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CP-201001489
Log # TXX-10152

Ref. # 10 CFR 50.90

December 1, 2010

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT (CPNPP)
DOCKET NOS. 50-445 AND 50-446
LICENSE AMENDMENT REQUEST 10-004, MODEL D5 STEAM GENERATOR TEMPORARY ALTERNATE REPAIR CRITERIA

Dear Sir or Madam:

Pursuant to 10CFR50.90, Luminant Generation Company LLC (Luminant Power) hereby requests an amendment to the CPNPP Unit 1 Operating License (NPF-87) and CPNPP Unit 2 Operating License (NPF-89) by incorporating the attached changes into the CPNPP Unit 1 and 2 Technical Specifications (TSs). This change request applies to both Units.

The proposed change revises TS 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program," to exclude portions of the Unit 2 Model D5 steam generator tubes below the top of the SG tubesheet from periodic steam generator tube inspections during Unit 2 Refueling Outage 12 and the subsequent operating cycle. In addition, this change proposes to revise TS 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report" to provide reporting requirements specific to Unit 2 for the temporary alternate repair criteria. This change is supported by the analysis described in Section 4 of Attachment 1 including WCAP-17330-P, Revision 0,"H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5)."

WCAP-17330-P, Revision 0,"H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5)," November 2010 (Proprietary) is provided in Enclosure 1. Enclosure 2 is WCAP-17330-NP, Revision 0,"H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5)," November 2010 (Non-Proprietary).

As Enclosure 1 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis for which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that information which is proprietary to Westinghouse be withheld from public disclosure in accordance 10 CFR Section 2.390 of the Commission's regulations. Enclosure 3 contains the Westinghouse authorization letter CAW-10-3000, "Application for Withholding Proprietary Information from Public Disclosure," with accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

A001
NRR

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-10-3000 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Attachment 1 provides a detailed description of the proposed changes, a technical analysis of the proposed changes, Luminant Power's determination that the proposed changes do not involve a significant hazard consideration, a regulatory analysis of the proposed changes and an environmental evaluation. Attachment 2 provides the affected Unit 1 and Unit 2 Technical Specification (TS) pages marked-up to reflect the proposed changes. Attachment 3 provides retyped TS pages which incorporate the requested changes.

Luminant Power requests approval of the proposed license amendment by April 5, 2011, to support the CPNPP Unit 2 Spring 2011 (2RF12) refueling outage. The proposed license amendment will be implemented prior to MODE 4 entry during startup from Unit 2 Refueling Outage 12.

On October 9, 2009, the NRC issued CPNPP Amendment Number 149 for a one-cycle steam generator alternate repair criterion. As a condition of approval, CPNPP made the following regulatory commitments:

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
3740015	Luminant Power commits to perform a one time verification of tube expansion locations to determine if any significant deviations exist from the top of the tubesheet to the beginning of expansion transition (BET). If any significant deviations are found, the condition will be entered into the Comanche Peak corrective action program and dispositioned.	Completed. No significant deviations found. See Page 14 of Attachment 1.
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.

Program/procedure changes will be completed for Commitment Number 3740011 prior to the start of Unit 2 Refueling Outage 12. Commitment Number 3740015 is closed. Program/procedure changes are complete for Commitment Number 3779679.

Commitments 3740011 and 3779679 are required for this amendment request.

In accordance with 10 CFR 50.91(b), Luminant Power is providing the State of Texas with a copy of the proposed license amendment.

Should you have any questions, please contact Mr. Jack Hicks at (254)897-6725.

I state under penalty of perjury that the foregoing is true and correct. Executed on the 1st of December, 2010.

Sincerely,

Luminant Generation Company LLC

Rafael Flores

By: Fred W. Madden

Fred W. Madden
Director, Oversight & Regulatory Affairs

Attachments -
1. Description and Assessment
2. Proposed Technical Specification Changes
3. Retyped Technical Specification Pages

Enclosure-
1. WCAP-17330-P, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)," November 2010
2. WCAP-17330-NP, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)," November 2010
3. Westinghouse Letter LTR-CAW-10-3000, "Application for Withholding Proprietary Information from Public Disclosure," dated November 5, 2010

c -
E. E. Collins, Region IV
B. K. Singal, NRR (2)
Resident Inspectors, CPNPP

Alice Hamilton Rogers, P.E.
Inspection Unit Manager
Texas Department of State Health Services
Mail Code 1986
P. O. Box 149347
Austin, TX 78714-9347

ATTACHMENT 1 TO TXX-10152

DESCRIPTION AND ASSESSMENT

LICENSEE'S EVALUATION

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90 and 10 CFR 50.91(a)(5), Luminant Generation Company, LLC (Luminant Power) hereby requests an amendment to the Comanche Peak Nuclear Power Plant (CPNPP) Unit 1 and Unit 2 Technical Specifications. Luminant Power proposes to revise Technical Specification (TS) 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program", to exclude portions of the Unit 2 Model D5 steam generator tubes below the top of the SG tubesheet from periodic steam generator tube inspections. In addition, this amendment proposes to revise TS 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report." Application of the supporting structural analysis and leakage evaluation results to exclude portions of the tubes from inspection and repair of tube indications is interpreted to constitute a redefinition of the primary to secondary pressure boundary. The proposed changes to the TS are based on the supporting structural analysis and leakage evaluation completed by Westinghouse Electric Company LLC. The documentation supporting the Westinghouse analysis is described in section 4 and provides the licensing basis for this change. Table 5-1 of Westinghouse Electric Company WCAP 17330-P, Revision 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5)" [Reference 8.1], provides the 95/95 H* value of 13.36 inches for plants with Model D5 Steam Generators which includes CPNPP Unit 2. However, Luminant Power has chosen to use an H* value of 16.95 inches for additional conservatism. This more conservative value was used in Amendment 149.

The NRC previously issued Amendment 149 [Reference 8.2] which revised TS 5.5.9 to eliminate inspection and repair of tubes more than 16.95 inches below the top of the tubesheet for Unit 2 Refueling Outage 11 and the subsequent operating cycle. Additionally, TS 5.6.9 was revised to provide reporting requirements specific to Unit 2 Refueling Outage 11 and the subsequent operating cycle.

No changes to the CPNPP Final Safety Analysis Report are anticipated at this time as a result of this License Amendment Request.

Approval of this amendment application is requested by April 5, 2011 to support CPNPP Unit 2 Refueling Outage 12 (Spring 2011) and the subsequent operating cycle.

2.0 PROPOSED CHANGE

TS 5.5.9.c. currently states:

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 11 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.

TS 5.5.9.c would be revised as follows, as noted in italic/underline type:

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.

TS 5.5.9.d currently states:

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 11 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 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 indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). For Unit 2 during Refueling Outage 11 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 5.5.9.d would be revised as follows, as noted in italic/underline type:

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 *<u>12</u>* 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 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 indication 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 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.h, 5.6.9.i, and 5.6.9.j currently state:

h. For Unit 2 only during Refueling Outage 11 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 11 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 leak rate, the report should describe how it was determined; and

j. For Unit 2 only during Refueling Outage 11 and the subsequent operating cycle, the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of discovery and corrective action shall be provided.

TS 5.6.9.h, 5.6.9.i, and 5.6.9.j would be revised as follows, as noted in italic/underline type:

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 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 leak 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 discovery and corrective action shall be provided.

3.0 BACKGROUND

CPNPP Unit 2 is a four loop Westinghouse designed plant with Model D5 Steam Generators (SGs) having 4570 tubes in each SG. A total of 78 tubes are currently plugged in all four Unit 2 SGs. The design of the Unit 2 SG includes Alloy 600 thermally treated tubing, full depth hydraulically expanded tubesheet joints, and stainless steel tube support plates with broached quatrefoil holes.

The steam generator inspection scope is governed by TS 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program"; Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," Revision 2, May 2, 2005, [Reference 8.3]; EPRI 1013706, "Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 7," October 2007 [Reference 8.4]; EPRI 1019038, "Generator Management Program: Steam Generator Integrity Assessment Guidelines," Revision 3, November 2009 [Reference 8.5]; CPNPP Procedure STA-733, "Steam Generator Reliability Program," Revision 12 [Reference 8.6]; and the results of the degradation assessments. Criterion IX, "Control of Special Processes" of 10 CFR Part 50, Appendix B, requires in part that nondestructive testing be accomplished by qualified personnel using qualified procedures in accordance with the applicable criteria. The inspection techniques and equipment are capable of reliably detecting the known and potential specific degradation mechanisms applicable to CPNPP Unit 2. The inspection techniques, essential variables and equipment are qualified to Appendix H, "Performance Demonstration for Eddy Current Examination" of Reference 8.4.

Catawba Nuclear Station, Unit 2, (Catawba) reported indication of cracking following nondestructive eddy current examination of the SG tubes during their Fall 2004 outage. NRC Information Notice (IN) 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," [Reference 8.7] provided industry notification of the Catawba issue. IN 2005-09 noted that Catawba reported crack like indications in the tubes approximately seven inches below the top of the hot leg tubesheet in one tube, and just above the tube-to-tubesheet welds in a region of the tube known as the tack expansion in several other tubes. Indications were also reported in the tube-end welds, also known as tube-to-tubesheet welds, which join the tube to the tubesheet.

Luminant Power policies and programs require the use of applicable industry operating experience in the operation and maintenance of CPNPP Unit 2. The recent experience at Catawba, as noted in IN 2005-09, shows the importance of monitoring all tube locations (such as bulges, dents, dings, and other anomalies from the manufacture of the steam generators) with techniques capable of finding potential forms of degradation that may be occurring at these locations (as discussed in Generic Letter 2004-001, "Requirements for Steam Generator Tube Inspections" [Reference 8.8]). Since the CPNPP Unit 2 Westinghouse Model D5 steam generators were fabricated with Alloy 600 thermally treated tubes similar to the Catawba Unit 2 Westinghouse Model D5 steam generators, a potential exists for CPNPP Unit 2 to identify tube indications similar to those reported at Catawba within the hot leg tubesheet region if similar inspections are performed during the Spring 2011 refueling outage.

Potential inspection plans for the tubes and tube welds underwent intensive industry discussions in March 2005. The findings in the Catawba SG tubes present three distinct issues with regard to the SG tubes at CPNPP Unit 2:

1) Indications in internal bulges and overexpansions within the hot leg tubesheet;

2) Indications at the elevation of the tack expansion transition; and

3) Indications in the tube-to-tubesheet welds and propagation of these indications into adjacent tube material.

Prior to each SG tube inspection, a degradation assessment, which includes a review of operating experience, is performed to identify degradation mechanisms that have a potential to be present in the CPNPP Unit 2 SGs. A validation assessment is also performed to verify that the eddy current techniques utilized are capable of detecting those flaw types that are identified in the degradation assessment. Based on the Catawba operating experience, CPNPP Unit 2 revised the SG inspection plan for the Spring 2008 refueling outage (2RF10) to include sampling of bulges and over expansions within the tubesheet region on the hot leg side. The sample was based on the guidance contained in EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 6, and TS 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program". According to EPRI SG examination guidelines, the inspection plan is expanded if necessary due to confirmed degradation in the region required to be examined (i.e., a tube crack). Axial and circumferential indications were reported in the tube end inspection program during 2RF10. As a result, the + Point inspection of the tube end (from THE to THE+2") was expanded from 50% to 100% of the hot leg tubes in all four Unit 2 steam generators. A total of 13 tubes were plugged during 2RF10 due to tube end indications on the hot leg side of the tubes. Prior to 2RF10, there were no active degradation mechanisms in the CPNPP Unit 2 SGs.

Based on these inspections, a limited number of tube flaws existed in the tubesheet area of the CPNPP Unit 2 steam generators. The flaws that have been found are associated with residual stress conditions at the tube ends. No indications of a 360 degree sever has been detected in any steam generator at CPNPP. Consequently, the level of degradation in the CPNPP Unit 2 steam generators is very limited compared to the assumption of "all tubes severed" that was utilized in the development of the H* alternate repair criterion. Consequently, structural integrity will be assured for the operating period between inspections allowed by TS 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program."

As a result of these potential issues and the possibility of unnecessarily plugging tubes in the CPNPP Unit 2 SGs, Luminant Power is proposing changes to TS 5.5.9 to limit the steam generator tube inspection and repair (plugging) to the portion of tubing from 16.95 inches below the top of the tubesheet.

4.0 ANALYSIS

4.1 Licensing Basis Analysis (H* Analysis)

On June 8, 2009, Westinghouse Electric Company WCAP-17072-P, Revision 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)," [Reference 8.9] was submitted as enclosure 5 of Luminant Power request to change Technical Specification (TS) 5.5.9.2, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program", and TS 5.6.10, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report" to support implementation of a permanent alternate repair criterion for steam generator tubes [Reference 8.10].

On July 23, 2009, Luminant Power received a request for additional information (RAI) letter, which contained twenty-four (24) questions [Reference 8.11]. As a result of a teleconference with NRC staff held on July 30, 2009, Luminant Power received a second request for additional information letter on August 11, 2009 [Reference 8.12]. The August 11, 2009 letter contained three (3) questions related to questions 4, 20 and 24 from RAI letter received on July 23, 2009. The August 11, 2009 letter also contained one (1) additional question.

On August 20, 2009, Luminant Power provided the response to all RAI questions except question 4 [Reference 8.13]. Enclosure 1 to Reference 8.13 is Westinghouse Letter LTR-SGMP-09-100 P-Attachment, Revision 0, "Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators," August 12, 2009 [Reference 8.14]. On October 7, 2010, Dominion Nuclear Connecticut, Inc. (Millstone Power Station Unit 3) submitted Westinghouse Letter LTR-SGMP-09-100 P-Attachment, Revision 1, "Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators," September 7, 2010, [Reference 8.29] to resolve editorial comments.

On August 27 2009, Luminant Power provided the response to RAI question 4 [Reference 8.15]. Enclosure 1 to Reference 8.15 is Westinghouse Letter LTR-SGMP-09-109 P-Attachment, Revision 0, "Response to NRC Request for Additional Information on H*; RAI #4; Model F and Model D5 Steam Generators," August 25, 2009 [Reference 8.16].

On August 28, 2009, Southern Nuclear Operating Company (Vogtle Electric Generating Plant – Units 1 and 2) submitted Westinghouse Letter LTR-SGMP-09-104 P-Attachment, Revision 1, "White Paper on Probabilistic Assessment of H*," dated August 13, 2009, [Reference 8.17] as supplemental information.

On September 14, 2009, Luminant Power submitted a request [Reference 8.18] to revise the permanent alternate repair criteria amendment request [Reference 8.10] to be an interim change applicable to Comanche Peak Unit 2 Refueling Outage 11 and the subsequent operating cycle. This request was made in response to a September 2, 2009 teleconference between NRC Staff and industry personnel, in which the NRC Staff indicated that their concerns with eccentricity of the tube sheet tube bore in normal and accident conditions (RAI question 4 of the July 10, 2009 letter and RAI question 1 of the August 5, 2009 letter) have not been resolved. The September 14, 2009 letter also requested the NRC staff to provide the specific questions concerning the tubesheet bore eccentricity issue which must be resolved to support a permanent alternate repair criteria amendment request.

On December 9, 2009, the NRC provided a letter [Reference 8.19] documenting the currently identified and unresolved issues relating to tubesheet bore eccentricity. This letter contained 14 questions which required resolution before the NRC could complete its review of a permanent amendment request.

WCAP-17330-P, Rev. 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5 Steam Generators)," November 2010, [Reference 8.1] LTR-SGMP-10-78, "Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and Their Relative Importance to H*," September 7, 2010, [Reference 8.20] and LTR-SGMP-10-33 P-Attachment, "H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity," September 13, 2010, [Reference 8.21] have been prepared by Westinghouse, to provide final resolution of the remaining questions identified in Reference 8.19 in support of a permanent H * amendment. Reference 8.18 is Enclosure 1 to this letter. Reference 8.20 was submitted to the NRC by Westinghouse Electric Company LLC Letter LTR-NRC-10-68 [Reference 8.22] dated November 9, 2010. Reference 8.21 was submitted to the NRC by Westinghouse Electric Company LLC Letter LTR-NRC-10-70 [Reference 8.23] dated November 11, 2010.

As a condition for approving Luminant Power Amendment 149 [Reference 8.2], One Cycle Alternate Repair Criterion, H* (H-star), the NRC required a commitment to measure the location of the bottom of the expansion transition (BET) relative to the top of the tubesheet (TTS) and report any significant deviations from the constant 0.3 inch value already included in the calculated value(s) of H*. Westinghouse Letter LTR-SGMP-09-111, Rev. 1, "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*," September 1, 2010, [Reference 8.24] was prepared to support plant determinations of BET measurements and their significant deviation assessment. Reference 8.24 was submitted to the NRC by Westinghouse Electric Company LLC Letter LTR-NRC-10-69 [Reference 8.25] dated November 10, 2010.

Westinghouse Letter LTR-SGMP-10-95 P-Attachment, "H*: Alternate Leakage Calculation Methods for H* for Situations When Contact Pressure at Normal Operating Conditions Exceeds Contact Pressure at Accident Conditions, Revision 1," September

2010 [Reference 8.26] considered the implication to the leak rate ratio between steam line break (SLB) conditions and normal operating pressure (NOP) conditions when the predicted contact pressure at SLB conditions are less than the predicted contact pressures at NOP conditions in the H* analysis. An evaluation was required because the NRC had applied a criterion that requires the SLB contact pressure exceed the NOP contact pressure at all elevations in the tubesheet. For the Model D5 SGs, this criterion could not be met. Reference 8.26 was submitted to the NRC by Westinghouse Electric Company LLC in letter LTR-NRC-10-60 [Reference 8.27] dated September 3, 2010.

The following table provides the list of licensing basis documents for H*.

Document Number	Revision Number	Title	Reference Number
WCAP-17330-P	0	H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/D5)	8.1
WCAP-17072-P	0	H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)	8.9
LTR -SGMP-09-109 P-Attachment	0	Response to NRC Request for Additional Information on H*; RAI #4; Model F and Model D5 Steam Generators	8.16
LTR-SGMP-09-104 P-Attachment	1	White Paper on Probabilistic Assessment of H*	8.17
LTR-SGMP-10-78 P-Attachment	0	Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and Their Relative Importance to H*	8.20
LTR-SGMP-10-33 P-Attachment	0	H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity	8.21
LTR-SGMP-10-95 P-Attachment	1	H*: Alternate Leakage Calculation Methods for H* for Situations When Contact Pressure at Normal Operating Conditions Exceeds Contact Pressure at Accident Conditions	8.26
LTR-SGMP-09-100 P-Attachment	1	Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators	8.29

4.2 Technical Analysis

To preclude unnecessarily plugging tubes in the CPNPP Unit 2 Steam Generators (SGs), tube inspections will be limited to identifying and plugging degradation in the portion of the tube within the tubesheet necessary to maintain structural and leakage integrity in both normal and accident conditions. The technical evaluation for the inspection and repair methodology is provided in the H* Analysis described in Section 4.1. This evaluation is based on the use of finite element model structural analysis and a bounding leak rate evaluation based on contact pressure between the tube and the tubesheet during normal and postulated accident conditions. The limited tubesheet inspection criteria were developed for the tubesheet region of the Comanche Peak Unit 2 Model D5 SGs considering the most stringent loads associated with plant operation, including transients and postulated accident conditions. The limited tubesheet inspection criteria were selected to prevent tube burst and axial separation due to axial pullout forces acting on the tube and to ensure that the accident induced leakage limits are not exceeded. The H* Analysis provides technical justification for limiting the inspection in the tubesheet expansion region to less than the full depth of the tubesheet.

The basis for determining the safety significant portion of the tube within the tubesheet is based upon evaluation and testing programs that quantified the tube-to-tubesheet radial contact pressure for bounding plant conditions as described in the H* Analysis. The tube-to-tubesheet radial contact pressure provides resistance to tube pullout and resistance to leakage during plant operation and transients.

Primary-to-secondary leakage from tube degradation in the tubesheet area is assumed to occur in several design basis accidents: feedwater line break (FLB), steam line break (SLB), locked rotor, and control rod ejection. The radiological dose consequences associated with this assumed leakage are evaluated to ensure that they remain within regulatory limits (e.g. 10 CFR Part 100, 10 CFR 50.67, GDC 19). The accident induced leakage performance criteria are intended to ensure the primary-to-secondary leak rate during any accident does not exceed the primary-to-secondary leak rate assumed in the accident analysis. Radiological dose consequences define the limiting accident condition for the H* justification.

The constraint that is provided by the tubesheet precludes tube burst for cracks within the tubesheet. The criteria for tube burst described in NEI 97-06 [Reference 8.3] and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," [Reference 8.28] are satisfied due to the constraint provided by the tubesheet. Through application of the limited tubesheet inspection scope as described below, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur. The accident induced leak rate limit for CPNPP is 1.0 gpm. The TS 3.4.13, "RCS Operational LEAKAGE," operational leak rate limit is 150 gpd (0.1 gpm) through any one steam generator. Consequently, there is significant margin between accident leakage and allowable operational leakage. The SLB/FLB overall leakage factor is only 3.16 resulting in significant margin between the conservatively estimated accident induced leakage and the allowable accident leakage (1.0 gpm).

Plant-specific operating conditions are used to generate the overall leakage factor ratios that are to be used in the condition monitoring and operational assessments. The plant-specific data provide the initial conditions for application of the transient input data. The results of the analysis of the plant-specific inputs, to determine the bounding plant for each model of steam generator are contained in Section 6 of Reference 8.9.

The leak rate ratio (accident induced leak rate to operational leak rate) is directly proportional to the change in differential pressure and inversely proportional to the dynamic viscosity. Since dynamic viscosity decreases with an increase in temperature, an increase in temperature results in an increase in leak rate.

However, for both the postulated SLB and FLB events, a plant cool down event would occur and the subsequent temperatures in the reactor coolant system (RCS) would not be expected to exceed the temperatures at plant no load conditions. Thus, an increase in leakage would not be expected to occur as a result of the viscosity change. The increase in leakage would only be a function of the increase in primary to secondary pressure differential.

The plant transient response following a full power double-ended main feedwater line rupture corresponding to "best estimate" initial conditions and operating characteristics, as generally presented in steam generator design transients and in the FSAR Chapter 15.0 safety analysis, indicates that the transient for a Model D5 SG exhibits a cooldown characteristic instead of a heat-up transient. The use of either the component design specification transient or the FSAR Chapter 15.0 safety analysis transient for leakage analysis for FLB is overly conservative because:

- The assumptions on which the FLB design transient is based are specifically intended to establish a conservative structural (fatigue) design basis for RCS components; however, H* does not involve component structural and fatigue issues. The best estimate transient is considered more appropriate for use in the H* leakage calculations.

- For the Model D5 SG FLB design transient, using the FLB design transient curve, the maximum RCS temperature can exceed the saturation temperature which is predicted to occur by the worst-case FLB heat-up FSAR Chapter 15 safety analysis transient response.

- The assumptions on which the FLB safety analysis is based are specifically intended to establish a conservative basis for minimum auxiliary feedwater (AFW) capacity and combines worst case assumptions, which are exceptionally more severe when the FLB occurs inside containment. For example, environmental errors that are applied to reactor trip and engineered safety features actuation would no longer be applicable. This would result in much earlier reactor trip and greatly increase the SG liquid mass available to provide cooling to the RCS.

A SLB event would have similarities to a FLB except that the break flow path would include the secondary separators, which could only result in an increased initial cooldown (because of retained liquid inventory available for cooling) when compared to the FLB transient. A SLB could not result in more limiting temperature conditions than a FLB.

In accordance with plant operating procedures, the operator would take action following a high energy secondary line break to stabilize the RCS conditions. The expectation for a SLB or FLB with credited operator action is to stop the system cooldown through isolation of the faulted steam generator and control of temperature using the AFW system. Steam pressure control would be established by either the steam generator safety valves or the atmospheric relief valves. For any of the steam pressure control operations, the maximum temperature would be approximately the no load temperature and would be well below normal operating temperature.

Since the best estimate FLB transient temperature would not be expected to exceed the normal operating temperature, the viscosity ratio for the FLB transient is set to 1.0.

The leakage factor of 3.16 for CPNPP Unit 2, for a postulated SLB/FLB, has been calculated as shown in Revised Table 9-7 of Reference 8.29. Specifically, for the condition monitoring (CM) assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.16 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 3.16 and compared to the observed operational leakage.

The other design basis accidents, such as the postulated locked rotor event and the control rod ejection event, are conservatively modeled using the design specification transients that result in increased temperatures in the SG hot and cold legs for a period of time. As previously noted, dynamic viscosity decreases with increasing temperature. Therefore, leakage would be expected to increase due to decreasing viscosity and increasing differential pressure for the duration of time that there is a rise in RCS temperature. For transients other than a SLB and FLB, the length of time that a plant with Model D5 SGs will exceed the normal operating differential pressure across the tubesheet is less than 30 seconds. As the accident induced leakage performance criteria is defined in gallons per minute, the leak rate for a locked rotor event can be integrated over a minute for comparison to the limit. Time integration permits an increase in acceptable leakage during the time of peak pressure differential by approximately a factor of two because of the short duration (less than 30 seconds) of the elevated pressure differential. This translates into an effective reduction in the leakage factor by the same factor of two for the locked rotor event. Therefore, for the locked rotor event, the leakage factor of 1.59 [Table 9-7, Reference 8.29] for Comanche Peak Unit 2 is adjusted downward to a factor of 0.79. Similarly, for the control rod ejection event, the duration of the elevated pressure differential is less than 10 seconds. Thus, the peak leakage factor is reduced by a factor of six, from 2.65 to 0.44. Due to the short duration of the transients above NOP differential, no leakage factor is required for the locked rotor and control rod ejection events (i.e., the leakage factor is under 1.0 for both transients).

The FLB heat-up transient definition is not a concern for the H* structural analysis. As shown in Figure 4-1 through Figure 4-6 of WCAP-17330-P [Reference 8.1], the FLB heatup event tube to tubesheet contact pressures are significantly higher than the SLB and NOP condition contact pressure. Additionally, the FLB cooldown event contact pressures would be similar to the SLB event which is also a cooldown event. Therefore, the FLB heatup event would not be a driving factor to limit the H* depth within the structural analysis.

Reference 8.9 redefines the primary pressure boundary. The tube-to-tubesheet weld no longer functions as a portion of this boundary. The hydraulically expanded portion of the tube into the tubesheet over the H* distance now functions as the primary pressure boundary in the area of the tube and tubesheet, maintaining the structural and leakage integrity over the full range of steam generator operating conditions, including the most limiting accident conditions. The evaluation in Reference 8.9 determined that degradation in tubing below this safety significant portion of the tube does not require inspection or repair (plugging). The inspection of the safety significant portion of the tubes provides a high level of confidence that the structural and leakage performance criteria are maintained during normal operating and accident conditions.

Although WCAP-17072-P [Reference 8.9] determined the H* inspection 95/50 whole plant depth of 13.8 inches from the top of the tubesheet and WCAP-17330-P [Reference 8.1] determined the 95/95 whole bundle depth of 13.36 inches from the top of the tubesheet, CPNPP is conservatively adding margin to this value and an H* inspection and plugging or repair depth of 16.95 inches from the top of the tubesheet is used. This additional margin is consistent with a 95/95 whole plant value used in the current licensing basis.

Section 9.8 of Reference 8.9 provides a review of leak rate susceptibility to tube slippage and concluded that the tubes are fully restrained against motion under very conservative design and analysis assumptions such that tube slippage is not a credible event for any tube in the bundle. As a condition of approval of Amendment 149, Luminant Power committed to monitor for tube slippage as part of the steam generator tube inspection program. This requirement will remain in place with the approval of this amendment request.

As a condition for approving Amendment 149 [Reference 8.2], the NRC staff requested that Luminant Power perform a validation of the tube expansion from the top of the tubesheet to the beginning of expansion transition (BET) to determine if there are any significant deviations that would invalidate assumptions in WCAP-17072-P. Luminant Power has completed the validation of the tube expansion from the top of tubesheet to the BET. Based on data review and LTR-SGMP-09-111, Rev. 1 [Reference 8.24], Luminant Power did not identify any significant deviations from the top of tubesheet to the BET for CPNPP Unit 2.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

This amendment application proposes to revise Technical Specification (TS) 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program," to exclude portions of the tubes within the tubesheet from periodic steam generator inspections. In addition, this amendment proposes to revise Technical Specification (TS) 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report" to provide reporting requirements specific to the temporary alternate repair criteria. Application of the structural analysis and leak rate evaluation results, to exclude portions of the tubes from inspection and repair, is interpreted to constitute a redefinition of the primary-to-secondary pressure boundary.

The proposed change defines the portion of the tube that must be inspected and repaired. A justification has been developed by Westinghouse Electric Company, LLC to identify the specific inspection depth below which any type of axial or circumferential primary water stress corrosion cracking can be shown to have no impact on Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," [Reference 8.3] performance criteria.

Luminant Power has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. *Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?*

   Response: No

   Of the accidents previously evaluated, the limiting transients with consideration to the proposed change to the SG tube inspection and repair criteria are the steam generator tube rupture (SGTR) event, the steam line break (SLB), and the feed line break (FLB) postulated accidents.

   The required structural integrity margins of the SG tubes and the tube-to-tubesheet joint over the H* distance will be maintained. Tube rupture in tubes with cracks within the tubesheet is precluded by the constraint provided by the presence of the tubesheet and the tube-to-tubesheet joint. Tube burst cannot occur within the thickness of the tubesheet. The tube-to-tubesheet joint constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet, differential pressure between the primary and secondary side, and tubesheet rotation. Based on this design, the structural margins against burst, as discussed in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," and TS 5.5.9 are maintained for both normal and postulated accident conditions.

   The proposed change has no impact on the structural or leakage integrity of the portion of the tube outside of the tubesheet. The proposed change maintains structural and leakage integrity of the SG tubes consistent with the performance criteria in TS 5.5.9. Therefore, the proposed change results in no significant increase in the probability of the occurrence of a SGTR accident.

   At normal operating pressures, leakage from tube degradation below the proposed limited inspection depth is limited by the tube-to-tubesheet crevice. Consequently, negligible normal operating leakage is expected from degradation below the inspected depth within the tubesheet region. The consequences of an SGTR event are not affected by the primary-to-secondary leakage flow during the event as primary-to-secondary leakage flow through a postulated tube that has been pulled out of the tubesheet is essentially equivalent to a severed tube. Therefore, the proposed change does not result in a significant increase in the consequences of a SGTR.

   The probability of a SLB is unaffected by the potential failure of a steam generator tube as the failure of tube is not an initiator for a SLB event.

   The leakage factor of 3.16 for CPNPP Unit 2, for a postulated SLB/FLB, has been calculated as described in Reference 8.29 and is shown in Revised Table 9-7 of this same reference. Specifically, for the condition monitoring (CM) assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 3.16 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 3.16 and compared to the observed operational leakage. The accident-induced leak rate limit for CPNPP Unit 2 is 1.0 gpm. The TS operational leak rate limit through any one steam generator is 150 gpd (0.1 gpm). Consequently, there is significant margin between accident leakage and allowable operational leakage. The SLB/FLB overall leakage factor is 3.16 resulting in significant margin between the conservatively estimated accident induced leakage and the allowable accident leakage.

No leakage factor was applied to the locked rotor or control rod ejection transients due to their short duration.

The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the SG inspection and reporting criteria does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated

2. *Do the proposed changes create the possibility of a new or different kind of accident from any previously evaluated?*

Response: No

The proposed change that alters the steam generator inspection and reporting criteria does not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident analyses. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. *Do the proposed changes involve a significant reduction in the margin of safety?*

Response: No

The proposed change that alters the steam generator inspection and reporting criteria maintains the required structural margins of the SG tubes for both normal and accident conditions. Nuclear Energy Institute 97-06, Rev.2, "Steam Generator Program Guidelines," and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, "Reactor Coolant

Pressure Boundary," GDC 15, "Reactor Coolant System Design," GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," and GDC 32, "Inspection of Reactor Coolant Pressure Boundary," by reducing the probability and consequences of a SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation, the probability and consequences of a SGTR are reduced. RG 1.121 uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, the H* Analysis documented in Section 4.1 defines a length of degradation-free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited hot and cold leg tubesheet inspection criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the proposed limited tubesheet inspection depth criteria.

Therefore, the proposed change does not involve a significant reduction in any margin of safety.

The safety significant portion of the tube is the length of tube that is engaged within the tubesheet to the top of the tubesheet (secondary face) that is required to maintain structural and leakage integrity over the full range of steam generating operating conditions, including the most limiting accident conditions. The H* Analysis determined that degradation in tubing below the safety significant portion of the tube does not require plugging and serves as the basis for the limited tubesheet inspection criteria, which are intended to ensure the primary-to-secondary leak rate during any accident does not exceed the leak rate assumed in the accident analysis.

Based on the above evaluations, Luminant Power concludes that the proposed amendment presents no significant hazards under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

General Design Criteria (GDC) 1, 2, 4, 14, 30, 31, and 32 of 10 CFR 50, Appendix A, define requirements for the reactor coolant pressure boundary (RCPB) with respect to structural and leakage integrity.

GDC 19 of 10 CFR 50, Appendix A, defines requirements for the control room and for the radiation protection of the operators working within it. Accidents involving the leakage or burst of SG tubing comprise a challenge to the habitability of the control room.

10 CFR 50, Appendix B, establishes quality assurance requirements for the design, construction, and operation of safety related components. The pertinent requirements

of this appendix apply to all activities affecting the safety related functions of these components. These requirements are described in Criteria IX, XI, and XVI of Appendix B and include control of special processes, inspection, testing, and corrective action.

10 CFR 100, Reactor Site Criteria, establishes reactor site criteria, with respect to the risk of public exposure to the release of radioactive fission products. Accidents involving leakage or tube burst of SG tubing may comprise a challenge to containment and therefore involve an increased risk of radioactive release.

Under 10 CFR 50.65, the Maintenance Rule, licensees classify SGs as risk significant components because they are relied upon to remain functional during and after design basis events. SGs are to be monitored under 10 CFR 50.65(a)(2) against industry established performance criteria. Meeting the performance criteria of NEI 97-06, Revision 2, provides reasonable assurance that the SG tubing remains capable of fulfilling its specific safety function of maintaining the reactor coolant pressure boundary. The NEI 97-06, Revision 2, SG performance criteria are:

- All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cool down, and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design and licensing basis shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial loads.

- The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG, except for specific types of degradation at specific locations when implementing alternate repair criteria as documented in the Steam Generator Program Technical Specifications.

- The RCS operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day.

The safety significant portion of the tube is the length of the tube that is engaged in the tubesheet from the secondary face that is required to maintain structural and leakage integrity over the full range of SG operating conditions, including the most limiting accident conditions. The evaluation in this attachment determined that the degradation in tubing below the safety significant portion of the tube does not require plugging and serves as the basis for the tubesheet inspection program. As such, the CPNPP inspection program provides a high level of confidence that the structural and leakage criteria are maintained during normal operating and accident conditions.

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

6.0 ENVIRONMENTAL CONSIDERATIONS

Luminant Power has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. Luminant Power has evaluated the proposed changes and has determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22 (c)(9). Therefore, pursuant to 10CFR51.22 (b), an environmental assessment of the proposed change is not required

7.0 PRECEDENTS

The changes to Technical Specifications 5.5.9 and 5.6.9 are similar to changes submitted by Vogtle Electric Generating Plant listed below.

7.1 Vogtle Electric Generating Plant License Amendment Request to Revise Technical Specification (TS) Sections 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" for Temporary Alternate Repair Criteria, dated November 23, 2010.

8.0 REFERENCES

8.1 Westinghouse Electric Company WCAP-17330-P, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)," November 2010.

8.2 Letter dated October 9, 2009, from Balwant K. Singal, USNRC, to Rafael Flores, Luminant Generation 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 (TAC NOS. ME1446 and ME1447)."

8.3 NEI 97-06, "Steam Generator Program Guidelines," Revision 2, May 2005.

8.4 EPRI 1013706, "Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 7," October 2007.

8.5 EPRI 1019038, "Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines," Revision 3, November 2009.

8.6 CPNPP Procedure STA-733 "Steam Generator Reliability Program," Revision 12.

8.7 NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," April 7, 2005.

8.8 NRC Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections," August 30, 2004.

8.9 Westinghouse Electric Company WCAP-17072-P," H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model D5)" May, 2009

8.10 Luminant Power Letter TXX-09075, "License Amendment Request 09-007, Model D5 Steam Generator Alternate Repair Criteria," dated June 8, 2009.

8.11 Letter dated July 23, 2009, from Balwant K. Singal, USNRC, to Rafael Flores, Luminant Generation Company LLC, "Comanche Peak Steam Electric Station, Units 1 and 2 – Request for Additional Information Regarding the Permanent Alternate Repair Criteria License Amendment Request (TAC NOS. ME1446 and ME1447)."

8.12 Letter dated August 11, 2009, from Balwant K. Singal, USNRC, to Rafael Flores, Luminant Generation Company LLC, "Comanche Peak Steam Electric Station, Units 1 and 2 – Request for Additional Information Regarding the Permanent Alternate Repair Criteria License Amendment Request (TAC NOS. ME1446 and ME1447)."

8.13 Luminant Power Letter TXX-09096, "Response to Request for Additional Information Regarding License Amendment Request 09-007, Model D5 Steam Generator Alternate Repair Criteria," dated August 20, 2009.

8.14 Westinghouse Letter LTR-SGMP-09-100 P-Attachment, Revision 0, "Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators," dated August 12, 2009.

8.15 Luminant Power Letter TXX-09105, "Response to Request for Additional Information Regarding License Amendment Request 09-007, Model D5 Steam Generator Alternate Repair Criteria," dated August 27, 2009.

8.16 Westinghouse Letter LTR-SGMP-09-109 P-Attachment, Revision 0, "Response to NRC Request for Additional Information on H*; RAI #4; Model F and Model D5 Steam Generators," dated August 25, 2009.

8.17 Westinghouse Letter LTR-SGMP-09-104 P-Attachment, Revision 1, "White Paper on Probabilistic Assessment of H*," dated August 13, 2009.

8.18 Luminant Power Letter TXX-09113, "Revision to License Amendment Request 09-007, Model D5 Steam Generator Alternate Repair Criteria," dated September 14, 2009.

8.19 Letter dated December 9, 2009, from Balwant K. Singal, USNRC, to Rafael Flores, Luminant Generation Company LLC, "Comanche Peak Steam Electric Station, Units 1 and 2 – Transmittal of Unresolved Issues Regarding Permanent Alternate Repair Criteria for Steam Generators (TAC NOS. ME1446 and ME1447)."

8.20 Westinghouse Letter LTR-SGMP-10-78 P-Attachment, Revision 0, "Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and Their Relative Importance to H*," dated September 7, 2010.

8.21 Westinghouse Letter LTR-SGMP-10-33 P-Attachment, Revision 0, "H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity," dated September 13, 2010.

8.22 Westinghouse Letter LTR-NRC-10-68," Submittal of LTR-SGMP-10-78 P-Attachment," dated November 9, 2010.

8.23 Westinghouse Letter LTR-NRC-10-70," Submittal of LTR-SGMP-10-33 P-Attachment," dated November 11, 2010.

8.24 Westinghouse Letter LTR-SGMP-09-111 P-Attachment, Revision 1, "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*," dated September 1, 2010.

8.25 Westinghouse Letter LTR-NRC-10-69," Submittal of LTR-SGMP-09-111 P-Attachment, Revision 1," dated November 10, 2010.

8.26 Westinghouse Letter LTR-SGMP-10-95 P-Attachment, "H*: Alternate Leakage Calculation Methods for H* for Situations When Contact Pressure at Normal Operating Conditions Exceeds Contact Pressure at Accident Conditions, Revision 1", dated September 2010.

8.27 Westinghouse Letter LTR-NRC-10-60," Submittal of LTR-SGMP-10-95 P-Attachment, Revision 1," dated September 3, 2010.

8.28 NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976.

8.29 Westinghouse Letter LTR-SGMP-09-100 P-Attachment, Revision 1, "Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators," dated September 7, 2010.

ATTACHMENT 2 TO TXX-10152

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARKUP)

5.5 Programs and Manuals (continued)

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

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as-found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

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

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 ~~11~~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 ~~11~~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 ~~11~~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 ~~11~~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 ~~11~~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 ~~11~~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.

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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.

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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."

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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.