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

April 6, 2011

Mr. Rafael Flores
Senior Vice President and
Chief Nuclear Officer
Attention: Regulatory Affairs
Luminant Generation Company LLC
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS TO MODIFY TECHNICAL SPECIFICATIONS TO ESTABLISH ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR PROGRAM (TAC NOS. ME5110 AND ME5111)

Dear Mr. Flores:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 154 to Facility Operating License No. NPF-87 and Amendment No. 154 to Facility Operating License No. NPF-89 for Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 1, 2010.

The amendments revise TS 5.5.9,"Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program," to exclude portions of the CPNPP, Unit 2 Model D5 SG tubes below the top of the SG tubesheet from periodic SG tube inspections. In addition, the amendments revise TS 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Reports," to provide reporting requirements specific to CPNPP, Unit 2, for the temporary alternate repair criteria. The proposed changes would be applicable only to CPNPP, Unit 2, during Refueling Outage 12 and the subsequent operating cycle.

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

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 154 to NPF-87
2. Amendment No. 154 to NPF-89
3. Safety Evaluation

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

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 154
License No. NPF-87

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

   A. The application for amendment by Luminant Generation Company LLC dated December 1, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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 Facility Operating License No. NPF-87 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. 154 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Luminant Generation Company LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan as indicated in the attachment to this license amendment.

3. The license amendment is effective as of its date of issuance and shall be implemented prior to Mode 4 entry during startup from Unit 2 Refueling Outage 12.

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 Facility Operating License No. NPF-87 and Technical Specifications

Date of Issuance: April 6, 2011

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

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 154
License No. NPF-89

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

   A. The application for amendment by Luminant Generation Company LLC dated December 1, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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 Facility Operating License No. NPF-89 is hereby amended to read as follows:

   (2) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A as revised through Amendment No. 154 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Luminant Generation Company LLC 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 Unit 2 Refueling Outage 12.

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 Facility Operating
License No. NPF-89 and
Technical Specifications

Date of Issuance: April 6, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 154

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 154

TO FACILITY OPERATING LICENSE NO. NPF-89

DOCKET NOS. 50-445 AND 50-446

Replace the following pages of the Facility Operating License Nos. NPF-87 and NPF-89, 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.

Facility Operating License No. NPF-87

REMOVE	INSERT
3	3

Facility Operating License No. NPF-89

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
5.5-6	5.5-6
5.5-7	5.5-7
5.6-5	5.6-5
5.6-6	5.6-6

(3) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, and described in the Final Safety Analysis Report, as supplemented and amended;

(4) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use, at any time, any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source, and 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) Luminant Generation Company LLC, 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 license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth 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) <u>Maximum Power Level</u>

Luminant Generation Company LLC is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal through Cycle 13 and 3612 megawatts thermal starting with Cycle 14 in accordance with the conditions specified herein.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A as revised through Amendment No. 154 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Luminant Generation Company LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, and described in the Final Safety Analysis Report, as supplemented and amended;

(4) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use, at any time, any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Luminant Generation Company LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source, and 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) Luminant Generation Company LLC, 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 license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth 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) <u>Maximum Power Level</u>

Luminant Generation Company LLC is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal through Cycle 11 and 3612 megawatts thermal starting with Cycle 12 in accordance with the conditions specified herein.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A as revised through Amendment No. 154 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Luminant Generation Company LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Antitrust Conditions</u>

DELETED

5.5 Programs and Manuals

5.5.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program (continued)

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

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

a. For Unit 2 only during Refueling Outage 12 and the subsequent operating cycle, tubes with service-induced flaws located greater than 16.95 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 16.95 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. For Unit 1, the number and portions of the tubes inspected and 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 repair criteria. For Unit 2 during Refueling Outage 12 and the subsequent operating cycle, 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 16.95 inches below the top of the tubesheet on the hot leg side to 16.95 inches below the top of the tubesheet on the cold leg side and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements below, the inspection scope, inspection methods and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

2. For the Unit 2 model D5 steam generators (Alloy 600 thermally treated) inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling

5.5 Programs and Manuals

5.5.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program (continued)

outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

3. For the Unit 1 model Delta-76 steam generators (Alloy 690 thermally treated) inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

4. For Unit 1, if crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indications shall not exceed 24 effective full power months or one refueling outage (whichever is less). For Unit 2 during Refueling Outage 12 and the subsequent operating cycle, if crack indications are found in any SG tube from 16.95 inches below the top of the tubesheet on the hot leg side to 16.95 inches below the top of the tubesheet on the cold leg side, then the next inspection for each SG for the degradation mechanism that caused the crack indications shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 Secondary Water Chemistry Program

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

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

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

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

1. WCAP-14040-NP-A; "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not used

5.6.8 PAM Report

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

5.6.9 Unit 1 Model D76 and Unit 2 Model D5 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. Active 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 active degradation mechanism,

f. Total number and percentage of tubes plugged to date, and

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

h. For Unit 2 only during Refueling Outage 12 and 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

5.6 Reporting Requirements

5.6.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report (continued)

secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

i. For Unit 2 only during Refueling Outage 12 and the subsequent operating cycle, the calculated accident induced leakage rate from the portion of the tubes below 16.95 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 3.16 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and

j. For Unit 2 only during Refueling Outage 12 and the subsequent operating cycle, the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 154 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 154 TO

FACILITY OPERATING LICENSE NO. NPF-89

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By letter dated December 1, 2010 (Reference 1), Luminant Generation Company, LLC (the licensee), submitted a license amendment request (LAR) to revise the technical specifications (TSs) of Comanche Peak Nuclear Power Plant (CPNPP), Units 1 and 2. The request proposed changes to the inspection scope and repair requirements of TS 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program," and to the reporting requirements of TS 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report." The proposed changes would be applicable only to CPNPP, Unit 2 during Refueling Outage 12 (2RF12) and the subsequent operating cycle. The proposed changes would establish temporary alternate repair criteria for portions of the CPNPP, Unit 2 SG tubes within the tubesheet, and would replace similar, existing criteria that were used in 2009 during the previous refueling outage, 2RF11.

2.0 BACKGROUND

CPNPP, Unit 2, has four Model D5 SGs that were designed and fabricated by Westinghouse. There are 4,570 Alloy 600 tubes in each SG, each with an outside diameter of 0.750 inches and a nominal wall thickness of 0.043 inches. The thermally treated tubes are hydraulically expanded for the full depth of the 21-inch tubesheet and are welded to the tubesheet at each tube end. Until the fall of 2004, no instances of stress-corrosion cracking (SCC) affecting the tubesheet region of thermally treated Alloy 600 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, Unit 2 (Catawba), which has Westinghouse Model D5 SGs. Like CPNPP, Unit 2, the Catawba SGs use thermally treated Alloy 600 tubing that is hydraulically expanded against the tubesheet. The crack-like indications at Catawba were found in a tube overexpansion (OXP), in the tack expansion region, and near the tube-to-tubesheet (T/TS) weld. 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 in the fall of 2004, other nuclear plants have found crack-like indications in tubes within the tubesheet as well. These plants include: Braidwood Station, Unit 2; Byron Station, Unit No. 2; CPNPP, Unit 2; Surry Power Station, Unit No. 2; Vogtle Electric Generating Plant, Unit 1; and Wolf Creek Generating Station (WCGS). Most of the indications were found in the tack expansion region near the tube-end welds and were a mixture of axial and circumferential primary-water stress-corrosion cracking (PWSCC).

On February 21, 2006, Wolf Creek Nuclear Operating Corporation (WCNOC), the licensee for WCGS, submitted an LAR that would permanently limit the scope of inspections required for tubes within the tubesheet (Reference 2). The LAR was based on an analysis performed by Westinghouse Electric Company LLC (Westinghouse) that provided a technical basis for permanently limiting the scope of inspections required for tubes within the tubesheet. After three requests for additional information (RAIs) and several meetings with WCNOC, the U.S. Nuclear Regulatory Commission (NRC) staff informed WCNOC during a phone call on January 3, 2008, that it had not provided sufficient information to allow the NRC staff to review and approve the permanent LAR. WCNOC withdrew the LAR by letter dated February 14, 2008 (Reference 3). Other plants had submitted permanent LARs similar to that for WCGS prior to 2008, which also were subsequently withdrawn. In a letter dated February 28, 2008 (Reference 4), the NRC staff identified the specific issues that needed to be addressed to support any future request for a permanent amendment, which included but were not limited to thermal expansion coefficients, crevice pressure assumptions, uncertainty models, acceptance standards for probabilistic assessment, and leakage resistance.

After withdrawal of the initial round of permanent LARs submitted prior to 2008, the licensees and their contractor, Westinghouse, worked with the NRC staff to address the issues posed in Reference 4. The NRC and industry held public meetings (References 5, 6, and 7) and phone calls to discuss resolution of these issues. The permanent LAR received from CPNPP on June 8, 2009 (Reference 8), resolved the issues identified by the NRC staff in Reference 4 but raised an additional technical issue that prevented approval of the permanent LAR. Responses to NRC staff RAIs were supplied in References 9, 10, and 11, and the licensee modified its LAR dated June 8, 2009 (via References 12, 13, and 14), to apply during 2RF11 and the subsequent operating cycle, instead of the permanent change originally requested.

The NRC staff approved the revised amendment in Reference 15. The accompanying safety evaluation concluded that the NRC staff did not have sufficient information to determine whether the tubesheet bore displacement eccentricity had been addressed in a conservative fashion and, thus, the NRC staff did not have an adequate basis to approve a permanent H* amendment at that time. The NRC staff further concluded that despite any potential non-conservatism in the calculated H* distance that may have been associated with the eccentricity issue, there was sufficient conservatism embodied in the proposed H* distance to ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity would 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.

Subsequent analyses by industry to address the NRC staff's concerns revealed that tubesheet bore eccentricity did not have a significant bearing on the outcome of the H* analyses. However, these analyses also revealed a significant shortcoming in how displacements from the three-dimensional (3-D) finite element model of the lower SG assembly were being applied to the T/TS interaction model, which was based on thick-shell equations. The industry developed a new T/TS interaction model to address this shortcoming and the H* analyses were updated accordingly. This more recent background is discussed in more detail as part of the NRC staff's technical evaluation in Section 4.0 of this safety evaluation. Details of these more recent analyses became available for NRC staff review too late to support applications for a permanent H* amendment in the spring or fall of 2011. For this reason, the subject amendment request by the licensee is for an interim H* amendment, applicable to 2RF12 and the subsequent operating cycle for Unit 2.

3.0 REGULATORY EVALUATION

In Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications," the requirements related to the content of the TSs are established. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TSs. 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 TSs to operate the facility in a safe manner. For CPNPP, Units 1 and 2, the requirements for performing SG tube inspections and repair are in TS 5.5.9, while the requirements for reporting the SG tube inspections and repair are in TS 5.6.9.

The TSs for all pressurized-water reactor (PWR) plants require that an SG program be established and implemented to ensure that SG tube integrity is maintained. For CPNPP, Units 1 and 2, SG tube integrity is maintained by meeting the performance criteria specified in TS 5.5.9.b for structural and leakage integrity, consistent with the plant design and licensing basis. TS 5.5.9.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. TS 5.5.9.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that may be present along the length of a tube, from the T/TS weld at the tube inlet to the T/TS weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The applicable tube repair criteria, specified in TS 5.5.9.c., are that tubes found during inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged, unless the tubes are permitted to remain in service through application of the proposed alternate repair criteria provided in TS 5.5.9.c.1.a.

The SG tubes are part of the reactor coolant pressure boundary (RCPB) and isolate fission products in the primary coolant from the secondary coolant. For the purposes of this safety evaluation, SG tube integrity means that the tubes are capable of performing this safety function in accordance with the plant design and licensing basis. The General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 provide regulatory requirements, which are applicable to CPNPP, and state that the RCPB shall have "an extremely low probability of abnormal leakage...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 GDCs for the CPNPP, in Section 3.1 of the Final Safety Analysis Report (FSAR) and does not identify any deviations from these GDCs for SG tube related issues. To this end, 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 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 PWR facilities (like CPNPP, Units 1 and 2), ASME Code Class 1 components meet the Section XI requirements of the ASME Code to the extent practical, except for design and access provisions, and pre-service examination requirements. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. The Section XI requirements pertaining to inservice inspection of SG tubing are augmented by additional requirements in the TSs.

As part of the plant's licensing bases, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents (DBAs), such as an SG tube rupture and a main steam line break (MSLB). These analyses consider primary-to-secondary leakage that may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," accident source term, GDC 19 for control room operator doses (or some fraction thereof as appropriate to the accident), or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analyses for CPNPP, Units 1 and 2, are being changed because of the proposed amendment and, therefore, no radiological consequences of any accident analysis are being changed. The use of the proposed alternate repair criteria does not impact the integrity of the SG tubes; therefore, the SG tubes still meet the requirements of the GDCs in Appendix A to 10 CFR Part 50, and the requirements for Class 1 components in Section III of the ASME Code. The proposed changes maintain the accident analyses and consequences that the NRC staff has reviewed and approved for the postulated DBAs for SG tubes.

License Amendment No. 149 (Reference 15) is currently approved at CPNPP, Unit 2, and the amendment modified TS 5.5.9 (5.5.9.2 at the time of the approval), "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program," and TS 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report," by incorporating interim alternate repair criteria and associated tube inspection and reporting requirements that are applicable during Unit 2 refueling outage 11 and the subsequent operating cycle. The proposed subject amendment maintains the same alternate repair criteria (i.e., 16.95 inches below the top of the tubesheet (TTS)), but would be applicable only to CPNPP, Unit 2 during 2RF12 (spring 2011) and the subsequent operating cycle.

4.0 TECHNICAL EVALUATION

4.1 Proposed Changes to the TSs

TS 5.5.9 would be revised as follows (new text in underline and bold):

5.5.9. Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program

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

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

a. For Unit 2 only during Refueling Outage **12** and the subsequent operating cycle, tubes with service-induced flaws located greater than 16.95 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 16.95 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. For Unit 1, the number and portions of the tubes inspected and 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 outlet, and that may satisfy the applicable tube repair criteria. For Unit 2 during Refueling Outage **12** and the subsequent operating cycle, 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 16.95 inches below the top of the tubesheet on the hot leg side to 16.95 inches below the top of the tubesheet on the cold leg side and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1.-3. [No change/not shown.]

4. For Unit 1, if crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). For Unit 2 during Refueling Outage **<u>12</u>** and the subsequent operating cycle, if crack indications are found in any SG tube from 16.95 inches below the top of the tubesheet on the hot leg side to 16.95 inches below the top of the tubesheet on the cold leg side, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

TS 5.6.9 would be revised as follows (new text in underline and bold):

5.6.9 <u>Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report</u>

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.-g. [No change/not shown.]

h. For Unit 2 only during Refueling Outage **<u>12</u>** and 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,

i. For Unit 2 only during Refueling Outage **12** and the subsequent operating cycle, the calculated accident induced leakage rate from the portion of the tubes below 16.95 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 3.16 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined; and

j. For Unit 2 only during Refueling Outage **12** and the subsequent operating cycle, the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

4.2 Technical Evaluation

The T/TS joints are part of the pressure boundary between the primary and secondary systems. Each T/TS joint consists of the tube, which is hydraulically expanded against the bore of the tubesheet, the T/TS weld located at the tube end, and the tubesheet. The joints were designed in accordance with Section III of the ASME Code 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. The axial force which could produce pullout comes from the primary-to-secondary pressure differentials associated with normal operating and DBA conditions, and is called the end-cap load. In addition, the welds serve to make the joints leak-tight.

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

To support the proposed TS changes, the licensee's contractor, Westinghouse, has defined a parameter called H* to be that distance below the TTS 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 TTS. For CPNPP, Unit 2, the proposed H* distance is 16.95 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 TSs 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 TTS to a distance below the TTS 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 license amendment is that the amended TSs should continue to ensure that tube integrity will be maintained consistent with the current design basis, as defined in the FSAR. This includes maintaining structural safety margins consistent with the structural integrity performance criteria in TS 5.5.9.b.1, as discussed in Section 4.3.1 of this safety evaluation. In addition, this includes limiting the potential for accident-induced primary-to-secondary leakage to values that do not exceed the accident-induced leakage performance criteria in TS 5.5.9.b.2, which are consistent with values assumed in the FSAR 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 Sections 4.3 and 4.4 of this safety evaluation, respectively.

4.3 Joint Structural Integrity

4.3.1 Acceptance Criteria

Westinghouse has conducted extensive analyses to establish the necessary H* distance to resist pullout under normal operating and DBA conditions. The NRC staff concludes that pullout is the structural failure mode of interest since the tubes are radially constrained against axial fishmouth rupture by the presence of the tubesheet. The axial force which could produce pullout comes from the primary-to-secondary pressure differentials associated with normal operating and DBA conditions, and is called the end-cap load. 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. These safety factors are consistent with the safety factors embodied in the structural integrity performance criteria in TS 5.5.9.b.1 and with the design basis including the stress limit criteria in the ASME Code, Section III, and, therefore, the NRC staff concludes they are acceptable.

4.3.2 3-D Finite Element Analysis

A detailed 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 separating 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 Section 4.3.3 below.

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.

A number of FEA mesh enhancements in the tubesheet region have been made since the reference analysis (Reference 16) was performed. The mesh near the plane of symmetry (perpendicular to the divider plate) was revised to permit obtaining displacements parallel to the direction of the divider plate directly from the 3-D finite element model, for application (as displacement boundary conditions) to the edges of the square-cell model discussed in Section 4.3.3.2. The mesh near the TTS was enhanced to accommodate high temperature gradients in this area during normal operating conditions.

This 3-D FEA replaces the 2-D axisymmetric FEA used to support H* amendment requests submitted prior to 2008. The NRC staff concludes that the 3-D analysis adequately addresses a concern cited by the NRC staff in Reference 4 concerning the validity of the axisymmetric model to conservatively bound significant non-axisymmetric features of the actual tubesheets. These non-axisymmetric features include the solid (non-bored) portion of the tubesheet between the hot- and cold-leg sides, and the divider plate which acts to connect the solid part of the tubesheet to the channel head.

Some non-U.S. units have experienced cracks in the weld between the divider plate and the stub runner attachment on the bottom of the tubesheet. Should such cracks ultimately cause the divider plate to become disconnected from the tubesheet, tubesheet vertical and radial displacements under operational conditions could be significantly increased relative to those for an intact divider plate weld. Although the industry believes that there is little likelihood that cracks such as those seen abroad could cause a failure of the divider plate weld, the 3-D FEA conservatively considered both the case of an intact divider plate weld and a detached divider plate weld to ensure a conservative analysis. The case of a detached divider plate weld was found to produce the most limiting H* values. In the reference analyses (Reference 16), a factor was applied to the 3-D FEA results to account for a non-functional divider plate, based on earlier sensitivity studies performed with the 2-D axisymmetric FEA model of the lower SG assembly. The 3-D FEA model now 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.

Separate 3-D FEA analyses were conducted for each loading condition considered (i.e., normal operating conditions, MSLB, feedwater line break (FLB)), rather than scaling unit load analyses to prototypic conditions as was done in analyses prior to 2008. The NRC staff concludes that this addresses (corrects) a significant source of error in analyses used by applicants to support permanent H* amendment requests submitted prior to 2008 and which were subsequently withdrawn (Reference 4). In addition, the temperature distributions throughout the lower SG assembly, including the tubesheet region, were calculated directly in the 3-D FEA from the assumed plant temperature conditions (e.g., from the assumed primary and secondary water temperatures) for each operating condition. The NRC staff concludes that this a more realistic approach than the reference analysis (Reference 16), where a linear distribution of temperature was assumed to exist through the thickness of the tubesheet, and an adjustment factor (based on sensitivity analyses) was applied to the H* calculations for normal operating conditions to account for the actual temperature distribution in the tubesheet.

4.3.3 T/TS Interaction Model

4.3.3.1 Thick-Shell Model

The resistance to pullout is the axial friction force developed between the expanded tube and the tubesheet over the H* distance. The friction force is a function of the radial contact pressure between the expanded tube and the tubesheet. In the analysis (Reference 16) for the interim H* amendment issued on October 9, 2009, for CPNPP, Unit 2 (Reference 15), 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 Section 4.3.2 above) 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 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 16, 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 600 thermally treated tubing. However, responses to NRC staff questions provided in References 10 and 11 did not contain sufficient information to allow the NRC staff to reach a conclusion on these matters and on the acceptability of a permanent H* amendment. However, for reasons discussed in the NRC staff's safety evaluation in Reference 15, the NRC staff concluded that there was an adequate technical basis to support issuance of an interim H* amendment.

In Reference 17, 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. 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 (Reference 18). 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 17, 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 a review of pertinent draft responses to the Reference 17, 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.

During the audit, Westinghouse presented preliminary details of a new T/TS interaction model developed as an alternative to the thick-shell interaction model. This model is termed the square-cell model and was developed in response to the difficulty encountered when applying the eccentricity adjustment to the Model D5 SG 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 Section 4.3.2 above, 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. The square-cell model was still under development at the time of the audit and no draft documentation of the model was available for NRC staff review. Although the NRC staff found that the objectives of the new model approach appeared reasonable, the NRC staff was unable to provide feedback on the details of the approach at that time. The NRC staff also observed (Reference 18) that the square-cell model approach might need to be applied to the Model F, 44F, and 51F SGs to confirm that the analyses for these plants were conservative.

4.3.3.2 Square-Cell Model

Documentation for the square-cell model is included with the subject amendment request for an interim H* at CPNPP, Unit 2. The square-cell model is a 2-D, plane stress, finite element 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 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 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 modeled to act independently of one another. Tube-to-tubesheet 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 TTS, is the average contact pressure over that incremental distance times the tubesheet bore surface area (equal to the tube outer diameter surface area) over the incremental axial distance times the coefficient of friction. The NRC staff reviewed the coefficient of friction used in the analysis and judges it to be a reasonable lower bound (conservative) estimate. The H* distance for each tube was determined by integrating the incremental friction forces from the TTS to the distance below the TTS where the friction force integral equaled the applied end-cap load, times the appropriate safety factor as discussed in Section 4.3.1.

The square-cell model assumes as an initial condition that each tube outer surface is in contact with the inner surface of the tubesheet bore, at room temperature and atmospheric pressure, with zero RCP associated with the hydraulic expansion process. The NRC staff concludes that the assumption of zero RCP in all tubes is a conservative assumption.

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 Section 4.3.2. 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 inside tube 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. Note the crevice pressure varies as a function of the elevation being analyzed, as discussed in Section 4.3.4.

The NRC staff has not completed its review of the square-cell model. This review will need to be completed before the NRC staff can approve any request for a permanent H* amendment. However, for reasons discussed in Section 4.6, the NRC staff concludes that the proposed H* distances will ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety.

4.3.4 Crevice Pressure Evaluation

As discussed in an earlier footnote, the H* analyses postulate that interstitial spaces exist between the hydraulically expanded tubes and tubesheet bore surfaces. These interstitial spaces are assumed to act as crevices between the tubes and the tubesheet bore surfaces. The NRC staff 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 TTS and the H* distance below the TTS where the tube is assumed to be severed. These distributions were used to determine the appropriate crevice pressure for each axial slice of the T/TS interaction model. 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 Section 4.3.4 of this safety evaluation for more information.

concludes that this approach acceptably addresses the NRC staff's concerns cited in Reference 4 concerning the use of the limiting median crevice pressure value of the normal operating and MSLB data, respectively, for each axial slice, in previous H* analyses in support of amendment applications submitted prior to 2008. The NRC staff concludes that the crevice pressure distributions used to support the current amendment request to be more realistic and more conservative than those used previously.

Because the crevice pressure distribution is assumed to extend from the H* location, where crevice pressure is assumed to equal primary pressure, to the TTS, 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 the initial estimate for the next H* iteration.

4.3.5 <u>H* Calculation Process</u>

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 and 3-D finite element models, assuming nominal geometric and material properties, and assuming that the tube is severed at the bottom of the tubesheet for purposes of defining the contact pressure distribution over the length of the T/TS crevice. This initial estimate did not consider the effect of the Poisson's contraction of the tube radius associated with application of the axial end-cap load (see Step 6 below).

2. In the reference analysis (Reference 16), 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 TTS, 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 of the subject amendment request, as discussed and evaluated in Section 4.3.5.1 of this safety evaluation.

3. In the reference analysis (Reference 16), 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 FEA. This adjustment is no longer necessary, as discussed in Section 4.3.2, since the tubesheet temperature distributions were calculated directly in the 3-D FEA, supporting the current request for an interim 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 Sections 4.3.6 and 4.3.7 of this safety evaluation.

5. Add a crevice pressure adjustment to the probabilistic estimate of H* to account for the crevice pressure distribution that 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 Section 4.3.5.2 of this safety evaluation.

6. This step has been added to the H* calculation process since the reference analysis, to support the subject interim 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 Section 4.3.5.3 of this safety evaluation.

4.3.5.1 BET Considerations

In the reference H* analysis (Reference 16), a 0.3-inch adjustment was added to the initial H* estimate to account for uncertainty in the BET location, relative to the TTS, based on a BET uncertainty analysis for Model F SGs conducted by Westinghouse. As discussed previously in Section 4.3.3.1, the reference analysis was based on the thick-shell model and the results of that analysis did not indicate a loss of contact pressure at the TTS during normal operating or steam line break conditions; therefore, this adjustment for the BET location was necessary. In response to NRC staff questions regarding the BET uncertainty analysis, Westinghouse performed an analysis (Reference 19) that showed BET locations as great as 1 inch below the TTS could be tolerated at any tube location. Because the limiting calculated H* value is in the most limiting tubesheet sector, that H* value provides greater than 1 inch of margin for most other tubesheet sectors. For those few sectors in the tubesheet where the local H* distance was within 1 inch of the maximum H* distance, Westinghouse showed that the contact pressure gradient was positive with increasing depth into the tubesheet, and therefore, an H* length reduced by 1 inch still met the pull out resistance requirements, including appropriate safety factors.

The new analysis performed in Reference 20 has made the need for this adjustment moot, as the square-cell model shows a loss of contact pressure at the TTS that is greater than the possible variation in the BET location. The loss of contact pressure at the TTS 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.

4.3.5.2 Crevice Pressure Adjustment

As discussed in Section 4.3.5, 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 are 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 during the reference analysis 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 Section 4.3.5) was plotted as a function of the initial H* estimate for the limiting loading case and tube radial location. The NRC staff concludes that this is an acceptable approach where the H* estimates are based on the thick-shell model;

however, the NRC staff has not yet reached a conclusion regarding the applicability of this adjustment to H* estimates that are based on the square-cell model. The NRC staff will need to reach a conclusion on this point before the NRC staff can approve any request for a permanent H* amendment. However, for reasons discussed in Section 4.6, the NRC staff concludes that the proposed H* distances will ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety.

4.3.5.3 Poisson Contraction Effect

The axial end-cap load acting on each tube is equal to the primary-to-secondary pressure difference times the tube 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 concludes is 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 Section 4.3.1.

This Poisson radial contraction effect was neglected in the reference analyses, but is accounted for in the analyses supporting the subject amendment request. A simplified approach was followed. 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 reacted by the tubesheet. The T/TS contact pressure distributions determined in Step 4 of the H* calculation process in Section 4.3.5 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 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 Section 4.3.5 were reduced only over the attenuation distance. The NRC staff has not completed its review of the applied adjustment to account for the Poisson radial contraction effect. However, for reasons discussed in Section 4.6, the NRC staff concludes that the proposed H* distances will ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety.

4.3.6 Acceptance Standard - Probabilistic Analysis

The purpose of the probabilistic analysis is to develop a safe H* distance that ensures with a probability of 0.95 that the population of tubes will retain margins against pullout consistent with criteria evaluated in Section 4.3.1 of this safety evaluation, assuming all tubes to be completely severed at their H* distance. The NRC staff concludes that this probabilistic acceptance standard is consistent with what the NRC staff has approved previously and is acceptable. For example, the upper voltage limit for the voltage based tube repair criteria in NRC Generic Letter 95-05 (Reference 21) employs a consistent criterion. The NRC staff also notes that use of the 0.95 probability criterion ensures that the probability of pullout of one or more tubes under normal operating conditions and conditional probability of pullout under accident conditions is well within tube rupture probabilities previously considered in probabilistic risk assessments (References 22 and 23).

In terms of the confidence level that should be attached to the 0.95 probability acceptance standard, it is industry practice for SG tube integrity evaluations, as embodied in industry guidelines, to calculate such probabilities at a 50 percent confidence level. The 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 (Reference 24). In the meantime, the calculated H* distances supporting the interim amendment currently being requested 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 FLB, the NRC staff and licensee both find that the tube population in the faulted SG is of interest, since it is the only SG that experiences a large increase in the primary-to-secondary pressure differential. For the Model D5 SGs in the subject amendment request, MSLB is the most limiting condition and the H* distances referenced herein are based on 0.95 probability/95 percent confidence estimates for the population of tubes in any one SG in the plant.

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.

4.3.7 Probabilistic Analyses

Sensitivity studies were conducted during the reference analyses (Reference 16) and demonstrated that H* was highly sensitive to the potential variability of the coefficients of thermal expansion (CTE) for the Alloy 600 thermally treated tubing material and the SA-508 Class 2a tubesheet material. Given that no credit was taken in the reference H* analyses (Reference 16) for RCP associated with the tube hydraulic expansion process[2], the sensitivity of H* to other geometry and material input parameters was judged by Westinghouse to be inconsequential and were ignored, with the exception of Young's modulus of elasticity for the tube and tubesheet materials. Although the Young's modulus parameters were included in the

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

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 agrees that the sensitivity studies adequately capture the input parameters which may significantly affect the value of H*. This conclusion is based, in part, on no credit being taken for RCP during the reference H* analyses.

These sensitivity studies were used to develop influence curves describing the change in H*, relative to the mean H* value estimate (see Section 4.3.5), 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 (Reference 16). One was a simplified statistical approach utilizing a "square root of the sum of the squares" method and the other was a detailed Monte Carlo sampling approach. The NRC staff's review of the reference analyses relied primarily on the Monte Carlo analysis, which provides the more realistic treatment of uncertainties.

The NRC staff reviewed the implementation of probabilistic analyses in the reference analyses (Reference 16) and questioned whether the H* influence curves had been conservatively treated. To address this concern, the licensee submitted new H* analyses as documented in References 9 and 10. These analyses made direct use of the H* influence curves in a manner the NRC staff concludes is acceptable.

The revised reference analyses in References 9 and 10 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 in Reference 4 concerning the appropriate way to sample material properties for the tubesheet, whose properties are unknown but do not vary significantly for a given SG, in contrast to the tubes whose properties tend to vary much more randomly from tube to tube in a given SG. This issue was addressed by a staged sampling process where the tubesheet properties were sampled once and then held fixed, while the tube properties were sampled a number of times equal to the SG tube population. This process was repeated 10,000 times, and the maximum H* value from each repetition was rank ordered. The final H* value was selected from the rank ordering to reflect a 0.95 probability value at the desired level of confidence for a single SG tube population or all SG population, as appropriate. The NRC staff concludes that this approach addresses the NRC staff's question in a realistic fashion and is acceptable.

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 in support of the subject interim amendments. Instead, the probabilistic analysis utilized the results of the

Monte Carlo from the reference analyses (References 16 and 9)[3], 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 high ranking 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 values of alpha. From these plots, paired sets of tube and tubesheet CTE values were selected such as to maximize the H* estimate and to upper and lower bound the rank orders corresponding to the appropriate probabilistic acceptance criteria described and evaluated in Section 4.3.6. These CTE values were then input to the lower SG assembly 3-D finite element model and the square-cell model to yield probabilistic H* estimates. These H* estimates were then plotted as a function of rank ordering, allowing the interpolation of H* values at the desired rank orders.

The limiting probabilistic H* value, evaluated at the appropriate acceptance standard as discussed in Section 4.3.6 and with the adjustments for crevice pressure and Poisson radial contraction effect discussed in Section 4.3.5, is bounded by the proposed H* value of 16.95 inches in the subject request for an interim amendment.

The NRC staff has not completed its evaluation of the above probabilistic analysis, which must be done before the NRC staff can approve any request for a permanent H* amendment. However, for reasons discussed in Section 4.6, the NRC staff concludes that the proposed H* distances will ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety.

4.3.8 <u>Coefficient of Thermal Expansion</u>

During operation, a large part of contact pressure in an SG T/TS joint is derived from the difference in the CTE between the tube and tubesheet. As discussed in Section 4.3.7, 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. In Reference 4, 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.

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[3] The NRC staff notes that because the reference Monte Carlo simulation for the Model D5 SGs was based on normal operating conditions, Westinghouse performed an additional reference Monte Carlo simulation on the Model D5 SGs using steam line break conditions, prior to performing the rank ordering of CTE values associated with the probabilistic H* values.

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 (Appendix A to 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 data base. Two additional sets of CTE test data (different from those addressed in the previous paragraph) had CTE offsets at low temperature that were not expected. Review of the test data showed that the first test, conducted in a vacuum, had proceeded to a maximum temperature of 700 degrees Celsius (°C), which changed the microstructure and the CTE of the steel during decreasing temperature conditions. As a result of the altered microstructure, the CTE test data generated in the second test, conducted in air, was also invalidated. As a result of the large "dead band" region and the altered microstructure, both data sets were excluded from the final CTE values obtained from the CTE testing program.

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

The data from the test program were combined with the ANL data that were found by the licensee to be acceptable, and with 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 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 4 and are acceptable.

4.4 Accident-induced Leakage Considerations

Operational leakage integrity is assured by monitoring primary-to-secondary leakage relative to the applicable TS LCO limits in TS 3.4.13, "RCS Operational LEAKAGE." However, it must also be demonstrated that the proposed TS changes do not create the potential for leakage during DBA to exceed the accident leakage performance criteria in TS 5.5.9.2.b.2, including the leakage values assumed in the plant licensing basis accident analyses.

If a tube is assumed to contain a 100 percent through-wall flaw some distance into the tubesheet, a potential leak path between the primary and secondary systems is introduced between the hydraulically expanded tubing and the tubesheet. The leakage path between the tube and tubesheet has been modeled by the licensee's contractor, Westinghouse, as a crevice consisting of a porous media. Using Darcy's model for flow through a porous media, leak rate is proportional to differential pressure and inversely proportional to flow resistance. Flow resistance is a direct function of viscosity, loss coefficient, and crevice length.

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

Using the Darcy model, the leakage factor for a given type accident is the product of four quantities. The first quantity is ratio of the maximum primary-to-secondary pressure difference during the accident divided by that for normal operating conditions. The second quantity is the ratio of viscosity under normal operating primary-water temperature divided by viscosity under the accident condition primary-water temperature. The third quantity is the ratio of crevice length under normal operating conditions to crevice length under accident conditions. This ratio equals 1, provided it can be shown that positive contact pressure is maintained along the entire H* distance for both conditions. The fourth quantity is the ratio of loss coefficient under normal operating conditions to loss coefficient under the accident condition. Although the absolute value of these loss coefficients is not known, Westinghouse has assumed that the loss coefficient is constant with contact pressure such that the ratio is equal to 1. The NRC staff agrees that this is a conservative assumption, provided there is a positive contact pressure for both conditions along the entire H* distance and provided that contact pressure increases at each axial location along the H* distance when going from normal operating to accident conditions. Both assumptions were confirmed to be valid in the original H* analyses submitted with Reference 16.

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, 3.16, 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 20) did not 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 Section 4.3.3.2 of this safety evaluation. Although T/TS contact pressure increased over some sections of the tubing under steam line break conditions, it decreased over other sections within the H* distance. This violated the assumed precondition for assuming that the ratio of loss coefficient under MSLB and normal operating conditions was at least equal to 1.

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

None of the potential functional relationships between loss coefficient and contact pressure considered by Westinghouse resulted in a leakage ratio value exceeding the value of 3.16 calculated for FLB. The description of the revised 3-D FEA of the lower SG assembly and the square-cell model was submitted to the NRC staff for evaluation, but there was insufficient time to review the new information in support of permanent H* amendments for the spring or fall of 2011. However, the NRC staff concludes that leakage is not a concern for the proposed period of the interim amendment for reasons discussed in Section 4.6 below.

As a condition of NRC approval of Amendment No. 149 (i.e., the currently approved alternate repair criteria (Reference 15)) for Unit 2, the licensee provided a commitment that described how the leakage factor would be used to satisfy TS 5.5.9.a for condition monitoring and TS 5.5.9.b.2 regarding performance criteria for accident-induced leakage:

> For the condition monitoring (CM) assessment, the component of operational leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.16 and added to the total accident leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowed accident induced leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 3.16 and compared to the observed operational leakage. An administrative limit will be established to not exceed the calculated value.

In the subject amendment request (Reference 1), the licensee stated the program/procedure changes needed to meet these commitments were completed in accordance with

Amendment No. 149 and that these changes remain in place and will also apply to the subject license amendment. The NRC staff concludes that these previously implemented program/procedural changes are acceptable, since they provide further assurance, in addition to the licensee's operational leakage monitoring processes, that accident-induced SG tube leakage will not exceed values assumed in the licensing bases accident analyses.

4.5 Proposed Change to TS 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report"

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

4.6 Technical Bases for Interim H* Amendment

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

As discussed in Sections 4.3.3.2, 4.3.5.2, 4.3.5.3, and 4.3.7, the NRC staff has not completed its review of certain elements of the technical basis for the proposed H* distance. Thus, in spite of the significant conservatisms embodied in the proposed H* distance, the NRC staff is unable to conclude at this time that the proposed H* distance is, on net, conservative from the standpoint of ensuring that all tubes will retain acceptable margins against pullout (i.e., structural integrity) and acceptable accident leakage integrity for the remaining lifetime of the steam generators, assuming all tubes to be severed at the H* location. This NRC staff will need to complete its review of these certain elements before it can approve any request for a permanent H* amendment. However, for the reasons below, the NRC staff concludes that the proposed H* distances will ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety.

From a fleet-wide perspective (for all Westinghouse plants with Alloy 600 thermally treated tubes), the NRC staff has observed from operating experience that the extent of cracking is at an early stage in terms of the number of tubes affected by cracking below the H* distance and the severity of cracks, compared to the idealized assumption that all tubes are severed at the H* distance. Most of these cracks occur in the lower-most 1 inch of tubing, which is a region of relatively high residual stress associated with the 1-inch tack roll expansion in that region. Although the extent of cracking can be expected to increase with time, it is the NRC staff's judgment based on experience that it will continue to limited to a small percentage of tubes, mostly near the tube ends, over the next operating cycle (approximately 18 months for CPNPP, Unit 2). The NRC staff's observations are based on the review of SG tube inspection reports from throughout the PWR fleet. These reports are reviewed and the NRC staff's conclusions are documented within a year of each SG tube inspection. Reference 25 provides a recent example of such a review for CPNPP, Unit 2 by the NRC staff.

At CPNPP, Unit 2, the most recent inspection of tubing below the proposed H* distance of 16.95 inches was performed in the spring of 2008. The licensee reported in Reference 26 that 13 tubes with flaw indications in the lower-most 0.5-inch of tubing were found, out of 18,200 hot-leg tube ends inspected. The licensee reported that nine of these indications were axially oriented and four were circumferentially oriented and that the maximum circumferential extent of the circumferential indications was less than 180 degrees. The NRC staff concludes that the extent and severity of cracking at CPNPP, Unit 2 is limited and within the envelope of industry experience with similar units.

The NRC staff concludes that there is sufficient conservatism embodied in the proposed H* distances to ensure acceptable margins against tube pullout for at least one operating cycle for the reasons discussed above. The NRC staff also concludes that there is reasonable assurance during the next operating cycle that any potential accident-induced leakage will not exceed the TS performance criteria for accident-induced leakage. This reflects current operating experience trends that cracking below the H* distance is occurring predominantly in the tack roll region near the bottom of the tube. At this location, it is the NRC staff's judgment that the total resistance to primary-to-secondary leakage will be dominated by the resistance of any "crevice" in the roll expansion region (due to very high T/TS contact pressures in this region), such that the leakage factors discussed in Section 4.4 will remain conservative even should there be a loss of T/TS contact near the TTS due to tubesheet bore eccentricity effects.

5.0 SUMMARY

The NRC staff has not completed its review of certain elements of the technical basis for the proposed H* distance and, thus, the NRC staff does not have an adequate basis to approve a permanent H* amendment.

The proposed license amendment applies only to 2RF12 and the subsequent operating cycle for CPNPP, Unit 2. The NRC staff concludes that there is sufficient conservatism embodied in the proposed H* distances to ensure for at least one operating cycle (one fuel cycle) that tube structural and leakage integrity will be maintained with structural safety margins consistent with the design basis and with leakage integrity within assumptions employed in the licensing basis accident analyses, without undue risk to public health and safety. Based on the above, the NRC staff further concludes that the proposed amendment meets 10 CFR 50.36 and, thus, the proposed amendment is acceptable.

6.0 REGULATORY COMMITMENTS

In its letter dated December 1, 2010, the licensee made the following regulatory commitments, which were also a condition for approval of Amendment No. 149. Commitments 3740011 and 3779679 were carried forward and apply to the current LAR:

Regulatory Commitment Number	Commitment	Due Date/Event
3740011	Luminant Power commits to monitor for tube slippage as part of the steam generator tube inspection program. Slippage monitoring will occur for each inspection of the Comanche Peak Unit 2 steam generators.	Required to be completed during each Unit 2 steam generator eddy current inspection starting in Refueling Outage 2RF12
3779679	For the condition monitoring (CM) assessment, the component of operational leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.16 and added to the total accident leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowed accident induced leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 3.16 and compared to the observed operational leakage. An administrative limit will be established to not exceed the calculated value.	During each inspection of the Unit 2 steam generators required by TS 5.5.9 starting in Refueling Outage 2RF12.

The NRC staff considers the above to be regulatory commitments and acceptable.

7.0 STATE CONSULTATION

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

8.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on February 1, 2011 (76 FR 5622). Accordingly, the amendments meet 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 amendments.

9.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

10.0 REFERENCES

1. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Nuclear Power Plant (CPNPP), Docket Nos. 50-445 and 50-446, License Amendment Request 10-004, Model D5 Steam Generators Temporary Alternate Repair Criteria," dated December 1, 2010 (ADAMS Accession No. ML103410267). This letter also transmitted Reference 20.

2. Garrett, T., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Revision to Technical Specification 5.5.9, 'Steam Generator Tube Surveillance Program'," dated February 21, 2006 (ADAMS Accession No. ML060600456).

3. Garrett, T., Wolf Creek Nuclear Operating Corporation, letter to U.S. Nuclear Regulatory Commission, "Withdrawal of License Amendment Request for a Permanent Alternate Repair Criteria in Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," dated February 14, 2008 (ADAMS Accession No. ML080580201).

4. Donohew, J. N., U.S. Nuclear Regulatory Commission, letter to R. 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).

5. Johnson, A. B., U.S. Nuclear Regulatory Commission, "Summary of the October 29 and 30, 2008, Category 2 Public Meeting with the Nuclear Energy Institute (NEI) and Industry to Discuss Modeling Issues Pertaining to the Steam Generator Tube-to-tubesheet Joints," dated November 24, 2008 (ADAMS Accession No. ML083300422).

6. Johnson, A. B., U.S. Nuclear Regulatory Commission, "Summary of the January 9, 2009, Category 2 Public Meeting with the U.S. Nuclear Industry Representatives to Discuss Steam Generator H*/B* Issues," dated February 6, 2009 (ADAMS Accession No. ML090370945).

7. Johnson, A. B., U.S. Nuclear Regulatory Commission, "Summary of the April 3, 2009, Category 2 Public Meeting with the U.S. Nuclear Industry Representatives to Discuss Steam Generator H* Issues," dated May 1, 2009 (ADAMS Accession No. ML091210437).

8. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Steam Electric Station, Docket Nos. 50-445 and 50-446, License Amendment Request 09-007, Model D5 Steam Generator Alternate Repair Criteria," dated June 8, 2009 (ADAMS Accession No. ML091670154).

9. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Steam Electric Station, Docket Nos. 50-445 and 50-446, Response to Request for Additional Information Regarding License Amendment Request 09-007, Model D5 Steam Generator Alternate Repair Criteria," dated August 20, 2009 (ADAMS Accession No. ML092370304). This letter also transmitted Westinghouse Electric Company LLC letter, LTR-SGMP-09-100-P (Proprietary) and LTR-SGMP-09-100-NP (Non-Proprietary), "Response to NRC Request for Additional Information on H*; Model F and D5 Steam Generators," dated August 12, 2009.

10. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Steam Electric Station, Docket Nos. 50-445 and 50-446, Response to Request for Additional Information Regarding License Amendment Request 09-007, Model D5 Steam Generator Alternate Repair Criteria," dated August 27, 2009 (ADAMS Accession No. ML092520324). This letter also transmitted Westinghouse Electric Company LLC letter, LTR-SGMP-09-109-P (Proprietary) and LTR-SGMP-09-109-NP (Non-Proprietary) "Response to NRC Request for Additional Information on H*; RAI #4; Model F and D5 Steam Generators," dated August 25, 2009.

11. 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 (ADAMS Accession No. ML092450029).

12. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Steam Electric Station, Docket Nos. 50-445 and 50-446, Revision to License Amendment Request 09-007, Model D5 Steam Generator Alternate Repair Criteria," dated September 14, 2009 (ADAMS Accession No. ML092650287).

13. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Steam Electric Station, Docket Nos. 50-445 and 50-446, Revision to License Amendment Request 09-007, Model D5 Steam Generator Alternate Repair Criteria," dated September 17, 2009 (ADAMS Accession No. ML092670205).

14. Flores, R., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Steam Electric Station, Docket Nos. 50-445 and 50-446, Revision to License Amendment Request 09-007, Model D5 Steam Generator Alternate Repair Criteria," dated September 28, 2009 (ADAMS Accession No. ML092790184).

15. Singal, B. K., U.S. Nuclear Regulatory Commission, letter to Rafael Flores, Luminant Generating Company LLC, "Comanche Peak Steam Electric Station, Units 1 and 2 – Issuance of Amendments to Modify Technical Specifications to Establish Alternate Repair Criteria and Include Reporting Requirements Specific to Alternate Repair Criteria for Steam Generator Program," dated October 9, 2009 (ADAMS Accession No. ML092740076).

16. Westinghouse Electric Company LLC report, WCAP-17072-P (Proprietary) and WCAP-17072-NP (Non-Proprietary), Revision 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)," dated May 2009 (ADAMS Accession No. ML091670172 (Non-Proprietary)).

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

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

19. Westinghouse Electric Company LLC, letter, LTR-SGMP-09-111 P-Attachment, Revision 1 (Proprietary) and LTR-SGMP-09-111 NP-Attachment, Rev. 1 (Non-Proprietary), "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*," (Proprietary/Non-Proprietary) for Review and Approval (ADAMS Accession Nos. ML103400083 (Proprietary) and ML103400084 (Non-Proprietary)).

20. Westinghouse Electric Company LLC, report, WCAP-17330-P (Proprietary) and WCAP-17330-NP (Non-Proprietary), Revision 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity) Model F/Model D5)," dated November 2010 (ADAMS Accession No. ML103410264 (Non-Proprietary)).

21. 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).

22. 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, dated September 1988 (ADAMS Accession No. ML082400710).

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

24. 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 (NRC ADAMS Accession No. ML090370782).

25. Singal, B. K., U.S. Nuclear Regulatory Commission, letter to R. Flores, Luminant Generation Company LLC, "Comanche Peak Steam Electric Station – Review of the 2008 Steam Generator Tube Inservice Inspections Performed During Refueling Outage 2RF10," dated August 20, 2009 (ADAMS Accession No. ML092250384).

26. Blevins, M., Luminant Generation Company LLC, letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Steam Electric Station (CPSES), Unit 2 Tenth Refueling Outage (2RG10) Steam Generator 180 Day Report," dated September 18, 2008 (ADAMS Accession No. ML082690600).

Principal Contributor: Andrew Johnson, CSGB/DCI/NRR

Date: April 6, 2011

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/

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 154 to NPF-87
2. Amendment No. 154 to NPF-89
3. Safety Evaluation

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ADAMS Accession No. ML110770322 *Memo dated March 15, 2011

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