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50-425

NL-07-1710

U. S. Nuclear Regulatory Commission  
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Washington, D. C. 20555-0001

Vogtle Electric Generating Plant Units 1 and 2  
License Amendment Request to Technical Specification (TS)  
Sections TS 5.5.9, "Steam Generator (SG) Program" and  
TS 5.6.10, "Steam Generator Tube Inspection Report"

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Southern Nuclear Operating Company (SNC), hereby requests an amendment to Facility Operating License Nos. NPF-68 and NPF-81 for Vogtle Electric Generating Plant (VEGP), Units 1 and 2, respectively.

The proposed amendment will revise VEGP Units 1 and 2 Technical Specification (TS) sections TS 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report." The proposed changes to TS 5.5.9 modify the inspection and plugging requirements for portions of SG tubes within the hot leg side of the tubesheet region of the SGs only. The proposed changes to TS 5.6.10 will add requirements to report specific data related to indications, leakage detected, and calculated accident leakage.

The attached amendment request is subdivided as shown below:

Enclosure 1 provides a basis for the proposed change.

Enclosure 2 includes the marked-up TS pages with the proposed changes.

Enclosure 3 includes the associated typed TS pages with the proposed changes incorporated for VEGP.

Enclosure 4 provides an application for withholding, affidavit, proprietary information notice, and copyright notice for information proprietary to Westinghouse Electric Company, LLC.

Enclosure 5 provides a non-proprietary version of Westinghouse Electric Company WCAP-16794-NP, "Steam Generator Tube Alternate Repair

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NRR

Enclosure 6 provides a proprietary version of Westinghouse Electric Company WCAP-16794-P, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Vogtle 1 & 2 Electric Generating Plant", dated October 2007.

Enclosure 6 contains information proprietary to Westinghouse Electric Company LLC; it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-07-2347 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.

Southern Nuclear Operating Company requests approval of the proposed license amendments by March 10, 2008, in order to support the VEGP-1 refueling outage that is currently scheduled to begin March 16, 2008. The proposed changes will reduce the potential for unnecessary plugging of SG tubes which could further adversely impact the ability of VEGP-1 and VEGP-2 to achieve their licensed power level. In addition, personnel responsible for the tube plugging activities will not be subject to additional radiation dose by having to unnecessarily plug SG tubes. The proposed changes would be implemented within 30 days of issuance of the amendment.

(Affirmation and signature are on the following page.)

Mr. Tom E. Tynan states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

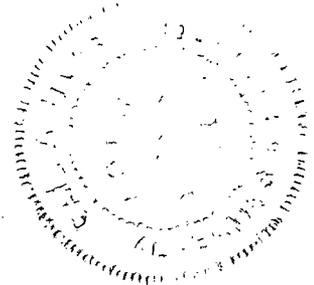
This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY



Tom E. Tynan  
Vice President - Vogtle



Sworn to and subscribed before me this 30<sup>th</sup> day of November, 2007.

  
Notary Public

My commission expires: July 5, 2010

TET/DRG/daj

- Enclosures:
1. Basis for Proposed Change
  2. Markup of Proposed Technical Specifications Page Changes for VEGP
  3. Typed Pages for Technical Specification Changes for VEGP
  4. Westinghouse Letter, CAW-07-2347, Application for Withholding and Affidavit, Proprietary Information Notice, and Copyright Notice
  5. Westinghouse Electric Company WCAP-16794-NP, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Vogtle 1 & 2 Electric Generating Plant," dated October 2007 - Non-Proprietary Version
  6. Westinghouse Electric Company WCAP-16794-P, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Vogtle 1 & 2 Electric Generating Plant," dated October 2007 - Proprietary Version

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cc: Southern Nuclear Operating Company

Mr. J. T. Gasser, Executive Vice President

Mr. L. M. Stinson, Vice President – Fleet Operations Support

Mr. D. H. Jones, Vice President – Engineering

RType: CVC7000

U. S. Nuclear Regulatory Commission

Mr. V. M. McCree, Acting Regional Administrator

Mr. S. P. Lingam, NRR Project Manager – Vogtle

Mr. G. J. McCoy, Senior Resident Inspector – Vogtle

State of Georgia

Mr. N. Holcomb, Commissioner – Department of Natural Resources

**Vogtle Electric Generating Plant  
License Amendment Request to Technical Specification 5.5.9  
Steam Generator Tube Surveillance Program**

**Enclosure 1**

**Basis for the Proposed Change**

**Vogtle Electric Generating Plant  
License Amendment Request to Technical Specification 5.5.9  
Steam Generator Tube Surveillance Program**

**Enclosure 1**

**Basis for the Proposed Change**

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## Enclosure 1

### Basis for the Proposed Change

#### 1.0 Summary Description

The proposed change would revise Technical Specification (TS) steam generator (SG) program requirements to incorporate changes to SG tubing inspection and plugging requirements and associated reporting requirements for Vogtle Electric Generating Plant (VEGP) Units 1 and 2. The proposed changes modify the inspection and plugging requirements for portions of SG tubes within the tubesheet region of the SGs by excluding approximately 4 inches at the tube ends on the hot leg side from inspection and/or plugging requirements. This change for both units is based on structural analysis and leak rate evaluation results and constitutes a redefinition of the primary-to-secondary pressure boundary. The change to reporting requirements adds specific data related to indications, leakage detected, and calculated accident leakage. This change is supported by Westinghouse Electric Company LLC as described in WCAP-16794-P, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Vogtle 1 & 2 Electric Generating Plant," dated October 2007 (Reference 1 and Enclosure 6 to this letter).

The NRC approved a similar one-time change for VEGP, Unit 1, to apply during Refueling Outage 13 and subsequent operating cycle and for Unit 2, to apply during Refueling Outage 12 and subsequent operating cycle, by letter from NRC to D. E. Grissette (Southern Nuclear Operating Company), "Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding the Steam Generator Tube Surveillance Program (TAC NOS. MD2642 and MD2643)," dated September 12, 2006.

#### 2.0 Detailed Description

Proposed changes to TS 5.5.9, "Steam Generator (SG) Program" are summarized below.

##### TS 5.5.9.c, "Provisions for SG tube repair criteria"

The word "may" has been replaced with the word "shall" in the following sentence:

*"The following alternate tube repair criteria shall be applied as an alternative to the 40% depth based criteria."*

Alternate repair criteria #1 and #2 have been replaced with the paragraph below:

*"Service-induced volumetric or crack-like flaws identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet do not require plugging. All tubes with service-induced volumetric or crack-like flaws identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the hot leg tubesheet shall be plugged upon detection."*

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### Basis for the Proposed Change

#### TS 5.5.9.d, "Provisions for SG tube inspections"

The following sentences will be deleted:

*"For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. For Unit 1 during Refueling Outage 13 and the subsequent operating cycle and for Unit 2 during Refueling Outage 12 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded."*

The above sentence will be replaced with the following sentence:

*"The portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded."*

Proposed changes to TS 5.6.10, "Steam Generator Tube Inspection Report" are summarized below.

The following paragraphs have been added:

- h. The number of indications and location, size, orientation, and whether initiated on primary or secondary side for each service-induced volumetric or crack-like flaw within the thickness of the tubesheet on the hot leg side,*
- i. The primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,*
- j. The calculated accident leakage rate from the portion of the tubes below the 17 inch depth from the top of the hot leg tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident leakage rate from the most limiting accident is less than 2 times the maximum primary to secondary LEAKAGE rate, the report should describe how it is determined.*

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### Basis for the Proposed Change

#### 3.0 Background

VEGP Units 1 and 2 are four loop Westinghouse-designed units with Model F steam generators (SG) having nominal 1 1/16 inch (OD) thermally treated A600TT tubes, full-depth hydraulically expanded tubesheet joints, and broached hole quatrefoil tube support plates constructed of stainless steel. The tubesheet is approximately 21 inches thick, and there are 5626 tubes in each SG. A total of 74 and 42 tubes are plugged in VEGP Units 1 and 2, respectively. VEGP Unit 1 is currently in Cycle 14 operation. VEGP Unit 2 is in Cycle 13 operation.

Indications of SG tube cracking were reported by Catawba Nuclear Station Unit 2 based on the results from the nondestructive, eddy current examination of the SG tubes during the fall 2004 outage as described in NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds" (Reference 2). The SGs at the Catawba 2 plant are type Westinghouse Model D5 with 3/4 inch nominal outside diameter (OD) Alloy 600 tubing (A600TT). The tube indications at Catawba 2 were reported approximately 7 inches below the top of the tubesheet (TTS) on the hot leg (HL) side in one tube, and just above the tube-to-tubesheet welds in a region of the tube known as the tack expansion (TE) in several other tubes. Finally, indications were also reported in the tube-end welds (TEWs), also known as tube-to-tubesheet welds, joining the tube to the tubesheet with a small number of those indications extending into the tubes.

As a result of the indications detected in tubing within the tubesheet at Catawba 2, additional rotating probe inspections were performed in overexpanded locations in tubing within the tubesheet at VEGP Unit 1 in eddy current inspections performed in the spring 2005 refueling outage. VEGP Unit 1 reported circumferential indications in two SG tubes in SG 4 (one tube had two indications) in overexpanded locations within the tubesheet. In fall 2005, eddy current inspection was performed in 2 of 4 SGs of the VEGP Unit 2 SG tubing. No degradation was detected in the tubesheet region in the sample inspections performed during this Unit 2 inspection. Based on the Catawba fall 2004 inspection results and the VEGP Unit 1 spring 2005 results, there are 3 distinct issues with regard to future inspections of A600TT SG tubes (including VEGP) which have been hydraulically expanded into the tubesheet:

1. indications in internal bulges and overexpansions within the hot leg tubesheet;
2. indications at the elevation of the TE transition; and
3. indications in the hot leg tube-to-tubesheet welds and propagation of these indications into the adjacent tube material.

The SG inspection scope is governed by TS 5.5.9, Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," Revision 2 dated May 2005 (Reference 3), Electric Power Research Institute (EPRI) "Pressurized Water

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### Basis for the Proposed Change

Reactor Steam Generator Examination Guidelines” (SG Examination Guidelines), SG degradation assessments which SNC prepares to support each SG tubing inspection, and SNC procedures. Criterion IX, “Control of Special Processes,” of 10 CFR Part 50, Appendix B, requires in part that nondestructive testing is to be accomplished by qualified personnel using qualified procedures in accordance with the applicable criteria. The inspection techniques and equipment were capable of reliably detecting the known and potential specific degradation mechanisms applicable to VEGP. The inspection techniques, essential variables, and equipment were qualified to Appendix H of the SG Examination Guidelines, “Performance Demonstration for Eddy Current Examination.”

The SG degradation assessment (DA) is prepared by SNC prior to each SG inspection. The DA is performed to identify degradation mechanisms that may be present, and includes a review of operating experience. A validation is performed to verify that the eddy current techniques utilized are capable of detecting those flaw types that are identified in the degradation assessment. Based on operating experience from VEGP and other plants, sample inspections of bulged and overexpanded locations within the tubesheet will be specified elements of VEGP SG eddy current inspections. These sample inspections are based on the guidance contained in the SG Examination Guidelines and TS 5.5.9. The inspection plan will be expanded according to industry guidelines if necessary due to confirmed degradation (i.e. tube crack indications).

Constraint provided by the hot leg tubesheet precludes tube burst for cracks within the tubesheet. The criteria for tube burst described in NEI 97-06 (Reference 3), and NRC Regulatory Guide (RG) 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes,” dated August 1976 (Reference 4), are satisfied due to the constraint provided by the tubesheet. Through application of the limited hot leg tubesheet inspection scope described herein, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur during a postulated SLB event.

Implementation of this proposed methodology involves limited inspection of the tubes within the tubesheet to depths of 17 inches from the top of the tubesheet on the hot leg side using specialized rotating eddy current probes. The limited tubesheet inspection length of tubing must be demonstrated to be non-degraded below the top of the tubesheet interface on the hot leg side. If cracks are found within the top of tubesheet to 17 inches below the hot leg side top of tubesheet, the tube must be removed from service (i.e., plugged).

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### Basis for the Proposed Change

#### 4.0 Technical Evaluation

The proposed TS change is intended to preclude unnecessarily plugging tubes in the VEGP Units 1 and 2 SGs. An analysis was performed as technical justification to identify the portion of the tube within the hot leg side tubesheet necessary to maintain structural and leakage integrity for both normal operating and accident conditions. The revised TS requirements will limit inspections to identifying and plugging degradation in this portion of the tubes. The technical justification for the inspection and repair methodology is provided in WCAP-16794-P (Reference 1 and Enclosure 6 to this letter). The evaluation is based on the use of finite element model structural analyses and a bounding leak rate evaluation based on the change in contact pressure between the tube and the tubesheet between normal operating and postulated accident conditions. The evaluation considered the requirements of the ASME Code, Regulatory Guides, NRC Generic Letters, NRC Information Notices, responses to NRC Request for Additional Information (RAI) letters, the Code of Federal Regulations, NEI 97-06 (Reference 3), and additional industry requirements. The content of the WCAP is structured with VEGP-specific evaluations and results being covered in Sections 1 through 10, VEGP-specific reviews with respect to the Wolf Creek responses to the relevant RAI questions addressed in the Wolf Creek response transmitted on September 27, 2007, being provided in Appendix A, and general technical bases applicable to the Westinghouse Model F steam generator is provided in Appendix B.

The following bullets are two of the conclusions of the evaluation:

- The structural integrity of the primary-to-secondary pressure boundary is unaffected by tube degradation of any magnitude below a tube location-specific depth ranging from 7.2 to 13.38 inches, depending on the tube leg and bundle zone being considered.
- The accident condition leak rate integrity can be bounded by twice the normal operational leak rate as a result of unlimited degradation below 17 inches from the top of the approximately 21-inch thick tubesheet.

Based on these conclusions, a redefinition of the pressure boundary can be effected while still assuring that the structural and leak rate performance criteria would be met during both normal operation and limiting postulated accident conditions. Implementation of the redefinition of the pressure boundary results in the elimination of the need for the inspection of the tubes below a depth on the order of 17 inches from the top of the hot leg tubesheet, which includes eliminating the need to inspect the region of the hot leg side SG tubes referred to as the tack expansion including tubing immediately adjacent to the tube-to-tubesheet weld, and the tack expansion transition near the bottom of the tubesheet. The tube-to-tubesheet weld is excluded from the definition of the SG tubing.

The determination of the required engagement depth was based on results from finite element model structural analyses and a steam line break to normal operation comparative leak rate evaluation.

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### Basis for the Proposed Change

The limited tubesheet inspection criteria were developed for the hot leg side tubesheet region of the VEGP Model F SGs considering the most stringent loads associated with plant operation, including transients and postulated accident conditions. The limited tubesheet inspection criteria were selected to prevent tube burst and axial separation due to axial pullout forces acting on the tube and to ensure that the steam line break (SLB) leakage limits are not exceeded. WCAP-16794-P (Reference 1 and Enclosure 6 to this letter) provides technical justification for allowing tubes with indications that are below 17 inches from the top of the hot leg tubesheet (i.e., within approximately four inches of the tube end on the hot leg side) to remain in service.

The portion of the tube in the tubesheet with the highest safety significance is the length of tube that is engaged in the tubesheet from the secondary face that is required to maintain structural and leakage integrity over the full range of SG operating conditions, including the most limiting accident conditions. The evaluation determined that degradation in tubing below the portion of the tube with the highest safety significance does not require repair and serves as the basis for the tubesheet inspection program. The determination of the portion of the tube within the tubesheet with the highest safety significance is based upon evaluation and testing programs that quantified the tube-to-tubesheet radial contact pressure for bounding plant conditions as described in WCAP-16794-P (Reference 1 and Enclosure 6 to this letter). The results provide the technical rationale to eliminate inspection of the region of the tube below the most limiting depth of either the structural requirement or leakage requirement. The WCAP evaluation addresses the leak rate depth for both hot leg and cold leg sides; however, this license amendment request pertains only to inspection and plugging requirements on the hot leg side. This request does not address cold leg side inspection and plugging requirements.

The tube-to-tubesheet radial contact pressure provides resistance to tube pull-out and resistance to leakage during plant operation and transients. Temperature effects and upward bending of the tubesheet, due to primary and secondary differential pressure during normal and transient conditions, result in the tube-to-tubesheet contact pressure increasing below the neutral plane of the tube sheet. Due to these effects, the tubesheet bore tends to dilate near the top of the tubesheet and constricts the tube near the bottom of the tubesheet.

The purpose of the tube-end weld is to ensure the hydraulically expanded tube-to-tubesheet joints in Model F SGs are leak-tight. Considerations were also made with regard to the potential for primary-to-secondary leakage during postulated faulted conditions. However, the leak rate during postulated accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet based on the WCAP-16794-P (Reference 1 and Enclosure 6 to this letter) evaluation which shows that while the driving pressure increases by about a factor of almost two, the flow resistance increases because the tube-to-tubesheet contact pressure also increases. Depending on the depth within the tubesheet, the relative increase in resistance could easily be larger than that of the pressure potential. Therefore, the leak rate under normal operating conditions could exceed its allowed value before the accident condition leak rate

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would be expected to exceed its allowed value. This approach is termed an application of the "bellwether principle." While such a decrease in the leak rate is expected, the postulated accident leak rate could conservatively be taken to be bounded by twice the normal operating leak rate if the increase in contact pressure is ignored.

Since normal operating leakage is limited by VEGP TS 3.4.13, "RCS Operational LEAKAGE" and by NEI 97-06 (Reference 3) to less than 150 gpd throughout one SG in the VEGP Units 1 and 2 SGs, the attendant accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by 0.20 gpm in the faulted SG which is less than the accident analysis assumption of 0.35 gpm to the affected SG included in Section 15.1.5 of the VEGP Updated Final Safety Analysis Report (FSAR). Hence, it is reasonable to omit any consideration of inspection of the tube, tube end weld, bulges/overexpansions, or other anomalies below 17 inches from the top of the hot leg tubesheet.

Testing and analyses have shown that tube-to-tubesheet engagement lengths of approximately 7.2 to 13.38 inches were sufficient to maintain structural integrity (i.e., resist tube pull-out resulting from loading considering differential pressures of three times the normal operating pressure difference and considering differential pressures of 1.4 times the limiting accident pressure difference). The variation of the required engagement length is a function of the radial tube location within the tube bundle. Additional conservatism is being added to the minimum structural distances of 7.2 to 13.38 inches by performing sampling inspections to depths of 17 inches below the top of the tubesheet, which traverses below the neutral plane. The increase in contact pressure at this depth significantly increases the tube structural strength and resistance to leakage.

Therefore, the proposed inspection sampling length of 17 inches from the top of the hot leg side tubesheet provides a high level of confidence that the structural and leakage criteria are maintained during normal operating and accident conditions, and degradation found in the portion of the tube below 17 inches from the top of the hot leg does not require plugging. The WCAP evaluation demonstrates the high level of conservatism which is maintained through the practice of inspecting to a depth of 17 inches from the top of the hot leg side tubesheet.

In accordance with the EPRI PWR Steam Generator Examination Guidelines and NEI 97-06, "Steam Generator Program Guidelines" (Reference 3), SNC will implement, as a minimum, the following inspection requirements for future outages on Units 1 and 2 in order to use the limited tubesheet inspection methodology:

1. For Unit 1, perform a 100% minimum inspection of the hot leg side tubes in all 4 SGs using rotating probe coil (RPC) technology from three inches above the top of the hot leg tubesheet to three inches below the top of the tubesheet. For Unit 2, perform a 40% minimum inspection of the hot leg side tubes of the two scheduled steam generators using RPC technology from three inches above the top of the hot leg tubesheet to three inches below the top of the tubesheet. Expand to 100% of the affected SG and

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20% of the unaffected SGs in this region only if cracking is found that is not associated with a bulge or overexpansion as described below.

2. Perform an inspection of the hot leg side tubes using RPC technology to a depth of 17 inches below the top of the tube sheet in order to inspect (1) for Unit 1, 100% of bulges and overexpansions in SG 4, and at least 20% of bulges and overexpansions in SGs 1, 2, and 3; and (2) for Unit 2, a 40% sample of bulges and overexpansions in the two scheduled SGs.
  - a. Bulge refers to a tube diameter deviation within the tubesheet of 18 mils or greater as measured by bobbin coil probe based on review of the previous cycle bobbin data; and
  - b. Overexpansion refers to a tube diameter deviation within the tubesheet of 1.5 mils or greater as measured by bobbin coil probe based on review of the previous cycle bobbin data.
3. If cracking is found in the sample population of bulges or overexpansions, the inspection scope will be increased to 100% of the bulges and overexpansions population for the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet in the affected SG and a 20% sample of each of the unscheduled SGs.
4. If cracking is reported at one or more tube locations not designated as either a top of the tubesheet expansion transition, a bulge, or an overexpansion, an engineering evaluation will be performed. This evaluation will determine the cause for the signal, e.g., some other tube anomaly, in order to identify a critical area for the expansion of the inspection. This expanded inspection will be limited to the identified critical area within 17 inches from the top of the hot leg tubesheet.
5. Cracks detected by inspections described in the immediately preceding paragraphs numbered 1 through 4 above will be plugged consistent with the following paragraph describing plugging criteria and acceptance criteria.

SNC will implement the following plugging criteria and acceptance criteria:

- Service-induced volumetric or crack-like flaws below 17 inches from the top of hot leg tubesheet are acceptable.
- Service-induced volumetric or crack-like flaws within 17 inches from the top of hot leg tubesheet must be plugged.

In summary:

- WCAP-16794-P (Reference 1 and Enclosure 6 to this letter) notes that the structural integrity requirements of NEI 97-06 (Reference 3), and Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR SG

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Tubes,” (Reference 4) are met by sound tube engagement lengths ranging from approximately 7.2 to 13.38 inches from the top of the tubesheet. The region of the tube below those elevations, including the tube-to-tubesheet weld, is not needed for structural integrity during normal operation or accident conditions. SNC will, however, perform sampling inspections to a depth of 17 inches from the top of the hot leg tubesheet.

- The leak rate during postulated faulted events would be bounded by twice the leak rate during normal operation.
- NEI 97-06 (Reference 3) defines the tube as extending from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, but specifically excludes the tube-to-tubesheet weld from the definition of the tube.
- The welds were originally designed and analyzed as the primary pressure boundary in accordance with the requirements of Section III of the 1971 edition of the ASME Code, Summer 1972 Addenda for the VEGP Units 1 and 2 SGs. This proposed license amendment request, in effect, redefines the hot leg side primary pressure boundary from the hot leg tube end weld to 17 inches below the top of the hot leg tube sheet.
- Section XI of the ASME Code deals with the in-service inspection of nuclear power plant components. The ASME Code (i.e., Editions 1971 through 2007) specifically recognizes that the SG tubes are under the purview of the NRC through the implementation of the requirements of the TS as part of the plant operating license.

## 5.0 Regulatory Evaluation

### 5.1 Significant Hazards Consideration

According to 10 CFR 50.92, “Issuance of amendment,” paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or
2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or
3. Involve a significant reduction in a margin of safety.

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of Amendment,” as discussed below:

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

No. The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed changes that alter the SG inspection criteria do not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed changes will not alter the operation of, or otherwise increase the failure probability of, any plant equipment that initiates an analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

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

During the SGTR event, the required structural integrity margins of the SG tubes will be maintained by the presence of the SG tubesheet. SG tubes are hydraulically expanded in the tubesheet area. Tube rupture in tubes with cracks in the tubesheet is precluded by the constraint provided by the tubesheet. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. Based on this design, the structural margins against burst discussed in RG 1.121 (Reference 4), are maintained for both normal and postulated accident conditions.

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

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

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

The consequences of a SLB are also not significantly affected by the proposed changes. During a SLB accident, the reduction in pressure

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above the tubesheet on the shell side of the SG creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet, thereby restricting primary-to-secondary leakage below the midplane.

The purpose of the tube-end weld is to ensure the hydraulically expanded tube-to-tubesheet joints in Model F SGs are leak-tight. Considerations were also made with regard to the potential for primary-to-secondary leakage during postulated faulted conditions. However, the leak rate during postulated accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet based on the evaluation (Reference 1) which shows that while the driving pressure increases by about a factor of almost two, the flow resistance increases, because the tube-to-tubesheet contact pressure also increases. Depending on the depth within the tubesheet, the relative increase in resistance could easily be larger than that of the pressure potential. Therefore, the leak rate under normal operating conditions could exceed its allowed value before the accident condition leak rate would be expected to exceed its allowed value. This approach is termed an application of the "bellwether principle." While such a decrease in the leak rate is expected, the postulated accident leak rate could conservatively be taken to be bounded by twice the normal operating leak rate if the increase in contact pressure is ignored.

Since normal operating leakage is limited by VEGP TS 3.4.13 and by NEI 97-06 (Reference 3) to less than 150 gpd throughout one SG in the VEGP Units 1 and 2 SGs, the attendant accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by 0.20 gpm in the faulted SG which is less than the accident analysis assumption of 0.35 gpm to the affected SG included in Section 15.1.5 of the VEGP FSAR. Hence, it is reasonable to omit any consideration of inspection of the tube, tube end weld, bulges/overexpansions or other anomalies below 17 inches from the top of the hot leg tubesheet.

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

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

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

## Enclosure 1

### Basis for the Proposed Change

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

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

No. The proposed changes maintain the required structural margins of the SG tubes for both normal and accident conditions. NEI 97-06 (Reference 3) and RG 1.121 (Reference 4), are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 (Reference 4) describes a method acceptable to the NRC for meeting the following General Design Criteria (GDC).

- GDC 14, "Reactor coolant pressure boundary,"
- GDC 15, "Reactor coolant system design,"
- GDC 31, "Fracture prevention of reactor coolant pressure boundary," and,
- GDC 32, "Inspection of reactor coolant pressure boundary."

RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation, 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.

Application of the limited tubesheet inspection depth criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the proposed limited tubesheet inspection depth criteria.

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

### 5.2 Applicable Regulatory Requirements / Criteria

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include technical specifications (TS) as part of the license.

The Commission's regulatory requirements related to the content of the TS are contained in Title 10, Code of Federal Regulations (10 CFR), Section 50.36, "Technical specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation (LCO), (3) surveillance requirements, (4) design features, and (5) administrative controls. The SG tube inspection requirements are included in the TS in accordance with 10 CFR 50.36(c)(5), "Administrative Controls."

## Enclosure 1

### Basis for the Proposed Change

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. SNC's submittal meets the requirements of 10 CFR 50.59(c)(1)(i) and 10 CFR 50.90.

RG 1.121 (Reference 4) describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, "Reactor coolant pressure boundary," GDC 15, "Reactor coolant system design," GDC 31, "Fracture prevention of reactor coolant pressure boundary," and GDC 32, "Inspection of reactor coolant pressure boundary." RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation, 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.

NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," (Reference 2) dated April 7, 2005, provides additional regulatory insight regarding SG tube degradation.

### 5.3 Precedent

Southern Nuclear Operating company was granted a similar TS change on September 12, 2006 (Reference 5) that involved a one-time change to TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," regarding the required SG inspection scope for Vogtle, Unit 1, during Refueling Outage 13 and the subsequent operating cycle and Vogtle, Unit 2, during Refueling Outage 12, and the subsequent operating cycle. The changes modified the inspection requirements for portions of the SG tubes within the hot leg tubesheet region of the SGs.

### 5.4 Conclusions

In 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.

### 6.0 Environmental Consideration

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as follows:

## Enclosure 1

### Basis for the Proposed Change

- (i) The amendment involves no significant hazards consideration.

As described above, the proposed change involves no significant hazards consideration.

- (ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed change does not involve the installation of any new equipment, or the modification of any equipment that may affect the types or amounts of effluents that may be released offsite. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

- (iii) There is no significant increase in individual or cumulative occupation radiation exposure.

The proposed change does not involve plant physical changes, or introduce any new mode of plant operation. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

Based on the above, SNC concludes that the proposed change meets the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.22 relative to requiring a specific environmental assessment by the Commission.

### 7.0 References

1. Westinghouse Electric Company WCAP-16794-P, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Vogtle 1 & 2 Electric Generating Plants," dated October 2007.
2. NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," dated April 7, 2005.
3. NEI 97-06, "Steam Generator Program Guidelines," Revision 2, dated May 2, 2005.
4. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976.
5. Vogtle Electric Generating Plant, Units 1 and 2, "Issuance of Amendments Regarding the Steam Generator Tube Surveillance Program," September 12, 2006, (TAC Nos. MD2642 and MD2643)

**Vogtle Electric Generating Plant  
License Amendment Request to Technical Specification 5.5.9  
Steam Generator Tube Surveillance Program**

**Enclosure 2**

**Markup of Proposed Technical Specification**

5.5 Programs and Manuals

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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 1 gpm 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:

Service-induced volumetric or crack-like flaws identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet do not require plugging. All tubes with service-induced volumetric or crack-like flaws identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the hot leg tubesheet shall be plugged upon detection.

~~1. For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging.~~

~~For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, degradation 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 upon detection.~~

~~2. For Unit 1 during Refueling Outage 13 and the subsequent operating cycle, and for Unit 2 during Refueling Outage 12 and the subsequent operating cycle, degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging.~~

~~For Unit 1 during Refueling Outage 13 and the subsequent operating cycle and for Unit 2 during Refueling Outage 12 and the subsequent operating cycle, degradation identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the hot leg tubesheet shall be plugged upon detection.~~

## 5.5 Programs and Manuals

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- 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 portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. ~~For Unit 2 during Refueling Outage 11 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. For Unit 1 during Refueling Outage 13 and the subsequent operating cycle and for Unit 2 during Refueling Outage 12 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded.~~ The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.
1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.
  2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.
  3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic nondestructive testing, or engineering evaluation indicates that a crack-like indication is not associated

## 5.6 Reporting Requirements

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5.6.9 Deleted.

### 5.6.10 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 5.5.9, 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 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,
  - h. The number of indications and location, size, orientation, and whether initiated on primary or secondary side for each service-induced volumetric or crack-like flaw within the thickness of the tubesheet on the hot leg side.
  - i. The primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report.
  - j. The calculated accident leakage rate from the portion of the tubes below the 17 inch depth from the top of the hot leg tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident leakage rate from the most limiting accident is less than 2 times the maximum primary to secondary LEAKAGE rate, the report should describe how it is determined.
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**Vogtle Electric Generating Plant  
License Amendment Request to Technical Specification 5.5.9  
Steam Generator Tube Surveillance Program**

**Enclosure 3**

**Typed Pages for Technical Specification**

5.5 Programs and Manuals

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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 1 gpm 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 shall be applied as an alternative to the 40% depth based criteria:

Service-induced volumetric or crack-like flaws identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet do not require plugging. All tubes with service-induced volumetric or crack-like flaws identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the hot leg tubesheet shall be plugged upon detection.

(continued)

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## 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. 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 tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.
  3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic nondestructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.
- e. Provisions for monitoring operational primary to secondary LEAKAGE.

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(continued)

## 5.6 Reporting Requirements

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5.6.9 Deleted.

### 5.6.10 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 5.5.9, 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 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,
- h. The number of indications and location, size, orientation, and whether initiated on primary or secondary side for each service-induced volumetric or crack-like flaw within the thickness of the tubesheet on the hot leg side,
- i. The primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,
- j. The calculated accident leakage rate from the portion of the tubes below the 17 inch depth from the top of the hot leg tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident leakage rate from the most limiting accident is less than 2 times the maximum primary to secondary LEAKAGE rate, the report should describe how it is determined.

**Vogtle Electric Generating Plant  
License Amendment Request to Technical Specification 5.5.9  
Steam Generator Tube Surveillance Program**

**Enclosure 4**

**Westinghouse Letter, CAW-07-2347, Application for Withholding and Affidavit,  
Proprietary Information Notice, and Copyright Notice**



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

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Our ref: CAW-07-2347

October 23, 2007

APPLICATION FOR WITHHOLDING PROPRIETARY  
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-16794-P, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Vogtle 1 and 2 Electric Generating Plants" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-07-2347 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 Southern Nuclear Operating Company (SNC).

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-07-2347, 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,

A handwritten signature in cursive script that reads "J. A. Gresham".

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  
E. C. Arnold, EC 557A

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

SS

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared C. Savage, 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:



C. Savage, Director  
Systems & Safety Analysis

Sworn to and subscribed before me  
this 23<sup>rd</sup> day of October, 2007



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 Director, Systems & Safety Analysis, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.
- (2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.
- (3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.
- (4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.
  - (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.
  - (ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

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

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

- (b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.
- (c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.
- (d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.
- (e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.
- (f) It contains patentable ideas, for which patent protection may be desirable.

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

- (a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.
- (b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.
- (c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.
- (d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component

may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

- (e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.
  - (f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.
- (iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.
- (iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.
- (v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-16794-P, "Steam Generator Tube Alternate Repair Criteria for the Portion of the Tube Within the Tubesheet at the Vogtle 1 and 2 Electric Generating Plants," October 2007 (Proprietary), for submittal to the Commission, being transmitted by Southern Nuclear Operating Company Application for Withholding Proprietary Information from Public Disclosure to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for the Vogtle 1 and 2 Electric Generating Plants is expected to be applicable to other licensee submittals in support of implementing a limited inspection of the tube joint within the tubesheet region of the steam generators

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

- (a) Provide documentation of the analyses, methods, and testing for the implementation of an alternate repair criteria for the portion of the tubes within the tubesheet of the Vogtle 1 and 2 Electric Generating Plants.

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

Further this information has substantial commercial value as follows:

- (a) Westinghouse plans to sell the use of similar information to its customers for the purposes of meeting NRC requirements for licensing documentation.
- (b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

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

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

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

Further the deponent sayeth not.

## **PROPRIETARY INFORMATION NOTICE**

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

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

## **COPYRIGHT NOTICE**

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.

**Vogtle Electric Generating Plant  
License Amendment Request to Technical Specification 5.5.9  
Steam Generator Tube Surveillance Program**

**Enclosure 5**

**Westinghouse Electric Company WCAP-16794-NP  
“Steam Generator Tube Alternate Repair Criteria for the  
Portion of the Tube Within the Tubesheet at the Vogtle 1 & 2  
Electric Generating Plant for One Cycle Application”  
dated October 2007 - Non-Proprietary Version**