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

December 11, 2012

Mr. Matthew W. Sunseri
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: STEAM GENERATOR TUBE PERMANENT ALTERNATE REPAIR CRITERIA (TAC NO. ME8350)

Dear Mr. Sunseri:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 201 to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 29, 2012.

The amendment revises TS 5.5.9, "Steam Generator (SG) Program," to exclude portions of the tube below the top of the steam generator tubesheet from periodic steam generator tube inspections. In addition, the proposed amendment revises TS 5.6.10, "Steam Generator Tube Inspection Report," to remove reference to the previous interim alternate repair criteria and provide reporting requirements specific to the permanent alternate repair criteria.

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

Sincerely,

Balwant K. Singal for

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures:
1. Amendment No. 201 to NPF-42
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 201
License No. NPF-42

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

   A. The application for amendment to the Wolf Creek Generating Station (the facility) Renewed Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated March 29, 2012, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, 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;

   D. The issuance of this license 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 amended by 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-42 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 201, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented prior to MODE 4 entry during startup from Refueling Outage 19, which is currently scheduled to commence on February 4, 2013.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: December 11, 2012

ATTACHMENT TO LICENSE AMENDMENT NO. 201

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following pages of the Renewed Facility Operating License No. NPF-42 and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating License

REMOVE	INSERT
4	4

Technical Specifications

REMOVE	INSERT
5.0-12	5.0-12
5.0-13	5.0-13
5.0-28	5.0-28

(5) The Operating Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) The Operating Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission, now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Operating Corporation is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 201, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Kansas Gas & Electric Company and Kansas City Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Environmental Qualification (Section 3.11, SSER #4, Section 3.11, SSER #5)*

Deleted per Amendment No. 141.

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

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

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

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

1. Tubes with service-induced flaws located greater than 15.21 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 15.21 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. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The portion of the tube below 15.21 inches from the top of the tubesheet is excluded from this requirement. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

2. After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period.

b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and

c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

3. If crack indications are found in any portion of the SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). 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 a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

(continued)

5.6 Reporting Requirements

5.6.10 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG;

b. Degradation mechanisms found;

c. Nondestructive examination techniques utilized for each degradation mechanism;

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;

e. Number of tubes plugged during the inspection outage for each degradation mechanism;

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator;

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

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

i. The calculated accident induced leakage rate from the portion of the tubes below 15.21 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.50 times the maximum operational primary to secondary leak rate, the report should describe how it was determined; and

j. The results of monitoring for the tube axial displacement (slippage). If slippage is discovered, the implications of discovery and corrective action shall be provided.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By application dated March 29, 2012 (Reference 1), Wolf Creek Nuclear Operating Corporation (the licensee) submitted a license amendment request (LAR) for changes to the Technical Specifications (TSs) for Wolf Creek Generating Station (WCGS). Portions of the letter dated March 29, 2012, contain proprietary information and, accordingly, those portions are withheld from public disclosure.

The proposed changes would establish permanent steam generator (SG) tube alternate repair criteria for tubing flaws located in the lower region of the tubesheet. The proposed changes would replace similar criteria for WCGS on an interim basis during Refueling Outage 18 (RFO18) and the subsequent operating cycle, which was approved by the U.S. Nuclear Regulatory Commission (NRC) staff on April 6, 2011, in License Amendment No. 195 (Reference 2).

The proposed changes revise TS 5.5.9, "Steam Generator (SG) Program," to exclude portions of the tube below the top of the steam generator tubesheet from periodic steam generator tube inspections. In addition, the proposed amendment revises TS 5.6.10, "Steam Generator Tube Inspection Report," to remove reference to previous interim alternate repair criteria and provide reporting requirements specific to the permanent alternate repair criteria.

1.1 Background

WCGS has four Model F SGs, which were designed and fabricated by Westinghouse Electric Company LLC (Westinghouse). There are 5,626 Alloy 600 thermally treated (Alloy 600TT) tubes in each SG, each with a nominal outside diameter of 0.688 inches and a nominal wall thickness of 0.040 inches. The 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 (SCC) affecting the tubesheet region of Alloy 600TT tubing had been reported at any nuclear power plants in the United States.

In the fall of 2004, crack-like indications were found in tubes in the tubesheet region of Catawba Nuclear Station (Catawba) Unit 2. These crack-like indications were found in a tube overexpansion (OXP) that was approximately 7 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 near 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 is an approximately 1-inch long expansion at each tube end. The purpose of the tack expansion is 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 Unit 2 in the fall of 2004, other nuclear plants with Alloy 600TT tubing have found crack-like indications in tubes within the tubesheet as well; most of these indications were in the tack expansion region near the tube-end welds and were a mixture of axial and circumferential primary-water SCC.

Over time, these cracks can be expected to become more and more extensive, necessitating more extensive inspections of the lower tubesheet region and more extensive tube plugging or repairs, with attendant increased cost and the potential for shortening the useful lifetime of the SGs. To avoid these impacts, the affected licensees and their contractor, Westinghouse, have developed proposed alternative inspection and repair criteria applicable to the tubes in the lowermost region of the tubesheets. These criteria are referred to as the "H*" criteria. H* is the minimum engagement distance between the tube and tubesheet, measured downward from the top of the tubesheet, that is proposed as needed to ensure the structural and leakage integrity of the T/TS joints. The proposed H* amendment would exclude the portions of tubing below the H* distance from inspection and plugging requirements on the basis that flaws below the H* distance are not detrimental to the structural and leakage integrity of the T/TS joints.

Requests for permanent H* amendments were proposed for a number of plants as early as 2005. The NRC staff identified a number of issues with these early LARs and in subsequent LARs submitted in 2009, including an LAR for WCGS dated June 2, 2009 (Reference 3), and the staff was, therefore, unable to approve H* amendments on a permanent basis pending resolution of these issues. The NRC staff found it did have a sufficient basis to approve H* amendments on an interim (temporary) basis, based on the relatively limited extent of cracking existing in the lower tubesheet region at the time the interim amendments were approved. The technical basis for approving the interim amendments was provided in detail in the NRC staff's safety evaluations accompanying issuance of those amendments. Interim H* amendments were approved for WCGS as early as October 19, 2009 (Reference 4), and most recently on April 6, 2011 (Reference 2).

2.0 <u>REGULATORY EVALUATION</u>

The SG tubes are part of the reactor coolant pressure boundary (RCPB) and isolate fission products in the primary coolant from the secondary coolant and the environment. 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, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50 provide regulatory requirements that are applicable to WCGS 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" (GDCs 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing...to assess...structural and leaktight integrity" (GDC 32). The licensee discusses compliance with each of these GDC for WCGS in Section 3.1 of the Updated Final Safety Analysis Report (UFSAR) and does not identify any deviations from these GDC for SG tube-related issues.

In 10 CFR 50.55a, "Codes and standards," specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III, "Rules for Construction of Nuclear Facility Components," 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). The regulations in 10 CFR 50.55a further require that throughout the service life of pressurized-water reactor (PWR) facilities like WCGS, ASME Code Class 1 components meet the requirements of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 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 TSs.

In 10 CFR 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.

In 10 CFR 50.36(c)(5), "Administrative controls," are, "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 WCGS, the SG program requirements, including requirements for SG tube inspection and repair, are in TS 5.5.9, while the reporting requirements for the SG Program are in TS 5.6.10.

The TSs for all PWR plants require that an SG program be established and implemented to ensure that SG tube integrity is maintained. For WCGS, 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 by 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 alternate repair criteria provided in TS 5.5.9.c. WCGS TS 3.4.13, "RCS [Reactor Coolant System] Operational LEAKAGE," includes a limit on operational primary-to-secondary leakage of 150 gallons per day, which if exceeded, requires the plant to be promptly shut down. Should a flaw exceeding the tube repair limit not be detected during the periodic tube surveillance required by the plant TSs, the operational leakage limit provides added assurance of timely plant shutdown before tube structural and leakage integrity, consistent with the design and licensing bases, are impaired.

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 an SG tube rupture and a main steamline break (MSLB). These analyses consider primary-to-secondary leakage that may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR 100.11," Determination of exclusion area, low population zone, and population center distance," accident source term, GDC 19, "Control room," for control room operator doses (or some fraction thereof as appropriate to the accident), or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analyses for WCGS are being changed because of the proposed amendment. Since there are no changes to the accident analyses, there are no changes to the radiological consequences of any accidents. The use of the proposed alternate repair criteria does not impact the integrity of the SG tubes; therefore, the SG tubes 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 has reviewed and approved for the postulated DBAs for SG tubes.

By letter dated April 6, 2011 (Reference 2), the NRC approved Amendment No. 195 for WCGS. This amendment modified TS 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator Inspection Report," incorporating interim alternate repair criteria and associated tube inspection and reporting requirements that were applicable during RFO18, and the subsequent operating cycle. The proposed permanent amendment uses the same tube inspection and reporting requirements that were approved for the interim amendment, but allows these requirements to be used on a permanent basis. The alternate repair criteria (i.e., the H* distance) associated with the permanent amendment is the same criteria used in the prior interim amendment.

3.0 TECHNICAL EVALUATION

3.1 Proposed Changes to the TS

Current TS 5.5.9.c.1 states that

> For Refueling Outage 18 and the subsequent operating cycle, tubes with service-induced flaws located greater than 15.2 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 15.2 inches below the top of the tubesheet shall be plugged upon detection.

Revised TS 5.5.9.c.1 would state that

> Tubes with service-induced flaws located greater than 15.21 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 15.21 inches below the top of the tubesheet shall be plugged upon detection.

Current TS 5.5.9.d states, in part, that

> Refueling Outage 18 and the subsequent operating cycle, the portion of the tube below 15.2 inches from the top of the tubesheet is excluded from this requirement.

Revised TS 5.5.9.d would state, in part, that

> The portion of the tube below 15.21 inches from the top of the tubesheet is excluded from this requirement.

Current TS 5.6.10.h states that

> Following completion of an inspection performed in Refueling Outage 18 (and any inspections performed in the subsequent operating cycle) the primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report;

Revised TS 5.6.10.h would state that

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

Current TS 5.6.10.i states, in part, that

> Following completion of an inspection performed in Refueling Outage 18 (and any inspections performed in the subsequent operating cycle) the calculated accident induced leakage rate from the portion of the tubes below 15.2 inches from the top of the tubesheet for the most limiting accident in the most limiting SG.

Revised TS 5.6.10.i would state, in part, that

> The calculated accident induced leakage rate from the portion of the tubes below 15.21 inches from the top of the tubesheet for the most limiting accident in the most limiting SG.

Current TS 5.6.10.j states, in part, that

> Following completion of an inspection performed in Refueling Outage 18 (and any inspections performed in the subsequent operating cycle) the results of monitoring for tube axial displacement (slippage).

Revised TS 5.6.10.j would state, in part, that

> The results of monitoring for tube axial displacement (slippage).

3.2 NRC Staff 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 load that is transmitted to the T/TS weld. 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 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 multiplied by 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 or pullout of the tube from the tubesheet, assuming the tube is fully severed at the H* distance below the top of the tubesheet. For WCGS, the licensee has proposed an H* distance of 15.21 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 and licensing basis. This includes maintaining structural safety margins consistent with the structural performance criteria in TS 5.5.9.b.1 and the design basis, as is discussed in SE Section 3.2.1.1. In addition, this includes limiting the potential for accident-induced primary-to-secondary leakage to values not exceeding the accident-induced leakage performance criteria in TS 5.5.9.b.2, which are consistent with values assumed in the licensing basis accident analyses. Maintaining tube integrity in this manner ensures that the amended TSs are in compliance with all applicable regulations. The NRC staff's evaluation of joint structural integrity and accident-induced leakage integrity is discussed in SE Sections 3.2.1 and 3.2.2, respectively.

3.2.1 Joint Structural Integrity

3.2.1.1 Acceptance Criteria

Westinghouse has conducted extensive analyses to establish the necessary H* distance to resist pullout under normal operating and DBA conditions. Based on the physical geometry of the SG tubesheet, the NRC staff finds that pullout is the structural failure mode of interest, since the tubes are radially constrained against axial 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 3 against pullout under normal operating conditions and a factor of 1.4 against pullout under DBA conditions. The NRC staff concludes that these are the appropriate safety factors to apply to demonstrate structural integrity, because they 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; namely the stress limit criteria in the ASME Code, Section III.

The above approach equates tube pullout to gross structural failure which is conservative. Should the pullout load be exceeded, tube slippage would be limited by the presence of adjacent tubes and support structures such that the tube would not be expected to pull out of the tubesheet.

The licensee has committed in Reference 1 to monitor for tube slippage as part of the SG inspection program. Under the proposed license amendment, TS 5.6.10.j will require that the results of slippage monitoring be included as part of the 180-day report, which is required by TS 5.6.10. In addition, TS 5.6.10.j requires that should slippage be discovered, the implications of the discovery and corrective action shall be included in the report. The NRC staff concludes that slippage is not expected to occur for the reasons discussed in this SE. However, in the unexpected event slippage does occur, it will be important to understand why it occurred so that the need for corrective action can be evaluated. The NRC staff concludes that the commitment to monitor for slippage and the accompanying reporting requirements are acceptable.

3.2.1.2 Three-dimensional Finite Element Analysis

A detailed three-dimensional (3-D) finite element analysis (FEA) of the lower SG assembly (consisting of the lower portion of the SG shell, the tubesheet, the channel head, and the divider plate that separates the hot- and cold-leg inlet plenums inside the channel head) was performed to calculate tubesheet displacements 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. The calculated tubesheet displacements were used as input to the T/TS interaction analysis evaluated in SE Section 3.2.1.3.

The tubesheet bore holes were not explicitly modeled. Instead, the tubesheet was modeled as a solid structure with equivalent material property values selected such that the solid model exhibited the same stiffness properties as the actual perforated tubesheet. This is a classical approach for analyzing perforated plates that the NRC staff concludes is acceptable.

Two versions of the 3-D FEA model were used to support the subject LAR, a "reference model" documented in Westinghouse WCAP-17071-P, Revision 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," April 2009 (Reference 5), which was submitted to support a previous request dated June 2, 2009 (Reference 3), for a permanent H* amendment for WCGS and a "revised model" described in the technical support document, Westinghouse WCAP-17330-P, Revision 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)," November 2010 (Reference 6). The revised model was submitted with a permanent H* amendment request from Duke Energy Carolinas, LLC for Catawba Unit 2. The revised model described in Reference 6 is applicable to the WCGS SGs. The reference 3-D FEA model was used to provide displacement input to the thick-shell T/TS interaction model described in SE Section 3.2.1.3.1. The revised 3-D FEA model was used to provide displacement input to the square cell T/TS interaction model described in SE Section 3.2.1.3.2. The revised 3-D model employs a revised mesh near the plane of symmetry (perpendicular to the divider plate) to be consistent with the geometry of the square cell model such that the displacement output from the 3-D model can be applied directly to the edges of the square cell model.

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. In the reference model (Reference 5), a factor was applied to the 3-D FEA results to account for a non-functional divider plate, based on earlier sensitivity studies. The revised 3-D FEA model in Reference 6 assumes the upper 5 inches of the divider plate to be non-existent. The NRC staff concludes that this further improves the accuracy of the 3-D FEA for the assumed condition of a non-functional divider plate and, therefore, is acceptable.

3.2.1.3 T/TS Interaction Model

3.2.1.3.1 Thick-shell Model

The licensee stated in its LAR that resistance to tube 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. In the reference analysis (Reference 5), 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). Calculated displacements from the 3-D FEA of the lower tubesheet assembly (see SE Section 3.2.1.2) were applied to the thick-shell model as input to account for the increment of tubesheet bore diameter change caused by 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. However, the tubesheet bore diameter change from the 3-D FEA tended to be non-uniform (eccentric) around the bore circumference. The thick-shell equations used in the T/TS interaction model are axisymmetric. Thus, the non-uniform diameter change from the 3-D FEA had to be adjusted to an equivalent uniform value before it could be used as input to the T/TS interaction analysis. A two-dimensional (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 finite element analyses.

In Reference 5, Westinghouse identified a difficultly in applying this relationship to Model D5 SGs under MSLB conditions. In reviewing the reasons for this difficulty, the NRC staff developed questions relating to the conservatism of the relationship and whether the tubesheet bore displacement eccentricities are sufficiently limited such as to ensure that T/TS contact is maintained around the entire tube circumference. This concern was applicable to all SG models with Alloy 600TT tubing. By letter dated November 23, 2009 (Reference 7), the NRC staff documented a list of questions that would need to be addressed satisfactorily before the NRC staff would be able to approve a permanent H* amendment for Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle). These questions related to the technical justification for the eccentricity adjustment, the distribution of contact pressure around the tube circumference, and a new model under development by Westinghouse to address the aforementioned issue encountered with the Model D5 SGs.

On June 14 and 15, 2010, the NRC staff conducted an audit at the Westinghouse Waltz Mill Site and provided a summary of the audit in a memorandum dated July 9, 2010 (Reference 8). The purpose of the audit was to gain a better understanding of the H* analysis pertaining to eccentricity, to review draft responses to the NRC staff's questions in Reference 7, and to determine which documents would need to be provided on the docket to support any future requests for a permanent H* amendment. Based on the audit, including review of pertinent draft responses to the questions in Reference 7, the NRC staff concluded that eccentricity does not appear to be a significant variable affecting either average T/TS contact pressure at a given elevation or calculated values of H*. The NRC staff found that average contact pressure at a given elevation is primarily a function of average bore diameter change at that elevation associated with the pressure and temperature loading of the tubesheet. Accordingly, the NRC staff concluded that no adjustment of computed average bore diameter change considered in the thick-shell model is needed to account for eccentricities computed by the 3-D FEA. The material reviewed during the audit revealed that computed H* values from the reference analyses continued to be conservative when the eccentricity adjustment factor is not applied.

3.2.1.3.2 Square Cell Model

Documentation for the square cell model is contained in Reference 6, and the licensee's response to the NRC staff request for additional information (RAI) regarding Reference 6 is included with the subject LAR for WCGS (Reference 1). The square cell model is a 2-D plane stress FEA model of a single square cell of the tubesheet with a bore hole in the middle and each of the four sides of the cell measuring one tube pitch in length. Displacement boundary conditions are applied at the edges of the cell, based on the displacement data from the revised 3-D FEA model. The model also includes the tube cross-section inside the bore. Displacement compatibility between the tube outer surface and bore inner surface is enforced except at locations where a gap between the tube and bore tries to occur.

The square cell model was originally developed in response to the above-mentioned difficulty encountered when applying the eccentricity adjustment to Model D5 SGs T/TS interaction analysis under MSLB conditions using the thick-shell model. Early results with this model indicated significant differences compared to the thick-shell model, irrespective of whether the eccentricity adjustment was applied to the thick-shell model. The square cell model revealed a fundamental problem with how the results of the 3-D FEA model of the lower SG assembly were being applied to the tubesheet bore surfaces in the thick-shell model. As discussed in SE Section 3.2.1.2, the perforated tubesheet is modeled in the 3-D FEA model as a solid plate whose material properties were selected such that the gross stiffness of the solid plate is equivalent to that of a perforated plate under the primary-to-secondary pressure acting across the thickness of the plate. This approach tends to smooth out the distribution of tubesheet displacements as a function of radial and circumferential location in the tubesheet, and ignores local variations of the displacements at the actual bore locations. These smoothed-out displacements from the 3-D FEA results were the displacements applied to the bore surface locations in the thick-shell model. The square cell model provides a means for post-processing the 3-D FEA results such as to account for localized variations of tubesheet displacement at the bore locations as part of T/TS interaction analysis. Based on these findings, square cell models were developed for each of the SG model types, including the Model F SGs at WCGS.

The square cell model is applied to nine different elevations, from the top to the bottom of the tubesheet, for each tube and loading case analyzed. The square cell slices at each elevation are assumed to act independently of one another. T/TS contact pressure results from each of the nine slices are used to define the contact pressure distribution from the top to the bottom of the tubesheet.

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) multiplied by both the tubesheet bore surface area (equal to the tube outer diameter surface area over the incremental axial distance) and the coefficient of friction. The NRC staff reviewed the coefficient of friction used in the analysis and concludes that it is 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 SE Section 3.2.1.1.

The square cell model 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 residual contact pressure associated with the hydraulic expansion process. Based on the licensee's response to the NRC staff RAI regarding Reference 6, which is included with the subject LAR for WCGS (Reference 1), the NRC staff considers that the assumption of zero residual contact pressure in all tubes is conservative.

The limiting tube locations in terms of H* were determined during the reference analysis to lie along the plane of symmetry perpendicular to the divider plate. The outer edges of the square cell model conform to the revised mesh pattern along this plane of symmetry in the 3-D FEA model of the lower SG assembly, as discussed in SE Section 3.2.1.2. Because the tubesheet bore holes were not explicitly modeled in the 3-D FEA, only the average displacements along each side of the square cell are known from the 3-D FEA. Three different assumptions for applying displacement boundary conditions to the edges of the square cell model were considered to allow for a range of possibilities about how local displacements might vary along the length of each side. The most conservative assumption, in terms of maximizing the calculated H* distance, was to apply the average transverse displacement uniformly over the length of each edge of the square cell.

Primary pressure acting on the tube inside surface, and crevice pressure[1] acting on both the tube outside surface and tubesheet bore surface, are not modeled directly as in the case of the thick-shell model. Instead, the primary side (inside) of the tube is assumed to have a pressure equal to the primary pressure minus the crevice pressure. The crevice pressure varies as a function of the elevation being analyzed, as discussed in SE Section 3.2.1.4.

The NRC staff concludes that the square cell model provides for improved compatibility between the 3-D FEA model of the lower SG assembly and the T/TS interaction model, more realistic and accurate treatment of the T/TS joint geometry, and added conservatism relative to the thick-shell model used in the reference analyses.

3.2.1.4 Crevice Pressure Evaluation

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 concludes 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 concludes 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 at each axial location of the T/TS interaction model and are concluded to be acceptable for this purpose by the NRC staff.

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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 SE Section 3.2.1.4 for more information.

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 guess 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 initial estimate for the next H* iteration.

3.2.1.5 H* Calculation Process

The calculation of H* consists of the following steps for each loading case considered:

1. Perform initial H* estimate (mean H* estimate) using the T/TS interaction model and 3-D FEA 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 contact pressure distribution over the length of the T/TS crevice. Two sets of mean H* estimates are pertinent to the proposed H* value, mean H* estimates calculated with the reference T/TS interaction and 3-D FEA models (Reference 5) and mean H* estimates calculated with the square cell T/TS interaction and revised 3-D FEA models (Reference 6). The highest calculate mean H* estimate (for the most limiting tube) from the reference analysis is 5.23 inches, for the most limiting case of normal operating conditions (with the associated factor of safety of 3 as evaluated in SE Section 3.2.1.1). This estimate includes the adjustments in items 2 and 3 below. The highest calculated mean H* estimate with the square cell T/TS interaction model, in conjunction with the revised 3-D lower SG FEA model, is 8.66 inches. The most limiting loading case for this revised analysis is still the normal operating condition (with its associated factor of safety of 3.0). The NRC staff concludes that the difference in mean H* estimates between the reference analysis and the revised analysis is due primarily to the improved post-processing of the 3-D FEA model displacements for application to the T/TS interaction model.

2. In the reference analysis (Reference 5), a 0.3-inch adjustment was added 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 for Model F SGs conducted by Westinghouse. This adjustment is not included in the revised H* analysis accompanying the subject LAR, as discussed and evaluated in SE Section 3.2.1.5.1.

3. In the reference analysis (Reference 5), for normal operating conditions only, an additional adjustment was added to the initial H* estimate to correct for the actual temperature distribution in the tubesheet compared to the linear distribution assumed in the reference 3-D FEA analysis. This adjustment is no longer necessary, as discussed in SE Section 3.2.1.2, since the temperature distributions throughout the tubesheet were calculated directly in the revised 3-D FEA supporting the current request for a permanent H* amendment.

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, associated with the potential variability of key input parameters for the H* analyses. This leads to a "probabilistic" estimate of H*, which includes the mean estimate. The NRC staff's evaluation of the probabilistic analysis is provided in SE Sections 3.2.1.6 and 3.2.1.7.

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. This step is discussed and evaluated in SE Section 3.2.1.5.2.

6. A new step, Step 6, has been added to the H* calculation process since the Reference 5 analysis to support the subject permanent amendment request. This step involves adding an additional adjustment to the probabilistic estimate of H* to account for the Poisson contraction of the tube radius due to the axial end-cap load acting on each tube. This step is discussed and evaluated in SE Section 3.2.1.5.3.

3.2.1.5.1 BET Considerations

The diameter of each tube transitions from its fully expanded value to its unexpanded value near the top of the tubesheet. The BET region is located a short distance below the top of tubesheet in order to avoid any potential for over-expanding the tube above the top of the tubesheet. In the reference H* analysis (Reference 5), a 0.3-inch adjustment was added to the mean H* estimate to account for the BET location being below the top of the tubesheet based on an earlier survey of BET distances conducted by Westinghouse. This adjustment was necessary since the reference analysis did not explicitly account for the lack of contact between the tube and tubesheet over the BET distance.

BET measurements, based on eddy current testing, have subsequently been performed for all tubes at WCGS. These measurements confirmed that the original 0.3-inch BET assumption is bounding on a 95th percentile basis; and that the maximum value at WCGS was 0.66 inches.

However, the most recent H* analyses using the square cell T/TS interaction model (Reference 6) has made the need for a BET adjustment unnecessary, as the square cell model shows a loss of contact pressure at the top of the tubesheet that is greater than the possible variation in the BET location. The loss of contact pressure at the top of the tubesheet shown in the square cell model (which is unrelated to BET location) is compensated for by a steeper contact pressure gradient than was shown previously in the thick-shell model H* analysis. The NRC staff concludes that the proposed H* value adequately accounts for the range of BET values at WCGS.

3.2.1.5.2 Crevice Pressure Adjustment

As discussed in SE Section 3.2.1.5, Steps 1 through 4 of the H* calculation process leading to a probabilistic H* estimate are performed with the assumption that the tube is severed at the bottom of the tubesheet for purposes of calculating the distribution of crevice pressure as a function of elevation. If the tube is assumed to be severed at the initially computed H* distance and Steps 1 through 4 repeated, a new H* may be calculated which will be incrementally larger than the first estimate. This process may be repeated until the change in H* becomes small (convergence). Sensitivity analyses conducted with the thick-shell model showed that the delta between the initial H* estimate and final (converged) estimate is a function of the initial estimate for the tube in question. This delta (i.e., the crevice pressure adjustment referred to in Step 5 of SE Section 3.2.1.5) was plotted as a function of the initial H* estimate for the limiting loading case and tube radial location. Although the sensitivity study was conducted with the thick-shell model, the deltas from this study were used in the Reference 6 (square cell model) analysis to make the crevice pressure adjustment to H*.

Updating this sensitivity study would have been very resource intensive, requiring many new 2-D FEA square cell runs. In response to an NRC staff question as to whether it is conservative to rely on the existing sensitivity study, as opposed to updating it to reflect the square cell model, Westinghouse submitted an analysis (Enclosure to Reference 1) demonstrating that if the sensitivity study were updated, it would show that the crevice pressure adjustment H* is negative, not positive as is shown by the existing study. This is because the square cell model predicts a much longer zone (approximately 4 inches) of no T/TS contact below the top of the tubesheet than does the thick-shell model. Therefore, the crevice pressure must reduce from primary side pressure (at the iterative H* location) to secondary side pressure approximately 4 inches below the top of the tubesheet.

This leads to higher predicted pressure differentials across the tube wall over the iterative H* distance than exists during the initial iteration, when crevice pressure is initially assumed to vary from primary pressure at the very bottom of the tubesheet to secondary pressure at the very top of the tubesheet. Based on its review of the Westinghouse analysis, the NRC staff concludes that the positive crevice pressure adjustment to H* in the Reference 6 analysis, which is based on the existing sensitivity study, is conservative and that an updated sensitivity analysis based on use of the square cell model would show that a negative adjustment can actually be justified. Thus, the NRC staff concludes the crevice pressure adjustment performed in support of the proposed H* amendment is conservative and acceptable.

3.2.1.5.3 Poisson Contraction Effect

The axial end-cap load acting on each tube is equal to the primary-to-secondary pressure difference multiplied by the tube's cross-sectional area. For purposes of resisting tube pullout under normal and accident conditions, the end-cap loads used in the H* analyses are based on the tubesheet bore diameter, which the NRC staff finds to be a conservative assumption. The axial end-cap load tends to stretch the tube in the axial direction, but causes a slight contraction in the tube radius due to the Poisson's Ratio effect. This effect, by itself, tends to reduce the T/TS contact pressure and, thus, to increase the H* distance. The axial end cap force is resisted by the axial friction force developed at the T/TS joint. Thus, the axial end cap force begins to decrease with increasing distance into the tubesheet, reaching zero at a location before the H* distance is reached. This is because the H* distances are intended to resist pullout under the end-cap loads with the appropriate factors of safety applied as discussed in SE Section 3.2.1.1.

A simplified approach was taken to account for the Poisson radial contraction effect. First, thick-shell equations were used to estimate the reduction in contact pressure associated with application of the full end-cap load, assuming none of this end-cap load has been reduced by friction between the tube and the tubesheet. The T/TS contact pressure distributions determined in Step 4 of the H* calculation process in SE Section 3.2.1.5 were reduced by this amount. Second, the friction force associated with these reduced T/TS contact pressures were integrated with distance into the tubesheet, and the length of engagement necessary to react one times the end-cap loading (i.e., no safety factor applied) was determined. At this distance (termed attenuation distance by Westinghouse), the entire end-cap loading was assumed to have been reacted by friction with the tubesheet, and the axial load in the tube below the attenuation distance was assumed to be zero. Thus, the T/TS contact pressures below the attenuation distance were assumed to be unaffected by the Poisson radial contraction effect. Finally, a revised H* distance was calculated, where the T/TS contact pressures from Step 4 of SE Section 3.2.1.5 were reduced only over the attenuation distance.

Regarding the safety factor of unity assumption, Westinghouse stated that it is unrealistic to apply a safety factor to a physical effect such as Poisson's ratio. The NRC staff has not reached a conclusion on this point; however, irrespective of whether a safety factor is applied to the Poisson's contraction effect (consistent with SE Section 3.2.1.1), the NRC staff concludes that there is ample conservatism embodied in the proposed H* distance to accommodate the difference. The NRC staff concludes that the simplified approach for calculating the H* adjustment for the Poisson contraction effect contains significant conservatism relative to a more detailed approach and is therefore acceptable.

3.2.1.6 Acceptance Standard - Probabilistic Analysis

The purpose of the probabilistic analysis is to develop an H* distance that ensures with a probability of 95 percent that the population of tubes will retain margins against pullout consistent with criteria evaluated in SE Section 3.2.1.1, 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, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," dated August 3, 1995 (Reference 9), employs a consistent criterion. The NRC staff also notes that use of the 95 percent 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 NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," September 1988 (Reference 10), and NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," March 1998 (Reference 11).

In terms of the confidence level that should be attached to the 95 percent 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 NRC staff has been encouraging the industry to revise its guidelines to call for calculating such probabilities at a 95 percent confidence level when performing operational assessments and a 50 percent confidence level when performing condition monitoring; see the summary of the January 8, 2009, NRC public meeting with the Nuclear Energy Institute and industry to discuss steam generator issues dated February 6, 2009 (Reference 12). In the meantime, the calculated H* distances supporting the subject amendment request have been evaluated at the 95 percent confidence level, as recommended by the NRC staff.

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 Feedline Break (FLB), the NRC staff and licensee agree that the tube population in the faulted SG is of interest, since it is the only SG 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 SE Section 3.2.1.1. 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, the calculated H* distances for normal operating conditions supporting the requested interim amendment are 95 percent probability/ 95 percent confidence estimates based on the entire tube population for the plant, consistent with the NRC staff's recommendation.

Based on the above, the NRC staff concludes that the proposed H* distance in the subject LAR is based on acceptable probabilistic acceptance standards evaluated at acceptable confidence levels.

3.2.1.7 Probabilistic Analyses

3.2.1.7.1 Reference Analyses

Sensitivity studies were conducted during the reference analyses (Reference 5) 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 the reference H* analyses (Reference 5) for residual contact pressure 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 (a measure of stiffness) 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 its engineering judgment, the NRC staff finds 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 residual contact pressure 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 SE Section 3.2.1.5), 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 were also created.

Two types of probabilistic analyses were performed independently in the reference analyses. One was a simplified statistical approach utilizing a "square root of the sum of the squares" method and the other was a detailed statistical (Monte Carlo) sampling approach. The NRC staff's review of the reference analysis relied on the Monte Carlo analysis, which provides the most realistic treatment of uncertainties. The NRC staff reviewed the implementation of probabilistic analyses in the reference analyses and questioned whether the H* influence curves had been conservatively treated. To address this concern, new H* analyses were performed as documented in Westinghouse LTR-SGMP-09-100-P, "Response to NRC Request for Additional Information on H*; Model F and D5 Steam Generators," dated August 12, 2009 (Reference 13), and Westinghouse LTR-SGMP-09-104-P, "White Paper on Probabilistic Assessment of H*," dated August 13, 2009 (Reference 14). These analyses made direct use of the H* influence curves in a manner the NRC staff concludes is acceptable.

The revised reference analyses in Reference 13 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 initial value of H* (based on nominal values of material and geometric input parameters) 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 revised reference analyses also addressed a question posed by the NRC staff 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 95 percent probability value at the desired level of confidence for a single SG tube population or all SG population, as appropriate. The NRC staff concludes that this approach addresses the staff's question in a realistic fashion and is, therefore, acceptable.

The reference analyses in References 5 and 13 indicated normal operating conditions (with associated safety factor of 3) to be the limiting case for determining H* for Model F SGs. As discussed earlier in SE Section 3.2.1.5, subsequent analyses with the more accurate square cell model and revised 3-D FEA model (due to the improved displacement compatibility between the two models) show that normal operating conditions (with associated safety factor of 3) to still be the limiting case for the Model F SGs.

3.2.1.7.2 Revised Analyses to Reflect Square Cell and Revised 3-D FEA Models

New Monte Carlo analyses using the square cell model to evaluate the statistical variability of H* due to the CTE variability for the tube and tubesheet materials were not performed. This was because such an approach would have been extremely resource intensive and because a simpler approach involving good approximation was available. The simplified approach involved using the results of the Monte Carlo analyses from the reference analysis, which are based on the thick-shell T/TS interaction model, to identify CTE values for the tube and tubesheet associated with the probabilistic H* values near the desired rank ordering. Tube CTE values associated with the upper 10 percent rank order estimates are generally negative variations from the mean value whereas tubesheet CTE values associated with the higher ranking order estimates are generally positive variations from the mean value. For the upper 10 percent of the Monte Carlo results ranking order, a combined uncertainty parameter, "alpha," was defined as the square root of the sum of the squares of the associated tube and tubesheet CTE values for each Monte Carlo sample. Alpha was plotted as a function of the corresponding H* estimate and separately as a function of rank order. Each of these plots exhibited well defined "break lines," representing the locus of maximum H* estimates and maximum rank orders associated with a given value of alpha. From these plots, three paired sets of tube and tubesheet CTE values, located near the break line, were selected. One of these pairs was for the rank order corresponding to an upper 95 percent probability and 95 percent confidence value for H* on a whole plant basis, which the NRC staff finds is appropriate for normal operating conditions (see SE Section 3.2.1.6). These CTE values were then input to the lower SG assembly 3-D FEA model and the square cell model to yield probabilistic H* estimates which approximate the H* values for these same rank orderings, had a full Monte Carlo been performed with the square cell and revised 3-D FEA models. These H* estimates were then plotted as a function of rank ordering, allowing the interpolation of H* values at the other rank orders. The resulting 95/95 upper bound H* estimate is 14.04 inches, which compares to the mean estimate of 8.66 inches as discussed in SE Section 3.2.1.5. With adjustments for Poisson's contraction (see SE Section 3.2.1.5.3) and crevice pressure (SE Section 3.2.1.5.2), the final 95/95 upper bound H* estimate is 15.21 inches.

The NRC staff judges that the above break line approach to be a very good approximation of what an actual Monte Carlo would show. A perfect approximation would mean that if, hypothetically, one were to perform a square cell analysis for each paired set of tube and tubesheet CTE values associated with the top 10 percent of rank orders, and plot the resulting H* values versus the original rank ordering associated with the CTE couple, the calculated H* values should monotonically increase from rank order to rank order. Westinghouse performed additional square cell analyses with CTE pairs for five consecutive rank orders for both Model D5 and Model F SGs. The results showed deviations from monotonically increasing values of H* with rank order to be on the order of only 0.3 inches for the Model D5 SGs and 0.1 inches for the Model F SGs. The NRC staff concludes that use of the break line approach adds little imprecision to the probabilistic H* estimates and is acceptable.

3.2.1.8 Coefficient of Thermal Expansion

During operation, a large part of contact pressure in an SG T/TS joint is derived from the difference in CTE between the tube and tubesheet. As discussed in SE Section 3.2.1.7, 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. By letter dated February 28, 2008, to the licensee (Reference 15), 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 16). 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 degrees Fahrenheit (°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 was 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 database. 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 1300 °F, 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 was combined with the ANL data that was found to be acceptable, and the 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 that was only present at low temperature. The CTE report is included as Appendix A to Reference 5.

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 concludes that this is 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 fully responsive to the concerns stated in Reference 15 and are, therefore, acceptable.

3.2.2 Leakage Considerations

Operational leakage integrity is assured by monitoring primary-to-secondary leakage relative to the applicable TS limiting condition for operation 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 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, the 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 differential during the accident to the normal operating condition pressure differential. The second quantity is the ratio of primary coolant viscosity at normal operating temperature to primary coolant viscosity at accident condition temperature. The third quantity is the ratio of crevice length under normal operating conditions to crevice length under accident condition. This third ratio equals 1, provided it can be shown that positive contact pressure is maintained along the entire H* distance for both normal operating and worst case accident condition. 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 is not known, Westinghouse has assumed that the loss coefficient is constant with contact pressure, as thus the ratio is equal to 1. The NRC staff finds 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.

Leakage factors were calculated for DBAs 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, 2.50, meaning that it is the transient expected to result in the largest increase in leakage relative to normal operating conditions.

The latest H* analyses by Westinghouse (Reference 6) continued to show an increasing T/TS contact pressure when going from normal operating to MSLB conditions. The new analyses used the revised 3-D finite element model of the lower SG assembly and the new square cell model, discussed in SE Section 3.2.1.3.2.

In Reference 1, the licensee provided a commitment describing 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 2.50 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the Operational Assessment (OA), the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.50 and compared to the observed operational leakage. An administrative limit will be established to not exceed the calculated value.

Details of how condition monitoring and operational assessments are performed are generally not included as part of the operating license, including the TSs. Extensive industry guidance on conducting condition monitoring and operational assessments is available as part of the industry NEI 97-06 (Rev. 3) initiative, "Steam Generator Program Guidelines," January 2011 (Reference 17). The above commitment helps ensure that plant procedures address the above leakage factor issue as well as industry guidelines.

The LAR includes reporting requirements (TS 5.6.10.h and 5.6.10.i) relating to operational leakage existing during the cycle preceding each SG inspection and condition monitoring assessment, and the associated potential for accident-induced leakage from the lower portion of the tubesheet below the H* distance. These reporting requirements will allow the NRC staff to monitor how the leakage factor is actually being used and are, therefore, acceptable.

The licensee provided another commitment in Reference 1 that states it will monitor for tube slippage as part of its SG Program.

> WCNOC will monitor for tube slippage as part of the steam generator tube inspection program. The results of this monitoring will be included in the report required by TS 5.6.10j.

This commitment will allow the NRC staff to monitor the results of the slippage inspections and is, therefore, acceptable.

3.3 Summary and Conclusions

Since the initial proposal for a permanent H* amendment in 2005, the supporting technical analyses have undergone substantial revision and refinement to address NRC staff questions and issues. The current analyses supporting the proposed permanent amendment still embody uncertainties and issues (e.g., should a factor of safety be applied to the Poisson's contraction effect) as discussed throughout this SE. However, it is important to acknowledge that there are significant conservatisms in the analyses. Some examples, also discussed elsewhere in this SE, include taking no credit for residual contact pressures associated with the hydraulic tube expansion process, the assumed value of 0.2 for coefficient of friction between the tube and tubesheet, and taking no credit for constraint against pullout provided by adjacent tubes and support structures. The NRC staff has evaluated the potential impact of the uncertainties and concludes that these uncertainties are adequately bounded by the significant conservatism within the analyses and proposed H* distance.

Based on the above, the NRC staff concludes that the proposed changes to the WCGS TSs ensure 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, without undue risk to public health and safety. Based on this conclusion, the NRC staff further concludes that the proposed amendment meets 10 CFR 50.36 and, therefore, is acceptable.

4.0 REGULATORY COMMITMENTS

In Attachment V to Reference 1, the licensee made the following regulatory commitments, to be implemented with the amendment, as discussed above in SE Section 3.2.2:

1. For the condition monitoring (CM) assessment, the component of operational leakage from the prior cycle from below the H* distance will be multiplied by a factor of 2.50 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the Operational Assessment (OA), the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.50 and compared to the observed operational leakage. An administrative limit will be established to not exceed the calculated value.

2. WCNOC will monitor for tube slippage as part of the steam generator tube inspection program. The results of this monitoring will be included in the report required by TS 5.6.10j.

The NRC staff considered these regulatory commitments as part of its review of the LAR and considers them acceptable. As stated earlier, the details of the condition monitoring and operational assessments are generally not included as part of the operating license, including the TSs.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Kansas 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 installation or use of a facility component 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 published in the *Federal Register* on July 3, 2012 (77 FR 39525). 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) there is reasonable assurance that 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. Broschak, J. P., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator Tube Inspection Report," for a Permanent Alternate Repair Criteria," dated March 29, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12102A080).

2. Hall, J. R., U.S. Nuclear Regulatory Commission, letter to Matthew W. Sunseri, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station - Issuance of Amendment Re: Changes To Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator Tube Inspection Report" (TAC No. ME5121)," dated April 6, 2011 (ADAMS Accession No. ML110840590).

3. Garrett, T. J., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Revision to Technical Specification 5.5.9, "Steam Generator

(SG) Program," and TS 5.6.10, "Steam Generator Tube Inspection Report," for a Permanent Alternate Repair Criterion," dated June 2, 2009 (ADAMS Accession No. ML091590170. This letter also transmitted Reference 5.

4. Singal, B. K., U.S. Nuclear Regulatory Commission, letter to Rick A. Muench, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station - Issuance of Amendment Re: Changes To Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator Tube Inspection Report" (TAC No. ME1393)," dated October 19, 2009 (ADAMS Accession No. ML092750606).

5. Westinghouse Electric Company LLC, WCAP-17071-P, Revision 0, "H*: Alternate Repair Criteria for the RESRAD Expansion Region in Steam Generators with Hydraulically, Expanded Tubes (Model F)", April 2009 (Proprietary). Non-proprietary version designated as WCAP-17071-NP, Revision 0, April 2009 (ADAMS Accession No. ML091590167).

6. Westinghouse Electric Company LLC, WCAP-17330-P, Revision 1, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)," November 2010 (Proprietary). Non-proprietary version designated as WCAP-17330-NP, Revision 1, November 2010 (ADAMS Accession No. ML11188A108). This report was enclosed with a permanent H* amendment request submitted by Duke Energy Carolinas, LLC for Catawba Nuclear Station Unit 2 dated June 30, 2011 (ADAMS Accession No. ML111880287).

7. Wright, D., U.S. Nuclear Regulatory Commission, letter to Mark J. Ajluni, Southern Nuclear Operating Company , Inc., "Vogtle Electric Generating Plant, Units 1 and 2 - Transmittal of Unresolved Issues Regarding Permanent Alternate Repair Criteria for Steam Generators (TAC Nos. ME1339 and ME1340)," dated November 23, 2009 (ADAMS Accession No. ML093030490).

8. U.S. Nuclear Regulatory Commission memorandum, Taylor, R. M., to Gloria J. Kulesa, "Vogtle Electric Generating Plant – Audit of Steam Generator H* Amendment Reference Documents," dated July 9, 2010 (ADAMS Accession No. ML101900227).

9. U.S. Nuclear Regulatory Commission, Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," dated August 3, 1995 (ADAMS Accession No. ML031070113).

10. U.S. Nuclear Regulatory Commission, "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).

11. U.S. Nuclear Regulatory Commission, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," NUREG-1570, March 1998 (ADAMS Accession No. ML070570094).

12. Johnson, A. B., U.S. Nuclear Regulatory Commission, "Summary of the January 8, 2009 Category 2 Public Meeting with the Nuclear Energy Institute (NEI) and Industry to Discuss Steam Generator Issues," dated February 6, 2009 (ADAMS Accession No. ML090370782).

13. Westinghouse Electric Company LLC, LTR-SGMP-09-100-P, "Response to NRC Request for Additional Information on H*; Model F and D5 Steam Generators," included as Attachment 10 to Duke Energy letter, dated April 28, 2010 (ADAMS Accession No. ML101730411). Non-proprietary version designated as LTR-SGMP-09-100-NP, dated April 28, 2010 (ADAMS Accession No. ML101730391).

14. Ajluni, M., Southern Nuclear Operating Company, Inc, letter to U.S. Nuclear Regulatory Commission, "Vogtle Electric Generating Plant, Supplemental Information for License Amendment Request to Revise Technical Specification (TS) 5.5.9, 'Steam Generator (SG) Program,' and TS 5.5.10, 'Steam Generator Tube Inspection Report" for Permanent Alternate Repair Criteria,'" dated August 28, 2009, transmitting Westinghouse Electric Company, LLC letter LTR-SGMP-09-104-P, "White Paper on Probabilistic Assessment of H*," dated August 13, 2009. Non-proprietary version designated as LTR-SGMP-09-104-NP dated August 13, 2009 (ADAMS Accession No. ML092450029).

15. Singal, B. K., U.S. Nuclear Regulatory Commission, letter to Rick A. Muench, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station – Withdrawal of License Amendment Request on Steam Generator Tube Inspections," dated February 28, 2008 (ADAMS Accession No. ML080450185).

16. Riley, J. H., Nuclear Energy institute, letter to U.S. Nuclear Regulatory Commission, "H*/B* Expert Panel Technical Evaluation – Re-assessment of Coefficient of Thermal Expansion Data for SA-508 Steel," dated July 7, 2008 (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," dated June 6, 2008 (ADAMS Accession No. ML082100097).

17. Nuclear Energy Institute, NEI 97-06 [Rev. 3], "Steam Generator Program Guidelines," January 2011 (ADAMS Accession No. ML111310708).

Principal Contributor: A. Johnson, NRR/DE/ESGB

Date: December 11, 2012

December 11, 2012

Mr. Matthew W. Sunseri
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: STEAM GENERATOR TUBE PERMANENT ALTERNATE REPAIR CRITERIA (TAC NO. ME8350)

Dear Mr. Sunseri:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 201 to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 29, 2012.

The amendment revises TS 5.5.9, "Steam Generator (SG) Program," to exclude portions of the tube below the top of the steam generator tubesheet from periodic steam generator tube inspections. In addition, the proposed amendment revises TS 5.6.10, "Steam Generator Tube Inspection Report," to remove reference to the previous interim alternate repair criteria and provide reporting requirements specific to the permanent alternate repair criteria.

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

Sincerely,
/RA by Balwant K. Singal for/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures:
1. Amendment No. 201 to NPF-42
2. Safety Evaluation

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AJohnson, NRR/DE/ESGB

ADAMS Accession No.: ML12300A309 ***via memo dated October 19, 2012**

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DATE	11/13/12	11/7/12	11/21/12	10/19/12
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