P-Attachment, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," (Proprietary).

Attachment 5 provides Westinghouse LTR-CDME-08-11, Revision 3, NP-Attachment, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," (Nonproprietary).

Attachment 6 provides affidavit CAW-08-2436 for withholding the proprietary information provided in Attachment 7. Also provided is the Westinghouse authorization letter, CAW-08-2436, "Application for Withholding Proprietary Information from Public Disclosure."

Attachment 7 provides Westinghouse LTR-CDME-08-43, Revision 3, P-Attachment, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11, Revision 3, P-Attachment."

Attachment 8 provides Westinghouse LTR-CDME-08-43, Revision 3, NP-Attachment, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11, Revision 3, P-Attachment."

Attachment 9 provides an EGC regulatory commitment regarding the multiplication factor to be used for Byron Unit 2 SG leakage assessments following implementation of the IARC TS amendment.

Attachments 4 and 7 contain information proprietary to Westinghouse and are supported by affidavits (Attachments 3 and 6 respectively) signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, it is requested that the information that is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390.

The proposed amendment has been reviewed by the Byron Station Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

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

EGC will continue to work with the industry, Westinghouse, and the NRC on this issue and supports the use of the expert panel to aid in the resolution of the open technical issues regarding the approval of the permanent ARC.

EGC requests that this proposed license amendment change be approved by October 6, 2008, to support the inspection activities for Byron Station Unit 2, refueling outage 14. The requested implementation would be prior to the return to service from the Byron Station Unit 2 Fall 2008 refueling outage.

If you have any questions about this letter, please contact Mr. David Chrzanowski at (630) 657-2816. As stated above, this submittal contains one regulatory commitment as detailed in Attachment 9.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 17th day of June 2008.

Respectfully,

Jeffrey L. Hansen
Manager – Licensing

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1.0 DESCRIPTION

This amendment application proposes a one cycle revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," for Byron Station to incorporate an interim alternate repair criteria (IARC) in the provisions for SG tube repair criteria during Byron Station Unit 2 (Byron 2) refueling outage 14 and the subsequent operating cycle. This IARC proposal requires full-length inspection of the SG tubes within the tubesheet but does not require plugging tubes if 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 plugging of SG tubes while maintaining the necessary structural and leakage integrity. The proposed amendment also revises the wording of reporting requirements provided in Byron Station TS 5.6.9, "Steam Generator (SG) Tube Inspection Report."

Approval of this amendment application is requested to support Byron 2 SG inspection activities during refueling outage 14 in Fall 2008 and the subsequent operating cycle. The approval of this IARC is requested as the existing one-cycle amendment (Reference 1) expires at the end of the current operating cycle for Byron 2.

2.0 PROPOSED CHANGE

EGC proposes to revise TS 5.5.9, "Steam Generator (SG) Program," by updating the current one-cycle ARC contained in the current wording in the Byron Station TS 5.6.9, "Steam Generator (SG) Tube Inspection Report," with a new one-cycle IARC.

Reference 1 approved a one-cycle revision to TS 5.5.9, for Byron Station Unit 2, to exclude from inspection and repair, that portion of the tube below 17 inches from the top of the hot leg tubesheet in the steam generators. Reference 2 proposed a permanent alternate repair criterion (ARC) to TS 5.5.9 to limit the inspection depth in the Byron 2 SG tube expansion zone, known as H*/B*. 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.

In a May 14, 2008, discussion between representatives of the Nuclear Energy Institute (NEI), U. S. Nuclear Regulatory Commission (NRC), and select Pressurized Water Reactor licensees, the NRC indicated that they would not be able to approve a permanent revision to TS 5.5.9 in time to support the Fall 2008 Byron Station Unit 2 refueling outage. In response, EGC is submitting this one-cycle amendment request.

Attachment 4, supplemented by Attachment 7, provides the technical justification for the IARC that requires full-length inspection of the tubes within the tubesheet (both hot leg and cold leg sides), 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 the tubesheet (TTS) is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads (i.e., the greater of 3 times the normal operating pressure (NOP) 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 changes from current TS wording are identified below with the affected sections italicized.

Byron Station TS 5.5.9 c. 1. currently states:	Byron Station proposed wording:
Tubes found by inservice inspection to contain flaws in a non-sleeved region with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate repair criteria discussed in TS 5.5.9.c.4. For Unit 2 only, during Refueling Outage *13* and the subsequent operating cycle, flaws identified in the portion of the tube from the top of the *hot leg* tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection.	Tubes found by inservice inspection to contain flaws in a non-sleeved region with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate repair criteria discussed in TS 5.5.9.c.4. For Unit 2 only, during Refueling Outage *14* and the subsequent operating cycle, flaws identified in the portion of the tube from the top of the tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection.
Byron Station TS 5.5.9 c. 4. i. currently states:	**Byron Station proposed wording:**
For Unit 2 only, during Refueling Outage *13* and the subsequent operating cycle, *flaws found in the portion of the tube below 17 inches from the top of the hot leg tubesheet do not require plugging or repair.*	For Unit 2 only, during Refueling Outage *14* 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 or repair. *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 axial indications found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging or repair.*

Byron Station TS 5.5.9 c. 4. i. currently states:	Byron Station proposed wording:
	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 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.*

Byron Station TS 5.5.9 d. currently states, in part:	Byron Station proposed wording:
Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. *For Unit 2 only, during Refueling Outage 13 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded.* The tube-to-tubesheet weld is not part of the tube.	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.

Byron Station TS 5.6.9 currently states in part:

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

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

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

Byron Station proposed j, k, and l changes:

j. For Unit 2, following completion of an inspection performed in Refueling Outage *14* (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, and whether initiated on primary or secondary side for each *service-induced flaw* detected *with*in *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 5.5.9 c.4.i,*

k. For Unit 2, following completion of an inspection performed in Refueling Outage *14* (and any inspections performed in the subsequent operating cycle), the operational primary to secondary leakage rate observed (greater than three gallons per day) in each steam generator (if it is not practical to assign the leakage to an individual steam generator, the entire primary to secondary leakage should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report, *and*

l. For Unit 2, following completion of an inspection performed in Refueling Outage *14* (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the lowermost 4-inches of tubing for the most limiting accident in the most limiting steam generator.

Also, editorial changes are proposed for Byron Station TS 5.5.9, paragraph c.2.i, adding "For Unit 2 only," and paragraph c.4, to change "may" to "shall." These changes provide consistency with the wording throughout TS 5.5.9.

The marked-up TS pages provided in Attachment 2 indicate the specific wording changes for Byron Station.

3.0 BACKGROUND

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

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

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

In order to preclude unnecessarily plugging tubes in the Byron 2 SGs, an analysis was performed to identify the maximum flaw size in the bottom 4 inches of the tube within the tubesheet necessary to maintain structural and leakage integrity for both normal operating and accident conditions. The analysis (provided in Attachment 4 and supplemented by Attachment 7) provides justification for an IARC that requires full-length inspection of the tubes within the tubesheet but does not require plugging tubes with axial indications or tubes with a certain arc length of circumferential cracking below 17 inches from the top of the tubesheet (i.e., the lower 4 inches). If flaws are found within the top of tubesheet to 17 inches below the top of tubesheet, the tube must be repaired or removed from service. The IARC methodology was developed for the tubesheet region of Model D5 SGs considering the most stringent loads associated with plant operation, including transients and postulated accident conditions.

4.0 TECHNICAL ANALYSIS

An evaluation has been performed by Westinghouse (provided as Attachments 4 and 7 to this submittal) 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 threefold:

1. Axial cracks in tubes below a distance of 17 inches below the top of the tubesheet can remain in service in the Byron 2 SGs as they are not a concern relative to tube pullout and leakage capability.

2. Circumferentially oriented cracks in tubes with an azimuthal extent of less than or equal to 203 degrees can remain in service for one cycle (approximately 18 months) of operation.

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 (approximately 18 months) of operation.

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 is used. The analysis includes the 0.75 inch diameter tube which is the tube size for the Byron 2 D5 model SGs. 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.

<u>Questions Relating to the IARC for Steam Generator Tubes</u>

As stated in the cover letter for this submittal, the Byron Station IARC request is based on those previously submitted and approved for Wolf Creek Nuclear Operating Corporation (WCNOC), Southern Nuclear Operating Company (SNC), and EGC's Braidwood Station. In the course of the review of those submittals the NRC requested additional information. The response to these questions was provided, for Braidwood Station, in Reference 3, with the majority of the questions addressed in Westinghouse LTR-CDME-08-43, which was provided as Attachment 3 to the Reference 3 submittal. Two questions regarding visual inspection capabilities for the detection of indications in tube to tubesheet welds (i.e., tube end welds) were addressed by EGC in Attachment 1 to Reference 3.

Since the visual examination technique for the tube-end welds is not, at this time, considered a qualified methodology, EGC will not use the visual examination technique to determine the acceptability of tube-end welds at Byron Station Unit 2 and instead will rely on an enhanced eddy current examination acceptance criteria for the lower 1 inch portion of the tube as discussed above and detailed in the Byron Station proposed TS.

Because of this approach, the two questions related to visual examination are no longer applicable to this Byron Station IARC request and therefore no response is being provided. The remainder of the responses to NRC questions from the Spring 2008 submittals are provided in Westinghouse LTR-CDME-08-43, Revision 3, provided as Attachment 7 to this submittal.

Discussion of Performance Criteria

The performance criteria of the Nuclear Energy Institute (NEI) (Reference 4) are the basis for these analyses. The performance criteria, also provided in the Byron Station TS, 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 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 Byron Station TS 5.5.9 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 SG. Leakage is not to exceed 1 gpm for all SGs.*

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 control room or offsite radiological dose consequences required by 10 CFR 50.67, "Accident source term," or GDC-19, "Control room," guidelines.

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 the Westinghouse analysis provided as Attachment 4 to this submittal, the bounding remaining structural ligament which meets the Reference 4 Performance Criterion described above and required for the tube to transmit the operational loads is 115 degrees arc length. 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 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.

Consideration of Non-Destructive Evaluation (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. The EPRI technique sheet ETSS 20510.1 (Reference 5) describes the qualified technique used to detect circumferential 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% 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 the 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 6). PDA takes into account the profile of the existing crack, including non-throughwall portions and shallow tails of the crack. Using the data from Reference 5 for cracks with a 90%, or greater, throughwall 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% throughwall 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 (EC) probes. 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% throughwall.

Based on the above, 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.

Consideration of Crack Growth

The growth of cracks due to PWSCC in this submittal request is dictated by four default growth rates from the Attachment 4 analysis. The distribution of growth rates is assumed to be log-normal. Typical values and conservative values are given, although it is recommended in Reference 7 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 Attachment 4 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 1.0
Calculation of Required Minimum Ligament for
18 Month Operating Period

	Bounding Structural Ligament	EFPY	Growth (In./EFPY) (1)	Growth (Deg./EFPY) (2)	Growth for Operating Period (degrees arc length)	Minimum Structural Ligament (degrees arc length)	Critical Ligament (degrees)
Tube	18 Calendar Month (CM) Operation	1.5	0.12	20.65	31	126	157
Tube End	18 Calendar Month (CM) Operation	1.5	0.12	20.65	31	235	266

1) 95% upper value of typical growth rates from the Attachment 4 analysis
2) Based on smallest mean tubesheet bore dimension for the limiting SG design (i.e., Model F)
3) Tube end area is the bottom 1 inch of the tube

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 Byron 2 is 18 calendar months (1.5 EFPY). For the Byron 2 SGs, referring to Table 1.0 above, the maximum allowable throughwall circumferential crack size in a SG tube is 203° (360° – 157°) for one cycle of operation. For the bottom 1 inch of the tube, the maximum allowable throughwall circumferential crack size is 94° (360° - 266°) for one cycle of operation. The critical ligament values were updated in Attachment 7 in the response to question 17. No additional uncertainty factor is necessary to be applied to the as-measured circumferential extent of the cracks.

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 Attachment 4. 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 factor of safety 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, for the limiting plant, if the normal operating leakage is within its allowable value, the accident-induced leakage will also be within the value assumed in the Byron 2 safety analysis.

As described in the regulatory commitment provided as Attachment 9 to this submittal, 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 of 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.

Inspection and Repair of Tube

The tube below the IARC depth (i.e., 17 inches below the TTS) 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, which exceed the maximum acceptable tube flaw size of 203 degrees, will be plugged. Circumferential indications in the bottom 1 inch of the tube, which exceed the maximum acceptable flaw size of 94 degrees, will be plugged. Detection of flaws will result in expansion per EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines," (Reference 8). Stress concentration areas may be used to define the extent of the expansion, e.g., if a repairable indication is located in a tube bulge (BLG) or overexpansion (OXP), the expansion may be limited to the non-inspected BLG/OXPs.

5.0 REGULATORY ANALYSIS

Steam Generator (SG) tube inspection and repair limits are specified in Section 5.5.9, "Steam Generator (SG) Program," of the Byron Station 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 throughwall. 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 General Design Criteria (GDC) 14, 15, 30, 31, and 32 of 10 CFR 50, Appendix A. Specifically, the GDC state that the Reactor Coolant Pressure Boundary (RCPB) shall have "an extremely low probability of abnormal leakage . . . and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDCs 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing . . . to assess . . .structural and leaktight integrity" (GDC 32). Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the steam generator tubing. Leakage integrity refers to limiting primary to secondary leakage during all plant conditions to within acceptable limits.

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 NRC 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 criteria discussed above.

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed amendment is 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 flaws observed in the region greater than 17 inches from the top of the tubesheet (TTS), that is, the lower 4 inches of the tube within the tubesheet is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads.

In addition, since axial flaws below 17 inches from the TTS do not degrade the axial load carrying capability of the tube, axial flaws do not require plugging if they are below 17 inches from the TTS.

The proposed amendment also modifies three reporting requirements in Byron Station TS 5.6.9, "Steam Generator (SG) Tube Inspection Report."

EGC has evaluated whether or not 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:

Criteria

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

   Response: No.

Of the various accidents previously evaluated, the proposed changes only affect the steam generator tube rupture (SGTR), 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 D5 steam generators has shown that axial loading of the tubes is negligible during an SSE.

At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below 17 inches from the top of the tubesheet 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 top of the tubesheet. 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, which takes into account eddy current uncertainty and crack growth rate. It has been shown that a circumferential crack with an azimuthal extent of 203 degrees meets the performance criteria of NEI 97-06, Rev. 2, "Steam Generator Program Guidelines" and the 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 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 0.10 gallons per minute (gpm) (or 150 gallons per day (gpd)), 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 0.5 gpm. This value is within the accident analysis assumptions for the limiting design basis accident for Byron 2, which is the postulated SLB event.

Based on the above, the performance criteria of NEI-97-06, Rev. 2 and RG 1.121 continue to be met and the proposed change does not involve a significant increase in the probability or consequences of the applicable accidents previously evaluated (i.e., SLB, the locked rotor and control rod ejection accidents).

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

   Response: No.

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

   Response: No.

The proposed change maintains the required structural margins of the steam generator tubes for both normal and accident conditions. NEI 97-06, Rev. 2 and RG 1.121 are used as the basis in the development of the interim alternate repair criteria (IARC) 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 General Design Criteria 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. 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-tubeshet weld, the Westinghouse analysis, provided in report "LTR-CDME-08-11 P-Attachment, Revision 3," supplemented by LTR-CDME-08-43 P-Attachment, Revision 3, defines 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 IARC will not result in unacceptable primary-to-secondary leakage during all plant conditions, including transients and postulated accident conditions.

Based on the above, it is concluded that the proposed changes do not result in any reduction in a margin of safety.

Based on the above, EGC 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.

5.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

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

As stated in 10 CFR 50.59, "Changes, tests, and experiments," paragraph (c)(1)(i), a licensee is required to submit a license amendment pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," if a change to the TS is required. Furthermore, the requirements of 10 CFR 50.59 necessitate that the NRC approve the TS changes before the TS changes are implemented. EGC's submittal revising the requirements of TS 5.5.9, "Steam Generator Program" to allow circumferentially oriented flaws in tubes with less than or equal to 203 degrees in extent and axial flaws in tubes below a distance of 17 inches below the top of the tubesheet to remain in service as well as the changes to the Byron Station TS 5.6.9, "Steam Generator (SG) Tube Inspection Report," meet the requirements of 10 CFR 50.59(c)(1)(i) and 10 CFR 50.90.

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

6.0 ENVIRONMENTAL CONSIDERATION

EGC 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.

7.0 PRECEDENTS

Wolf Creek Nuclear Operating Corporation, Vogtle Electric Generating Plant Unit 1, and Braidwood Station Unit 2 were granted similar TS changes as indicated below.

(1) Letter from J. N. Donohew (U. S. Nuclear Regulatory Commission) to R. A. Muench (Wolf Creek Generating Station) " Wolf Creek Generating Station - Issuance of Amendment re: Revision to Technical Specification 5.5.9 on the Steam Generator Program (TAC No. MD8054)", dated April 4, 2008

(2) Letter from S. P. Lingam (U. S. Nuclear Regulatory Commission) to T. E. Tynan (Vogtle Electric Generating Plant), "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. MD75450 and MD7451)", dated April 9,2008

(3) Letter from M. J. David (U. S. Nuclear Regulatory Commission) to C. G. Pardee (Exelon Generation Company), "Braidwood Station, Units 1 and 2, Issuance of Amendments Re: Revision of Technical Specifications for the Steam Generator Program (TAC Nos. MD8158 and MD 8159)," dated April 18, 2008

8.0 REFERENCES

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

(2) Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Application for Steam Generator Tube Alternate Repair Criteria Technical Specification Amendment," dated November 29, 2007

(3) Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Application for Steam Generator Tube Interim Alternate Repair Criteria Technical Specification," dated May 27, 2008

(4) NEI 97-06, Rev. 2, "Steam Generator Program Guidelines," May 2005

(5) ETSS #20510.1, "Technique for Detection of Circumferential PWSCC at Expansion Transitions"

(6) EPRI TR-107197, "Depth Based Structural Analysis Methods for Steam Generator Circumferential Indications," November 1997

(7) EPRI 1012987, "Steam Generator Integrity Assessment Guidelines," July 2006

(8) EPRI 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 7, October 2007

Attachment 2

Application for Steam Generator Tube Alternate Repair Criteria Technical Specification Amendment

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

Byron Station

Marked-up Technical Specifications Pages

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

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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

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

c. Provisions for SG tube repair criteria.

1. Tubes found by inservice inspection to contain flaws in a non-sleeved region with a depth equal to or exceeding 40% of the nominal wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate repair criteria discussed in TS 5.5.9.c.4. For Unit 2 only, during Refueling Outage ~~13~~ *14* and the subsequent operating cycle, flaws identified in the portion of the tube from the top of the ~~*hot leg*~~ tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection.

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

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

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

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

   i. For Unit 2 only, during Refueling Outage ~~13~~ *14* and the subsequent operating cycle, ~~*flaws found in the portion of the tube below 17 inches from the top of the hot leg tubesheet do not require plugging or repair.*~~

INSERT A →

INSERT A

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 or repair. 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 axial indications found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging or repair.

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 or 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 exceeds 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.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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

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

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

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

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

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

5.6.9 Steam Generator (SG) Tube Inspection Report

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

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

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

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

j. For Unit 2, following completion of an inspection performed in Refueling Outage ~~*13*~~ *14* (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, and whether initiated on primary or secondary side for each ~~*indication*~~ *service-induced flaw* detected *with*in ~~*the upper 17 inches of*~~ 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 5.5.9 c.4.i,* 5.6 Reporting Requirements

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

k. For Unit 2, following completion of an inspection performed in Refueling Outage ~~13~~ *14* (and any inspections performed in the subsequent operating cycle), the operational primary to secondary leakage rate observed (greater than three gallons per day) in each steam generator (if it is not practical to assign the leakage to an individual steam generator, the entire primary to secondary leakage should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report~~.~~*, and*

l. For Unit 2, following completion of an inspection performed in Refueling Outage ~~13~~ *14* (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the lowermost 4-inches of tubing for the most limiting accident in the most limiting steam generator. ~~*In addition, if the calculated accident leakage rate from the most limiting accident is less than 2 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined.*~~

Attachment 3

Application for Steam Generator Tube Alternate Repair Criteria Technical Specification Amendment

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

Westinghouse Affidavit and Authorization Letter

CAW-08-2435

Application for Withholding Proprietary Information from Public Disclosure

Westinghouse

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Our ref: CAW-08-2435

June 4, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-CDME-08-11, Rev. 3 P-Attachment, "Interim Alternate Repair Criterion (IARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated June 3, 2008 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-08-2435 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

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

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-08-2435, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

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

Enclosures

cc: Jon Thompson (NRC O-7E1A)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

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

Sworn to and subscribed before me
this 4th day of June, 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, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

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

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

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

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

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system 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. 3 P-Attachment, "Interim Alternate Repair Criterion (IARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated June 3, 2008 (Proprietary), for submittal to the Commission, being transmitted by Exelon Generation Company, LLC Application for Withholding Proprietary Information from Public Disclosure to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Byron Unit 2 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 Byron Unit 2 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.

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Attachment 5

Application for Steam Generator Tube Alternate Repair Criteria Technical Specification Amendment

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

Westinghouse LTR-CDME-08-11, Revision 3, NP-Attachment

Non-Proprietary Version