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April 28, 2010 10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Carolinas, LLC (Duke)
Catawba Nuclear Station, Unit 2
Docket Number 50-414
Proposed Technical Specifications (TS) Amendment
TS 5.5.9, "Steam Generator (SG) Program"
TS 5.6.8, "Steam Generator (SG) Tube Inspection Report"
License Amendment Request to Revise TS for Alternate Repair Criteria

Pursuant to 10 CFR 50.90, Duke is requesting an amendment to Catawba Facility Operating License NPF-52 and the subject TS. This amendment request proposes to revise TS 5.5.9 to exclude portions of the tube below the top of the SG tubesheet from periodic SG tube inspections and plugging or repair. In addition, reporting requirement changes are proposed to TS 5.6.8. This change is supported by Westinghouse Electric Company, LLC, (Westinghouse) WCAP-17072-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)". Note that the WCAP was originally prepared to support a permanent alternate repair criteria; however, ongoing technical issues resulted in Westinghouse plants with Fall 2009 and Spring 2010 refueling outages requesting one-cycle approvals until all of the issues could be resolved. Therefore, this submittal is also requesting a one-cycle approval for the Catawba Unit 2 End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation. Please note that in the event that all ongoing technical issues are resolved within a time frame supporting a permanent alternate repair criteria, Duke reserves the right to modify this submittal to request a permanent amendment. Duke will discuss this with the NRC prior to modifying this submittal.

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This request includes attachments as noted in the following table:

Attachment	Subject
1	Technical and Regulatory Evaluations
2	Marked-Up TS Pages
3	Westinghouse Authorization Letter CAW-09-2585 with Accompanying Affidavit, Proprietary Information Notice, and Copyright Notice (WCAP-17072-P)
4	Westinghouse WCAP-17072-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)" (Proprietary)
5	Westinghouse WCAP-17072-NP, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)" (Non-Proprietary)
6	Westinghouse Letter LTR-SGMP-09-79, "WCAP-17072 Errata and Clarifications"
7	Westinghouse Letter LTR-RCPL-09-133, "WCAP-17072-P, Rev. 0 Proprietary Information Clarification"
8	Westinghouse Authorization Letter CAW-09-2637 with Accompanying Affidavit, Proprietary Information Notice, and Copyright Notice (LTR-SGMP-09-100 P-Attachment)
9	Westinghouse Letter LTR-SGMP-09-100 P-Attachment, "Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators" (questions 1 through 20 and 24 of the NRC RAI) (Proprietary)
10	Westinghouse Letter LTR-SGMP-09-100 NP-Attachment, "Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators" (questions 1 through 20 and 24 of the NRC RAI) (Non-Proprietary)
11	Westinghouse Letter LTR-SGMP-09-121, "Replacements for Illegible Pages in Prior RAI Response (LTR-SGMP-09-100)"
12	Catawba Unit 2 Site Specific Response to (Industry) NRC RAI Questions 21, 22, and 23
13	Westinghouse Authorization Letter CAW-09-2664 with Accompanying Affidavit, Proprietary Information Notice, and

	Copyright Notice (LTR-SGMP-09-109 P-Attachment)
14	Westinghouse Letter LTR-SGMP-09-109 P-Attachment, "Response to NRC Request for Additional Information on H*; RAI #4; Model F and Model D5 Steam Generators" (Proprietary)
15	Westinghouse Letter LTR-SGMP-09-109 NP-Attachment, "Response to NRC Request for Additional Information on H*; RAI #4; Model F and Model D5 Steam Generators" (Non-Proprietary)
16	Westinghouse Letter LTR-SGMP-10-34 Rev. 2, "An Assessment of the Impact of Revised Normal Operating Conditions on the Catawba Unit 2 H* Calculations" (Non-Proprietary)
17	Summary of Regulatory Commitments

As the attached reports contain information proprietary to Westinghouse Electric Company LLC, they are supported by affidavits signed by Westinghouse, the owner of the information. The attached affidavits set 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. Accordingly, it is requested that the information that 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 information listed above or the supporting Westinghouse affidavits should reference the applicable CAW letters 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.

This proposed amendment has been reviewed and approved by the Catawba Plant Operations Review Committee and by the corporate Nuclear Safety Review Board in accordance with the requirements of the Duke Quality Assurance Program.

Duke requests approval of this proposed amendment by August 31, 2010, to support implementation during the Catawba Unit 2 Fall 2010 End of Cycle 17 Refueling Outage. Once approved, the amendment will be implemented prior to requiring the SGs to be operable at the completion of the outage.

In accordance with 10 CFR 50.91, Duke is notifying the State of South Carolina of this application for license amendment by transmitting a copy of this letter and its non-proprietary attachments to the designated state official.

Should you have any questions concerning this information, please contact L.J. Rudy at (803) 701-3084.

Very truly yours,

James R. Morris

LJR/s

Attachments

James R. Morris affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

James R. Morris, Vice President

Subscribed and sworn to me: April 28, 2010
Date

Notary Public

BETTY W. ALDRIDGE
YORK COUNTY
NOTARY PUBLIC
MY COMMISSION EXPIRES 1/31/2018
STATE OF SOUTH CAROLINA

My commission expires: 1/31/2018
Date

SEAL

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ATTACHMENT 1

Technical and Regulatory Evaluations

Subject: License Amendment Request to Revise TS for Alternate Repair Criteria

1. DESCRIPTION

2. PROPOSED CHANGE

3. BACKGROUND

4. TECHNICAL EVALUATION

5. REGULATORY EVALUATION

   5.1 Applicable Regulatory Requirements/Criteria

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6. ENVIRONMENTAL CONSIDERATION

1. DESCRIPTION

This evaluation supports a request to amend Facility Operating License NPF-52 (Catawba Nuclear Station Unit 2).

This amendment application proposes to revise TS 5.5.9, "Steam Generator (SG) Program" to exclude portions of the tube below the top of the SG tubesheet from periodic SG tube inspections and plugging or repair. The application also revises the wording of reporting requirements contained in TS 5.6.8, "Steam Generator (SG) Tube Inspection Report".

This change is for the Unit 2 End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, and is supported by Westinghouse Electric Company, LLC, WCAP-17072-P, Revision 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)", May 2009, as supplemented by documentation issued in response to NRC Requests for Additional Information (RAIs). WCAP-17072-P recommends an H* value of 13.8 inches based on the statistical confidence limits of 95/50; however, Duke has chosen to use an H* value of 16.95 inches for additional conservatism. This more conservative value was discussed between the NRC and industry representatives on May 27, 2009.

Note that the WCAP was initially prepared to support a permanent alternate repair criteria; the reason and justification for the request for a one-cycle change is discussed below.

The NRC has previously granted similar H* amendments to other Westinghouse plants for their Fall 2009 and Spring 2010 refueling outages, as indicated in this submittal.

On September 2, 2009, in a teleconference between NRC Staff and industry personnel, NRC Staff indicated that their concerns with eccentricity of the tubesheet tube bore in normal and accident conditions (RAI question 4 as transmitted by a July 10, 2009 letter and RAI question 1 as transmitted by a August 5, 2009 letter) have not been completely resolved to the satisfaction of the NRC. The NRC further indicated that there was insufficient time to resolve these issues to support approval of a permanent amendment request to support upcoming refueling outages. Consequently, Duke is proposing changes to the indicated TS as a one-cycle change for the Unit 2 End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation.

Note that throughout this submittal, references to WCAP-17072-P include not only the originally issued WCAP, but also all changes to the WCAP as documented by the responses to the NRC RAIs.

The H* analysis is based on maintaining structural and leakage integrity in the event of an accident. From a structural perspective, the value of H* ensures that tube rupture or tube pullout from the tubesheet will not occur in the event of an accident during the entire life of the plant. Even in the event that all tubes in the SG have a 360 degree sever at the H* location, structural integrity of the SG tube bundle will be maintained. This assumption bounds the current status of the Catawba Unit 2 SGs with significant margin. Tubesheet inspections with probes capable of detecting crack-like flaws have been extensively performed by several utilities with SGs similar to those installed at Catawba Unit 2 (i.e., fabricated with Alloy 600 Thermally Treated (TT) tubing). These inspections included the top of the tubesheet region, expansion anomalies within the tubesheet, and the tube end region near the weld. The industry inspections have demonstrated that flaws in the tubesheet are negligible when considering the number of tubes inspected, the severity of the degradation detected, and when compared to the conservative H* assumption that all tubes are severed.

Catawba Unit 2 reported indication of cracking following non-destructive eddy current examination of the SG tubes during the Fall 2004 Refueling Outage. NRC Information Notice (IN) 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds", provided industry notification of this 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.

Based on overall industry inspections, a limited number of flaws exist in the tubesheets of SGs. The flaws that have been found are associated with residual stress conditions at either the tube ends or bulges/overexpansions within the tubesheet. No indication of a 360 degree sever has been detected in any SG. Consequently, the level of degradation in the SGs is very limited compared to the H* assumption of "all tubes severed". Therefore, structural integrity will be assured for the operating period between inspections allowed by TS 5.5.9.

From a leakage perspective, projections of accident induced SG tube leakage are based on leakage rate factors applied to leakage detected during normal operation. The acceptance criteria for Catawba Unit 2 SG tube leak rates as operated upon by the associated multiplication factor is bounded by the SG tube leak rate assumed in the relevant accident analyses. The projected accident induced leakage remains the same for both the temporary one-cycle and the permanent H* amendments. No quantifiable primary to secondary SG tube leakage has been detected during the current operating cycle at Catawba Unit 2.

For Catawba Unit 2, the number of SG tubes identified with flaws within the tubesheet is small in comparison to the input assumptions used in the

development of the permanent H*. Consequently, significant margin exists between the current state of the Catawba Unit 2 SGs and the conservative assumptions used as the basis for the permanent H*. Structural and leakage integrity will continue to be assured for the operating period between inspections allowed by TS 5.5.9 with the implementation of the proposed one-cycle H*. WCAP-17072-P recommends the 95% probability/50% confidence H* value of 13.8 inches; however, Duke has chosen to use an H* value of 16.95 inches for additional conservatism.

2. PROPOSED CHANGE

The proposed changes to the TS are as follows:

TS 5.5.9c currently states:

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

1. *For the Unit 2 End of Cycle 16 Refueling Outage and subsequent Cycle 17 operation only, tubes with flaws having a circumferential component less than or equal to 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet do not require plugging. Tubes with flaws having a circumferential component greater than 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet shall be removed from service.*

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

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

   When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet, and the total of the circumferential components found in the tube exceeds 94 degrees, then the tube shall be removed from service. When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet and within 1 inch axial separation distance of a flaw above 1 inch from the bottom of the tubesheet, and the total of the circumferential components found in the tube exceeds 94 degrees, then the tube shall be removed from service.

When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

TS 5.5.9d currently states:

Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and method of inspection shall be performed with the objective of detecting flaws of any type (for example, volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting requirements d.1, d.2, d.3, and d.4 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. *For Unit 1, inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 Effective Full Power Months (EFPM). The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 EFPM or three refueling outages (whichever is less) without being inspected.*

3. *For Unit 2, inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 EFPM. 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 EFPM or two refueling outages (whichever is less) without being inspected.*

4. *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 EFPM or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that*

a crack-like indication is not associated with crack(s), then the indication need not be treated as a crack.

TS 5.6.8 currently states:

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of the inspection. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Non-destructive 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 2, following completion of an inspection performed during the End of Cycle 16 Refueling Outage (and any inspections performed during subsequent Cycle 17 operation), the number of indications and location, size, orientation, whether initiated on the primary or secondary side for each service-induced flaw within the thickness of the tubesheet, and the total of the circumferential components and any circumferential overlap below 17 inches from the top of the tubesheet as determined in accordance with TS 5.5.9c.1,

i. For Unit 2, following completion of an inspection performed during the End of Cycle 16 Refueling Outage (and any inspections performed during subsequent Cycle 17 operation), the primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign leakage to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report, and

j. For Unit 2, following completion of an inspection performed during the End of Cycle 16 Refueling Outage (and any inspections performed during

subsequent Cycle 17 operation), the calculated accident leakage rate from the portion of the tubes below 17 inches from the top of the tubesheet for the most limiting accident in the most limiting SG.

The proposed changes to TS 5.5.9c are as follows (additions or changes are in **bold type**):

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

1. ***For Unit 2 only, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, tubes with service-induced flaws located greater than 16.95 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 16.95 inches below the top of the tubesheet shall be plugged upon detection.***

The proposed changes to TS 5.5.9d are as follows (additions or changes are in **bold type**):

Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. ***For Unit 1, the*** *number and portions of the tubes inspected and method of inspection shall be performed with the objective of detecting flaws of any type (for example, volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube.* ***For Unit 2, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, the number and portions of the tubes inspected and method of inspection shall be performed with the objective of detecting flaws of any type (for example, volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from 16.95 inches below the top of the tubesheet on the hot leg side to 16.95 inches below the top of the tubesheet on the cold leg side, and that may satisfy the applicable tube repair criteria.*** *In addition to meeting requirements d.1, d.2, d.3, and d.4 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. *For Unit 1, inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 Effective Full Power Months (EFPM). The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 EFPM or three refueling outages (whichever is less) without being inspected.*

3. *For Unit 2, inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 EFPM. 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 EFPM or two refueling outages (whichever is less) without being inspected.*

4. ***For Unit 1, 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 EFPM or one refueling outage (whichever is less). **For Unit 2, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, if crack indications are found in any SG tube from 16.95 inches below the top of the tubesheet on the hot leg side to 16.95 inches below the top of the tubesheet on the cold leg side, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 EFPM or one refueling outage (whichever is less).** If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with crack(s), then the indication need not be treated as a crack.*

The proposed changes to TS 5.6.8 are as follows (additions or changes are in **bold type**):

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of the inspection. The report shall include:

a. *The scope of inspections performed on each SG,*

b. *Active degradation mechanisms found,*

c. *Non-destructive 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 2,* ***following completion of an inspection performed during the End of Cycle 17 Refueling Outage (and any inspections performed during subsequent Cycle 18 operation),*** *the primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign leakage to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,*

i. *For Unit 2,* ***following completion of an inspection performed during the End of Cycle 17 Refueling Outage (and any inspections performed during subsequent Cycle 18 operation),*** *the calculated accident leakage rate from the portion of the tubes below* ***16.95*** *inches from the top of the 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 3.27 times the maximum primary to secondary LEAKAGE rate, the report shall describe how it was determined, and***

j. ***For Unit 2, following completion of an inspection performed during the End of Cycle 17 Refueling Outage (and any inspections performed during subsequent Cycle 18 operation), the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.***

The Facility Operating License condition associated with Unit 2 Amendment 244 currently states:

Additional Condition: *For steam generator (SG) integrity assessments, the ratio of 2.5 will be used in completion of both the Condition Monitoring*

	(CM) and the Operational Assessment (OA) upon implementation of the Interim Alternate Repair Criterion (IARC). For example, for the CM assessment, the component of leakage from the lower 4 inches of the most limiting SG during the prior cycle of operation will be multiplied by a factor of 2.5 and added to the total leakage from any other source and compared to the allowable accident analysis leakage assumption. For the OA, the difference in leakage from the allowable limit during the limiting design basis accident minus the leakage from the other sources will be divided by 2.5 and compared to the observed leakage. An administrative limit will be established to not exceed the calculated value.
Implementation Date:	*Prior to any entry into Mode 4 during Cycle 17 operation*

This Facility Operating License condition is deleted in favor of the third regulatory commitment in Attachment 17.

3. BACKGROUND

TS 5.5.9 requires that a SG program be established and implemented to ensure that SG tube integrity is maintained. SG tube integrity is maintained by meeting specified performance criteria for structural and leakage integrity, consistent with the plant design and licensing bases. TS 5.5.9 requires a condition monitoring assessment to be performed during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met. TS 5.5.9 also includes provisions regarding the scope, frequency, and methods of SG tube inspections. Of relevance to the amendment application, these provisions require that the number and portions of tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type that may be present along the length of a tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet (excluding the welds themselves), and that may satisfy the applicable tube repair criteria. The applicable tube repair criteria are that 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.

On March 31, 2006, the NRC issued Amendment 224 for Catawba Unit 2. This amendment involved a one-cycle change regarding required SG tube repair criteria during the End of Cycle 14 Refueling Outage and subsequent Cycle 15 operation. The amendment also added a license condition requiring a reduction in the allowable normal operating primary to secondary leakage rate through any one SG and through all SGs. On October 31, 2007, the NRC issued Amendment 233 for Catawba Unit 2. This amendment involved a second one-cycle change for the End of Cycle 15 Refueling Outage and subsequent Cycle 16 operation. On April 13, 2009, the NRC issued Amendment 244 for Catawba Unit 2. This amendment involved a one-cycle change for the End of Cycle 16 Refueling Outage and subsequent Cycle 17 operation and incorporated an interim alternate repair criteria for SG tube repair.

The industry has been actively working toward a permanent solution to this issue using the Westinghouse H* methodology. However, due to outstanding technical issues, no permanent TS change has yet been approved by the NRC.

Catawba Unit 2 is a four loop Westinghouse designed plant with Model D5 SGs having 4570 tubes in each SG (for a total of 18,280 tubes). A total of 328 tubes are currently plugged in all four SGs. The design of the SGs includes Alloy 600 thermally treated tubing, full depth hydraulically expanded tubesheet joints, and stainless steel tube support plates with broached hole quatrefoils.

In addition to TS 5.5.9, the SG inspection scope is currently governed by the following documents:

- Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines", Revision 2, May 2005
- Electric Power Research Institute (EPRI) 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines", Revision 7
- EPRI 1012987, "Steam Generator Integrity Assessment Guidelines", Revision 2
- Duke Energy's SG Management Program

10 CFR 50, Appendix B, Criterion IX, "Control of Special Processes", requires in part that non-destructive 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 Catawba Unit 2. The inspection techniques, essential variables, and equipment are qualified to Appendix H, "Performance Demonstration for Eddy Current Examination", of EPRI 1013706 or to Appendix I, "NDE System Measurement Uncertainties for Tube Integrity Assessment".

4. TECHNICAL EVALUATION

To preclude unnecessarily plugging tubes in the Catawba Unit 2 SGs, tube inspections will be limited to identifying and plugging degradation in the portion of the tube within the tubesheet necessary to maintain structural and leakage integrity during both normal and accident conditions. The technical evaluation for the inspection and repair methodology is provided in WCAP-17072-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)". The 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 Catawba Unit 2 Model D5 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 pullout from the tubesheet due to axial end cap loads acting on the tube and to ensure that the accident induced leakage limits are not exceeded. WCAP-17072-P 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 portion of the tube which requires eddy current inspection within the tubesheet is evaluation and testing programs that quantified the tube-to-tubesheet radial contact pressure for bounding plant conditions as described in WCAP-17072-P. The tube-to-tubesheet radial contact pressure provides resistance to tube pullout.

Primary to secondary leakage from tube degradation is assumed to occur in several design basis accidents: Main Steam Line Break, Locked Rotor, and Control Rod Ejection. (In addition, for the SG Tube Rupture, primary to secondary leakage is also assumed to occur in the intact SGs.) The radiological dose consequences associated with this assumed leakage are evaluated to ensure that they remain within regulatory limits (e.g., 10 CFR 50.67, General Design Criterion (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 for cracks within the tubesheet. The criteria for tube burst described in NEI 97-06 and in NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes", 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 analysis calculations assume a primary to secondary leakage equivalent to the TS operational leak rate limit of 150 gallons per day through any one SG and 600 gallons per day through all SGs. The maximum accident leak rate ratio for Catawba Unit 2 is 2.65 (Attachment 9, Table RAI24-2). Per Attachment 16, the leak rate ratio has been increased to 3.27.

Plant-specific operating conditions are used to generate the overall leakage factor ratios that are 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 SG and to assure that the design basis accident contact pressures are greater than the normal operating pressure contact pressure are contained in Section 6 of WCAP-17072-P.

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 Steam Line Break and Feed Line Break events, a plant cooldown event would occur and the subsequent temperatures in the reactor coolant system 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 temperature 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 Steam Line Break and Feed Line Break events is 2.65. The leak rate ratio has been increased to 3.27 per Attachment 16.

The other design basis accidents, such as the postulated Locked Rotor and the Control Rod Ejection events, are conservatively modeled using the design specification transients that result in increased temperatures in the SG 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 and increasing differential pressure for the duration of time that there is a rise in reactor coolant system temperature. For transients other than a Steam Line Break and a Feed Line Break, the length of time that a plant with Model D5 SGs 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 for comparison to the limit. Time integration permits an increase in acceptable leakage during the time of peak pressure differential by 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 the leakage factor by the same factor of two for the Locked Rotor event. Therefore, for the Locked Rotor event, the leakage factor of 1.48 (Attachment 9, Table RAI24-2) for Catawba Unit 2 is adjusted downward to a

factor of 0.74. Similarly, for the Control Rod Ejection event, the duration of the elevated pressure differential is less than 10 seconds. Thus, the peak leakage factor is reduced by approximately a factor of six, from 2.19 to 0.37. Due to the short duration of the transients above Normal Operating Pressure (NOP) differential, no leakage factor is required for the Locked Rotor and Control Rod Ejection events (i.e., the leakage factor is under 1.0 for both transients).

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 exhibits a cooldown characteristic instead of a heatup transient as is generally presented in the SG design transients and in the Chapter 15 safety analyses. The use of either the component design specification transient or the Chapter 15 safety analysis transient for the leakage analysis for Feed Line Break is overly conservative because:

- The assumptions on which the Feed Line Break design transient is based are specifically intended to establish a conservative structural (fatigue) design basis for reactor system components; however, since 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 D5 Feed Line Break SG design transient, using the Feed Line Break design transient curve, the maximum reactor coolant system temperature can exceed the saturation temperature which is predicted to occur by the worst-case Feed Line Break heatup Chapter 15 Safety Analysis transient response.

- The assumptions on which the Feed Line Break safety analysis is based are specifically intended to establish a conservative basis for minimum auxiliary feedwater capacity requirements and combines worst-case assumptions which are exceptionally more severe when the Feed Line Break occurs inside containment. For example, environmental errors that are applied to reactor trip and emergency safety feature actuations would no longer be applicable. This would result in a much earlier reactor trip and greatly increase the SG liquid mass available to provide cooling to the reactor coolant system.

A Steam Line Break event would have similarities to a Feed Line Break 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 Feed Line Break transient. A Steam Line Break could not result in more limiting temperature conditions than a Feed Line Break.

In accordance with plant emergency operating procedures, it is expected that the operator would take action following a high energy secondary line break to stabilize the reactor coolant system conditions. The expectation for a Steam Line Break or Feed Line Break with credited operator action is to stop the system cooldown through isolation of the faulted SG and control temperature by the auxiliary feedwater system. Steam pressure control would be established by either the SG safety valves or steam dump or power operated relief valves. For any of the steam pressure control options, the maximum temperature would be approximately the no load temperature and would be well below the normal operating temperature for the plant.

Subsequently, the operator would initiate a cooldown and depressurization of the reactor coolant system which would continue to be well bounded by the selected conditions for the H* leakage calculations.

Precedent exists to credit operator action. The SG Tube Rupture event in the Updated Final Safety Analysis Report (UFSAR) permits operator action to mitigate the expected leakage. No operator action to reduce SG tube leakage is credited in the analyses of any accident scenario including fission product releases with SG boiloff. The analyses for all of these accident scenarios demonstrate that the radiological consequences are within the appropriate NRC acceptance criteria.

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

As a conservative basis for calculating the leakage for the Feed Line Break transient, the maximum Feed Line Break design basis transient pressure is used in the calculation of H* Feed Line Break leakage.

The leakage factor of 2.65 for Catawba Unit 2 for a postulated Steam Line Break/Feed Line Break has been calculated in WCAP-17072-P. The leakage factor has been increased to 3.27 per Attachment 16. Specifically, for the Condition Monitoring assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.27 (refer to the third regulatory commitment of Attachment 17) and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the Operational Assessment, the difference between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 3.27 (refer to the third regulatory commitment of Attachment 17) and compared to the observed operational leakage.

WCAP-17072-P redefines the primary pressure boundary. The tube-to-tubesheet weld no longer functions as a portion of this boundary. The hydraulic expansion 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 SG operating conditions, including the most limiting accident conditions. The evaluation in WCAP-17072-P determined that degradation in tubing below 13.8 inches from the top of the tubesheet does not require inspection or repair (plugging). The inspection of the portion of the tubes above 13.8 inches from the top of the tubesheet for tubes that have been hydraulically expanded in the tubesheet provides a high level of confidence that the structural and leakage performance criteria are maintained during normal operating and accident conditions.

WCAP-17072-P recommended a final value of H* of 13.8 inches below the top of the tubesheet for the entire bundle of tubes. However, Duke has chosen to use a more conservative value of 16.95 inches. This more conservative value was discussed between the NRC staff and industry representatives on April 24, 2009 and May 1, 2009.

WCAP-17072-P provides a review of leak rate susceptibility 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. However, in response to an NRC staff request, Duke has included monitoring for tube slippage as part of the SG tube inspection program. (Refer to the first regulatory commitment of Attachment 17.)

In addition, the NRC staff has requested that licensees determine if there are any significant deviations in the location of the bottom of the expansion transition (BET) relative to the top of the tubesheet that would invalidate assumptions in WCAP-17072-P. Therefore, Duke commits to perform a one-time verification of the tube expansion to locate any significant deviations in the distance from the top of the tubesheet to the BET. If any deviations are found, the condition will be entered into the corrective action program and dispositioned. Additionally, Duke commits to notify the NRC of significant deviations. (Refer to the second regulatory commitment of Attachment 17.)

5. REGULATORY EVALUATION

5.1 Applicable Regulatory Requirements/Criteria

SG tube inspection and repair limits are specified in Section 5.5.9, "Steam Generator (SG) Program" of the Catawba TS. The current TS require that flawed tubes be repaired if the depths of the flaws are greater than or equal to 40% through wall. The TS repair limits ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with GDC 14, 15, 30, 31, and 32 of 10 CFR 50, Appendix A. Specifically, the GDC state that the Reactor Coolant Pressure Boundary (RCPB) shall have "an extremely low probability of abnormal leakage ... and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing ... to assess ... structural and leaktight integrity" (GDC 32). Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the SG tubing. Leakage integrity refers to limiting primary to secondary leakage during all plant conditions to within acceptable limits.

The following NEI 97-06, Revision 2 performance criteria, which are included in the TS for Catawba Unit 2, are the basis for the WCAP-17072-P analysis. (Note: The actual performance criteria as stated in the Catawba Unit 2 TS are shown below.)

The structural integrity performance criterion is:

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

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

The accident induced leakage performance criterion is:

The primary to secondary accident induced leakage rate for any design basis accident, other than a 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 150 gallons per day through each SG for a total of 600 gallons per day through all SGs.

Primary to secondary leakage is a factor in the calculated dose due to releases outside containment resulting from a limiting design basis accident. The potential primary to secondary leak rate during postulated design basis accidents shall not result in exceeding the offsite radiological dose consequences as limited by 10 CFR 50.67 or the radiological consequences to control room personnel as limited by GDC 19.

The H* distance as documented in WCAP-17072-P for the tubesheet region has been developed to meet the above criteria. The structural criterion regarding tube burst is inherently satisfied because the constraint provided by the tubesheet to the tube prohibits burst.

The proposed change defines the portion of the 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 in WCAP-17072-P determined that degradation in tubing below 13.8 inches from the top of the tubesheet portion of the tube does not require plugging and serves as the bases for the SG tube inspection program. Duke has chosen to use an H* value of 16.95 inches for additional conservatism. As such, the Catawba Unit 2 inspection program provides a high level of confidence that the structural and leakage criteria are maintained during normal operating and accident conditions.

5.2 Precedent

This amendment request is similar to amendments that the NRC granted for other Westinghouse plants. The following three precedents are cited. These amendments incorporated on a one-cycle basis similar TS changes supported by the Westinghouse H* methodology.

1. Comanche Peak Steam Electric Station, Units 1 and 2 - Issuance of Amendments to Modify Technical Specifications to Establish Alternate Repair Criteria and Include Reporting Requirements Specific to Alternate

Repair Criteria for Steam Generator Program (TAC Nos. ME1446 and ME1447), October 9, 2009.

2. Seabrook Station, Unit No. 1 - Issuance of Amendment Re: Changes to the Steam Generator Inspection Scope and Repair Requirements (TAC No. ME1386), October 13, 2009.

3. Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2 - Issuance of Amendments Re: Revision to Technical Specifications for the Steam Generator Program (TAC Nos. ME1613, ME1614, ME1615, and ME1616), October 16, 2009.

5.3 No Significant Hazards Consideration

This amendment request proposes to revise TS 5.5.9 to exclude portions of the tubes within the tubesheet from periodic SG inspection and repair (plugging). In addition, this amendment request proposes to revise TS 5.6.8 to provide reporting requirements specific to the alternate repair criteria. Finally, this amendment request proposes to delete a Facility Operating License condition that was applicable to the previous operating cycle in favor of new NRC commitments that will be applicable to the next operating cycle. 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 to identify the specific inspection depth below which any type of degradation can be shown to have no impact on the NEI 97-06, Revision 2 performance criteria.

Duke has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by analyzing the three standards set forth in 10 CFR 50.92(c) as discussed below:

Criterion 1:

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

Response: No.

The proposed changes to TS 5.5.9, TS 5.6.8, and the Facility Operating License have no significant effect upon accident probabilities or consequences. Of the various accidents previously evaluated, the following are limiting with respect to the proposed changes as discussed in this amendment request:

- SG Tube Rupture evaluation
- Steam Line Break/Feed Line Break evaluation
- Locked Rotor evaluation
- Control Rod Ejection evaluation

Loss of Coolant Accident conditions cause a compressive axial load to act on the tube. Therefore, since this accident tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a Safe Shutdown Earthquake; however, the seismic analysis of Model D5 SGs (the SGs at Catawba) has shown that axial loading of the tubes is negligible during this event.

At normal operating pressures, leakage from Primary Water Stress Corrosion Cracking (PWSCC) below 16.95 inches from the top of the tubesheet is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

For the SG Tube Rupture event, tube rupture is precluded for cracks in the hydraulic expansion region due to the constraint provided by the tubesheet. Therefore, the margin against tube burst/pullout is maintained during normal and postulated accident conditions and the proposed change does not result in a significant increase in the probability of a tube rupture. SG Tube Rupture consequences are not affected by the primary to secondary leakage flow during the event, as primary to secondary leakage flow through a postulated tube that has been pulled out of the tubesheet is essentially equivalent to that from a severed tube. Therefore, the proposed change does not result in a significant increase in the consequences of a tube rupture.

The probability of a Steam Line Break/Feed Line Break, Locked Rotor, and Control Rod Ejection are not affected by the potential failure of a SG tube, as the failure of a tube is not an initiator for any of these events. In WCAP-17072-P, leakage is modeled as flow through a porous medium via the use of the Darcy equation. The leakage model is used to develop a relationship between operational leakage and leakage at accident conditions that is based on differential pressure across the tubesheet and the viscosity of the fluid. A leak rate ratio was developed to relate the leakage at operating conditions to leakage at accident conditions. The fluid viscosity is based on fluid temperature and it has been shown that for the most limiting accident, the fluid temperature does not exceed the normal operating temperature. Therefore, the viscosity ratio is assumed to be 1.0 and the leak rate ratio is a function of the ratio of the accident differential pressure and the normal operating differential pressure.

The leakage factor of 2.65 for Catawba Unit 2 for a postulated Steam Line Break/Feed Line Break has been calculated as shown in WCAP-17072-P, as

supplemented. The leakage factor has been increased to 3.27 per additional Westinghouse analysis specific to Catawba. Therefore, Catawba Unit 2 will apply a factor of 3.27 to the normal operating leakage associated with the tubesheet expansion region in the Condition Monitoring assessment and Operational Assessment. Through application of the limited tubesheet inspection scope, the proposed operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur. No leakage factor will be applied to the Locked Rotor or Control Rod Ejection due to their short duration, since the calculated leak rate ratio is less than 1.0. Therefore, the proposed change does not result in a significant increase in the consequences of these accidents.

For the Condition Monitoring assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.27 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the Operational Assessment, 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 3.27 and compared to the observed operational leakage.

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

Criterion 2:

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

Response: No.

The proposed changes to TS 5.5.9, TS 5.6.8, and the Facility Operating License do not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the one-cycle alternate repair criteria. The proposed change does not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

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

Criterion 3:

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

Response: No.

The proposed changes to TS 5.5.9, TS 5.6.8, and the Facility Operating License maintain the required structural margins of the SG tubes for both normal and accident conditions. NEI 97-06, Revision 2 and RG 1.121 are used as the basis in the development of the limited tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting GDC 14, 15, 31, and 32 by reducing the probability and consequences of a SG Tube Rupture. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation, the probability and consequences of a SG Tube Rupture are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, WCAP-17072-P 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 as described in WCAP-17072-P shows that significant margin exists between an acceptable level of leakage during normal operating conditions that ensures meeting the accident induced leakage assumption and the TS leakage limit.

Based on the above, it is concluded that the proposed change does not result in any reduction of margin with respect to plant safety as defined in the UFSAR or Bases of the plant TS.

Based on the above, Duke concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

5.4 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. ENVIRONMENTAL CONSIDERATION

Duke has determined that the proposed amendment does change requirements with respect to the installation or use of a facility component located within the restricted area, as defined by 10 CFR 20. It also represents a change to an inspection or surveillance requirement. Duke has evaluated the proposed amendment and has determined that it does not involve: (1) a significant hazards consideration, (2) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (3) a significant increase in individual or cumulative occupational radiation exposures. Accordingly, the proposed amendment meets 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.

ATTACHMENT 2

Marked-Up TS Pages

INSERT 1

For Unit 2 only, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, tubes with service-induced flaws located greater than 16.95 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 16.95 inches below the top of the tubesheet shall be plugged upon detection.

INSERT 2

For Unit 2, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, the number and portions of the tubes inspected and method of inspection shall be performed with the objective of detecting flaws of any type (for example, volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from 16.95 inches below the top of the tubesheet on the hot leg side to 16.95 inches below the top of the tubesheet on the cold leg side, and that may satisfy the applicable tube repair criteria.

INSERT 3

For Unit 2, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, if crack indications are found in any SG tube from 16.95 inches below the top of the tubesheet on the hot leg side to 16.95 inches below the top of the tubesheet on the cold leg side, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 EFPM or one refueling outage (whichever is less).

INSERT 4

In addition, if the calculated accident leakage rate from the most limiting accident is less than 3.27 times the maximum primary to secondary LEAKAGE rate, the report shall describe how it was determined, and

INSERT 5

j. For Unit 2, following completion of an inspection performed during the End of Cycle 17 Refueling Outage (and any inspections performed during subsequent Cycle 18 operation), the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 248, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than February 24, 2026, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

Duke Energy Carolinas, LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program (Section 9.5.1, SER, SSER #2, SSER #3, SSER #4, SSER #5)*

Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, as amended, for the facility and as approved in the SER through Supplement 5, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

*The parenthetical notation following the title of this renewed operating license condition denotes the section of the Safety Evaluation Report and/or its supplements wherein this renewed license condition is discussed.

(6) Mitigation Strategies

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(7) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 248 are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of Appendix J to 10 CFR Part 50, as delineated below and pursuant to evaluations contained in the referenced SER and SSERs. These include, (a) partial exemption from the requirement of paragraph III.D.2(b)(ii) of Appendix J, the testing of containment airlocks at times when the containment integrity is not required (Section 6.2.6 of the SER, and SSERs # 3 and #4). (b) exemption from the requirement of paragraph III.A.(d) of Appendix J, insofar as it requires the venting and draining of lines for type A tests (Section 6.2.6 of SSER #3), and (c) partial exemption from the requirements of paragraph III.B of Appendix J, as it relates to bellows testing (Section 6.2.6 of the SER and SSER #3). These exemptions are authorized by law, will not present an undue risk to the public health and safety, are consistent

Amendment Number	Additional Condition		Implementation Date
165	The schedule for the performance of new and revised surveillance requirements shall be as follows: For surveillance requirements (SRs) that are new in Amendment No. 165 the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment No. 165. For SRs that existing prior to Amendment No. 165, including SRs with modified acceptance criteria and SRs who intervals of performance are being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of amendment No. 165. For SRs that existed prior to Amendment No. 165, whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of Amendment No. 165		By January 31, 1999
172	The maximum rod average burnup for any rod shall be limited to 60 GWd/mtU until the completion of an NRC environmental assessment supporting an increased limit.		Within 30 days of date of amendment.
244 Deleted	For steam generator (SG) integrity assessments, the ratio of 2.5 will be used in completion of both the Condition Monitoring (CM) and the Operational Assessment (OA) upon implementation of the Interim Alternate Repair Criterion (IARC). For example, for the CM assessment, the component of leakage from the lower 4 inches of the most limiting SG during the prior cycle of operation will be multiplied by a factor of 2.5 and added to the total leakage from any other source and compared to the allowable accident analysis leakage assumption. For the OA, the difference in leakage from the allowable limit during the limiting design basis accident minus the leakage from the other sources will be divided by 2.5 and compared to the observed leakage. An administrative limit will be established to not exceed the calculated value.		Prior to any entry into Mode 4 during Cycle 17 operation

NO CHANGES THIS PAGE.
FOR INFORMATION ONLY

5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the

(continued)

5.5.9 Steam Generator (SG) Program (continued)

condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

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

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 150 gallons per day through each SG for a total of 600 gallons per day through all SGs.

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.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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

1. For the Unit 2 End of Cycle 16 Refueling Outage and subsequent Cycle 17 operation only, tubes with flaws having a circumferential component less than or equal to 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet do not require plugging. Tubes with flaws having a circumferential component greater than 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet shall be removed from service.

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

   INSERT 1

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

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

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

For Unit 1, the

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and method of inspection shall be performed with the objective of detecting flaws of any type (for example, volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting requirements d.1, d.2, d.3, and d.4 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.

INSERT 2

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

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

3. For Unit 2, inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 EFPM. 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 EFPM or two refueling outages (whichever is less) without being inspected.

For Unit 1, if

4. 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 EFPM or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with crack(s), then the indication need

INSERT 3

(continued)

5.5.9 Steam Generator (SG) Program (continued)

not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with exceptions as noted in the UFSAR.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows the following penetration and system bypass when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the flowrate specified below ± 10%.

(continued)

5.6 Reporting Requirements (continued)

5.6.7 PAM Report

When a report is required by LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.8 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of the inspection. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Non-destructive 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 2, following completion of an inspection performed during the End of Cycle 16 Refueling Outage (and any inspections performed during subsequent Cycle 17 operation), the number of indications and location, size, orientation, whether initiated on the primary or secondary side for each service-induced flaw within the thickness of the tubesheet, and the total of the circumferential components and any circumferential overlap below 17 inches from the top of the tubesheet as determined in accordance with TS 5.5.9c.1,

(continued)

5.6 Reporting Requirements

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

h. For Unit 2, following completion of an inspection performed during the End of Cycle 16 Refueling Outage (and any inspections performed during subsequent Cycle 17 operation), the primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign leakage to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report, and

i. For Unit 2, following completion of an inspection performed during the End of Cycle 16 Refueling Outage (and any inspections performed during subsequent Cycle 17 operation), the calculated accident leakage rate from the portion of the tubes below 17 inches from the top of the tubesheet for the most limiting accident in the most limiting SG.

INSERT 5

INSERT 4

ATTACHMENT 3

Westinghouse Authorization Letter CAW-09-2585 with Accompanying Affidavit, Proprietary Information Notice, and Copyright Notice (WCAP-17072-P)

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

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

Direct tel: (412) 374-4643
Direct fax: (412) 374-4011
e-mail: greshaja@westinghouse.com

Our ref: CAW-09-2585

May 21, 2009

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-17072-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)," dated May 2009 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-09-2585 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 Duke Energy.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-09-2585, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

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

Enclosures

cc: G. Bacuta (NRC OWFN 12E-1)

bcc: J. A. Gresham (ECE 4-7A) 1L
R. Bastien, 1L (Nivelles, Belgium)
C. Brinkman, 1L (Westinghouse Electric Co., 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) 1L (letter and affidavit only)
G. W. Whiteman, Waltz Mill
H. O. Lagally, Waltz Mill
C. D. Cassino, Waltz Mill
J.T. Kandra, Waltz Mill
C. Hammer, Waltz Mill
D.C. Beddingfield, ECE 558 B

CAW-09-2585

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

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

Sworn to and subscribed before me
this 21st day of May, 2009

Sharon L Markle
Notary Public

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-17072-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)," dated May 2009 (Proprietary), for submittal to the Commission, being transmitted by Duke Energy Application for Withholding Proprietary Information from Public Disclosure to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Catawba Unit 2 is expected to be applicable to other licensee submittals in support of implementing an alternate repair criterion, called H*, that does not require an eddy current inspection and plugging of the tubes below a distance of 13.8 inches from the top of the tubesheet.

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

(a) Provide documentation of the analyses, methods, and testing which support the implementation of an alternate repair criterion, designated as H*, for a portion of the tubes within the tubesheet of the Catawba Unit 2 steam generators.

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

Further this information has substantial commercial value as follows:

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

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

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

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

Duke Energy
Letter for Transmittal to the NRC

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

Enclosed are:

1. 1 copy of WCAP-17072-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)," dated May 2009 (Proprietary)

2. 1 copy of WCAP-17072-NP, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)," dated May 2009 (Non-Proprietary).

Also enclosed is Westinghouse authorization letter CAW-09-2585 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

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

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

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-09-2585 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.