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ENCLOSURES 4, 7, 10 AND 13 CONTAINS INFORMATION EXEMPT FROM PUBLIC DISCLOSURE IN ACCORDANCE WITH 10 CFR 2.390

SOUTHERN COMPANY

November 23, 2010

Docket Nos.: 50-424
50-425

NL-10-2104

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

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

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. This amendment request proposes to revise VEGP Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," to exclude portions of the tube below the top of the steam generator tubesheet from periodic steam generator tube inspections for Unit 1 during Refueling Outage 16 and the subsequent operating cycle and for Unit 2 during Refueling Outage 15 and the subsequent operating cycle. In addition, this amendment proposes to revise Technical Specification (TS) 5.6.10, "Steam Generator Tube Inspection Report" to remove reference to previous interim alternate repair criteria and provide reporting requirements specific to the temporary alternate repair criteria. This change is supported by the analysis described in section 4 of Enclosure 1.

The documents contained in enclosures 4, 7, 10 and 13 provide proprietary Westinghouse Electric Company LLC information which supports the analysis described in Enclosure 1. Enclosures 5, 8, 11 and 14 provide the non- proprietary version of the same Westinghouse Electric Company LLC documents. As enclosures 4, 7 ,10 and 13 contain information proprietary to Westinghouse Electric Company LLC, enclosures 6, 9, 12 and 15 contain the supporting affidavits signed by Westinghouse Electric Company LLC, the owner of the information. These affidavits set 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 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 2.390 of the Commission's regulations.

ENCLOSURES 4, 7, 10 AND 13 CONTAINS INFORMATION EXEMPT FROM PUBLIC DISCLOSURE IN ACCORDANCE WITH 10 CFR 2.390

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On September 24, 2009, the NRC issued Vogtle Electric Generating Plant Amendment Numbers 157 and 138 (Units 1 and 2, respectively) for SG Interim Alternate Repair Criteria (Reference 18 of Enclosure 1). As a condition of approval, SNC made regulatory commitments:

- To revise the Steam Generator Program Strategic Plan procedure to include monitoring for tube slippage as part of the steam generator tube inspection program for Unit 1 and Unit 2. Slippage monitoring will occur for each inspection of the Vogtle 1 and 2 steam generators.

- For the condition monitoring (CM) assessment, the component of operational leakage from the prior cycle from below the H* distance will be multiplied by a factor of 2.48 and added to the total accident leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable accident induced leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.48 and compared to the observed operational leakage. An administrative limit will be established to not exceed the calculated value.

The program/procedure changes needed to meet these commitments were completed in accordance with the NRC approval of Amendment Numbers 157 and 138 (Units 1 and 2, respectively). The changes will also apply to this License Amendment Request. As such, these changes will remain in place. Therefore no new NRC commitments are required.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

SNC requests approval of the proposed license amendments by March 12, 2011 to support implementation during Vogtle Unit 1 spring refueling outage. The proposed changes would be implemented within 30 days of issuance of the amendment.

This letter contains no NRC commitments. If you have any questions, please contact Mr. Jack Stringfellow at (205) 992-7037.

Mr. M. J. Ajluni states he is Nuclear Licensing Director 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.

Respectfully submitted,

Mark J Ajluni

M. J. Ajluni
Nuclear Licensing Director

Sworn to and subscribed before me this 23 *day of* November*, 2010.*

Charlotte A. Graham
Notary Public

My commission expires: 6/9/12

MJA/TAH/<>

Enclosures:

1. Basis for Proposed Change
2. Markup of Proposed Technical Specification
3. Typed Pages for Technical Specification
4. Westinghouse Electric Company LLC WCAP-17330, Rev. 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5), November 2010. (Proprietary)
5. Westinghouse Electric Company LLC WCAP-17330, Rev. 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5), November 2010. (Non-Proprietary)
6. Westinghouse Electric Company LLC LTR-CAW-10-2993, "Application for Withholding Proprietary Information from Public Disclosure"
7. LTR-SGMP-10-78, "Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and Their Relative Importance to H*," September 7, 2009. (Proprietary)
8. LTR-SGMP-10-78, "Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and Their Relative Importance to H*," September 7, 2009. (Non-Proprietary)
9. Westinghouse Electric Company LLC LTR-CAW-10-2939, "Application for Withholding Proprietary Information from Public Disclosure"
10. LTR-SGMP-10-33 P-Attachment, "H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity," September 13, 2010. (Non-Proprietary)
11. LTR-SGMP-10-33 P-Attachment, "H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity," September 13, 2010. (Proprietary)
12. Westinghouse Electric Company LLC LTR-CAW-10-2955, "Application for Withholding Proprietary Information from Public Disclosure"

13. LTR-SGMP-09-111, Rev. 1, "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*," September 1, 2010. (Proprietary)
14. LTR-SGMP-09-111, Rev. 1, "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*," September 1, 2010. (Non-Proprietary)
15. Westinghouse Electric Company LLC LTR-CAW-10-2936, "Application for Withholding Proprietary Information from Public Disclosure"

cc: <u>Southern Nuclear Operating Company</u>
Mr. J. T. Gasser, Executive Vice President W/O Enclosure
Mr. T. E. Tynan, Vice President – Vogtle W/O Enclosure
Ms. P. M. Marino, Vice President – Engineering W/O Enclosure
RType: CVC7000

<u>U. S. Nuclear Regulatory Commission</u>
Mr. L. A. Reyes, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Vogtle
Mr. M. Cain, Senior Resident Inspector – Vogtle
Mr. P.G. Boyle, NRR Project Manager

<u>State of Georgia</u>
Mr. Allen Barnes, Director Environmental Protection Division

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

Enclosure 1

Basis for Proposed Change

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

Enclosure 1

Basis for Proposed Change

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1.0 Summary Description

Southern Nuclear Operating Company (SNC) proposes to revise the Vogtle Electric Generating Plant (VEGP) Technical Specifications (TS) 5.5.9, "Steam Generator (SG) Program," to exclude portions of the steam generator tube below the top of the steam generator tubesheet from periodic tube inspections for Unit 1 during Refueling Outage 16 and the subsequent operating cycle and for Unit 2 during Refueling Outage 15 and the subsequent operating cycle. Application of the supporting structural analysis and leakage evaluation results to exclude portions of the tubes from inspection and repair of tube indications is interpreted to constitute a redefinition of the primary-to-secondary pressure boundary. In addition, this amendment proposes to revise Technical Specification (TS) 5.6.10, "Steam Generator Tube Inspection Report" to remove reference to previous interim alternate repair criteria and provide reporting requirements specific to the temporary alternate repair criteria. The proposed changes to the TS are based on the supporting structural analysis and leakage evaluation completed by Westinghouse Electric Company, LLC. The documentation supporting the Westinghouse analysis is described in section 4 and provides the licensing basis for this change. Table 5-1 of WCAP 17330-P (Reference 16) provides the 95/95 H* value of 15.2 inches for plants with Model F Steam Generators which includes VEGP 1 and 2.

The NRC previously issued the following amendments revising steam generator tube inspection requirements:

- Amendment Numbers 138 and 117 (Units 1 and 2, respectively) (Reference 1) to exclude degradation found in the portion of the tubes below 17 inches from the top of the hot leg tubesheet from the requirement to plug for Unit 2 Refueling Outage 11 and the subsequent operating cycle.
- Amendments 146 and 126 (Units 1 and 2, respectively) (Reference 2) to exclude the portion of the tubes below 17 inches from the top of the hot leg tubesheet from the requirement to plug for Unit 1 Refueling Outage 13 and Unit 2 Refueling Outage 12 and the subsequent operating cycles.
- Amendment Numbers 150 and 130 (Units 1 and 2, respectively) (Reference 3) which approved an interim alternate repair criteria for Unit 1 Refueling Outage 14 and the subsequent operating cycle that requires full-length inspection of the tubes within the tubesheet but does not require plugging tubes if any circumferential cracking observed in the region greater than 17 inches from the top of the tubesheet is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads.
- Amendment Numbers 152 and 133 (Units 1 and 2, respectively) (Reference 4) applied the criteria of Amendments 150 and 130 to Unit 2 Refueling Outage 12 and the subsequent operating cycle. Amendment Numbers 150 and 152 (Unit1) and 130 and 133 (Unit 2) are only applicable for Refueling Outages 14 (Unit 1) and 13 (Unit 2) and the subsequent operating cycles.
- Amendment Numbers 157 and 138 (Units 1 and 2, respectively) (Reference 18) revised TS 5.5.9, "Steam Generator (SG) Program," to exclude portions of the tubes within the tubesheet from periodic SG inspections (establish alternate repair criteria). In addition, these amendments revise TS 5.6.10, "Steam Generator Tube Inspection Report," to remove reference to previous interim alternate repair criteria and provide reporting requirements specific to Refueling Outage 15 and the subsequent operating cycle and for Unit 2 during Refueling Outage 14 and the subsequent operating cycle.

Approval of this amendment application is requested by March 12, 2011 to support VEGP 1 Refueling Outage 16 (Spring 2011), since the existing one-cycle amendment expires at the end of the current operating cycle.

2.0 Detailed Description

Proposed Changes to Current TS

TS 5.5.9.c. currently states:

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

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

For Unit 1 during Refueling Outage 15 and the subsequent operating cycle and for Unit 2 during Refueling Outage 14 and the subsequent operating cycle, tubes with service-induced flaws located greater than 13.1 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 13.1 inches below the top of the tubesheet shall be plugged upon detection.

This section would be revised as follows, as noted in italic type:

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:

For Unit 1 during Refueling Outage ***16*** and the subsequent operating cycle and for Unit 2 during Refueling Outage ***15*** and the subsequent operating cycle, tubes with service-induced flaws located greater than ***15.2*** inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to ***15.2*** inches below the top of the tubesheet shall be plugged upon detection.

TS 5.5.9.d currently states:

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Unit 1 during Refueling Outage 15 and the subsequent operating cycle and for Unit 2 during Refueling Outage 14 and the subsequent operating cycle, portions of the tube below 13.1 inches below the top of the tubesheet are excluded from this requirement.

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 portions of the SG tube not excluded above, 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.

This section would be revised as follows, as noted in italic type:

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Unit 1 during Refueling Outage ***16*** and the subsequent operating cycle and for Unit 2 during Refueling Outage ***15*** and the subsequent operating cycle, portions of the tube below ***15.2*** inches below the top of the tubesheet are excluded from this requirement.

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 portions of the SG tube not excluded above, 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.

TS 5.6.10 h., 5.6.10 i., and 5.6.10 j. currently state:

h. For Unit 1 during Refueling Outage 15 and the subsequent operating cycle and for Unit 2 during Refueling Outage 14 and the subsequent operating cycle, 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; and

i. For Unit 1 during Refueling Outage 15 and the subsequent operating cycle and for Unit 2 during Refueling Outage 14 and the subsequent operating cycle, the calculated accident induced leakage rate from the portion of the tubes below 13.1 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.48 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined.

j. For Unit 1 during Refueling Outage 15 and the subsequent operating cycle and for Unit 2 during Refueling Outage 14 and the subsequent operating cycle, the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

TS 5.6.10 h., 5.6.10 i and 5.6.10 j. would be revised as follows, as noted in italic type:

h. For Unit 1 during Refueling Outage ***16*** and the subsequent operating cycle and for Unit 2 during Refueling Outage ***15*** and the subsequent operating cycle, 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; and

i. For Unit 1 during Refueling Outage ***16*** and the subsequent operating cycle and for Unit 2 during Refueling Outage ***15*** and the subsequent operating cycle, the calculated accident induced leakage rate from the portion of the tubes below ***15.2*** inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.48 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined.

j. For Unit 1 during Refueling Outage ***16*** and the subsequent operating cycle and for Unit 2 during Refueling Outage ***15*** and the subsequent operating cycle, the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

3.0 Background

VEGP consist of two four-loop Westinghouse designed plants with Model F steam generators, having 5626 tubes in each steam generator. A total (for all four steam generators per unit) of 146 tubes are currently plugged on Unit 1 and 45 on Unit 2. The design of the steam generator includes Alloy 600 thermally treated tubing, full depth hydraulically expanded tubesheet joints, and stainless steel tube support plates with broached hole quatrefoil.

The steam generator inspection scope is governed by TS 5.5.9, “Steam Generator (SG) Program;” Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines,” (Reference 6); EPRI 1003138, “Pressurized Water Reactor Steam Generator Examination Guidelines,” (Reference 7); EPRI 1012987, “Steam Generator Integrity Assessment Guidelines,” (Reference 8); NMP-ES-004, "Steam Generator Program", and the results of the degradation assessments required by the Steam Generator Program. Criterion IX, "Control of Special Processes" of 10 CFR Part 50, Appendix B, requires in part that nondestructive testing be accomplished by qualified personnel using qualified procedures in accordance with the applicable criteria. The inspection techniques and equipment are capable of reliably detecting the known and potential specific degradation mechanisms applicable to VEGP. The inspection techniques, essential variables and equipment are qualified to Appendix H, "Performance Demonstration for Eddy Current Examination" of the EPRI Steam Generator Examination Guidelines.

Catawba Nuclear Station, Unit 2, (Catawba) reported indication of cracking following nondestructive eddy current examination of the steam generator tubes during their fall 2004 outage. NRC Information Notice (IN) 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," (Reference 9), provided industry notification of the Catawba issue. IN 2005-09 noted that Catawba reported crack like indications in the tubes approximately seven inches below the top of the hot leg tubesheet in one tube, and just above the tube-to-tubesheet welds in a region of the tube known as the tack expansion in several other tubes. Indications were also reported in the tube-end welds, also known as tube-to-tubesheet welds, which join the tube to the tubesheet.

SNC policies and programs require the use of applicable industry operating experience in the operation and maintenance of VEGP. The experience at Catawba, as noted in IN 2005-09, shows the importance of monitoring all tube locations (such as bulges, dents, dings, and other anomalies from the manufacture of the steam generators) with techniques capable of finding potential forms of degradation that may be occurring at these locations (as discussed in Generic Letter 2004-001, "Requirements for Steam Generator Tube Inspections"). Since the VEGP Westinghouse Model F steam generators were fabricated with Alloy 600 thermally treated tubes similar to the Catawba Unit 2 Westinghouse Model D5 steam generators, a potential exists for VEGP to identify tube indications similar to those reported at Catawba within the hot leg tubesheet region if similar inspections are performed during the 1R16 and 2R15 refueling outages.

Potential inspection plans for the tubes and tube welds underwent intensive industry discussions in March 2005. The findings in the Catawba steam generator tubes present three distinct issues with regard to the steam generator tubes at VEGP:

1) Indications in internal bulges and overexpansions within the hot leg tubesheet;

2) Indications at the elevation of the tack expansion transition; and

3) Indications in the tube-to-tubesheet welds and propagation of these indications into adjacent tube material.

Prior to each steam generator tube inspection, a degradation assessment, which includes a review of operating experience, is performed to identify degradation mechanisms that have a potential to be present in the VEGP steam generators. A validation assessment is also 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 discussed above, VEGP revised the steam generator inspection plan to include sampling of bulges and overexpansions within the tubesheet region on the hot leg side during Refueling Outage 1R14 and 2R13 for Units 1 and 2, respectively. The sample is based on the guidance contained in EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 7, and TS 5.5.9, 'Steam Generator (SG) Program." The inspection plan was expanded according to EPRI steam generator examination guidelines due to confirmed degradation in the region required to be examined (i.e., a tube crack). Degradation was not detected in the tubesheet region in Refueling Outage 1R14 and 2R13 for Units 1 and 2, respectively.

At Vogtle, tube flaw indications within the tube sheet have only been found at the hot leg tube ends and in bulges/overexpansions. Approximately 18,274 tube ends have been recently inspected at Vogtle. Twenty-seven flaw indications have been found in the inspections within 1 inch of the tube end. All of these indications were small and none met the tube repair criteria in the current technical specifications.

While flaws in bulges/overexpansions have been found at Vogtle 1, a separate inspection program for these indications has been implemented at both Vogtle 1 and 2. This inspection program is in accordance with Vogtle's current technical specifications and industry guidance.

Based on these inspections, a limited number of flaws existed in the tube sheets of Vogtle steam generators. The flaws that have been found are associated with residual stress conditions at either the tube ends or bulges/overexpansions within the tube sheet. No indications of a 360 degree sever has been detected in any steam generator at Vogtle. Consequently, the level of degradation in the Vogtle steam generators is very limited compared to the assumption of "all tubes severed" that was utilized in the development of the permanent H*. Consequently, structural integrity will be assured for the operating period between inspections allowed by TS 5.5.9, "Steam Generator (SG) Program".

As a result of these potential issues and the possibility of unnecessarily plugging tubes in the VEGP steam generators, SNC is proposing changes to TS 5.5.9 to limit the steam generator tube inspection and repair (plugging) to the safety significant portion of the tubes.

4.0 <u>Summary of Licensing Basis Analysis (H* Analysis)</u>

On May 19, 2009, Westinghouse WCAP-17071-P, Revision 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," (Reference 5) was submitted as enclosure 5 of Southern Nuclear Operating Company (SNC) request to change Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program", and TS 5.6.10, "Steam Generator Tube Inspection Report" to support implementation of a permanent alternate repair criterion for steam generator tubes. (Reference 19)

On July 10, 2009, SNC received a request for additional information (RAI) letter, which contained twenty-four (24) questions (Reference 22). As a result of a teleconference with NRC staff held on July 30, 2009, SNC received a second request for additional information letter on August 5, 2009 (Reference 23). The August 5, 2009 letter contained three (3) questions related to questions 4, 20 and 24 from RAI letter received on July 10, 2009. The August 5, 2009 letter also contained one (1) additional question.

On August 28, 2009, SNC provided the following documents in response to questions 1 through 24 of the July 10, 2009 letter and questions 1 through 4 of the August 5, 2009 letter. Westinghouse letter LTR-SGMP-09-100 P-Attachment, Revision 0, "Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators," August 12, 2009 (Reference 11) and Westinghouse letter SGMP-09-109-P Attachment, Revision 0 "Response to NRC Request for Additional Information on H*; RAI #4; Model F and Model D5 Steam Generators," August 25, 2009 (Reference 13)

On August 28, 2009, SNC submitted Westinghouse letter LTR-SGMP-09-104-P Attachment "White Paper on Probabilistic Assessment of H*" dated August 13, 2009, (Reference 12) as supplemental information.

On September 11, 2009, SNC submitted a request to revise the May 19, 2009 amendment request to be an interim change applicable to Unit 1 during refueling outage 15 and the subsequent operating cycle, and to Unit 2 during refueling outage 14 and the subsequent operating cycle (Reference 20). This request was made in response to a September 2, 2009 teleconference between NRC Staff and industry personnel, in which the NRC Staff indicated that their concerns with eccentricity of the tube sheet tube bore in normal and accident conditions (RAI question 4 of the July 10, 2009 letter and RAI question 1 of the August 5, 1009 letter) have not been resolved. The September 11, 2009 letter also requested the NRC staff to provide the specific questions concerning the tubesheet bore eccentricity issue which must be resolved to support a permanent alternate repair criteria amendment request.

On November 23, 2009, the NRC provided a letter (Reference 21) documenting the currently identified and unresolved issues relating to tubesheet bore eccentricity. This letter contained 14 questions which required resolution before the NRC could complete its review of a permanent amendment request.

WCAP-17330-P, Rev. 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5), November 2010 (Reference 16), LTR-SGMP-10-78, "Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and Their Relative Importance to H*," September 7, 2009 (Reference 14) and LTR-SGMP-10-33 P-Attachment, "H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity," September 13, 2010 (Reference 15) have been prepared by Westinghouse, to provide final resolution of the remaining questions identified in the November 23, 2009 NRC letter in support of the permanent H*amendment.

WCAP-17330-P, Rev. 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5), November 2010 (Reference 16) makes reference to Revision 2 of WCAP-17071-P and Revision 1 of LTR-SGMP-09-100 P-Attachment. As described above, SNC has previously submitted Revision 0 of these documents. These revisions (Revisions 1 and 2 of WCAP-17071-P, Revision 1 of LTR-SGMP-09-100 P-Attachment) were created to resolve editorial comments. The technical information contained in WCAP-17071-P, Revision 0 and LTR-SGMP-09-100 P-Attachment, Revision 0, remains valid and provides part of the licensing basis for the requested amendment change.

As a condition for approving VEGP Amendments157 and 138 (Units 1 and 2, respectively) (Reference 18), One Cycle Alternate Repair Criterion, H* (H-star), the NRC required a commitment to measure the location of the bottom of the expansion transition (BET) relative to the top of the tubesheet (TTS) and report any significant deviations from the constant 0.3 inch value already included in the calculated value(s) of H*. Enclosure 14, LTR-SGMP-09-111, Rev.

1, "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*," September 1, 2010, was prepared to support plant determinations of BET measurements and their significant deviation assessment.

The following table provides the list of licensing basis documents for H*.

Document Number	Revision Number	Title	Reference Number
WCAP-17071-P	0	H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F),"	5
LTR-SGMP-09-100 P-Attachment	0	Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators	11
LTR -SGMP-09-109-P Attachment	0	Response to NRC Request for Additional Information on H*; RAI #4; Model F and Model D5 Steam Generators	13
WCAP-17330-P	0	H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5)	16
LTR-SGMP-10-78	0	Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and Their Relative Importance to H*	14
LTR-SGMP-10-33 P-Attachment	0	H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity	15

5.0 Technical Evaluation

To preclude unnecessarily plugging tubes in the VEGP steam generators, an evaluation was performed to identify the safety significant portion of the tube within the tubesheet necessary to maintain structural and leakage integrity in both normal and accident conditions. Tube inspections will be limited to identifying and plugging degradation in the safety significant portion of the tubes. The technical evaluation for the inspection and repair methodology is provided in the H* Analysis described in section 4. This evaluation is based on the use of finite element model structural analysis and a bounding leak rate evaluation based on contact pressure between the tube and the tubesheet during normal and postulated accident conditions. The limited tubesheet inspection criteria were developed for the tubesheet region of the VEGP Model F steam generator 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 accident induced leakage limits are not exceeded. The H* Analysis provides technical justification for limiting the inspection in the tubesheet expansion region to less than the full depth of the tubesheet.

The basis for determining the safety significant portion of the tube within the tubesheet is based upon evaluation and testing programs that quantified the tube-to-tubesheet radial contact pressure for bounding plant conditions as described in the H* Analysis. The tube-to-tubesheet radial contact pressure provides resistance to tube pullout and resistance to leakage during plant operation and transients.

Primary-to-secondary leakage from tube degradation in the tubesheet area is assumed to occur in several design basis accidents: feedwater line break (FLB), steam line break (SLB), locked rotor, and control rod ejection. The radiological dose consequences associated with this assumed leakage are evaluated to ensure that they remain within regulatory limits (e.g. 10 CFR Part 100, 10 CFR 50.67, GDC 19). The accident induced leakage performance criteria are intended to ensure the primary-to-secondary leak rate during any accident does not exceed the primary-to-secondary leak rate assumed in the accident analysis. Radiological dose consequences define the limiting accident condition for the H* justification.

The constraint that is provided by the tubesheet precludes tube burst from cracks within the tubesheet. The criteria for tube burst described in NEI 97-06 and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," (Reference 10) are satisfied due to the constraint provided by the tubesheet. Through application of the limited tubesheet inspection scope as described below, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur. The accident induced leak rate limit is 1.0 gpm. The TS operational leak rate limit is 150 gpd (0.1 gpm) through any one steam generator. Consequently, there is significant margin between accident leakage and allowable operational leakage. The SLB/FLB leak rate ratio is only 2.48 resulting in significant margin between the conservatively estimated accident leakage and the allowable accident leakage (1.0 gpm).

Plant-specific operating conditions are used to generate the overall leakage factor ratios that are to be used in the condition monitoring and operational assessments. The plant-specific data provide the initial conditions for application of the transient input data. The results of the analysis of the plant-specific inputs, to determine the bounding plant for each model of steam generator are contained in Section 6 of Reference 5.

The leak rate ratio (accident induced leak rate to operational leak rate) is directly proportional to the change in differential pressure and inversely proportional to the dynamic viscosity. Since dynamic viscosity decreases with an increase in temperature, an increase in temperature results in an increase in leak rate.

However, for both the postulated SLB and FLB events, a plant cool down event would occur and the subsequent temperatures in the reactor coolant system (RCS) would not be expected to exceed the temperatures at plant no load conditions. Thus, an increase in leakage would not be expected to occur as a result of the viscosity change. The increase in leakage would only be a function of the increase in primary to secondary pressure differential. The resulting leak rate ratio for the SLB and FLB events is 2.48.

The other design basis accidents, such as the postulated locked rotor event and the control rod ejection event, are conservatively modeled using design specification transients which result in increased temperatures in the steam generator hot and cold legs for a period of time. As previously noted, dynamic viscosity decreases with increasing temperature. Therefore, leakage would be expected to increase due to decreasing viscosity, as well as due to the increasing differential pressure, for the duration of time that there is a rise in RCS temperature. For transients other than a SLB and FLB, the length of time that a plant with Model F steam generators will exceed the normal operating differential pressure across the tubesheet is less than 30 seconds. As the accident induced leakage performance criteria is defined in gallons per minute, the leak rate for a locked rotor event can be integrated over a minute to compare to the limit. Time integration permits an increase in acceptable leakage during the time of peak pressure differential by approximately a factor of two because of the short duration (less than 30 seconds) of the elevated pressure differential. This translates into an effective reduction in leakage factor by the same factor of two for the locked rotor event. Therefore, for the locked rotor event, the leakage factor of 1.74 (Revised Table 9-7, Reference 11) for VEGP is adjusted downward to a factor of 0.87. Similarly, for the control rod ejection event, the duration of the elevated pressure differential is less than 10 seconds. Thus, the peak leakage factor may be reduced by a factor of six from 2.62 to 0.44.

The plant transient response following a full power double-ended main feedwater line rupture corresponding to "best estimate" initial conditions and operating characteristics indicates that the transient for a Model F steam generator exhibits a cool down characteristic instead of a heat-up transient as generally presented in steam generator design transients and in the UFSAR Chapter 15.0 safety analysis. The use of either the component design specification transient or the Chapter 15.0 safety transient for leakage analysis for FLB is overly conservative because:

- The assumptions on which the FLB design transient is based are specifically intended to establish a conservative structural (fatigue) design basis for reactor coolant system components; however, H* does not involve component structural and fatigue issues. The best estimate transient is considered more appropriate for use in the H* leakage calculations.

- For the Model F steam generator, the FLB transient curve (Figure 9-5, Reference 5) represents a double-ended rupture of the main feedwater line concurrent with both loss of offsite power (loss of main feedwater and reactor coolant pump coast down) and turbine trip.

- The assumptions on which the FLB safety analysis is based are specifically intended to establish a conservative basis for minimum auxiliary feedwater (AFW) capacity requirements and combines worst case assumptions which are exceptionally more severe when the FLB occurs inside containment. For example, environmental errors that are applied to reactor trip and engineered safety feature actuation would be less severe. This would result in much earlier reactor trip and greatly increase the steam generator liquid mass available to provide cooling to the RCS.

A SLB event would have similarities to a FLB except that the break flow path would include the secondary separators, which could only result in an increased initial cooldown (because of retained liquid inventory available for cooling) when compared to the FLB transient. A SLB could not result in more limiting RCS temperature conditions than a FLB.

In accordance with plant operating procedures, the operator would take action following a high energy secondary line break to stabilize the RCS conditions. The expectation for a SLB or FLB with credited operator action is to stop the system cooldown through isolation of the faulted steam generator and control of temperature by the AFW system. Steam pressure control would be established by either the steam generator safety valves or control system (atmospheric relief valves). For any of the steam pressure control operations, the maximum RCS temperature would be approximately the no load temperature and would be well below normal operating temperature.

Since the best estimate FLB transient temperature would not be expected to exceed the normal operating temperature, the viscosity ratio for the FLB transient is set to 1.0.

The leakage factor of 2.48 for VEGP, for a postulated SLB/FLB, has been calculated as shown in Revised Table 9-7 of Reference 11. Specifically, for the condition monitoring (CM) assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 2.48 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.48 and compared to the observed operational leakage.

Reference 5 redefines the primary pressure boundary. The tube-to-tubesheet weld no longer functions as a portion of this boundary. The hydraulically expanded portion of the tube into the tubesheet over the H* distance now functions as the primary pressure boundary in the area of the tube and tubesheet, maintaining the structural and leakage integrity over the full range of steam generator operating conditions, including the most limiting accident conditions. The evaluation in Reference 5 determined that degradation in tubing below this safety significant portion of the tube does not require inspection or repair (plugging). The inspection of the safety significant portion of the tubes provides a high level of confidence that the structural and leakage performance criteria are maintained during normal operating and accident conditions.

WCAP-17071-P, section 9.8, provides a review of leak rate susceptibility due to tube slippage and concluded that the tubes are fully restrained against motion under very conservative design and analysis assumptions such that tube slippage is not a credible event for any tube in the bundle. As a condition of approval of Amendment Numbers 157 and 138 (Units 1 and 2, respectively) (Reference 18), SNC committed to monitor for tube slippage as part of the steam generator tube inspection program. This requirement will remain in place to support this License Amendment Request.

As a condition for approving VEGP One Cycle Alternate Repair Criterion, H* (Reference 18), the NRC staff requested that SNC perform a validation of the tube expansion from the top of tubesheet to the beginning of expansion transition (BET) to determine if there are any significant deviations that would invalidate assumptions in WCAP-17071-P (Reference 5). SNC has completed the validation of the tube expansion from the top of tubesheet to the beginning of expansion transition (BET) for VEGP Units 1 and 2. Based on data review and LTR-SGMP-09-111, Rev. 1 (Reference 17), SNC did not identify any significant deviations from the top of tubesheet to the beginning of expansion transition (BET) on VEGP Units 1 and 2.

6.0 <u>Regulatory Evaluation</u>

6.1 Applicable Regulatory Requirements/Criteria

General Design Criteria (GDC) 1, 2, 4, 14, 30, 31, and 32 of 10 CFR 50, Appendix A, define requirements for the reactor coolant pressure boundary (RCPB) with respect to structural and leakage integrity.

GDC 19 of 10 CFR 50, Appendix A, defines requirements for the control room and for the radiation protection of the operators working within it. Accidents involving the leakage or burst of steam generator tubing comprise a challenge to the habitability of the control room.

10 CFR 50, Appendix B, establishes quality assurance requirements for the design, construction, and operation of safety related components. The pertinent requirements of this appendix apply to all activities affecting the safety related functions of these components. These requirements are described in Criteria IX, XI, and XVI of Appendix B and include control of special processes, inspection, testing, and corrective action.

10 CFR 100, Reactor Site Criteria, established reactor siting criteria, with respect to the risk of public exposure to the release of radioactive fission products. Accidents involving leakage or tube burst of steam generator tubing may comprise a challenge to containment and therefore involve an increased risk of radioactive release.

Under 10 CFR 50.65, the Maintenance Rule, licensees classify steam generators as risk significant components because they are relied upon to remain functional during and after design basis events. Steam generators are to be monitored under 10 CFR 50.65(a)(2) against industry established performance criteria. Meeting the performance criteria of NEI 97-06, Revision 2, provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining the reactor coolant pressure boundary. The NEI 97-06, Revision 2, steam generator performance criteria are:

- All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design 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 loads.

- The primary to secondary accident induced leakage rate for any design basis accident, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. Leakage is not to exceed 1.0 gpm per steam generator, except for specific types of degradation at specific locations when implementing alternate repair criteria as documented in the Steam Generator Program technical specifications.

- The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day.

The safety significant portion of the tube is the length of tube that is engaged in the tubesheet from the secondary face that is required to maintain structural and leakage integrity over the full range of steam generator operating conditions, including the most limiting accident conditions. The evaluation in this Enclosure determined that degradation in tubing below the safety significant portion of the tube does not require plugging and serves as the bases for the tubesheet inspection program. As such, the VEGP inspection program provides a high level of confidence that the structural and leakage criteria are maintained during normal operating and accident conditions.

6.2 No Significant Hazards Consideration

This amendment application proposes to revise Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," to exclude portions of the tubes within the tubesheet from periodic steam generator inspections. In addition, this amendment proposes to revise Technical Specification (TS) 5.6.10, "Steam Generator Tube Inspection Report" to remove reference to previous interim alternate repair criteria and provide reporting requirements specific to the temporary alternate repair criteria. Application of the structural analysis and leak rate evaluation results, to exclude portions of the tubes from inspection and repair is interpreted to constitute a redefinition of the primary-to-secondary pressure boundary.

The proposed change defines the safety significant portion of the tube that must be inspected and repaired. A justification has been developed by Westinghouse Electric Company, LLC to identify the specific inspection depth below which any type of axial or circumferential primary water stress corrosion cracking can be shown to have no impact on Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," (Reference 6) performance criteria.

SNC 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:

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

   Response: No

   The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the steam generator inspection criteria and the steam generator inspection reporting criteria does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident.

   Of the applicable accidents previously evaluated, the limiting transients with consideration to the proposed change to the steam generator tube inspection and repair criteria are the steam generator tube rupture (SGTR) event and the feedline break (FLB) postulated accidents.

   During the SGTR event, the required structural integrity margins of the steam generator tubes and the tube-to-tubesheet joint over the H* distance will be maintained. Tube rupture in tubes with cracks within the tubesheet is precluded by the constraint provided by the tube-to-tubesheet joint. 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, as discussed in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes,"(Reference 10) are maintained for both normal and postulated accident conditions.

The proposed change has no impact on the structural or leakage integrity of the portion of the tube outside of the tubesheet. The proposed change maintains structural integrity of the steam generator tubes and does not affect other systems, structures, components, or operational features. Therefore, the proposed change results 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 below the proposed limited inspection depth is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. However, primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed changes 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. Therefore, the proposed changes do not result in a significant increase in the consequences of a SGTR.

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

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., a SLB) is limited by flow restrictions. These restrictions result from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications.

The leakage factor of 2.48 for Vogtle Electric Generating Plant (VEGP), for a postulated SLB/FLB, has been calculated as shown in Revised Table 9-7 of Reference 11. Specifically, for the condition monitoring (CM) assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 2.48 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.48 and compared to the observed operational leakage.

The probability of a SLB is unaffected by the potential failure of a steam generator tube as the failure of the tube is not an initiator for a SLB event. SLB leakage is limited by leakage flow restrictions resulting from the leakage path above potential cracks through the tube-to-tubesheet crevice. The leak rate during postulated accident conditions (including locked rotor) has been shown to remain within the accident analysis assumptions for all axial and or circumferentially orientated cracks occurring 15.2 inches below the top of the tubesheet. The accident induced leak rate limit is 1.0 gpm. The TS operational leak rate is 150 gpd (0.1 gpm) through any one steam generator. Consequently, there is significant margin between accident leakage and allowable operational leakage. The SLB/FLB leak rate ratio is only 2.48 resulting in significant margin between the conservatively estimated accident leakage and the allowable accident leakage (1.0 gpm).

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed change that alters the steam generator inspection criteria and the steam generator inspection reporting criteria does not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident analyses.

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

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

Response: No

The proposed change that alters the steam generator inspection criteria and the steam generator inspection reporting criteria maintains the required structural margins of the steam generator tubes for both normal and accident conditions. NEI 97-06, Revision 2, "Steam Generator Program Guidelines" (Reference 6) and RG 1.121, are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting GDC 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," and GDC 32, "Inspection of Reactor Coolant Pressure Boundary," by reducing the probability and consequences of a SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall 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 American Society of Mechanical Engineers (ASME) Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, the H* analysis, documented in section 4, defines a length of degradation free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited hot and cold leg tubesheet inspection 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 change does not involve a significant reduction in any margin of safety.

6.3 Conclusion

The safety significant portion of the tube is the length of tube that is engaged within the tubesheet to the top of the tubesheet (secondary face) that is required to maintain structural and leakage integrity over the full range of steam generating operating conditions, including the most limiting accident conditions. The H* Analysis determined that degradation in tubing below the safety significant portion of the tube does not require plugging and serves as the basis for the limited tubesheet inspection criteria, which are intended to ensure the primary-to-secondary leak rate during any accident does not exceed the leak rate assumed in the accident analysis.

Based on the considerations above, 1) there is a 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.

7.0 Environmental Considerations

SNC has evaluated the proposed amendment for environmental considerations. The review has resulted in the determination 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 forth 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.

8.0 References

1. Vogtle Electric Generating Plant Units 1 and 2, "RE: Issuance of Amendments Regarding the Steam Generator Tube Surveillance Program," September 21, 2005, (TAC Nos. MC8078 and MC8079).

2. 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).

3. 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," April 9, 2008, (TAC Nos. MD7450 and MD7451).

4. Vogtle Electric Generating Plant Units 1 and 2, "Issuance of Amendments Regarding Steam Generator Tube Inspection Program," September 16, 2008, (TAC Nos. MD9148 and MD9149).

5. Westinghouse Electric Company LLC, WCAP-17071-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)." (ADAMS Accession Nos. ML091470699 (Introduction through Chapter 5) and ML091470700 (Chapter 6 to end))

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

7. EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines."

8. EPRI 1012987; "Steam Generator Integrity Assessment Guidelines".

9. NRC Information Notice (IN) 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds."

10. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976, (ADAMS Accession No. ML003739366).

11. LTR-SGMP-09-100, "LTR-SGMP-09-100 P-Attachment, "Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators," August 12, 2009. (ADAMS Accession No. ML092450102)

12. LTR-SGMP-09-104 P-Attachment, "White Paper on Probabilistic Assessment of H*," August 13, 2009. (ADAMS Accession No. ML092450030)

13. LTR-SGMP-09-109 P-Attachment, "Response to NRC Request For Additional Information on H*; RAI #4; Model F and Model D5 Steam Generators," August 25, 2009. (ADAMS Accession No. ML092450333)

14. LTR-SGMP-10-78, "Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and Their Relative Importance to H*," September 7, 2009.

15. LTR-SGMP-10-33 P-Attachment, "H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity," September 13, 2010.

16. WCAP-17330, Rev. 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5), November 2010.

17. LTR-SGMP-09-111, Rev. 1, "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*," September 1, 2010.

18. Vogtle Electric Generating Plant, Units 1 And 2, Issuance Of Amendments Regarding Technical Specification (TS) Section 5.5.9, "Steam Generator Program," And TS 5.6.10, "Steam Generator Tube Inspection Report," For Interim Alternate Repair Criteria September 24, 2009 (TAC Nos. ME1339 And ME1340) (ADAMS Accession No. ML092170782)

19. Southern Nuclear Operating Company, Inc. (SNC), letter NL-09-0547, "Vogtle Electric Generating Plant License Amendment Request to Revise Technical Specification (TS) Sections 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" for Permanent Alternate Repair Criteria," May 19, 2009, (ADAMS Accession No. ML091470701).

20. SNC letter NL-09-1411, September 11, 2009, amending its H* application to apply only to the next operating cycle, (ADAMS Accession No. ML092540511)

21. NRC Letter, "Vogtle Electric Generating Plant, Units 1 and 2 – Transmittal of Unresolved Issues Regarding Permanent Alternate Repair Criteria for Steam Generators (TAC Nos. ME 1339 and ME 1340)," November 23, 2009. (ADAMS Accession No. ML093030490)

22. NRC Letter, "Vogtle Electric Generating Plant, Units 1 And 2 -- Request for Additional Information Regarding Steam Generator Program (TAC Nos. ME 1339 and ME 1340)," July 10, 2009. (ADAMS Accession No. ML091880384)

23. NRC Letter, "Vogtle Electric Generating Plant, Units 1 And 2 -- Request for Additional Information Regarding Steam Generator Program (TAC Nos. ME 1339 and ME 1340)," August 5, 2009. (ADAMS Accession No. ML092150057)

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

Enclosure 2

Markup of Proposed Technical Specifications

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 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. 16 15

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

For Unit 1 during Refueling Outage ~~15~~ and the subsequent operating
15.2 cycle and for Unit 2 during Refueling Outage ~~14~~ and the subsequent operating cycle, tubes with service-induced flaws located greater than ~~13.1~~ inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to ~~13.1~~ inches below the top of the tubesheet shall be plugged upon detection.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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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. For Unit 1 during Refueling Outage ~~15~~ and the subsequent operating cycle and for Unit 2 during Refueling Outage ~~14~~ and the subsequent operating cycle, portions of the tube below ~~13.1~~ inches below the top of the tubesheet are excluded from this requirement.

15.2

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(continued)

5.6 Reporting Requirements

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:

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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,

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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. For Unit 1 during Refueling Outage ~~15~~ and the subsequent operating cycle and for Unit 2 during Refueling Outage ~~14~~ and the subsequent operating cycle, 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; and

15 16 15.2

i. For Unit 1 during Refueling Outage ~~15~~ and the subsequent operating cycle and for Unit 2 during Refueling Outage ~~14~~ and the subsequent operating cycle, the calculated accident induced leakage rate from the portion of the tubes below ~~13.1~~ inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.48 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined.

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j. For Unit 1 during Refueling Outage ~~15~~ and the subsequent operating cycle and for Unit 2 during Refueling Outage ~~14~~ and the subsequent operating cycle, the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

16

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. For Unit 1 during Refueling Outage ~~15~~ and the subsequent operating cycle and for Unit 2 during Refueling Outage ~~14~~ and the subsequent operating cycle, portions of the tube below ~~13.1~~ inches

15

15.2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.2 (continued)

criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 100.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

7. License Amendment Nos. XXX ~~157~~ and XXX ~~138~~, "Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding Technical Specification (TS) Section 5.5.9, "Steam Generator Program," and TS 5.6.10, "Steam Generator Tube Inspection Report," for Temporary ~~Interim~~ Alternate Repair Criteria (Approval Date) ~~(Tac Nos. ME1339 and ME1340): September 29, 2009.~~

AMENDMENT NUMBERS AND APPROVAL DATE WILL BE ADDED AFTER NRC APPROVAL OF LICENSE AMENDMENT REQUEST.

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

Enclosure 3

Typed Pages for Technical Specification

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 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:

For Unit 1 during Refueling Outage 16 and the subsequent operating cycle and for Unit 2 during Refueling Outage 15 and the subsequent operating cycle, tubes with service-induced flaws located greater than 15.2 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 15.2 inches below the top of the tubesheet shall be plugged upon detection.

(continued)

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 1 during Refueling Outage 16 and the subsequent operating cycle and for Unit 2 during Refueling Outage 15 and the subsequent operating cycle, portions of the tube below 15.2 inches below the top of the tubesheet are excluded from this requirement.

(continued)

5.6 Reporting Requirements

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. For Unit 1 during Refueling Outage 16 and the subsequent operating cycle and for Unit 2 during Refueling Outage 15 and the subsequent operating cycle, 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; and

i. For Unit 1 during Refueling Outage 16 and the subsequent operating cycle and for Unit 2 during Refueling Outage 15 and the subsequent operating cycle, the calculated accident induced leakage rate from the portion of the tubes below 15.2 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.48 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined.

j. For Unit 1 during Refueling Outage 16 and the subsequent operating cycle and for Unit 2 during Refueling Outage 15 and the subsequent operating cycle, the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

BASES (continued)

APPLICABLE SAFETY ANALYSES

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.2 (continued)

criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

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