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

September 27, 2010

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 THE STEAM GENERATOR PROGRAM (TAC NO. ME4108)

Dear Mr. Morris:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 257 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 April 28, 2010, as supplemented by letter dated September 9, 2010.

The amendment revises Technical Specification (TS) 5.5.9 to exclude portions of the Steam Generator (SG) tube from periodic SG tube inspections and plugging or repair. In addition, reporting requirement changes are proposed to TS 5.6.8. This amendment is effective for one-cycle for the Catawba Nuclear Station, Unit 2, beginning at the End of Cycle 17 Refueling Outage and extending through subsequent Cycle 18 operation. Your application also requested deletion of a related license condition. The NRC staff's determination regarding deletion of the related license condition will be provided separately.

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 301-415-1119.

Sincerely,

Jon Thompson

Jon Thompson, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-414

Enclosures:
1. Amendment No. 257 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 257.
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 April 28, 2010, as supplemented by letter dated September 9, 2010, 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. 257 , 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. This license amendment is effective as of its date of issuance and shall be implemented prior to requiring the Steam Generators to be operable at the completion of the End of Cycle 17 Refueling Outage. Prior to implementation of the license amendment, the licensee shall complete the following commitment:

   For the Condition Monitoring (CM) assessment, the component of operational leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.27 and added to the total accident leakage from any other source and compared to the allowable accident induced leakage limit. For the Operational Assessment (OA), the difference between the allowable accident induced leakage from sources other than the tubesheet expansion region will be divided by 3.27 and compared to the observed operational leakage. An administrative operational limit will be established to not exceed the calculated value. Applicable to Catawba Unit 2 End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation.

FOR THE NUCLEAR REGULATORY COMMISSION

Gloria Kulesa, 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: September 27, 2010

ATTACHMENT TO

LICENSE AMENDMENT NO. 257

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	License
NPF-52, page 4	NPF-52, page 4
TSs	TSs
5.5-7a	5.5-7a
5.5-8	5.5-8
5.6-5	5.6-5
5.6-6	5.6-6

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 257 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.

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 Unit 2 only, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, tubes with service-induced flaws located greater than 20 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 20 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. For Unit 1, the number and portions of the tubes inspected and method of inspection shall be performed with the objective of detecting flaws of any type (for example, volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. For Unit 2, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, the number and portions of the tubes inspected and method of inspection shall be performed with the objective of detecting flaws of any type (for example, volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from 20 inches below the top of the tubesheet on the hot leg side to 20 inches below the top of the tubesheet on the cold leg side, and that may satisfy the applicable tube repair criteria. In addition to meeting requirements d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

(continued)

5.5.9 Steam Generator (SG) Program (continued)

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

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

3. For Unit 2, inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 EFPM. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 EFPM or two refueling outages (whichever is less) without being inspected.

4. For Unit 1, if crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 EFPM or one refueling outage (whichever is less). For Unit 2, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, if crack indications are found in any SG tube from 20 inches below the top of the tubesheet on the hot leg side to 20 inches below the top of the tubesheet on the cold leg side, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 EFPM or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with crack(s), then the indication need

(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,

(continued)

5.6 Reporting Requirements

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

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

i. For Unit 2, following completion of an inspection performed during the End of Cycle 17 Refueling Outage (and any inspections performed during subsequent Cycle 18 operation), the calculated accident leakage rate from the portion of the tubes below 20 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident leakage rate from the most limiting accident is less than 3.27 times the maximum primary to secondary LEAKAGE rate, the report shall describe how it was determined, and

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 257 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 April 28, 2010 (Reference 1), as supplemented by letter dated September 9, 2010, (Reference 2), Duke Energy Carolinas, LLC (Duke, the licensee), requested changes to the Technical Specifications (TSs) for the Catawba Nuclear Station, Unit 2 (Catawba 2). The supplement dated September 9, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staffs original proposed no significant hazards consideration determination as published in the *Federal Register* on July 13, 2010 (75 FR 39977).

The proposed changes would revise TS 5.5.9 to exclude portions of the Steam Generator (SG) tube from periodic SG tube inspections and plugging or repair. In addition, reporting requirement changes were proposed to TS 5.6.8. This amendment would be effective for one-cycle for Catawba 2, beginning at the End of Cycle (EOC) 17 Refueling Outage and extending through subsequent Cycle 18 operation. As a result of refinements to the SG model used in its analyses, the licensee issued a letter dated September 9, 2010, which clarified and further restricted the application of the inspection and repair requirements proposed in its letter of April 28, 2010. The letters dated September 9, 2010, and April 28, 2010, also transmitted References 3, 4, 5, 6, and 7. The licensee also proposed deletion of a related license condition. The U.S. Nuclear Regulatory Commission (NRC, the Commission) staffs determination regarding the deletion of the related license condition will be provided separately.

2.0 BACKGROUND

Catawba 2 has four Model D5 SGs, which were designed and fabricated by Westinghouse Electric Company (Westinghouse). There are 4,578 Alloy 600 tubes in each SG and each tube has a nominal outside diameter of 0.750 inches and a nominal wall thickness of 0.043 inches. The thermally-treated tubes are hydraulically expanded for the full depth of the 21-inch tubesheet and are welded to the tubesheet at each tube end. Until the fall of 2004, no instances of stress corrosion cracking affecting the tubesheet region of thermally treated Alloy 600 tubing had been reported at any nuclear power plant in the United States (U.S.).In the fall of 2004, crack-like indications were found in tubes within the tubesheet region of Catawba 2. These crack-like indications were found in a tube overexpansion (OXP) that was approximately seven inches below the hot-leg tubesheet in one tube, and just above the tube-to-tubesheet (T/TS) weld in a region of the tube known as the tack expansion region in several other tubes. Indications were also reported in the T/TS welds, which join the tube to the tubesheet. An OXP is created when the tube is expanded into a tubesheet bore hole that is not perfectly round. These out-of-round conditions were created during the tubesheet drilling process by conditions such as drill bit wandering or chip gouging. The tack expansion region is an approximately 1-inch long expansion at each tube end. The tack expansion is created to facilitate performing the T/TS weld, which is made prior to the hydraulic expansion of the tube over the full tubesheet depth. Since the initial findings at Catawba 2 in the fall of 2004, other nuclear plants have found crack-like indications in tubes within the tubesheet as well. These plants include Braidwood Station, Unit 2, Byron Station, Unit 2, Comanche Peak Steam Electric Station, Unit 2, Surry Nuclear Power Station, Unit 2, Vogtle Electric Generating Plant, Unit 1 (Vogtle 1), and Wolf Creek Generating Station, Unit 1 (Wolf Creek 1). Most of the indications were found in the tack expansion region near the tube-end welds and were a mixture of axial and circumferential primary water stress corrosion cracking.

On February 21, 2006, Wolf Creek Nuclear Operating Corporation (WCNOC), the licensee for Wolf Creek 1, submitted a license amendment request (LAR) that would permanently limit the scope of inspections required for tubes within the tubesheet (Reference 8). The LAR was based on an analysis performed by Westinghouse that provided a technical basis for permanently limiting the scope of inspections required for tubes within the tubesheet. After issuing three request for information (RAI) letters and holding several meetings with WCNOC, the NRC staff informed WCNOC during a telephone call on January 3, 2008, that it had not provided sufficient information to allow the NRC staff to review and approve the permanent LAR. WCNOC withdrew the LAR by letter dated February 14, 2008 (Reference 9). In a letter dated February 28, 2008 (Reference 10), the NRC staff identified the specific issues that needed to be addressed to support any future request for a permanent amendment, which included but were not limited to thermal expansion coefficients, crevice pressure assumptions, uncertainty models, acceptance standards for probabilistic assessment, and leakage resistance.

The Southern Nuclear Operating Company (SNC) had submitted a permanent LAR for Vogtle 1 and 2 (Reference 11) that used the same technical basis as the WCNOC LAR. Upon learning of the NRC staffs denial of the WCNOC permanent LAR, SNC substituted its permanent LAR for Vogtle 1 and 2 by letter dated February 13, 2008 (Reference 12), with one-cycle LARs that used a conservative interim alternate repair criteria (IARC) approach. The NRC staff sent additional RAIs related to the IARC, and SNC responded satisfactorily to these. The IARCs for Vogtle 1 and 2 were approved by the NRC staff on April 9, 2008 (Reference 13) and September 16, 2008 (Reference 14). After SNC received approval for the IARC amendments, several licensees submitted and gained approval for one-cycle IARC amendments that used the more conservative approach. Catawba 2 currently has an approved one-cycle IARC amendment (Reference 15).

Subsequently, several licensees and Westinghouse worked to address the issues posed in Reference 10. The NRC staff and nuclear industry representatives held public meetings (References 16, 17 and 18) and phone calls to discuss resolution of these issues.

A permanent LAR that addressed the issues identified by the NRC staff in Reference 10 was submitted by SNC in the spring of 2009 (Reference 19). Responses to NRC staff RAIs were supplied in References 4 and 5. During the RAI response review, the NRC staff identified an additional technical issue that could not be resolved in time to support approval of the permanent LAR; consequently, SNC modified its LAR in a letter dated September 18, 2009, such that the proposed changes would only be applicable for one inspection cycle (Reference 20).

The LAR submitted by Duke for Catawba 2, on April 28, 2010 (Reference 1), addressed the issues previously identified by the NRC staff in Reference 10 and was a one-cycle amendment similar to those approved for other licensees in the fall of 2009 (e.g., Reference 20). However, on July 27, 2010, Westinghouse informed the NRC that when the D5 SGs were analyzed with the Square Cell Model, the T/TS contact pressures during the main steam line break (MSLB) accident were not always higher than the normal operating condition T/TS contact pressures. As a result, Westinghouse and Duke developed the revised LAR that was submitted to the NRC on September 9, 2010 (Reference 2). The revised LAR provided an alternate leakage calculation method for H*, in situations where normal operating condition T/TS contact pressure exceeds T/TS contact pressure at accident conditions.

3.0 REGULATORY EVALUATION

In Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.36 "Technical specifications", the requirements related to the content of the TS are established. Pursuant to 10 CFR 50.36, TS are required to include items in the following five 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 plant's TS. In 10 CFR 50.36(c)(5), administrative controls are defined as "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." Programs established by the licensee, including the SG program, are listed in the administrative controls section of the TS to operate the facility in a safe manner. For Catawba 2, the requirements for performing SG tube inspections and repair are in TS 5.5.9, while the requirements for reporting the SG tube inspections and repair are in TS 5.6.8.

The TS 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 the performance criteria specified in TS 5.5.9.b for structural and leakage integrity, consistent with the plant design and 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.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. 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 T/TS weld at the tube inlet to the T/TS weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The applicable tube repair criteria, specified in TS 5.5.9.c., are that tubes found during inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged, unless the tubes are permitted to remain in service through application of the proposed alternate repair criteria provided in TS 5.5.9.c.1.

The SG tubes are part of the reactor coolant pressure boundary (RCPB) and isolate fission products in the primary coolant from the secondary coolant. For the purposes of this safety evaluation (SE), SG tube integrity means that the tubes are capable of performing this safety function in accordance with the plant design and licensing basis. The General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 provide regulatory requirements which are applicable to Catawba 2, and state that the RCPB shall have "an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), and that RCPB components shall be of "the highest quality standards practical" (GDC 30) and shall be designed to permit "periodic inspection and testing to assess their structural and leaktight integrity" (GDC 32). The licensee discusses compliance with each of these GDC for Catawba 2 in Section 3.1 of the Updated Final Safety Analysis Report (UFSAR) and does not identify any deviation from these GDC for SG tube-related issues. 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), except as provided in 10 CFR 50.55a(c)(2), (3), and (4). Section 50.55a further requires that throughout the service life of PWR facilities (like Catawba 2), ASME Code Class 1 components shall meet the Section XI requirements of the ASME Code to the extent practical, except for design and access provisions, and pre-service examination requirements. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. The Section XI requirements pertaining to inservice inspection of SG tubing are augmented by additional requirements in the TS.

As part of the plant's licensing bases, applicants for PWR licenses are required to analyze the consequences of postulated design basis accidents (DBA), such as a SG tube rupture and a 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, control room habitability standards (GDC 19), for control room operator doses (or some fraction thereof, as appropriate to the accident), or the NRC-approved licensing basis (i.e., a small fraction of these limits).

The licensee's compliance with the GDC and regulatory requirements described above is not affected by the proposed amendment. No accident analyses or dose limits for Catawba 2 are being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed. The use of the proposed alternate repair criteria does not impact the integrity of the SG tubes, and the SG tubes, therefore, still meet the requirements of the GDC in Appendix A to 10 CFR Part 50, and the requirements for Class 1 components in Section III of the ASME Code. The proposed changes maintain the accident analyses and consequences that the NRC staff has reviewed and approved for the Catawba 2 postulated DBAs for SG tubes.

The proposed amendment eliminates inspection and repair of tubes more than 20 inches below the top of the tubesheet (TTS). Tubes with service-induced flaws located in the portion of the tube from the TTS to 20 inches below the TTS shall be plugged upon detection. The proposed amendment would be applicable to Catawba 2 during Refueling Outage 17 (fall 2010) and the subsequent Cycle 18 operation.

4.0 TECHNICAL EVALUATION

4.1 Proposed Changes to the TS–Catawba 2

TS 5.5.9. is being revised as follows (new text in underline):

5.5.9 Steam Generator (SG) Program

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

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

   1. For Unit 2 only, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, tubes with service-induced flaws located greater than 20 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 20 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. For Unit 1, the number and portions of the tubes inspected and method of inspection shall be performed with the objective of detecting flaws of any type (for example, volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. For Unit 2, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, the number and portions of the tubes inspected and method of inspection shall be performed with the objective of detecting flaws of any type (for example, volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from 20 inches below the top of the tubesheet on the hot leg side to 20 inches below the top of the tubesheet on the cold leg side, and that may satisfy the applicable tube repair criteria. In addition to meeting requirements d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations

   1. [No change; not shown].

2. [No change; not shown].

3. [No change; not shown].

4. For Unit 1, if crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 EFPM or one refueling outage (whichever is less). For Unit 2, during the End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation, if crack indications are found in any SG tube from 20 inches below the top of the tubesheet on the hot leg side to 20 inches below the top of the tubesheet on the cold leg side, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 EFPM [effective full-power month] or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with crack(s), then the indication need not be treated as a crack.

TS 5.6.8. is being revised as follows (new text in underline):

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 an inspection performed in accordance with Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a.–g. [No change; Not shown]

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

i. For Unit 2, following completion of an inspection performed during the End of Cycle 17 Refueling Outage (and any inspections performed during subsequent Cycle 18 operation), the calculated accident induced leakage rate from the portion of the tubes below 20 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident leakage rate from the most limiting accident is less than 3.27 times the maximum primary to secondary LEAKAGE rate, the report shall describe how it was determined, and

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

4.2 Facility Operating License Condition Change

The licensee's application also requested deletion of a related license condition. The NRC staff's determination regarding deletion of the related license condition will be provided by separate correspondence.

The licensee submitted a regulatory commitment with their application which addressed similar technical requirements. The regulatory commitment is evaluated in Section 4.3.2 of this SE.

4.3 Technical Evaluation

The T/TS joints are part of the pressure boundary between the primary and secondary systems. Each T/TS joint consists of the tube, which is hydraulically expanded against the bore of the tubesheet, the T/TS weld located at the tube end, and the tubesheet. The joints were designed in accordance with the ASME Code, Section III, as welded joints, not as friction joints. The T/TS welds were designed to transmit the tube end cap pressure loads, during normal operating and DBA conditions, from the tubes to the tubesheet, with no credit taken for the friction developed between the hydraulically-expanded tube and the tubesheet. In addition, the welds serve to make the joints leak tight.

This design-basis is a conservative representation of how the T/TS joints actually work, since it conservatively ignores the role of friction between the tube and tubesheet in reducing the tube end cap loads. The initial hydraulic expansion of the tubes against the tubesheet produces an "interference fit" between the tubes and the tubesheet; thus, producing a residual contact pressure (RCP) between the tubes and tubesheet, which acts normally to the outer surface of the tubes and the inner surface of the tubesheet bore holes. Additional contact pressure between the tubes and tubesheet is induced by operational conditions as will be discussed in detail below. The amount of friction force that can be developed between the outer tube surface and the inner surface of the tubesheet bore is a direct function of the contact pressure between the tube and tubesheet times the applicable coefficient of friction.

To support the proposed TS changes, the licensee's contractor, Westinghouse, has defined a parameter called H* to be that distance below the top of the tubesheet over which sufficient frictional force, with acceptable safety margins, can be developed between each tube and the tubesheet-under-tube-end-cap pressure loads associated with normal operating and DBA conditions to prevent significant slippage and pullout of the tube from the tubesheet, assuming the tube is fully severed at the H* distance below the top of the tubesheet. For Catawba 2, the proposed H* distance is 20 inches. Given that the frictional force developed in the T/TS joint over the H* distance is sufficient to resist the tube end cap pressure loads, it is the licensee's and Westinghouse's position that the length of tubing between the H* distance and the T/TS weld is not needed to resist any portion of the tube end cap pressure loads. Thus, the licensee is proposing to change the TS to not require inspection of the tubes below the H* distance and to exclude tube flaws located below the H* distance (including flaws in the T/TS weld) from the application of the TS tube repair criteria. Under these changes, the T/TS joint would now be treated as a friction joint extending from the top of the tubesheet to a distance below the top of the tubesheet equal to H* for purposes of evaluating the structural and leakage integrity of the joint.

The regulatory standard by which the NRC staff has evaluated the subject LAR is that the amended TSs should continue to ensure that tube integrity will be maintained consistent with the current design basis, as defined in the UFSAR. This includes maintaining structural safety margins consistent with the structural performance criteria in TS 5.5.9.b.1 discussed in Section 4.3.1.1 below. In addition, this includes limiting the potential for accident-induced primary-to-secondary leakage to values that do not exceed the accident-induced leakage performance criteria in TS 5.5.9.b.2, which are consistent with values assumed in the UFSAR accident analyses. Maintaining tube integrity in this manner assures that the amended TS are in compliance with all applicable regulations. The NRC staffs evaluation of joint structural integrity and accident-induced leakage integrity is discussed in Sections 4.3.1 and 4.3.2 of this SE, respectively.

4.3.1 Joint Structural Integrity

4.3.1.1.1 Acceptance Criteria

Westinghouse has conducted extensive analyses to establish the necessary H* distance to resist pullout under normal operating and DBA conditions. The NRC staff finds that pullout is the structural failure mode of interest since the tubes are radially-constrained against axial (fishmouth) rupture by the presence of the tubesheet. The axial force which could produce pullout derives from the pressure end cap loads due to the primary-to-secondary pressure differentials associated with normal operating and DBA conditions. Westinghouse determined the needed H* distance on the basis of maintaining a factor of three against pullout under normal operating conditions and a factor of 1.4 against pullout under DBA conditions. The NRC staff finds that these are appropriate safety factors to apply to demonstrate structural integrity. These safety factors are consistent with the safety factors embodied in the structural integrity performance criteria in TS 5.5.9.b.1 and with the design basis including the stress limit criteria in the ASME Code, Section III.

4.3.1.2 T/TS Interaction Model

The resistance to pullout is the axial friction force developed between the expanded tube and the tubesheet over the H* distance. The friction force is a function of the radial contact pressure between the expanded tube and the tubesheet. Westinghouse used classical thick-shell equations to model the interaction effects between the tubes and tubesheet under various pressure and temperature conditions for purposes of calculating contact pressure (T/TS interaction model). For each tube, the tubesheet was modeled as an equivalent cylinder. The thickness of this equivalent cylinder was calculated to provide a stiffness equivalent to the actual tubesheet geometry in terms of the amount of tubesheet bore radial displacement that is associated with a given amount of radial pressure on the surface of the bore. Two-dimensional (2-D) finite element analyses (FEAs) of portions of the perforated tubesheet geometry were used to determine the thickness of the equivalent tubesheet cylinder that provided the necessary stiffness, as a function of tube location within the bundle. These analyses directly modeled a spectrum of possibilities concerning the pressure loads acting on nearby bore surfaces, instead of employing a beta factor adjustment as was done to support previous H* amendment requests submitted prior to 2008. Based on its review, the NRC staff concludes that the equivalent tubesheet cylinder thicknesses calculated by Westinghouse are conservative since they provide for lower bound stiffness estimates, leading to lower (conservative) estimates of contact pressure and resistance to pullout.

The shell model representing the tube was used to determine the relationship between the tube outer surface radial displacement and the applied axial end cap load (due to the primary-to-secondary pressure differential), primary pressure acting on the tube inner surface, crevice pressure[1] acting on the tube outer surface, contact pressure between the tube and tubesheet bore, and tube thermal expansion. However, the equivalent shell model representing the tubesheet was used only to determine the relationship between the tubesheet bore surface radial displacement with the applied crevice pressure and contact pressure. Radial displacements of the tubesheet bore surfaces are also functions of the primary pressure acting on the primary face of the tubesheet and SG channel head, secondary pressure acting on the secondary face of the tubesheet and SG shell, and the temperature distribution throughout the entire lower SG assembly. These displacements are a function of tube location within the tube bundle and, also, a function of axial location within the tubesheet. To calculate these displacements, 3-D FEAs were performed. The NRC staffs evaluation of these FEAs is provided in Section 4.3.1.3, below. The tubesheet bore radial displacements from the 3-D FEAs were added to those from the tubesheet equivalent shell model to yield the total displacement of the tubesheet bore surface as a function of tube radial and axial location. As discussed more fully in Sections 4.3.1.3 and 4.3.1.8, there are outstanding technical issues with respect to how to apply asymmetric displacement output from the 3-D FEAs to the axisymmetric thick shell model.

The reference T/TS interaction model (Reference 3) assumes as an initial condition that each tube is fully expanded against the tubesheet bore such that the outer tube surface is in contact with the inner surface of the tubesheet bore under room temperature, atmospheric pressure conditions, with zero RCP associated with the hydraulic expansion process. The NRC staff finds the assumption of zero RCP in the reference analysis to be a very conservative assumption.

The thick shell equations used in the T/TS interaction model allow calculation of the tube radial displacements and the tubesheet equivalent cylinder radial displacements for a given set of pressure and temperature conditions. Under normal operational and DBA pressures and temperatures, the tube outer surface undergoes a higher radial displacement than the tubesheet bore surface if interaction between the tube and tubesheet is ignored. The T/TS interaction effects demand continuity of displacements (i.e., the radial displacement of the tube outer surface must equal the radial displacement of the bore surface) at each axial location. Therefore, pressure of sufficient magnitude to ensure equal radial displacements is developed between the two surfaces and can be determined directly. The NRC staff has reviewed the development of the T/TS interaction model and finds that it conservatively approximates the actual T/TS interaction effects and the resulting contact pressures.

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1 Although the tubes are in tight contact with the tubesheet bore surfaces, surface roughness effects are conservatively assumed to create interstitial spaces, which are effectively crevices, between these surfaces. See section 4.3.1.5 for more information.

The classical thick shell equations used in the interaction model were developed for cylindrical shells whose geometry and applied loads are uniform along the cylindrical axis. As discussed above, radial deflections of the tubesheet bores are non-uniform from the top to the bottom of the tubesheet, due to the temperature and pressure loadings acting on the various components of the SG lower assembly. In addition, the crevice pressure may vary in the axial direction, as discussed below. The interaction model essentially divides the T/TS joint into a series of horizontal slices, where each slice is assumed to behave independently of the slices above and below. The NRC staff concludes this to be conservative since it adds radial flexibility to the T/TS joint leading to lower contact pressures and tube pullout resistance.

The resisting force to the applied end cap load, which is developed over each incremental axial distance from the top of the tubesheet, is the average contact pressure over that incremental distance times the tubesheet bore surface area (equal to the tube outer diameter surface area) over the incremental axial distance times the coefficient of friction. The NRC staff reviewed the coefficient of friction used in the analysis and judges it to be a reasonable lower bound (conservative) estimate. The H* distance for each tube was determined by integrating the incremental friction forces from the top of the tubesheet to the distance below the top of the tubesheet where the friction force integral equaled the applied end cap load times the appropriate safety factor as discussed in Section 4.3.1.1.

In summary, the NRC staff has evaluated the T/TS interaction model and finds it to be a reasonable and conservative approach for the calculation of H* distances. However, as discussed more fully in Sections 4.3.1.3 and 4.3.1.8, there are outstanding issues with respect to how to apply asymmetric displacement output from the 3-D FEAs to the axisymmetric thick shell T/TS interaction model, which is the reason the licensee is requesting only an interim amendment at this time.

4.3.1.3 3-D Finite Element Analysis

A 3-D FEA of the lower SG assembly (consisting of the lower portion of the SG shell, the tubesheet, the channel head, and the divider plate separating the hot- and cold-leg inlet plenums inside the channel head) was performed to calculate the diameter changes of the tubesheet bore surfaces due to primary pressure acting on the primary face of the tubesheet and SG channel head, secondary pressure acting on the secondary face of the tubesheet and SG shell, and the temperature distribution throughout the entire lower SG assembly. These calculated diameter changes tended to be non-uniform around the circumference of the bore. The thick-shell equations used in the T/TS interaction model are axisymmetric, and thus, the non-uniform diameter change from the 3-D FEA had to be adjusted (i.e., post-processed) to an equivalent uniform value before it could be used as input to the T/TS interaction analysis. A 2-D plane stress finite element model was used to define a relationship for determining a uniform diameter change that would produce the same change to average T/TS contact pressure as would the actual non-uniform diameter changes from the 3-D FEAs. As documented in Reference 3, Westinghouse identified a difficultly in applying this relationship to Model D5 SGs, for the case of MSLB, during spring 2009. Westinghouse attributed this difficulty to the relatively low primary water temperature that exists in Model D5 SGs under MSLB conditions and which is less than the range of temperatures that the eccentricity relationship is intended to address.

To address this problem, Westinghouse developed an alternative model for the eccentricity effect, which it applied to the Model D5 MSLB case, but continued to apply the original eccentricity relationship for the Model D5 normal operating conditions case. This alternative model was an early version of the Square Cell Model discussed later in Section 4.3.1.8.1. In reviewing the reasons for the difficulty in applying the eccentricity relationship to the Model D5 SGs, the NRC staff issued RAIs relating to the conservatism of the relationship and whether the tubesheet bore displacement eccentricities were sufficiently limited to ensure that T/TS contact was maintained around the entire tube circumference. The RAI responses in References 4 and 5 did not provide sufficient information to allow the NRC staff to reach a conclusion on the concerns relating to tubesheet bore displacement eccentricity. The NRC staff informed Westinghouse and the industry that the questions regarding eccentricity would prevent approval of a permanent H* amendment. Therefore, on April 28, 2010, the licensee requested an interim amendment (Reference 1) that would be applicable only to the EOC 17 Refueling Outage and the subsequent Cycle 18 Operation.

This 3-D FEA replaced the 2-D axisymmetric FEA used to support H* LARs submitted prior to 2008. The NRC staff finds that the 3-D analysis adequately addresses a concern cited by the NRC staff in Reference 10 concerning the validity of the axisymmetric model to conservatively bound significant non-axisymmetric features of the actual tubesheets. These non-axisymmetric features include the solid (non-bored) portion of the tubesheet between the hot- and cold-leg sides, and the divider plate which acts to connect the solid part of the tubesheet to the channel head.

Some non-U.S. units have experienced cracks in the weld between the divider plate and the stub runner attachment on the bottom of the tubesheet. Should such cracks ultimately cause the divider plate to become disconnected from the tubesheet, tubesheet vertical and radial displacements under operational conditions could be significantly increased relative to those for an intact divider plate weld. Although the industry believes that there is little likelihood that cracks such as those seen abroad could cause a failure of the divider plate weld, the 3-D FEA conservatively considered both the case of an intact divider plate weld and a detached divider plate weld to ensure a conservative analysis. The case of a detached divider plate weld was found to produce the most limiting H* values.

Separate 3-D FEAs were conducted for each loading condition considered (i.e.; normal operating conditions, MSLB, feedwater line break (FLB)). The NRC staff finds that this adequately addresses (corrects) a significant source of error in analyses used by applicants to support permanent H* LARs submitted prior to 2008 and which were subsequently withdrawn (Reference 10).

4.3.1.4 Coefficient of Thermal Expansion (CTE)

During operation, a large part of contact pressure in a SG T/TS joint is derived from the difference in the CTE between the tube and tubesheet. As discussed in Section 4.3.1.8, the calculated value of H* is highly sensitive to the assumed values of these CTE parameters. However, CTE test data acquired by an NRC contractor, Argonne National Laboratory (ANL), suggested that CTE values may vary substantially from values listed in the ASME Code for design purposes. In Reference 10, the NRC staff highlighted the need to develop a rigorous technical basis for the CTE values, and their potential variability, to be employed in future H* analyses.

In response, Westinghouse had a subcontractor review the CTE data in question, determine the cause of the variance from the ASME Code CTE values, and provide a summary report (Reference 21). Analysis of the CTE data in question revealed that the CTE variation with temperature had been developed using a polynomial fit to the raw data, over the full temperature range from 75 °F to 1300 °F. The polynomial fit chosen resulted in mean CTE values that were significantly different from the ASME Code values from 75 °F to about 300 °F. When the raw data were reanalyzed using the locally weighted least squares regression (LOWESS) method, the mean CTE values determined were in good agreement with the established ASME Code values.

Westinghouse also formed a panel of licensee experts to review the available CTE data in open literature, review the ANL-provided CTE data, and perform an extensive CTE testing program on Alloy 600 and SA-508 steel material to supplement the existing data base. Two additional sets of CTE test data (different from those addressed in the previous paragraph) had CTE offsets at low temperature, that were not expected. Review of the test data showed that the first test, conducted in a vacuum, had proceeded to a maximum temperature of 700 °C, which changed the microstructure and the CTE of the steel during decreasing temperature conditions. As a result of the altered microstructure, the CTE test data generated in the second test, conducted in air, was also invalidated. As a result of the large "dead band" region and the altered microstructure, both data sets were excluded from the final CTE values obtained from the CTE testing program.

The test program included multiple material heats to analyze chemistry influence on CTE values and repeat tests on the same samples were performed to analyze for test apparatus influence. Because the tubes are strain hardened when they are expanded into the tubesheet, strain hardened samples were also measured to check for strain hardening influence on CTE values.

The data from the test program were combined with the ANL data that were found by the licensee to be acceptable and with data obtained from the open literature search. A statistical analysis of the data uncertainties was performed by comparing deviations to the mean values obtained at the applicable temperatures. The correlation coefficients obtained indicated a good fit to a normal distribution, as expected. Finally, an evaluation of within-heat variability was performed due to increased data scatter at low temperatures. The within-heat variability assessment determined that the increase in data scatter was a testing accuracy limitation and only present at low temperature. The CTE report is included as Appendix A to Reference 3.

The testing showed that the nominal ASME Code values for Alloy 600 and SA-508 steel were both conservative relative to the mean values from all the available data. Specifically, the CTE mean value for Alloy 600 was greater than the ASME Code value, and the CTE mean value for SA-508 steel was smaller than the ASME Code value. Thus, the H* analyses utilized the ASME Code values as mean values in the H* analyses. The NRC staff finds this to be conservative because it tends to lead to an over-prediction of the expansion of the tubesheet bore and an under-prediction of the expansion of the tube, thereby resulting in an increase in the calculated H* distance. The statistical variances of the CTE parameters from the combined data base were utilized in the H* probabilistic analysis.

Based on its review of Westinghouse CTE program, the NRC staff concludes that the CTE values used in the H* analyses are responsive to the concerns in Reference 10 and are acceptable.

4.3.1.5 Crevice Pressure Evaluation

As discussed in an earlier footnote, the H* analyses postulate that interstitial spaces exist between the hydraulically expanded tubes and tubesheet bore surfaces. These interstitial spaces are assumed to act as crevices between the tubes and the tubesheet bore surfaces. The NRC staff finds that the assumption of crevices is conservative since the pressure inside the crevices acts to push against both the tube and the tubesheet bore surfaces, thus reducing contact pressure between the tubes and tubesheet.

For tubes which do not contain through-wall flaws within the thickness of the tubesheet, the pressure inside the crevice is assumed to be equal to the secondary system pressure. For tubes that contain through-wall flaws within the thickness of the tubesheet, a leak path is assumed to exist, from the primary coolant inside the tube, through the flaw, and up the crevice to the secondary system. Hydraulic tests were performed on several tube specimens that were hydraulically expanded against tubesheet collar specimens to evaluate the distribution of the crevice pressure from a location where through-wall holes had been drilled into the tubes to the top of the crevice location. The T/TS collar specimens were instrumented at several axial locations to permit direct measurement of the crevice pressures. Tests were run for both normal operating and MSLB pressure and temperature conditions.

The NRC staff finds that the use of the drilled holes, rather than through-wall cracks, is conservative since it eliminates any pressure drop between the inside of the tube and the crevice at the hole location. This maximizes the pressure in the crevice at all elevations, thus reducing contact pressure between the tubes and tubesheet.

The crevice pressure data from these tests were used to develop a crevice pressure distribution as a function of normalized distance between the top of the tubesheet and the H* distance below the top of the tubesheet where the tube is assumed to be severed. These distributions were used to determine the appropriate crevice pressure for each axial slice of the T/TS interaction model. Based on its review of the tests and test results, the NRC staff finds the assumed crevice pressure distributions to be realistic and acceptable.

Because the crevice pressure distribution is assumed to extend from the H* location, where crevice pressure is assumed to equal primary pressure, to the top of the tubesheet, where crevice pressure equals secondary pressure, an initial selection as to the H* location must be made before solving for H* using the T/TS interaction model and 3-D finite element model. The resulting new H* estimate becomes the initial estimate for the next H* iteration.

4.3.1.6 H* Calculation Process

The calculation of H* in the reference analyses (Reference 3) consisted of the following steps for each loading case considered:

1. Perform initial H* estimate using the interaction and 3-D finite element models, assuming nominal geometric and material properties, and assuming that the tube is severed at the bottom of the tubesheet for purposes of defining the pressure distribution over the length of the T/TS crevice.

2. Add 0.3-inch adjustment to the initial H* estimate to account for uncertainty in the bottom of the tube expansion transition (BET) location relative to the top of the tubesheet, based on an uncertainty analysis on the BET conducted by Westinghouse.

3. For normal operating conditions only, add an additional adjustment to correct for the actual temperature distribution in the tubesheet compared to the linear distribution assumed in the FEA. (In later analyses (Reference 4), this step was shown to be conservative when replaced by direct modeling; see Section 4.3.1.8.).

4. Steps 1 through 3 yield a so-called "mean" estimate of H*, which is deterministically based. Step 4 involves a probabilistic analysis of the potential variability of H*, relative to the mean estimate, based on the combined potential variability of key input parameters for the H* analyses. This leads to a "probabilistic" estimate of H*, which is greater than the "mean" estimate calculated in steps 1 through 3.

5. Add a crevice pressure adjustment to the probabilistic estimate of H* to account for the crevice pressure distribution which results from the tube being severed at the final H* value, rather than at the bottom of the tubesheet. The value of this adjustment was determined iteratively.

The NRC staffs evaluation of the probabilistic analysis is provided in Section 4.3.1.8 of this SE. Regarding Step 2 above, the NRC staff did not review the Westinghouse BET uncertainty analysis. Therefore, at the NRC staffs request, the licensee has committed to a one-time inspection of the actual BET locations to confirm that there are no significant deviations from the assumed BET value. Any such deviations will be entered into the corrective actions program for disposition. The NRC staff finds this to be acceptable, since the BET inspections are a one-time action that is reviewable during routine NRC staff regional oversight activities. Any deviations are likely to be small (less than a few tenths of an inch) and not likely to impact the overall conservatism of the proposed H* distance.

4.3.1.7 Acceptance Standard - Probabilistic Analysis

The purpose of the probabilistic analysis is to develop a safe H* distance that ensures with a probability of 0.95 that the population of tubes will retain margins against pullout consistent with criteria evaluated in Section 4.3.1.1 of this SE, assuming all tubes to be completely severed at their H* distance. The NRC staff finds this probabilistic acceptance standard is consistent with what the NRC staff has approved previously and is acceptable. For example, the upper voltage limit for the voltage-based tube repair criteria in NRC Generic Letter 95-05 (Reference 22) employs a consistent criterion. The NRC staff also notes that use of the 0.95 probability criterion ensures that the probability of pullout of one or more tubes under normal operating conditions and conditional probability of pullout under accident conditions is well within tube rupture probabilities that have been considered in probabilistic risk assessments (References 23 and 24).

In terms of the confidence level that should be attached to the 0.95 probability acceptance standard, it is industry practice for SG tube integrity evaluations, as embodied in industry guidelines, to calculate such probabilities at a 50 percent confidence level. The Westinghouse recommended H* value of 13.8 inches in Reference 3 for model D5 SGs is based on probabilistic estimates performed at a 50 percent confidence value. However, as discussed in Section 4.3.1.8, the NRC staff finds that the 20 inch H* value proposed by the licensee conservatively bounds an H* value based on probabilistic estimates performed at a 95 percent confidence value.

Another issue relating to the acceptance standard for the probabilistic analysis is determining what population of tubes needs to be analyzed. For accidents such as MSLB or FLB, the NRC staff and licensee both find that the tube population in the faulted SG is of interest, since it is the only SG population that experiences a large increase in the primary-to-secondary pressure differential. However, normal operating conditions were found to be the most limiting in terms of meeting the tube pullout margins in Section 4.3.1.1. For normal operating conditions, tubes in all SGs at the plant are subject to the same pressures and temperatures. Although there is not a consensus between the NRC staff and industry on which population needs to be considered in the probabilistic analysis for normal operating conditions, and although the Westinghouse recommended H* value in Reference 3 is based on the population of just one SG, the NRC staff finds that the 20 inch H* value proposed by the licensee conservatively bounds an H* value based on probabilistic estimates performed at a 95 percent confidence level for the entire tube population (i.e., for all SGs) at the plant, as discussed in Section 4.3.1.8 below.

4.3.1.8 Probabilistic Analyses

Sensitivity studies were conducted and demonstrated that H* was highly sensitive to the potential variability of the coefficients of thermal expansion (CTE) for the Alloy 600 tubing material and the SA-508 Class 2a tubesheet material. Given that no credit was taken in any of the H* analyses, for RCP associated with the tube hydraulic expansion process[2], the sensitivity of H* to other geometry and material input parameters was judged by Westinghouse to be inconsequential and were ignored, with the exception of Young's modulus of elasticity for the tube and tubesheet materials. Although the Young's modulus parameters were included in the reference H* analyses sensitivity studies, these parameters were found to have a weak effect on the computed H*. Based on its review of the analysis models and engineering judgment, the NRC staff concurs that the sensitivity studies adequately capture the input parameters which may significantly affect the value of H*. This conclusion is based, in part, on no credit being taken for RCP during the reference H* analyses.

2 Residual contact pressures are sensitive to variability of other input parameters.

These sensitivity studies were used to develop influence curves describing the change in H*, relative to the mean H* value estimate (see Section 4.3.1.6), as a function of the variability of each CTE parameter and Young's modulus parameter, relative to the mean values of CTE and Young's Modulus. Separate influence curves were developed for each of the four input parameters. The sensitivity studies showed that of the four input parameters, only the CTE parameters for the tube and tubesheet material had any interaction with one another. A combined set of influence curves containing this interaction effect was also created.

Two types of probabilistic analyses were performed independently. One was a simplified statistical approach utilizing a square root of the sum of the squares method and the other was a detailed Monte Carlo sampling approach. The NRC staffs review relies primarily on the Monte Carlo analysis which provides the more realistic treatment of uncertainties.

The NRC staff reviewed the implementation of probabilistic analyses in the reference analyses (Reference 3) and questioned whether the H* influence curves had been conservatively treated. The NRC staff concluded that the reference analysis was insufficient to support the amendment request. To address this concern, the licensee submitted new H* analyses as documented in Reference 4. These analyses made direct use of the H* influence curves in a manner the NRC staff finds to be acceptable. To show that the proposed H* value in the April 28, 2010, LAR was conservative, the revised Reference 4 analyses eliminated some of the conservatisms in the Reference 3 analyses as follows:

1. The Reference 3 analyses assumed that all tubes were located at the location in the tube bundle where the mean value estimate of H* was at its maximum value. The revised Reference 4 analyses divided the tubes by sector location within the tube bundle and all tubes were assumed to be at the location in their respective sectors where the mean value estimate of H* was at its maximum value for that sector. The H* influence curves discussed above, developed for the most limiting tube location in the tube bundle, were conservatively used for all sectors. The NRC staff concludes that the sector approach in the revised Reference 4 analyses results in a more realistic, but still conservative H* estimate.

2. The original Reference 3 analyses add an incremental distance to H* to account for the actual distribution of temperatures in the tubesheet under normal operating conditions versus the linear distribution assumed in the reference FEAs (see step 3 in Section 4.3.1.6). The revised Reference 4 analyses included new FEAs which considered the actual distribution of temperatures under normal operating conditions. The revised analyses confirmed the conservatism of the adjustment made in the original Reference 3 analyses. The revised FEAs, in conjunction with the sector analyses in item 1 above, result in an H* value which was significantly less than the originally proposed 16.95 inch H* based on a 0.95 probability/50 percent confidence, single SG basis. The NRC staff concluded that direct modeling of the actual temperature distribution in the tubesheet provided a more realistic, but still conservative estimate of H*, albeit on a 50 percent confidence, single SG basis. No H* estimate was provided on a 0.95 probability/95 percent confidence, all SGs basis for this specific case. However, the sensitivity of the calculated H* when evaluated at a 95 percent versus 50

percent confidence level, and when evaluated on an all SGs versus a single SG basis, was determined for other cases in Reference 25. Based on its review of these sensitivities, the NRC staff concluded that an H* value for the case based on a 0.95 probability/95 percent confidence all SGs basis, was less than the originally proposed H* distance of 16.95 inches.

3. The Reference 3 analyses take no credit for RCP due to hydraulic expansion of tubes against the respective tubesheet bores during SG manufacture. The revised Reference 4 analyses include consideration of recently completed pullout tests and analyses. The licensee states that the tests confirm a significant level of RCP, and showed that within a small degree of slippage, the forces required to continue to move the tube by far exceeded the maximum pullout forces that could be generated under very conservative assumptions. The licensee finds that crediting this latest information, in conjunction with the sector analysis discussed in Item 1 and the updated correction discussed in Item 2 based on direct modeling of the temperature distribution in the tubesheet, leads to a further, significant reduction in the calculated H* value relative to values calculated in Items 1 and 2. This information, including the latest pullout test data, has not been reviewed in detail by the NRC staff. However, the NRC staff concludes that H* estimates that include no credit for RCP (e.g., the estimates in Items 1 and 2 above) are very conservative, as evidenced by the high pullout forces needed to overcome the RCP.

The revised Reference 4 analyses, Items 1, 2, and 3 above, also address a question posed by the NRC staff in Reference 10 concerning the appropriate way to sample material properties for the tubesheet, whose properties are unknown but do not vary significantly for a given SG, in contrast to the tubes whose properties tend to vary much more randomly from tube to tube in a given SG. This issue was addressed by a staged sampling process where the tubesheet properties were sampled once and then held fixed, while the tube properties were sampled a number of times equal to the SG tube population. This process was repeated 10,000 times, and the maximum H* value from each repetition was rank ordered. The final H* value was selected from the rank ordering to reflect a 0.95 probability value at the desired level of confidence. The NRC staff concludes that this approach addresses the NRC staffs question in a realistic fashion and is acceptable.

The licensee has committed to monitor for tube slippage as part of the SG inspection program. Under the proposed license amendment, TS 5.6.8.j will require that the results of slippage monitoring be included as part of the 180-day report required by TS 5.6.8. TS 5.6.8.j will also require that should slippage be discovered, the implications of the discovery and corrective action shall be included in the report. The NRC staff finds that slippage is not expected to occur for the reasons discussed previously. In the unexpected event it should occur, it will be important to understand why it occurred so that the need for corrective action can be evaluated. The NRC staff concludes the commitment to monitor for slippage and the accompanying reporting requirements are acceptable.

4.3.1.8.1 Recent Analyses/Revised Value of Proposed H* Distance

New analyses by Westinghouse (References 6 and 7), performed after the submittal of Reference 1 for the interim H* LAR, indicate that although the above 3-D FEA provides a realistic treatment of the tubesheet behavior on a global basis, it was not providing realistic displacements at the local tubesheet bore surfaces. This was because the 3-D model of the tubesheet did not explicitly model the tube bores, but rather the tubesheet was modeled as a solid plate whose stiffness had been adjusted to reflect the perforated tubesheet geometry.

The new analyses (called the Square Cell (or C^2) Model) employed a 2-D finite element model of a single tube that is centered in a square tubesheet section with one tube pitch dimensions (1.0625 inches). As with the thick-shell approach, the tube was assumed to be in full contact with the TS bore surface in the unloaded condition, with zero residual contact pressure. The loading conditions applied to the Square Cell Model are temperature, which varies axially through the tubesheet, internal tube pressure, axially-dependent crevice pressure, and planar displacements at the model boundaries. The planar displacement boundary conditions were applied to the edges of the Square Cell Model, based on the results of a revised version of the 3-D FEA discussed in Section 4.3.1.3. The revised 3-D model incorporates a revised mesh intended to conform to the geometry of the Square Cell Model.

This recent work resulted in significant changes to the calculated values of T/TS contact pressure as a function of tube radial location and elevation within the thickness of the tubesheet relative to those in References 3, 4, and 5. In general, this led to a reduction in contact pressure over the upper portion of the H* distance and an increase in contact pressure over the lower portions of the H* distance, at the limiting tube locations. The net effect was an increase the nominal H* distance, relative to the earlier analyses in References 3, 4, and 5. Westinghouse is currently redoing the probabilistic H* analyses to determine H* at the appropriate statistically-bounding value, but has not yet completed that analysis. Westinghouse provided a qualitative assessment in Reference 7, which concluded that the ongoing probabilistic analysis is expected to confirm that the proposed H* distance in Reference 1 is conservative. This conclusion is based primarily on the higher contact pressures now being estimated in the lower portion of the H* distance. However, the licensee has revised its proposed interim H* value from 16.95 inches to 20 inches in order to reflect the 3-inch increase in the nominal H* estimate (Reference 7).

The NRC staff finds that the recent analyses by Westinghouse should, in concept, be more realistic than the earlier analyses in References 3, 4, and 5 and, in addition, the NRC staff expects these analyses to address the NRC staffs concerns relating to the eccentricity adjustment relationship discussed in 4.3.1.3. The NRC staff finds that the Square Cell Model has allowed comparison of both DBA and normal operating conditions with the same model, which adequately addresses a topic raised by the NRC staff in Reference 26. However, Westinghouse has only provided summary descriptions of its recent work (Reference 7), including the latest version of the Square Cell Model and 3-D finite element model of the lower SG assembly. A complete description of theses analyses and the probabilistic analysis will need to be submitted to support a request for a permanent H* amendment. In the meantime, the NRC staffs basis for approving the requested interim amendment is provided in Section 4.3.4.

4.3.2 Accident-induced Leakage Considerations

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 a DBA 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.

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. 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 T/TS joint mockups to establish loss coefficient as a function of contact pressure. A large amount of data scatter, however, precluded quantification of such a correlation. In the absence of such a correlation, Westinghouse has developed a leakage factor relationship between accident induced leak rate and operational leakage rate, where the source of leakage is from flaws located at or below the H* distance.

Using the Darcy model, the leakage factor for a given type accident is the product of four quantities. The first quantity is the ratio of the maximum primary-to-secondary pressure difference during the accident divided by that for normal operating conditions. The second quantity is the ratio of viscosity under normal operating primary water temperature divided by viscosity under the accident condition primary water temperature. The third quantity is the ratio of crevice length under normal operating conditions to crevice length under accident conditions. This ratio equals 1, provided it can be shown that positive contact pressure is maintained along the entire H* distance for both conditions. The fourth quantity is the ratio of loss coefficient under normal operating conditions to loss coefficient under the accident condition. Although the absolute value of these loss coefficients isn't known, Westinghouse has assumed that the loss coefficient is constant with contact pressure such that the ratio is equal to 1. The NRC staff agrees that this is a conservative assumption, provided there is a positive contact pressure for both conditions along the entire H* distance and provided that contact pressure increases at each axial location along the H* distance when going from normal operating to accident conditions. Both assumptions were confirmed to be valid in the H* analyses submitted with Reference 1.

Leakage factors were calculated for design basis accidents exhibiting a significant increase in primary-to-secondary pressure differential, including MSLB, FLB, locked rotor, and control rod ejection. The design basis FLB heat-up transient was found to exhibit the highest leakage factor, 3.27, meaning that it is the transient expected to result in the largest increase in leakage relative to normal operating conditions.

Subsequent to the Reference 1 LAR submittal, new analyses were performed by Westinghouse (Reference 7); these analyses did not show an increasing T/TS contact pressure when going from normal operating to MSLB conditions. These new analyses utilized the newly revised 3-D finite element model of the lower SG assembly and the newly revised Square Cell Model, discussed in Section 4.3.1.8.1 of this SE. Although T/TS contact pressure increased over some sections of the tubing under SLB conditions, it decreased over other sections within the H* distance. This violated the assumed precondition for assuming that the ratio of loss coefficient under MSLB and normal operating conditions was at least equal to 1.

As discussed above, the large scatter of the loss coefficient versus contact pressure data prevented direct use of this data in applying Darcy's leakage model. Instead, Westinghouse considered a number of mathematical functions that represented the potential functional relationship between loss coefficient and contact pressure. For each potential functional relationship, Westinghouse evaluated the ratio of loss coefficient under MSLB and normal operating conditions, at each elevation and radial location within the tubesheet. For each tube, this ratio was integrated over the length of the H* distance yielding a ratio of flow resistances for MSLB and normal operating conditions. This ratio, in conjunction with the differential pressure and viscosity ratios, was then used to compute the ratio of leakage under MSLB and normal operating conditions, at each radial location within the tubesheet.

None of the potential functional relationships between loss coefficient and contact pressure considered by Westinghouse resulted in a leakage ratio value exceeding the value of 3.27 calculated for FLB. A formal description of the revised 3-D FEA of the lower SG assembly and the Square Cell Model needs to be submitted for NRC staff evaluation before the NRC staff will be able to approve the 3.27 leakage factor as part of a permanent H* amendment. In addition, the NRC staff finds that both the loss coefficient database and the robustness of the assumed functional relationships between loss coefficient and contact pressure need further review before the NRC staff can approve a permanent H* amendment. However, the NRC staff finds that leakage is not a concern for the proposed period of the interim LAR for reasons discussed in Section 4.3.4 below.

The licensee provided the following commitment in Reference 1 that describes how the leakage factor will be used to satisfy TS 5.5.9.a for condition monitoring and TS 5.5.9.b.2 regarding performance criteria for accident induced leakage:

> *For the Condition Monitoring (CM) assessment, the component of operational leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.27 and added to the total accident leakage from any other source and compared to the allowable accident induced leakage limit. For the Operational Assessment (OA), the difference between the allowable accident induced leakage from sources other than the tubesheet expansion region will be divided by 3.27 and compared to the observed operational leakage. An administrative operational limit will be established to not exceed the calculated value. Applicable to Catawba Unit 2 End of Cycle 17 Refueling Outage and subsequent Cycle 18 operation.*

The NRC staff finds this license commitment acceptable, since it provides further assurance, in addition to the licensee's operational leakage monitoring processes, that accident-induced SG tube leakage will not exceed values assumed in the licensing bases accident analyses. The NRC staff finds that the leakage factor of 3.27 conservatively bounds the increase in leakage from locations below the H* distance that may be induced by accident conditions relative to leakage from the same locations under normal operating conditions during Catawba 2 Cycle 18 operations. The NRC staff has conditioned the implementation of the proposed LAR upon completion of this commitment.

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

The NRC staff has reviewed the proposed revised reporting requirements and finds that they, in conjunction with existing reporting requirements, are sufficient to allow the NRC staff to monitor the condition of the SG tubing. Based on this conclusion, the NRC staff finds that the proposed revised reporting requirements are in accordance with 10 CFR 50.36(c)(5) and are acceptable.

4.3.4 Technical Bases for Interim H* Amendment

The proposed H* value is based on the conservative assumption that all tubes in all SGs are severed at the H* location. This is a bounding, but necessary, assumption for purposes of supporting a permanent H* amendment because the tubes will not be inspected below the H* distance for the remaining life of the SGs, which may range up to 30 years or more from now depending on the plant, and because the tubes are susceptible to stress corrosion cracking below the H* distance. In addition, the proposed H* distance conservatively takes no credit for RCP associated with the tube hydraulic expansion process.

As discussed in Section 4.3.1.3, the NRC staff does not have sufficient information to support a conclusion on whether the newly calculated T/TS contact pressures and the corresponding H* distances (nominal values and probabilistic values) are conservative. In addition, as discussed in Section 4.3.2, the NRC staff does not have sufficient information at this time to support a conclusion regarding the conservatism of the proposed leakage factor of 3.27. Thus, in spite of the significant conservatisms embodied in the proposed H* distance, the NRC staff is unable to conclude at this time that the proposed H* distance is, on balance, conservative from the standpoint of ensuring that all tubes will retain acceptable margins against pullout (i.e., structural integrity) and acceptable accident leakage integrity for the remaining lifetime of the SGs, assuming all tubes to be severed at the H* location. However, the licensee is requesting an interim amendment that is applicable only to Catawba 2 during Refueling Outage 17 (fall 2010) and the subsequent Cycle 18 operation, rather than an amendment that is applicable to the remaining life of the plant. The NRC staff finds that assuming all tubes will be severed at the H* distance over the next operating cycle to be unrealistic and that the proposed H* distance is conservative for the next operating cycle for the reasons cited below.

From a fleet-wide perspective (for all Westinghouse plants with tubes fabricated from thermally treated Alloy 600), the NRC staff has observed from operating experience that the extent of cracking is at an early stage in terms of the number of tubes affected by cracking below the H* distance and the severity of cracks, compared to the idealized assumption that all tubes are severed at the H* distance. Most of these cracks occur in the lowest one inch of tubing, which is a region of relatively high residual stress associated with the 1 inch tack roll expansion in that region. Although the extent of cracking can be expected to increase with time, it is the NRC staffs judgment, based on experience, that it will continue to be limited to a small percentage of tubes, mostly near the tube ends, over the next operating cycle. The NRC staffs observations are based on the review of SG tube inspection reports from throughout the PWR fleet. These reports are reviewed and the NRC staffs conclusions are documented generally within 18 months of each SG tube inspection. Reference 27 provides a recent example of a review by the NRC staff.

In Reference 28, the licensee stated that during the most recent Catawba 2 refueling outage (spring 2009), 100 percent (17,952) of the inservice hot leg tubes were inspected for the full tubesheet depth, and 60 percent (10,776) of the cold leg inservice tubes were inspected for the full tubesheet depth. A total of 208 tube end indications were identified within 1.0 inch of the hot leg tube end and no indications were identified in the cold leg tube ends. Only six hot leg tubes were required to be plugged because Catawba 2 currently has an approved amendment with interim alternate repair criteria (IARC) for the SG tubes (Reference 15). All of the flaws were small and did not challenge structural or leakage integrity performance criteria. The total number of tube end flaws found was small compared to the number of tubes inspected. One tube in SG C was reported with a bulge just above the hot leg tubesheet, so the tube was plugged and removed from service. No other flaws were found above the tube end (and within the tubesheet) region during this inspection. The NRC staff finds the extent and severity of cracking at Catawba 2 to be limited and within the envelope of industry experience with similar units.

The NRC staff concludes that there is sufficient conservatism embodied in the proposed H* distances to ensure acceptable margins against tube pullout for Catawba 2 Operating Cycle 18 for the reasons discussed above. The revised LAR (Reference 2) for an H* distance of 20 rather than 16.95 inches (to within 1 inch of the tube end) provides added assurance that acceptable margins against tube pullout will be maintained during this interim period. The NRC staff also concludes there is reasonable assurance during the next operating cycle that any potential accident induced leakage will not exceed the technical specification performance criteria for accident induced leakage. This reflects current operating experience trends that cracking below the H* distance is occurring predominantly in the tack roll region near the bottom of the tube. At this location, it is the NRC staffs judgment that the total resistance to primary-to-secondary leakage will be dominated by the resistance of any "crevice" in the roll expansion region (due to very high T/TS contact pressures in this region), such that the leakage factors discussed in Section 4.3.2 will remain conservative.

4.3.5 Technical Evaluation Summary

The NRC staff finds that the proposed LAR acceptably addresses all issues identified by the NRC staff in Reference 10 relating to H* amendment requests submitted prior to 2008 (which were subsequently withdrawn). As discussed in section 4.3.1.3, the NRC staff does not have sufficient information to support a conclusion on whether the newly calculated T/TS contact pressures and the corresponding H* distances (nominal values and probabilistic values) are conservative. In addition, as discussed in Section 4.3.2, the NRC staff does not have sufficient information at this time to support a conclusion regarding the conservatism of the proposed leakage factor of 3.27. Thus, the NRC staff does not have an adequate basis to approve a permanent H* amendment. Accordingly, the licensee has requested an interim amendment that will only be applicable to Catawba 2 during the End of Cycle 17 Refueling Outage and the subsequent Cycle 18 operation.

The NRC staff concludes that, given the current state of the tubes, there is sufficient conservatism embodied in the proposed H* distances and leakage factor to ensure, for one operating cycle, that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses. Based on this finding, the NRC staff further concludes that the proposed amendment meets 10 CFR 50.36 and, thus, the proposed LAR is acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (75 FR 39977). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). 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.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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.

8.0 REFERENCES

1. J. R. Morris, Duke, letter to Document Control Desk (DCD), NRC, April 28, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML101730388). This letter also transmitted References 3, 4, and 5.

2. G. T. Hamrick, Duke, letter to DCD, NRC, September 9, 2010 (ADAMS Accession No. ML102560144). This letter also transmitted References 6 and 7.

3. WCAP-17072-P (Proprietary) and WCAP-17072-NP (Non-Proprietary), Rev. 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)," May 2009 (ADAMS Accession No. ML101730389 (Non- Proprietary)).

4. LTR-SGMP-09-100-P (Proprietary) and LTR-SGMP-09-100-NP (Non-Proprietary), "Response to NRC Request for Additional Information on H*; Model F and D5 Steam Generators," August 12, 2009 (ADAMS Accession No. ML101730391 (Non-Proprietary)).

5. LTR-SGMP-09-109-P (Proprietary) and LTR-SGMP-09-109-NP (Non-Proprietary) "Response to NRC Request for Additional Information on H*; RAI #4; Model F and D5 Steam Generators," August 25, 2009 (ADAMS Accession No. ML101730384 (Non-Proprietary)).

6. LTR-SGMP-10-95 NP-Attachment, Rev. 1 (Non-Proprietary), "H*: Alternate Leakage Calculation Methods for H* Situations When Contact Pressure at Normal Operating Conditions Exceeds Contact Pressure at Accident Conditions," September 9, 2010 (ADAMS Accession No. ML102560144 (Non-Proprietary)).

7. LTR-SGMP-10-107 NP-Attachment (Non-Proprietary), "Catawba Unit 2: Assessment of Probabilistic Value of H*," September 9, 2010 (ADAMS Accession No. ML102560144 (Non-Proprietary)).

8. T. J. Garrett, WCNOC, letter to DCD, U.S. NRC, ET 06-0004, "Revision to Technical Specification 5.5.9, "Steam Generator Tube Surveillance Program,"" February 21, 2006, (ADAMS Accession No. ML060600456).

9. T. J. Garrett, WCNOC, letter to DCD, U.S. NRC, ET 08-0010, "Withdrawal of License Amendment Request for a Permanent Alternate Repair Criteria in Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," February 14, 2008 (ADAMS Accession No. ML080580201).

10. B. K. Singal, NRC, letter to R. A. Muench, WCNOC, "Wolf Creek Generating Station – Withdrawal of License Amendment Request on Steam Generator tube Inspections," February 28, 2008 (ADAMS Accession No. ML080450185).

11. T. E. Tynan, SNC, letter to DCD, NRC, NL-07-1710, "Vogtle Electric Generating Plant Units 1 and 2 License Amendment Request to Technical Specification (TS) Sections TS 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report,"" November 30, 2007 (ADAMS Accession No. ML073380100).

12. T. E. Tynan, SNC, letter to DCD, NRC, NL-08-0148, "Vogtle Electric Generating Plant Units 1 and 2 License Amendment Request to Revise Technical Specification (TS) Sections TS 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report for Interim Alternate Repair Criterion,"" February 13, 2008 (ADAMS Accession No. ML080500223).

13. S. P. Lingam, NRC, letter to T. E. Tynan, SNC, "Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding Changes to Technical Specification (TS) Sections TS 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report,"" April 9, 2008 (ADAMS Accession No. ML080950247).

14. R. E. Martin, NRC, letter to T. E. Tynan, SNC, "Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding Steam Generator Tube Inspection Program," September 16, 2008 (ADAMS Accession No. ML082530044).

15. J. F. Stang, NRC, letter to J. R. Morris, Duke, "Catawba Nuclear Station, Unit 2, Issuance of Amendment Regarding Interim Alternate Repair Criterion for Stem Generator Tube Repair," April 13, 2009 (ADAMS Accession No. ML091030088).

16. A. B. Johnson, NRC, memorandum to A. L. Hiser, NRC, "Summary of the October 29 and 30, 2008, Category 2 Public Meeting with the Nuclear Energy Institute (NEI) and Industry to Discuss Modelling Issues Pertaining to the Steam Generator Tube-to-Tubesheet Joints" (ADAMS Accession No. ML083300422).

17. A. B. Johnson, NRC, memorandum to A. L. Hiser, NRC, "Summary of the January 9, 2009, Category 2 Public Meeting with the U.S. Nuclear Industry Representatives to Discuss Steam Generator H*/B* Issues," (ADAMS Accession No. ML090370945).

18. A. B. Johnson, NRC, memorandum to M. Gavrilas, NRC, "Summary of the April 3, 2009, Category 2 Public Meeting with the U.S. Nuclear Industry Representatives to Discuss Steam Generator H* Issues," April 30, 2009 (ADAMS Accession No. ML091210437).

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

20. M. J. Ajluni, SNC, letter to DCD, NRC, NL-09-1411, Vogtle Electric Generating Plant License Amendment Request to Revise Technical Specification (TS) Sections 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" for Interim Alternate Repair Criteria," September 11, 2009 (ADAMS Accession No. ML092540511).

21. J. H. Riley, NEI, letter to DCD, NRC, "H*/B* Expert Panel Technical Evaluation - Re-assessment of Coefficient of Thermal Expansion Data for SA-508 Steel," July 7, 2009 (ADAMS Accession No. ML082100086), transmitting Babcock and Wilcox Limited Canada letter 2008-06-PK-001, "Re-assessment of PMIC measurements for the determination of CTE of SA 508 steel," June 6, 2009 (ADAMS Accession No. ML082100097).

22. NRC, "Voltage Based Alternate Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," Generic Letter 95-05, August 3, 1995 (ADAMS Accession No. ML031070113).

23. NRC, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," NUREG-0844, September 1988 (ADAMS Accession No. ML082400710).

24. NRC, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," NUREG-1570, March 1998 (ADAMS Accession No. ML070570094).

25. M. J. Ajluni, SNC, letter to DCD, NRC, NL-09-1317, "Vogtle Electric Generating Plant Supplemental Information for License Amendment Request to Revise Technical Specification (TS) Sections 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" for Permanent Alternate Repair Criteria," August 28, 2009 (ADAMS Accession No. ML092450029 (Non-Proprietary)).

26. R. M. Taylor, NRC, memorandum to G. J. Kulesa, NRC, "Vogtle Electric Generating Plant – Audit of Steam Generator H* Amendment Reference Documents," July 9, 2010 (ADAMS Accession No. ML101900227).

27. J. H. Thompson, letter to J. R. Morris, Duke, "Catawba Nuclear Station, Unit 2 – Review of the 2009 Steam Generator (SG) Tube Inspections During Refueling Outage 16 End of Cycle," July 7, 2010 (ADAMS Accession No. ML101880047).

28. J. R. Morris, Duke, letter to DCD, NRC, July 14, 2009 (ADAMS Accession No. ML092010498).

Principal Contributors: A. Johnson
E. Murphy

Date: September 27, 2010

September 27, 2010

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 THE STEAM GENERATOR PROGRAM (TAC NO. ME4108)

Dear Mr. Morris:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 257 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 April 28, 2010, as supplemented by letter dated September 9, 2010.

The amendment revises Technical Specification (TS) 5.5.9 to exclude portions of the Steam Generator (SG) tube from periodic SG tube inspections and plugging or repair. In addition, reporting requirement changes are proposed to TS 5.6.8. This amendment is effective for one-cycle for the Catawba Nuclear Station, Unit 2, beginning at the End of Cycle 17 Refueling Outage and extending through subsequent Cycle 18 operation. Your application also requested deletion of a related license condition. The NRC staffs determination regarding deletion of the related license condition will be provided separately.

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 301-415-1119.

Sincerely,
/RA/
Jon Thompson, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-414

Enclosures:
1. Amendment No. 257 to NPF-52
2. Safety Evaluation

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ADAMS Accession No. ML102640537 ***transmitted via memo ML102660052**

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NAME	JThompson	SRohrer	RElliott	RTaylor*	STurk	GKulesa	JThompson
DATE	09/21/10	09/21/10	09/22/10	09/22/10	09/24/10	09/27/10	09/27/10

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