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November 13, 2008

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"

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 a one-cycle revision to the subject TS to incorporate an Interim Alternate Repair Criterion (IARC) in the provisions for SG tube repair criteria during the End of Cycle 16 Refueling Outage and subsequent Cycle 17 operation. This change is supported by Westinghouse Electric Company LLC, LTR-CDME-08-11, Rev. 3, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone", and LTR-CDME-08-43, Rev. 3, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11, Rev. 3, P-Attachment".

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.

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Subsequent to the issuance of Amendment 233, another utility submitted a permanent TS 5.5.9 amendment request to support its Spring 2008 refueling outage. During their review of this request, the NRC indicated that they would not be able to approve a permanent revision to TS 5.5.9 in time to support plants with Spring 2008 refueling outages. The NRC indicated that they would entertain one-cycle amendment requests (IARC) that differed from the previously approved one-cycle amendments. IARC amendments were approved for several plants to support Spring 2008 and Fall 2008 refueling outages.

The contents of this amendment request package are as follows:

Attachment 1 provides the technical and regulatory evaluations of the proposed changes. Attachment 2 contains a marked-up version of the affected TS pages. Reprinted (clean) TS pages will be provided to the NRC prior to issuance of the approved amendment. This amendment request contains NRC commitments as discussed in Attachment 3. Duke requests NRC approval of these proposed changes by February 28, 2009.

Duke is requesting a 30-day implementation period in conjunction with this amendment. Implementation of the approved amendment will not require changes to the Catawba Updated Final Safety Analysis Report (UFSAR).

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, this proposed amendment has been reviewed and approved by the Catawba Plant Operations Review Committee and by the Corporate Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91, a copy of this proposed amendment is being sent to the designated official of the State of South Carolina.

Enclosure 1 provides the proprietary Westinghouse Electric Company LLC LTR-CDME-08-11, Rev. 3, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone", and LTR-CDME-08-43, Rev. 3, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11, Rev. 3, P-Attachment". Enclosure 2 provides the non-proprietary versions of these documents. As Enclosure 1 contains information proprietary to Westinghouse Electric Company LLC, it is supported by affidavits signed by Westinghouse Electric Company LLC, the owner of the information. The affidavits set forth the basis on which the information may be withheld from public disclosure by the Commission and address with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with Section 2.390 of the Commission's regulations. These affidavits, along with Westinghouse authorization letters CAW-08-2437 and CAW-08-2438, "Application for Withholding Proprietary Information from Public Disclosure", are contained in Enclosure 3.

Inquiries on this matter should be directed to L.J. Rudy at (803) 701-3084.

Very truly yours,

James R. Morris

LJR/s

Attachments and Enclosures

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

James R. Morris, Vice President

Subscribed and sworn to me: 11/13/08
Date

Anthony P. Jackson

Notary Public

My commission expires: 7/2/2014
Date

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Document Control File 801.01
RGC File
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ATTACHMENT 1

TECHNICAL AND REGULATORY EVALUATIONS

Subject: Application for License Amendment for Interim Alternate Repair Criterion

1. SUMMARY DESCRIPTION

2. DETAILED DESCRIPTION

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1. SUMMARY DESCRIPTION

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

This amendment application proposes a one-cycle revision to TS 5.5.9, "Steam Generator (SG) Program" and TS 5.6.8, "Steam Generator (SG) Tube Inspection Report" to incorporate an IARC in the provisions for SG tube repair criteria during the End of Cycle 16 Refueling Outage and subsequent Cycle 17 operation. This amendment application requests approval of an IARC that requires full-length inspection of the tubes within the tubesheet but does not require plugging tubes if any circumferential cracking observed in the region greater than 17 inches from the top of the tubesheet (TTS) is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads. In addition, axial cracks in tubes in this region would be allowed to remain in service. This amendment application is required to preclude unnecessary plugging while still maintaining structural and leakage integrity.

Approval of this amendment application is requested to support the End of Cycle 16 Refueling Outage (Spring 2009) and the subsequent eddy current inspection interval, as the existing one-cycle amendment expires at the end of the current operating cycle.

2. DETAILED DESCRIPTION

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 (Reference 1). 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 (Reference 2). This amendment involved a second one-cycle change for the End of Cycle 15 Refueling Outage and subsequent Cycle 16 operation.

The industry has been actively working toward a permanent solution to this issue using the Westinghouse H*/B* methodology. The premise of the H*/B* methodology is that the expansion joint provides sufficient structural restraint to prevent the tube from pulling out of the tubesheet under normal operating and accident conditions, and that the accident induced leakage during accident conditions is bounded by twice the observed normal operating leakage. Another utility submitted a permanent amendment request using the H*/B* methodology to support its Spring 2008 refueling outage.

During their review of this permanent amendment request, the NRC indicated that they would not be able to approve the proposed permanent revision to TS 5.5.9 (using the H*/B* methodology) in time to support plants with Spring 2008 refueling outages. The NRC indicated that they would entertain one-cycle amendment requests (IARC) that differed from the previously approved one-cycle amendments. IARC amendments were approved for several plants to support Spring 2008 and Fall 2008 refueling outages.

Enclosure 1 provides the technical justification for an IARC that requires full-length inspection of the tubes within the tubesheet, but does not require plugging tubes if the extent of any circumferential cracking observed in the region greater than 17 inches from the TTS is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads (the greater of 3 times the normal operating loads or 1.4 times the steam line break end cap loads). Axial cracks below 17 inches from the TTS are not relevant to the tube pullout arguments because axial cracks do not degrade the axial load carrying capability of the tube. Axial cracks do not require plugging if they are below 17 inches from the TTS.

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

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

Although the tube-to-tubesheet weld is not part of the tube, as indicated in TS 5.5.9d, it is, nevertheless, also necessary to consider the capability of a degraded weld to prevent tube pullout for this IARC. Because of the underlying assumption of zero friction load between the tubes and the tubesheet, the weld must provide the IARC's ultimate structural restraint of the tube within the tubesheet. Therefore, a limiting ligament size has also been determined for the tube-to-tubesheet weld and is discussed below.

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

1. *For the Unit 2 End of Cycle 15 Refueling Outage and Cycle 16 operation only, the 40% depth based criterion does not apply to degradation identified in the portion of the tube below 17 inches from the top of the tubesheet. If degradation is identified in the portion of the tube from the top of the tubesheet to 17 inches below the top of the tubesheet, the tube shall be removed from service. If degradation is found in the portion of the tube below 17 inches from the top of the tubesheet, the tube does not require plugging.*

TS 5.5.9c is being revised as follows (revisions 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 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 5.6.8 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, and*

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

TS 5.6.8 is being revised to add the following three reporting criteria for Catawba Unit 2 (additions are in **bold type**):

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 the Facility Operating License are as follows:

The Facility Operating License condition associated with Unit 2 Amendment 233 currently states:

Additional Condition: This amendment requires the licensee to use administrative controls, as described in the licensee's letter of April 30, 2007, and evaluated in the Staff's Safety Evaluation dated October 31, 2007, to restrict the primary to secondary leakage through any one steam generator to 75 gallons per day and through all steam generators to 300 gallons per day (in lieu of the limits in TS Sections 3.4.13d. and 5.5.9b.3.), for Cycle 16 operation.

Implementation Date: Prior to any entry into Mode 4 during Cycle 16 operation

This Facility Operating License condition is being revised as follows (revisions are in **bold type**):

*Additional Condition: This amendment requires the licensee to use administrative controls, as described in the licensee's letter of **November 13, 2008**, and evaluated in the Staff's Safety Evaluation dated **[DATE]**, to restrict the primary to secondary leakage through any one steam generator to **60** gallons per day and through all steam generators to **240** gallons per day (in lieu of the limits in TS Sections 3.4.13d. and 5.5.9b.3.), for Cycle **17** operation.*

*Implementation Date: Prior to any entry into Mode 4 during Cycle **17** operation*

3. TECHNICAL EVALUATION

The enclosed evaluation has been performed by Westinghouse to assess the need for removing tubes from service due to the occurrence of circumferentially or axially oriented cracks within the tubesheet. The primary conclusions of the evaluation are:

1. Axial cracks in tubes below a distance of 17 inches below the TTS are allowed to remain in service in the Catawba Unit 2 SGs as they are not a concern relative to tube pullout and leakage capability.

2. Circumferentially oriented cracks in tubes below a distance of 17 inches below the TTS with an azimuthal extent of less than or equal to 203 degrees are allowed to remain in service for one cycle of operation.

3. Circumferentially oriented cracks in the bottom 1 inch of the tube or in the TTS weld with an azimuthal extent of less than or equal to 94 degrees are allowed to remain in service for one cycle of operation.

A bounding analysis approach is utilized in the enclosed Westinghouse evaluation for both the minimum ligament calculation and the leakage ratio calculation. "Bounding" means that the most challenging conditions from the plants with hydraulically expanded Alloy 600 thermally treated tubing are used. Three different tube diameters are represented by the affected plants (11/16 inch diameter Model F, 3/4 inch diameter Model D5, and 7/8 inch diameter Model 44F). Catawba Unit 2 has Model D5 SGs. The most limiting conditions for structural evaluation depend on tube geometry and applied normal operating loads; thus, the conditions from the plant that result in the highest stress in the tube are used to define the minimum required circumferential ligament. The limiting leak rate ratio depends on the leak rate values assumed in the safety analysis and allowable normal operating leakage that result in the longest length of undegraded tube.

<u>Questions Related to IARC for SG Tubes</u>

This amendment request is based upon the precedent amendments approved by the NRC as indicated in Section 4.2. The responses to the NRC Requests for Additional Information associated with these precedent amendments have been incorporated into this amendment request.

Discussion of Performance Criteria

The following NEI (Nuclear Energy Institute) 97-06, Rev. 2, "Steam Generator Program Guidelines" performance criteria, which are included in the TS for Catawba Unit 2, are the basis for the enclosed Westinghouse 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 General Design Criterion (GDC) 19.

The IARC for the tubesheet region have 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.

Limiting Structural Ligament Discussion

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

Consideration of Non-Destructive Examination (NDE) Uncertainty

The NDE uncertainty must be addressed to assure the as-indicated circumferential arc of the reported crack is a reliable estimate of the actual crack. ETSS 20510.1, "Technique for Detection of Circumferential PWSCC at Expansion Transitions", describes the qualified technique used to detect circumferential Primary Water Stress Corrosion Cracking (PWSCC) in the expansion transitions and in the tubesheet expansion zone. The qualification data is provided in the ETSS.

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

EPRI (Electric Power Research Institute) TR-107197, "Depth Based Structural Analysis Methods for Steam Generator Circumferential Indications", has correlated the axial strength of the tube to the Percent Degraded Area (PDA) of the flaw. PDA takes into account the profile of the existing crack, including non-through wall portions and shallow tails of the crack. Using the data from ETSS 20510.1 for cracks with a 90% or greater through wall condition from both NDE and destructive examination, a comparison of the actual crack lengths and corresponding PDA for the cracks to a theoretical PDA which assumes that cracks are 100% through wall has been made. All points with a PDA of 60% or greater fall below the theoretical PDA line. As the crack lengths increase, the separation of the actual PDA from the theoretical PDA tends to increase.

The conclusion that the as-indicated crack angle is conservative is further supported by considering the characteristics of the eddy current probes. Each probe has a "field of view", that is, a window of finite dimension in which it detects flaws. The field of view for the +Point probe typically varies between 0.1 inch to 0.2 inch depending on the specific characteristics of the probe. Therefore, as the probe traverses its path, a flaw will be detected as the leading edge of the field of view first crosses the location of the flaw, continuing until the trailing edge of the field of view passes the opposite end of the flaw. This is known as "lead-in" and "lead-out" of the probe and the effect of these are to render the indicated flaw length greater than the actual flaw length. Therefore, it is concluded that the indicated flaw length will be conservative relative to the actual flaw length, especially when it is assumed that the entire length of the indicated flaw is 100% through wall.

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

Consideration of Crack Growth

According to the enclosed Westinghouse analysis, the growth of cracks due to PWSCC is dictated by four default growth rates. The distribution of growth rates is assumed to be lognormal. Typical values and conservative values are given. However, EPRI 1012987, "Steam Generator Integrity Assessment Guidelines", recommends using the default values only when the historical information is not available and not using the typical values unless the degradation is mild. (No significant crack growth data exists for the circumferential cracking in the tubesheet expansion region.) Both sets provided in the Westinghouse analysis have mean values and 95% upper bound values. For this analysis, the typical 95% upper bound growth rate is used. The circumferential growth rates are expressed as inches per Effective Full Power Year (EFPY).

Calculation of Required Minimum Ligament for 18-Month Operating Period

	Bounding Structural Ligament	EFPY (1)	Growth (in/EFPY) (2)	Growth (deg/EFPY) (3)	Growth for Operating Period (deg)	Minimum Structural Ligament (deg)	Critical Ligament (deg)
Tube	18 Calendar Month (CM) Operation	1.5	0.12	20.65	31	126	157

(1) It is conservatively assumed that 1 EFPY = 1 Calendar Year.
(2) 95% upper value of typical growth rates from Westinghouse analysis.
(3) Based on smallest (Model F) mean tubesheet bore dimension.

The residual structural ligament must be adjusted for growth during the anticipated operating period between the current and the next planned inspection. For the Catawba Unit 2 SGs, referring to the above table, the maximum allowable through wall circumferential crack size in a SG tube is 203 degrees (360 degrees - 157 degrees) for one cycle of operation (nominal 18-month SG tube eddy current inspection interval).

Primary to Secondary Leakage Discussion

A basis using the D'Arcy formula for flow through a porous medium is provided to assure the accident induced leakage for the limiting accident will not exceed the value assumed in the safety analysis for the plant if the observed leakage during normal operation is within its limits for the bounding plant as discussed in the enclosed Westinghouse analysis. The bounding plant envelopes all plants with recirculating SGs with Inconel 600 thermally treated tubes. The D'Arcy formulation was previously compared to other potential models such as the Bernoulli equation or orifice flow formulation and was found to provide the most conservative results. Assuming zero contact pressure in the tube joint, the length of undegraded crevice required to limit the accident induced leakage to less than the value assumed in the safety analysis for the limiting plant is calculated to be 3.78 inches. By definition of the IARC, a tube that can remain in service has an undegraded crevice of 17 inches. Therefore, a safety factor of 4.5 is available (17 inches/3.78 inches). Expressed in terms of length, the margin in the crevice is 13.22 inches. Significant margin on crevice length is available even if only the distance below the neutral axis of the tubesheet is considered. This distance is approximately 6.5 inches. A factor of safety of 1.72 is available. Expressed in terms of length, the margin in the crevice is 2.72 inches below the neutral axis of the tubesheet. During normal operating conditions, the tubesheet flexes due to differential pressure loads, causing the tubesheet holes above the neutral axis to dilate, and below the neutral axis to constrict. No mechanical benefit is assumed in the analysis due to tubesheet bore constriction below the neutral axis of the tubesheet; however, first principles dictate the tubesheet bore and crevice must decrease. Therefore, the leakage analysis provided is conservative.

Based on the above, with a length of undegraded crevice of 17 inches, it is concluded that if the normal operating leakage is within its allowable value, the accident induced leakage will also be within the value assumed in the Catawba Unit 2 safety analysis. The total increase in leakage during a postulated accident condition would be less than a factor of 2.5 (600 gpd allowable leakage through all SGs during a Steam Line Break event/240 gpd allowable leakage through all SGs during normal operating conditions).

For 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 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 multipled by a factor of 2.5 and added to the total leakage from any other source and compared to the allowable accident analysis leakage assumption. For the OA, the difference in leakage from the allowable limit during the limiting design basis accident minus the leakage from the other sources will be divided by 2.5 and compared to the observed leakage. An administrative limit will be established to not exceed the calculated value.

Reporting Requirements

For Catawba Unit 2, Duke proposes to report the following additional information associated with the IARC following the End of Cycle 16 Refueling Outage inspections and any additional inspections 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 TTS

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

- The calculated accident leakage rate from the portion of the tubes below 17 inches from the TTS for the most limiting accident in the most limiting SG (as indicated above, a factor of 2.5 shall be used to relate the accident leakage to the related operational leakage)

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

Inspection and Repair of Tube

The tube below the IARC depth will be examined with a qualified technique (e.g., +Point probe). Axial flaws have no impact on the structural integrity of the tube in this region and may be left in service. Circumferential indications that exceed the maximum acceptable tube flaw size of 203 degrees will be plugged. The detection of flaws will result in sample expansion per EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines". Stress concentration areas may be used to define the extent of the expansion (e.g., if a repairable indication is located in a bulge/overexpansion, the expansion may be limited to the non-inspected bulges/overexpansions). The circumferential components of multiple flaws within 1 inch of each other axially will be combined in accordance with TS 5.5.9c.1. Furthermore, the circumferential component of flaws within the bottom 1 inch of the SG tubes and within 1 inch axial separation distance of a flaw above 1 inch from the bottom of the SG tubes is limited to 94 degrees.

4. REGULATORY EVALUATION

4.1 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 General Design Criteria (GDC) 14, 15, 30, 31, and 32 of 10 CFR 50, Appendix A. Specifically, the GDC state that the Reactor Coolant Pressure Boundary (RCPB) shall have "an extremely low probability of abnormal leakage ... and gross rupture" (GDC 14), "shall be designed with sufficient margin" (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.

4.2 Precedent

This amendment request is similar to amendments that the NRC granted for Wolf Creek Generating Station on April 4, 2008 (Reference 3), Vogtle Electric Generating Plant on April 9, 2008 (Reference 4), and Braidwood Station on April 18, 2008 (Reference 5). These amendments also allowed the use of an IARC on a one-cycle basis.

4.3 Significant Hazards Consideration

Duke has evaluated whether or not a significant hazard 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.

Of the various accidents previously evaluated, the following are limiting with respect to the proposed changes to TS 5.5.9, TS 5.6.8, and the Facility Operating License:

- SG Tube Rupture (SGTR) evaluation
- Steam Line Break (SLB) evaluation
- Locked Rotor Accident (LRA) evaluation
- Rod Ejection Accident (REA) evaluation

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

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

For the SGTR event, the required structural margin of the SG tubes is maintained by limiting the allowable ligament size for a circumferential crack to remain in service to 203 degrees below 17 inches from the TTS and above 1 inch from the bottom of the tubesheet. Tube rupture is precluded for cracks in the hydraulic expansion region due to the constraint provided by the tubesheet. The potential for tube pullout is mitigated by limiting the allowable crack size to 203 degrees. This allowable crack size takes into account eddy current uncertainty and crack growth rate. It has been shown that a circumferential crack with an azimuthal extent of 203 degrees meets the performance criteria of NEI (Nuclear Energy Institute) 97-06, Rev. 2, "Steam Generator Program Guidelines" and NRC draft

Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes". Therefore, the margin against tube burst/pullout is maintained during normal and postulated accident conditions and the proposed change does not result in a significant increase in the probability or consequence of a SGTR.

The probability of a SLB, LRA, and REA 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. SLB leakage is limited by leakage flow restrictions resulting from the leakage path above potential cracks through the TTS crevice. The leak rate during postulated accident conditions has been shown to remain within the accident analysis assumptions for all axially or circumferentially oriented cracks occurring 17 inches below the TTS. Since normal operating leakage is limited to 60 gpd through any one SG and 240 gpd through all SGs, the attendant accident condition leak rate, assuming all leakage to be from indications below 17 inches from the TTS, would be bounded by 150 gpd through any one SG and 600 gpd through all SGs. This value is within the accident analysis assumptions for these design basis accidents for Catawba Unit 2.

Based on the above, the performance criteria of NEI 97-06, Rev. 2 and draft 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 IARC. 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, Rev. 2 and draft RG 1.121 are used as the basis in the development of a methodology for determining that SG tube integrity considerations are maintained within acceptable limits. Draft 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 SGTR. Draft RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking in a tube or the TTS weld, the supporting Westinghouse analysis defines a length of remaining tube ligament that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factors applied).

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.

4.4 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.

5. 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.

6. REFERENCES

1. Letter from John Stang, NRC to Mr. Dhiaa Jamil, Duke Energy Corporation, "Catawba Nuclear Station, Unit 2, Issuance of Amendments Regarding the Steam Generator Program (TAC No. MC9430)", dated March 31, 2006.

2. Letter from John Stang, NRC to Mr. J.R. Morris, Duke Power Company LLC, "Catawba Nuclear Station, Unit 2, Issuance of Amendment Regarding Technical Specification 5.5.9 "Steam Generator Tube Surveillance Program" (TAC No. MD5554)", dated October 31, 2007.

3. Letter from Jack N. Donohew, NRC to Mr. Rick A. Muench, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station - Issuance of Amendment Re: Revision to Technical Specification 5.5.9 on the Steam Generator Program (TAC No. MD8054)", dated April 4, 2008.

4. Letter from Siva P. Lingam, NRC to Mr. Tom E. Tynan, Vogtle Electric Generating Plant, "Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding Changes to Technical Specification (TS) Sections TS 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" (TAC Nos. MD7450 and MD7451)", dated April 9, 2008.

5. Letter from Marshall J. David, NRC to Mr. Charles G. Pardee, Exelon Generation Company, LLC, "Braidwood Station, Units 1 and 2 - Issuance of Amendments Re: Revision to Technical Specifications for the Steam Generator Program (TAC Nos. MD8158 and MD8159)", dated April 18, 2008.

ATTACHMENT 2

MARKED-UP TS PAGES

INSERT 1

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.

INSERT 2

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.

INSERT 3

Additional Condition: This amendment requires the licensee to use administrative controls, as described in the licensee's letter of November 13, 2008, and evaluated in the Staff's Safety Evaluation dated [DATE], to restrict the primary to secondary leakage through any one steam generator to 60 gallons per day and through all steam generators to 240 gallons per day (in lieu of the limits in TS Sections 3.4.13d. and 5.5.9b.3.), for Cycle 17 operation.

Implementation Date: Prior to any entry into Mode 4 during Cycle 17 operation

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

~~1. For the Unit 2 End of Cycle 15 Refueling Outage and Cycle 16 operation only, the 40% depth based criterion does not apply to degradation identified in the portion of the tube below 17 inches from the top of the tubesheet. If degradation is identified in the portion of the tube from the top of the tubesheet to 17 inches below the top of the tubesheet, the tube shall be removed from service. If degradation is found in the portion of the tube below 17 inches from the top of the tubesheet, the tube does not require plugging.~~

REPLACE WITH INSERT 1

(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, ~~and~~

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

INSERT 2

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 238 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. 233 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.
~~233~~	~~This amendment requires the licensee to use administrative controls, as described in the licensee's letter of April 30, 2007, and evaluated in the Staff's Safety Evaluation dated October 31, 2007, to restrict the primary to secondary leakage through any one steam generator to 75 gallons per day and through all steam generators to 300 gallons per day (in lieu of the limits in TS Sections 3.4.13d. and 5.5.9b.3.), for Cycle 16 operation.~~		~~Prior to any entry into Mode 4 during Cycle 16 operation~~

REPLACE WITH INSERT 3 ADDITIONAL CONDITION

REPLACE WITH INSERT 3 IMPLEMENTATION DATE

ATTACHMENT 3

NRC COMMITMENTS

The following NRC commitments are being made in support of this amendment request:

1. The approved amendment will be implemented within 30 days from the date of NRC approval. "Implemented" means that the approved amendment will have been placed into the control room copies of the TS. However, the provisions afforded by the approved amendment will not actually be utilized until such time as they are needed.

2. Prior to actually utilizing the provisions afforded by the approved amendment, Catawba will have in place all required document and process changes necessary to support these provisions.