UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 13, 2009

Mr. J. R. Morris
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNIT 2, ISSUANCE OF AMENDMENT REGARDING INTERIM ALTERNATE REPAIR CRITERION FOR STEAM GENERATOR TUBE REPAIR (TAC NO. ME0236)

Dear Mr. Morris:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 244 to Renewed Facility Operating License NPF-52 for the Catawba Nuclear Station, Unit 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 13, 2008, as supplemented by letters dated February 5, 2009, and February 19, 2009.

The amendment adds a one-cycle revision to the TSs to incorporate an interim alternate repair criterion for steam generator tube repair criteria during the end of cycle 16 refueling outage and subsequent operating cycle 17.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

If you have any questions, please call me at 301-415-1345.

Sincerely,

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-414

Enclosures:
1. Amendment No. 244 to NPF-52
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CAROLINAS, LLC

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 244
Renewed License No. NPF-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. NPF-52 filed by the Duke Energy Carolinas, LLC, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated November 13, 2008, as supplemented by letters dated February 5, 2009, and February 19, 2009, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (I) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 244 , which are attached hereto, are hereby incorporated into this license. Duke Energy Carolinas, LLC, shall operate the facility in accordance with the Technical Specifications.

3. The license is also amended to remove the third license condition on page 2 of Appendix B of Renewed Facility Operating License No. NPF-52 and add a new condition as indicated in the attachment to this license amendment. The new condition reads as follows;

   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.

4. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance. However, the new license condition on page 2 of Appendix B shall be implemented prior to any entry into Mode 4 during Cycle 17 operation.

FOR THE NUCLEAR REGULATORY COMMISSION

Melanie

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-52
and the Technical Specifications

Date of Issuance: April 13, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 244

RENEWED FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Renewed Facility Operating License and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License Pages	License Pages
4	4
5	5
Appendix B, page 2	Appendix B, page2
TSs	TSs
5.5-7a	5.5-7a
5.6-5	5.6-5
-	5.6-6

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 244 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) <u>Mitigation Strategies</u>

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) <u>Additional Conditions</u>

The Additional Conditions contained in Appendix B, as revised through Amendment No. 244 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	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

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.

   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)

5.6 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)

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 244 TO RENEWED FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNIT 2

DOCKET NO. 50-414

1.0 INTRODUCTION

By application dated November 13, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML083260010), as supplemented by letters dated February 5, 2009 (ADAMS Accession No. ML ML090430135), and February19, 2009 (ADAMS Accession No. ML ML090540252), Duke Energy Carolinas, LLC (Duke, the licensee), requested changes to the Technical Specifications (TSs) for the Catawba Nuclear Station, Unit 2 (Catawba 2). The supplements dated February 5, 2009 and February 19, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 24, 2009 (74 FR 8278).

The proposed amendment adds a one-cycle revision to the TSs to incorporate an interim alternate repair criterion (IARC) in the provisions for steam generator (SG) tube repair criteria during the end of cycle 16 refueling outage and subsequent operating cycle 17.

2.0 BACKGROUND

Catawba 2 has four Westinghouse Model D5 SGs. There are 4570 thermally-treated Alloy 600 tubes in each SG, each with an outside diameter of 0.750 inches and a nominal wall thickness of 0.043 inches. The tubes are hydraulically expanded for the full depth of the tubesheet at each end and are welded to the tubesheet at the bottom of each expansion.

Until the fall of 2004, no instances of stress corrosion cracking (SCC) affecting the tubesheet region of thermally treated Alloy 600 tubing had been reported at any nuclear power plants in the United States. As a result, most plants, including Catawba 2, had been using bobbin probes for inspecting the length of tubing within the tubesheet. Since bobbin probes are not capable of reliably detecting SCC in the tubesheet region, supplementary rotating coil probe inspections were used in a region extending from 3 inches above the top of the tubesheet (TTS) to 3 inches below the TTS. This zone includes the tube-expansion transition, which contains significant residual stress, and was considered a likely location for SCC to develop.

During the fall 2004 refueling outage, three crack-like indications were found in a tube in the tubesheet region of CNS Unit 2. These crack-like indications were found in an overexpansion (OXP) in the tubesheet region. An OXP is created when the tube is expanded into a region of the tubesheet that is not perfectly round. This out-of-round condition results from anomalies in the tubesheet drilling process (e.g., drill bit wandering). At the time the cracking was found, Catawba 2 had accumulated 14.7 effective full power years (EFPY) of service.

Based on the findings, the licensee expanded the scope of rotating coil inspections to include 100 percent of the OXPs in the hot-leg of all SGs. The licensee reported that they found no additional degradation in overexpansions; however, 196 indications were found in tube to tubesheet welds at tube-ends, and nine indications were found in tack expansion regions. The tack expansion is an approximately 1-inch long expansion at each tube end. The purpose of the tack expansion is to facilitate performing the tube-to-tubesheet weld, which is made prior to the hydraulic expansion of the tube over the full tubesheet depth.

As a result of these findings, the licensee further expanded the scope of rotating coil inspections to include 100 percent of the tubes in SG B and 20 percent of the tubes in SGs A, C, and D from the tube-end through the tack expansion region (approximately 2 to 3 inches from the tube-end).

During the spring 2006 refueling outage, Catawba 2 performed an array probe inspection of 100 percent of historical OXPs within the tubesheet and 20 percent of newly identified OXPs within the tubesheet (a new criterion was developed to more consistently identify overexpansions within the tubesheet, which lead to the identification of new OXPs). These inspections were performed from 2 inches above the TTS through the tube-end. Additionally, a 20-percent random sample and a 100-percent sample of the periphery tubes were inspected with the array probe, from 2 inches above the TTS through the tube-end.

The licensee believes that flaws located more than 17 inches below the TTS (i.e., in the bottom 4 inches of the tubesheet region, including the tack expansion region and the tubing in the vicinity of the welds) have no potential to impair tube integrity, and thus, do not pose a safety concern. To avoid unnecessary plugging or repair of SG tubes, the licensee requested, and the NRC staff approved, TS change License Amendment No. 224, dated March 31, 2006 (ADAMS Accession No. ML060760111) for Catawba 2 that excluded degradation in the lowermost 4 inches of the tubesheet from application of the 40-percent depth-based tube repair criterion during operating cycle 15.

During refueling outage 15 in the fall of 2007, the licensee performed a 20-percent sample of the tubesheet region in all four SGs from 3 inches above the TTS through the tube-end. This 20-percent sample was expanded to 100 percent for TTS indications in SG B. An inspection of 100 percent of tubesheet OXP's and bulges in the steam generator B hot-leg (above the ARC elevation) and an inspection of 20 percent of tubesheet OXP's and bulges in the remaining SGs (above ARC elevation) was also performed. To avoid unnecessary plugging or repair of tubes, the licensee requested, and the NRC staff approved, a TS change License Amendment No. 233, dated October 31, 2007 (ADAMS Accession No. ML072820018) for Catawba 2 which excluded degradation in the lowermost 4 inches of the tubesheet from application of the 40-percent depth-based, tube repair criterion during refueling outage 15 and the subsequent operating cycle 16. Amendment No. 233 was identical to Amendment No. 224.

Wolf Creek Nuclear Operating Corporation (WCNOC) also obtained a TS license amendment on April 28, 2005 (ADAMS Accession No. ML051160100), that excluded degradation in the lowermost 4 inches of the tubesheet from application of the 40-percent depth-based tube repair criterion, for Wolf Creek Generating Station during refueling outage 14 and the subsequent operating cycle. On February 21, 2006 (ADAMS Accession No. ML060600456), WCNOC submitted a permanent amendment request for the Wolf Creek Generating Station that would have limited the applicability of the TS tube inspection and plugging requirements to the upper 2.7 to 7 inches of the tubesheet thickness, depending on the tube location. This amendment, would have replaced the then current, one-cycle, SG tube inspection and plugging requirements. After three requests for additional information (RAIs) and several meetings with WCNOC, the NRC staff informed WCNOC during a phone call on January 16, 2008, that it had not provided sufficient information to allow the NRC staff to review and approve the permanent license amendment request.

Since the lack of information in the technical analysis mentioned above prevented the NRC staff from approving a permanent amendment to the TS inspection and reporting criteria, WCNOC submitted a revised application with a more conservative IARC approach. After WCNOC responded to the NRC Request for Additional Information (RAI) regarding the IARC, the NRC staff approved the IARC amendment by letter dated April 4, 2008 (ADAMS Accession No. ML080840004). Subsequent to the approval of the IARC amendment for WCNOC, similar amendments were approved for Surry Unit 2, Vogtle Unit 1, Millstone Unit 3, Vogtle Unit 2, Point Beach Unit 1, Byron Unit 2, and Braidwood Unit 2.

3.0 REGULATORY EVALUATION

In Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.36, the NRC established its regulatory requirements related to the content of the TS. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a licensee's TSs. In 10 CFR 50.36(d)(5), administrative controls are stated to be "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." This also includes the programs established by a licensee and listed in the administrative controls section of the TSs for the licensee to operate the facility in a safe manner. For Catawba 2, the requirements for performing SG tube inspections and repair are in TS 3.4.18 and TS 5.5.9, while the requirements for reporting the SG tube inspections and repair are in TS 5.6.8.

The TSs for all pressurized-water reactor (PWR) plants require that an SG program be established and implemented to ensure that SG tube integrity is maintained. For Catawba 2, SG tube integrity is maintained by meeting specified performance criteria (in TS 5.5.9.b) for structural and leakage integrity, consistent with the Catawba 2 licensing basis. TS 5.5.9.a requires that a condition monitoring assessment be performed during each outage in which the

SG tubes are inspected, 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 subject amendment request, these provisions require that the inspections 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, and that may satisfy the applicable tube repair criteria (except as indicated above regarding the application of a limited inspection scope in the tubesheet region). The applicable tube repair criteria, specified in TS 5.5.9.c., are that tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged, except if permitted to remain in service through application of the alternate repair criteria provided in TS 5.5.9.c.1.

The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, isolate fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this safety evaluation, SG tube integrity means that the tubes are capable of performing these safety functions in accordance with the plant design and licensing basis.

The General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 contain regulatory requirements which state that the RCPB shall have "an extremely low probability of abnormal leakage and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDCs 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing...to assess...structural and leaktight integrity" (GDC 32). To this end, 10 CFR 50.55a specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers, Boiler and *Pressure Vessel Code* (ASME Code). Section 50.55a further requires, in part, that throughout the service life of a PWR facility like Catawba 2, ASME Code Class 1 components meet the requirements (except design and access provisions; and pre-service examination requirements in Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components") of the ASME Code to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. The Section XI requirements pertaining to ISI of SG tubing are augmented by additional requirements in the TS.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents (DBAs), such as a SG tube rupture and a main steam line break (MSLB). These analyses consider primary to secondary leakage that may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR Part 50.67 accident source term, GDC 19 for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analysis for Catawba 2 is being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed. The licensee-proposed changes to TS 5.5.9 stay within the GDC requirements for the SG tubes and maintain the accident analysis and consequences that the NRC staff has reviewed and approved for the postulated DBAs for SG tubes.

The currently-approved License Amendment No. 233 modified the TS wording for Catawba 2 in order to exclude degradation in the lowermost 4 inches of the tubesheet from application of the

40 percent, depth-based, tube-repair criterion during refueling outage 15 and the subsequent operating cycle 16. The proposed amendment is applicable to refueling outage 16 2R16 and the subsequent operating cycle 17. The proposed amendment differs from License Amendment No. 233 in a number of ways. First, the lowermost 4 inches of the tube in the tubesheet would no longer be automatically excluded from application of the 40-percent, depth-based, tube-repair criterion. Under the proposed amendment, flaws found in the lowermost 4 inches of tubing would be subject to the IARC in lieu of the aforementioned 40-percent depth-based tube-repair criterion. Additionally, the proposed amendment includes new reporting requirements to allow the NRC staff to monitor the implementation of the amendment. As with License Amendment No. 233, the proposed amendment requires the plugging of all tubes found with flaws in the upper 17 inches of the tubesheet region.

4.0 TECHNICAL EVALUATION

4.1 Proposed Changes to the TSs

TS 5.5.9.c. 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 5.5.9.c. is being revised as follows (revisions are in **bold type**) to state:

> 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.

4.2 Proposed Change to the Facility Operating License Condition

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 NRC 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 in bold type) to state:

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.**

4.3 Detailed Technical Evaluation

The tube-to-tubesheet joint consists of the tube, which is hydraulically expanded against the bore of the tubesheet; the tube-to-tubesheet weld, located at the tube end; and the tubesheet. The joint was designed as a welded joint and not as a friction or expansion joint. The weld itself was designed as a pressure boundary element. It was designed to transmit the entire end-cap pressure load during normal and DBA conditions from the tube to the tubesheet with no credit taken for the friction developed between the hydraulically expanded tube and the tubesheet. In addition, the weld serves to make the joint leak tight.

The one-cycle amendments approved for Catawba 2 (Amendment Nos. 224 and 233) and other plants (such as Vogtle and Braidwood) prior to 2008, exempted the lower 4-inch portion of the tube within the 21-inch-deep tubesheet from inspection and exempted tubes with flaw indications in this region from being removed from service (i.e., plugged). These one-cycle amendments, in effect, redefined the pressure boundary at the tube-to-tubesheet joint as consisting of a friction or expansion joint with the tube hydraulically expanded against the tubesheet over the top 17 inches of the tubesheet. These amendments took no credit for the lower portion of the tube or the tube-to-tubesheet weld as contributing to the structural or leakage integrity of the joint.

The proposed amendment that is the subject of this Safety Evaluation (SE) (and similar amendments approved in 2008 for Wolf Creek, Vogtle, Braidwood, and Surry) differs fundamentally from the one-cycle amendments approved prior to 2008 and is a more conservative approach. The proposed amendment treats the tube-to-tubesheet joint as a welded joint in a manner consistent with the original design basis, with no credit taken for the friction developed between the hydraulically expanded tube and the tubesheet. The proposed amendment is intended to ensure that the aforementioned end-cap loads can be transmitted down the tube, through the tube-to-tubesheet weld, and into the tubesheet.

4.3.1 Proposed Change to TS 5.5.9.c, "Provisions for SG tube repair criteria"

The 40-percent depth-based tube repair criterion in TS 5.5.9.c is intended to ensure, in conjunction with other elements of TS 5.5.9, that tubes accepted for continued service (i.e., not plugged) satisfy the structural integrity performance criteria in TS 5.5.9.b.1 and the accident induced leakage performance criteria in TS 5.5.9.b.2. The criteria include allowances for eddy current measurement error and incremental flaw growth prior to the next inspection of the tube.

The alternate tube repair criteria in the existing TS and the proposed IARC in this amendment are an alternative to this 40-percent depth-based criterion.

4.3.1.1 Structural Integrity Considerations

The 40-percent depth-based criterion was developed to be conservative for flaws located anywhere in the SG, including free span regions. In the tubesheet, however, the tubes are constrained against radial expansion by the tubesheet and, therefore, are constrained against an axial (fish-mouth) rupture failure mode. The only potential structural failure mode within the tubesheet is a circumferential failure mode, leading to tube severance.

The proposed IARC would permit tubes with 100-percent through-wall flaws, in the portion of the tube from 17 inches below the TTS to 1 inch above the bottom of the tubesheet, to remain in service provided the circumferential component of these flaws does not exceed 203 degrees. The 203-degree criterion was determined by calculating the minimum tube cross-sectional area needed to resist both the limiting axial end-cap load and the pressure load on the flaw cross-section, using limit-load analysis, with the required TS structural integrity performance criteria safety factors. Because the 203-degree criterion was determined on this basis, the NRC staff finds this approach acceptable.

For the portion of the tube from the bottom of the tubesheet to 1 inch above the bottom of the tubesheet, the proposed IARC would permit tubes with 100-percent through-wall flaws to remain in service, provided the circumferential component of these flaws does not exceed 94 degrees. This 94-degree criterion was determined by calculating the minimum tube-to-tubesheet weld cross-sectional area needed to resist both the limiting axial end-cap load and the pressure load on the flaw cross-section, using limit load analysis, with the required TS structural integrity performance criteria safety factors. A 203-degree crack in the tube wall immediately above the weld would concentrate the entire end-cap load on a 157-degree segment of the weld, and would result in an inadequate safety margin. A minimum 266-degree segment (i.e., 360 minus 94 degrees) of weld is needed to resist the end-cap load with adequate safety margin. Thus, the 94-degree criterion for the tube in the lowermost 1-inch region is required to ensure that the weld is not overstressed. Although the NRC staff did not complete its review of the specific limit-load methodology used to calculate the 94-degree criterion, it reviewed the results of the stress analysis of the weld, which was performed to demonstrate that the weld complied with the stress limits of the ASME Code, Section III. The TS structural integrity performance criteria are intended to ensure the tube safety margins are consistent with the ASME Code, Section III, stress limits. Based on a comparison of the calculated maximum design stress to the ASME Code-allowable stress, the NRC staff concludes that the proposed 94-degree criterion ensures that the weld can carry the end-cap loads with margins to failure consistent with the margins ensured by the ASME stress limits and is, therefore, acceptable.

The 203- and 94-degree criteria include an allowance for incremental flaw growth in the circumferential direction prior to the next inspection. The licensee states that no significant growth rate data exists for the specific case of circumferential cracking in the tubesheet expansion region. The licensee's growth rate estimate is based on a 95-percent upper-bound value of available primary water stress corrosion crack (PWSCC) growth rate data for other tube locations. Given the lack of actual growth rate data for cracks that may potentially initiate in the lowermost 4 inches of the tube, the NRC staff attaches only a low level of confidence in the conservatism of the licensee's growth rate estimate. However, the NRC staff notes that the effect of any lack of conservatism in the licensee's estimate is mitigated somewhat by the fact that TS 5.5.9.d.4 requires inspections to be performed at Catawba Unit 2 during 2R17 (fall 2010), should any crack indications be found during 2R16 (spring 2009). In addition, the 203- and 94-degree criteria conservatively take no credit for the effects of friction in the tube-to-tubesheet joint. Any friction in the tube-to-tubesheet joint would reduce the amount of axial end-cap load that reaches the cracked tube cross-section. Thus, the NRC staff concludes that the 203- and 94-degree criteria are conservative, irrespective of growth rate uncertainties.

The 203- and 94-degree criteria do not include an explicit allowance for eddy current measurement error. The licensee will be utilizing an inspection technique that has been qualified for the detection of circumferential PWSCC in tube expansion transitions and in the tack expansion region just above the tube-to-tubesheet weld. The tack expansion is an approximately 1-inch long expansion of the tube in the tubesheet that is performed before the tube is hydraulically expanded for the entire depth of the tubesheet. A fundamental assumption behind the proposed 203- and 94-degree repair criteria is that all detected circumferential flaws in the lowermost 4 inches of the tube are 100-percent through-wall, irrespective of the actual flaw depth. With this assumption, the licensee referenced an Electric Power Research Institute (EPRI) sponsored study that indicated the eddy current measurement of the crack arc length was conservative (i.e., larger than the actual crack size), and resulted in an estimate of the remaining cross sectional area that was always smaller than values obtained through direct measurement of cracks. Although the NRC staff has not reviewed the EPRI study in detail, it finds, based on the results of the study, that any uncertainties relating to measured arc length of the flaw are not expected to impair the conservatism of the 203- and 94-degree criteria.

The proposed IARC also accounts for the interaction effects of multiple circumferential flaws that are in close proximity. The proposed IARC treats multiple circumferential flaws located within 1 inch of one another as all occurring at the same axial location. The total arc length of the combined flaws is the sum of the individual flaw arc lengths, with overlapping arc lengths counted only once. The licensee stated that flaws located more than 17 inches below the TTS and more than 1 inch above the bottom of the tubesheet will be compared to the 203-degree criterion. If one flaw is located less than 1 inch from the bottom of the tubesheet and another flaw is within 1 inch of the first flaw (or if both flaws are within 1 inch of the bottom of the tubesheet) these flaws would be compared to the 94-degree criterion. Flaws located more than 1 inch apart are assumed to act independently of each other. This 1-inch criterion was determined using a fracture mechanics approach to determine the axial distance from an individual crack tip at which the stress distribution reverts to a nominal stress distribution for an uncracked section. The 1-inch criterion is twice the calculated distance since twice this distance is the necessary separation between two cracks for the cracks to act independently of each other. The NRC staff reviewed the basis for the 1-inch criterion and the fracture mechanics approach to determining the criterion. Because the criterion is based on a valid fracture mechanics approach, the NRC staff finds it acceptable.

The proposed IARC would permit tubes with axial cracks in the lower most 4 inches of the tube to remain in service, irrespective of crack depth. The NRC staff finds this acceptable because axial cracks do not impair the ability of the tube or the weld to resist axial load and because the tube is fully constrained by the tubesheet against an axial failure mode.

Finally, the proposed IARC includes a requirement to plug all tubes in which flaws are detected in the upper 17-inch portion of the tube within the tubesheet. This adds to the conservatism of the 203- and 94-degree criteria since it mitigates any loss of tightness and, thus, any loss of friction between the tube and tubesheet due to flaws in the upper 17-inch region of the joint.

4.3.1.2 Accident-Induced Leakage Considerations

If a tube is assumed to contain a 100-percent through-wall flaw some distance into the tubesheet, a potential leak path between the primary and secondary systems is introduced between the hydraulically expanded tubing and the tubesheet. Operational leakage integrity is assured by monitoring primary-to-secondary leakage relative to the applicable TS LCO limits in TS 3.4.13, "RCS Operational LEAKAGE." However, it must also be demonstrated that the proposed TS changes do not create the potential for leakage during DBAs to exceed the accident leakage performance criteria in TS 5.5.9.b.2, including the leakage values assumed in the plant licensing basis accident analyses. The licensee states that this is ensured for Catawba 2 by limiting primary-to-secondary leakage to 0.10 gpm in the faulted SG during an MSLB accident.

The leakage path between the tube and tubesheet has been modeled by the licensee's contractor, Westinghouse, as a crevice consisting of a porous media. Using Darcy's model for flow through a porous media, leak rate is proportional to differential pressure and inversely proportional to flow resistance. Flow resistance is a direct function of viscosity, loss coefficient, and crevice length. Westinghouse performed leak tests of tube-to-tubesheet joint mockups to establish loss coefficient as a function of contact pressure. Westinghouse states that the flow resistance varies as a log normal linear function of joint contact pressure, but due to the large scatter of the flow resistance test data, has been assumed to be constant with joint contact pressure at a value which provides a conservative lower bound for the data.

Using the above model, a "modified B*" approach for calculating accident leakage was initially proposed in the amendment request. The proposed modified B* approach relies to some extent on an assumed, constant value of loss coefficient, based on a lower bound of the data. This contrasts with the "nominal B*" approach which, in its latest form, is not directly impacted by the assumed value of loss coefficient since this value is assumed to be constant with increasing contact pressure between the tube and tubesheet. The NRC staff is not able to make a conclusion as to whether the assumed value of loss coefficient in the "modified B*" approach is conservative at this time. However, the NRC staff has performed some evaluations regarding the potential for the normal operating leak rate to increase under steam-line break conditions. Making the conservative assumption that loss coefficient and viscosity are constant under both normal operating and steam-line break conditions, the ratio of steam-line break leakage rate to normal operating leak rate is equal to the ratio of steam-line break differential pressure to normal operating differential pressure times the ratio of effective crevice length under normal operating conditions (l_{NOP}) to effective crevice length under steam-line break conditions (l_{SLB}). Effective crevice length is the crevice length over which there is contact between the tube and tubesheet.

Using various values of (l_{NOP}/ l_{SLB}) determined from the "nominal B*" approach (which does not rely on an assumed value of loss coefficient) and recognizing the issues associated with some of these previous H*/B* analyses, the NRC staff concludes that a factor of 2.5 reasonably bounds the potential increase in leakage from the lowermost 4 inches of tubing that would be realized in going from normal operating to steam-line break conditions.

The licensee provided a facility operating license condition that stated:

> 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.

Since this properly addresses the factor of 2.5 that bounds the potential increase in leakage in the lowermost 4 inches of tubing, the NRC staff finds this acceptable.

4.3.2 Proposed Change to TS 5.6.8, "Steam Generator Tube Inspection Report"

The NRC staff has reviewed the proposed new reporting requirements and finds that they are sufficient to allow the NRC staff to monitor the implementation of the proposed amendment. Based on this conclusion, the NRC staff finds that the proposed new reporting requirements are acceptable.

4.3.3 Considerations Relating to Tube-to-Tubesheet Welds

The standard technical specifications and the Catawba 2 TS state specifically that the tube-to-tubesheet welds are not part of the tube. Therefore, the requirements of TS 5.5.8 do not apply to these welds. However, licensees typically visually inspect the tube ends (including the welds) for evidence of leakage while the SG primary manways are open to permit eddy current inspection of the tubes.

Eddy-current inspection of the SG tubes at Catawba 2 in 2007 revealed indications interpreted as cracks at or near the tube-to-tubesheet weld, suggesting the potential for such cracks in similar SGs at other nuclear power plants. An industry peer review was recently conducted for the Catawba 2 cold-leg tube-end indications to establish whether the reported indications are in the tube material or the welds. A consensus was reached that the indications most likely exist within the tube material. However, some of the indications extend close enough to the tube end that the possibility that the flaws extend into the weld could not be ruled out. An NRC staff member and an expert consultant from Argonne National Laboratory also reviewed these indications and concluded that the industry's position was reasonable. The peer review group and the NRC consultant also reviewed eddy-current signals from a tube-to-tubesheet mockup, which included a circumferential notch in one of the welds, and they concluded that this notch did not produce a detectable signal.

4.4 Summary of Technical Evaluation

Based on the above evaluation, the NRC staff finds that the proposed license amendment, which is applicable only to 2R16 and the subsequent Cycle 17 operation, ensures that SG tube structural and leakage integrity will be maintained during this period, with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses. Additionally, there will be no adverse impact on the ability of the tube-to-tubesheet welds to perform their safety-related function. Based on this finding, the NRC staff further concludes that the proposed amendment meets 10 CFR 50.36 and, thus, the proposed amendment is acceptable.

The current TS and the proposed amendment do not address inspection requirements for the tube-to-tubesheet welds. There are no safety issues with respect to hypothetical cracks in the weld if it can be demonstrated, such as with the H*/B* strategies discussed in this SE, that the axial end-cap loads in the tube are reacted by frictional forces developed between the tube and tubesheet before any portion of the end-cap load is transmitted to the weld.

Currently, all industry requests for a permanent H*/B* amendment have been withdrawn; however, the industry is still pursuing development of the information needed by the NRC staff to support future amendment requests for H*/B*. The licensee has concluded that cracking exclusively in the weld is not a potential damage mechanism on the basis of the peer review findings. Should it not be possible for the NRC staff to approve an acceptable H*/B* amendment within a reasonable time period, it is the NRC staff's position that the industry will need to develop inspection techniques (e.g., visual, eddy-current) capable of detecting weld cracks to ensure that the welds are capable of performing their safety-related function. It should be noted that the NRC staff observed a demonstration of an available visual inspection technique for inspecting the welds, but raised questions on whether this technique was sufficiently reliable.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission's regulations in 10 CFR 50.92(c), "Issuance of amendment," state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The following analysis was provided by the licensee in its letter dated November 13, 2008:

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 aire 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.

The NRC staff has reviewed the licensee's analysis and, based on this review, has concluded that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that the proposed amendment involves no significant hazards consideration.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final finding that the amendment involves no significant hazards consideration. The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b) ,no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) involve a significant increase in the probability or consequences of an accident previously evaluated or, (b) create the possibility of a new or different kind of accident from any previously evaluated or, (c) involve a significant reduction in a margin of safety and therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (3) such activities will be conducted in compliance with the Commission's regulations, and (4) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Johnson, DCI/CSGB

Date: April 13, 2009

April 13, 2009

Mr. J. R. Morris
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNIT 2, ISSUANCE OF AMENDMENT REGARDING INTERIM ALTERNATE REPAIR CRITERION FOR STEAM GENERATOR TUBE REPAIR (TAC NO. ME0236)

Dear Mr. Morris:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 244 to Renewed Facility Operating License NPF-52 for the Catawba Nuclear Station, Unit 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 13, 2008, as supplemented by letters dated February 5, 2009, and February 19, 2009.

The amendment adds a one-cycle revision to the TSs to incorporate an interim alternate repair criterion for steam generator tube repair criteria during the end of cycle 16 refueling outage and subsequent operating cycle 17.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

If you have any questions, please call me at 301-415-1345.

Sincerely,

/RA/

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-414

Enclosures:
1. Amendment No. 244 to NPF-52
2. Safety Evaluation

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ADAMS Accession No. ML091030088 * no significant change to SE input sent by memo dated 3/23/09

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