WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President Engineering

November 30, 2010

ET 10-0030

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

Subject: Docket No. 50-482: Revision to Technical Specifications 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator Tube Inspection Report," for a Temporary Alternate Repair Criterion

Gentlemen:

Pursuant to 10 CFR 50.90, Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Renewed Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS). This amendment request proposes to revise Wolf Creek Generating Station (WCGS) Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," to exclude portions of the tube below the top of the steam generator tubesheet from periodic steam generator tube inspections during Refueling Outage 18 and the subsequent operating cycle. In addition, this amendment proposes to revise Technical Specification (TS) 5.6.10, "Steam Generator Tube Inspection Report" to remove reference to previous interim alternate repair criteria and provide reporting requirements specific to the temporary alternate repair criteria. This change is supported by the analysis described in Section 3 of Attachment I including Westinghouse Electric Company LLC WCAP-17330-P, Revision 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)."

WCNOC requests the approval of the proposed license amendment by April 8, 2011 to support implementation during the WCGS spring 2011 refueling outage. Once approved, the amendment will be implemented prior to MODE 4 entry during startup from Refueling Outage 18.

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Enclosure I provides the proprietary Westinghouse Electric Company LLC WCAP-17330-P, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)." Enclosure II provides the non- proprietary Westinghouse Electric Company LLC WCAP-17330-NP, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)." As Enclosure I contains information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse Electric Company LLC, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 2.390 of the Commission's regulations. This affidavit, along with Westinghouse authorization letter, CAW-10-3001, "Application for Withholding Proprietary Information from Public Disclosure," is contained in Enclosure III.

Attachment I through IV provide the Evaluation, Markup of TSs, proposed TS Bases changes, and Retyped TS pages, respectively, in support of this amendment request. Attachment III, proposed changes to the TS Bases, is provided for information only. Final TS Bases changes will be implemented pursuant to TS 5.5.14, "Technical Specification (TS) Bases Control Program," at the time the amendment is implemented.

On October 19, 2009, the NRC issued WCGS Amendment Number 186 for a one-cycle alternate repair criterion. As a condition of approval, WCNOC made regulatory commitments to:

- WCNOC commits to monitor for tube slippage as part of the steam generator tube inspection program. Slippage monitoring will occur for each inspection of the WCGS steam generators.

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

The program/procedure changes needed to meet these commitments were completed in accordance with the NRC issuance of Amendment Number 186. The changes will also apply to this license amendment request. As such, these changes will remain in place. Therefore, no new NRC commitments are required.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

The Plant Safety Review Committee reviewed this amendment application. In accordance with 10 CFR 50.91, a copy of this amendment application, with attachments, is being provided to the designated Kansas State official.

If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Richard D. Flannigan at (620) 364-4117.

Sincerely,

Terry J. Garrett

TJG/rlt

Attachments: I - Evaluation
II - Markup of Technical Specification pages
III - Markup of Technical Specification Bases pages (for information only)
IV - Retyped Technical Specification pages

Enclosure I - Westinghouse Electric Company LLC WCAP-17330-P, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)"
II - Westinghouse Electric Company LLC WCAP-17330-NP, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)"
III - Westinghouse Electric Company LLC LTR-CAW-10-3001, "Application for Withholding Proprietary Information from Public Disclosure"

cc: E. E. Collins (NRC), w/a, w/e
T. A. Conley (KDHE), w/a, w/e (Enclosure II only)
G. B. Miller (NRC), w/a, w/e
B. K. Singal (NRC), w/a, w/e
Senior Resident Inspector (NRC), w/a, w/e

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Terry J. Garrett, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By________________________
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 30th day of November, 2010.

RHONDA L. TIEMEYER
OFFICIAL SEAL
MY COMMISSION EXPIRES
Rhonda L. Tiemeyer

Rhonda L. Tiemeyer
Notary Public

Expiration Date January 11, 2014

EVALUATION

Subject: Revision to Technical Specifications 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator Tube Inspection Report," for a Temporary Alternate Repair Criterion

1. SUMMARY DESCRIPTION

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1. SUMMARY DESCRIPTION

Wolf Creek Nuclear Operating Corporation (WCNOC) proposes to revise Wolf Creek Generating Station (WCGS) Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," to exclude portions of the tube below the top of the steam generator tubesheet from periodic steam generator tube inspections. In addition, this amendment proposes to revise Technical Specification (TS) 5.6.10, "Steam Generator Tube Inspection Report" to remove reference to previous interim alternate repair criteria and provide reporting requirements specific to the temporary alternate repair criteria. 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 WCAP 17330-P (Reference 19) provides the 95/95 H* value of 15.2 inches for plants with Model F Steam Generators which includes WCGS.

The NRC previously issued the following amendments revising steam generator tube inspection requirements:

- Amendment Number 162 (Reference 1) to exclude degradation found in the portion of the tubes below 17 inches from the top of the hot leg tubesheet from the requirement to plug for Refueling Outage 14 and the subsequent operating cycle.

- Amendment Number 169 (Reference 2) to exclude the portion of the tubes below 17 inches from the top of the hot leg tubesheet from the requirement to plug for Refueling Outage 15 and the subsequent operating cycle.

- Amendment Number 178 (Reference 3) which approved an interim alternate repair criteria for Refueling Outage 16 and the subsequent operating cycle that requires full-length inspection of the tubes within the tubesheet but does not require plugging tubes if any circumferential cracking observed in the region greater than 17 inches from the top of the tubesheet is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads. Three new reporting requirements were added to TS 5.6.10.

- Amendment Number 186 (Reference 4) revised TS 5.5.9, "Steam Generator (SG) Program," to eliminate inspection and repair of tubes more than 13.1 inches below the top of the tubesheet for Refueling Outage 17 and the subsequent operating cycle. Additionally TS 5.6.10 was revised to provide reporting requirements specific to Refueling Outage 17 and the subsequent operating cycle.

Approval of this amendment application is requested by April 8, 2011 to support the WCGS Refueling Outage 18 (Spring 2011), since the existing one-cycle amendment expires at the end of the operating cycle.

2. DETAILED DESCRIPTION

Proposed Changes to Current TS

TS 5.5.9c. 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.

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

1. For Refueling Outage 17 and the subsequent operating cycle, tubes with service-induced flaws located greater than 13.1 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 13.1 inches below the top of the tubesheet shall be plugged upon detection.

This section 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.

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

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

TS 5.5.9d. currently states:

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Refueling Outage 17 and the subsequent operating cycle, the portion of the tube below 13.1 inches from the top of the tubesheet is excluded from this requirement. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. 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. 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. If crack indications are found in any portion of the SG tube not excluded above, 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.

This section would be revised as follows, as noted in italic/underline type:

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Refueling Outage *18* and the subsequent operating cycle, the portion of the tube below *15.2* inches from the top of the tubesheet is excluded from this requirement. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. 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. 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. If crack indications are found in any portion of the SG tube not excluded above, 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 5.6.10h., 5.6.10i., and 5.6.10j. currently state:

h. Following completion of an inspection performed in Refueling Outage 17 (and any inspections performed in the subsequent operating cycle) the primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report;

i. Following completion of an inspection performed in Refueling Outage 17 (and any inspections performed in the subsequent operating cycle) the calculated accident induced leakage rate from the portion of the tubes below 13.1 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.50 times the maximum operational primary to secondary leak rate, the report should describe how it was determined; and

j. Following completion of an inspection performed in Refueling Outage 17 (and any inspections performed in the subsequent operating cycle) the results of monitoring for the tube axial displacement (slippage). If slippage is discovered, the implications of discovery and corrective action shall be provided.

TS 5.6.10h., 5.6.10i., and 5.6.10j. would be revised as follows, as noted in italic/underline type:

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

i. Following completion of an inspection performed in Refueling Outage _18_ (and any inspections performed in the subsequent operating cycle) the calculated accident induced leakage rate from the portion of the tubes below _15.2_ inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.50 times the maximum operational primary to secondary leak rate, the report should describe how it was determined; and

j. Following completion of an inspection performed in Refueling Outage _18_ (and any inspections performed in the subsequent operating cycle) the results of monitoring for the tube axial displacement (slippage). If slippage is discovered, the implications of discovery and corrective action shall be provided.

3. TECHNICAL EVALUATION

Background

WCGS is a four loop Westinghouse designed plant with Model F steam generators having 5626 tubes in each steam generator. A total of 251 tubes are currently plugged in all four steam generators. The design of the steam generator includes Alloy 600 thermally treated tubing, full depth hydraulically expanded tubesheet joints, and stainless steel tube support plates with broached hole quatrefoils.

The steam generator inspection scope is governed by TS 5.5.9, "Steam Generator (SG) Program;" Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," (Reference 5); EPRI 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines," (Reference 6); EPRI 1019038; "Steam Generator Integrity Assessment Guidelines," (Reference 7); WCNOC procedure AP 29A-003, "Steam Generator Management;" and the results of the degradation assessments required by the Steam Generator Program. 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 WCGS. The inspection techniques, essential variables and equipment are qualified to Appendix H, "Performance Demonstration for Eddy Current Examination" of the EPRI Steam Generator Examination Guidelines.

Catawba Nuclear Station, Unit 2, (Catawba) reported indication of cracking following nondestructive eddy current examination of the steam generator 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), 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.

WCNOC policies and programs, as well as TS 5.5.9, require the use of applicable industry operating experience in the operation and maintenance of WCGS. The 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"). Since the WCGS Westinghouse Model F 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 WCGS to identify tube indications similar to those reported at Catawba within the hot leg tubesheet region.

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

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 steam generator 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 WCGS steam generators. 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 operating experience discussed above, WCNOC revised the steam generator inspection plan to include sampling of bulges and overexpansions within the tubesheet region on the hot leg side in Refueling Outage 14 (Spring 2005) and Refueling Outage 15 (Fall 2006). The sample is based on the guidance contained in EPRI 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 7, and TS 5.5.9, "Steam Generator (SG) Program." The inspection plan is expanded according to EPRI steam generator examination guidelines if necessary due to confirmed degradation in the region required to be examined (i.e. a tube crack). Degradation was not detected in the tubesheet region in Refueling Outage 14 and Refueling Outage 15.

At WCGS, tube flaw indications within the tube sheet have only been found at the hot leg tube ends. Approximately 18,414 tube ends were inspected at WCGS during Refueling Outage 16. Seventy-six flaw indications have been found in the inspections within 1 inch of the tube end. Of these seventy-six flaw indications, only eight met the tube repair criteria in the technical specifications.

Based on these inspections, a limited number of tube flaws existed in the tubesheet area of the WCGS 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 have been detected in any steam generator at WCGS. Consequently, the level of degradation in the WCGS steam generators is very limited compared to the assumption of "all tubes severed" that was utilized in the development of the permanent H* alternate repair criterion. Consequently, structural integrity will be assured for the operating period between inspections allowed by the proposed TS 5.5.9, "Steam Generator (SG) Program."

As a result of these potential issues and to prevent the unnecessarily plugging of additional tubes in the WCGS steam generators, WCNOC is proposing changes to TS 5.5.9 to limit the steam generator tube inspection and repair (plugging) to the safety significant portion of the tubes.

<u>Summary of Licensing Basis Analysis (H* Analysis)</u>

On June 2, 2009, Westinghouse WCAP-17071-P, Revision 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," (Reference 10) was submitted as Enclosure I of WCNOC request to change Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program", and TS 5.6.10, "Steam Generator Tube Inspection Report" to support implementation of a permanent alternate repair criterion for steam generator tubes. (Reference 11)

On August 11, 2009, WCNOC received a request for additional information (RAI) letter (Reference 12), which contained twenty-five (25) questions.

On August 25, 2009 (Reference 13) and September 3, 2009 (Reference 14), WCNOC provided the responses to questions 1 through 25 of the August 10, 2009 letter and included the following documents:

- 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 15) and

- Westinghouse letter 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 16)

On September 15, 2009, WCNOC submitted a request (Reference 17) to revise the permanent alternate repair criteria amendment request (Reference 11) to be an interim change applicable to Refueling Outage 17 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 August 11, 2009 letter) have not been resolved. The September 15, 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 18) 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/Model D5), November 2010 (Reference 19), LTR-SGMP-10-78 P-Attachment, "Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and Their Relative Importance to H*," September 7, 2009 (Reference 20) and LTR-SGMP-10-33 P-Attachment, "H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity," September 13, 2010 (Reference 21) have been prepared by Westinghouse Electric Company LLC, to provide final resolution of the remaining questions identified in the December 9, 2009 NRC letter in support of a permanent H* amendment request. LTR-SGMP-10-78 P-Attachment was submitted to the NRC by Westinghouse Electric Company LLC in letter LTR-NRC-10-68 on November 9, 2010. LTR-SGMP-10-33 P-Attachment was submitted to the NRC by Westinghouse Electric Company LLC in letter LTR-NRC-10-70 on November 11, 2010.

WCAP-17330-P, Rev. 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5), November 2010 (Reference 19) makes reference to Revision 2 of WCAP-17071-P and Revision 1 of LTR-SGMP-09-100 P-Attachment. As described above, WCNOC has previously submitted Revision 0 of these documents. These revisions (Revisions 1 and 2 of WCAP-17071-P, Revision 1 of LTR-SGMP-09-100 P-Attachment) were created to resolve editorial comments. The technical information contained in WCAP-17071-P, Revision 0 and LTR-SGMP-09-100 P-Attachment, Revision 0, remains valid and provides part of the licensing basis for the requested amendment.

As a condition for approving WCGS Amendment Number 178 (Reference 3), 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*. LTR-SGMP-09-111 P-Attachment, Rev. 1, "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*" (Reference 22), was prepared to support plant determinations of BET measurements and their significant deviation assessment. LTR-SGMP-09-111 P-Attachment was submitted to the NRC by Westinghouse Electric Company LLC in letter LTR-NRC-10-69 on November 10, 2010.

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

Document Number	Revision Number	Title	Reference Number
WCAP-17071-P	0	H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F),"	10
LTR-SGMP-09-100 P-Attachment	0	Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators	15
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	16
WCAP-17330-P	0	H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)	19
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*	20
LTR-SGMP-10-33 P-Attachment	0	H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity	21

Evaluation

To preclude unnecessarily plugging tubes in the WCGS steam generators, an evaluation was performed to identify the safety significant portion of the tube within the tubesheet necessary to maintain structural and leakage integrity in both normal and accident conditions. Tube inspections will be limited to identifying and plugging degradation in the safety significant portion of the tubes. The technical evaluation for the inspection and repair methodology is provided in the H* Analysis as described above. 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 WCGS Model F steam generator 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 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* Analysis.

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 and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," (Reference 9) 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 WCGS 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, accident leakage is approximately 10 times the allowable leakage, if only one steam generator is leaking. Using a SLB/FLB overall leakage factor of 2.50, accident induced leakage is approximately 0.5 gpm, if all 4 steam generators are leaking at 150 gpd at the beginning of the accident. Therefore, significant margin exists 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 10.

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.

For both the postulated SLB and FLB events, a plant cooldown 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. However, per Reference 15, the FLB transient was evaluated as a heatup event. The resulting leak rate ratio for the SLB and FLB events is 2.50.

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 to result in increased temperatures in the steam generator 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 a FLB, the length of time that a plant with Model F steam generators 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 ejection event can be integrated over a minute to compare 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 leakage factor by the same factor of two for the locked rotor event. Therefore, for the locked rotor event, the leakage factor of 1.77 (Revised Table 9-7, Reference 15) for WCGS is adjusted downward to a factor of 0.89. Similarly, for the control rod ejection event, the duration of the elevated pressure differential is less than 10 seconds. Thus, the peak leakage factor may be reduced by a factor of six from 2.65 to 0.44.

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 the Updated Safety Analysis Report (USAR) Chapter 15.0 safety analysis, indicates that the transient for a Model F steam generator exhibits a cooldown characteristic instead of a heatup transient. The use of either the component design specification transient or the Chapter 15.0 safety 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 F steam generator, the FLB transient curve (Figure 9-5, Reference 10) represents a double-ended rupture of the main feedwater line concurrent with both station blackout (loss of main feedwater and reactor coolant pump coast down) and turbine trip.

- The assumptions on which the FLB safety analysis is based are specifically intended to establish a conservative basis for minimum auxiliary feedwater (AFW) capacity requirements 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 feature actuation would no longer be applicable. This would result in much earlier reactor trip and greatly increase the steam generator 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 by the AFW System. Steam pressure control would be established by either the steam generator safety valves or control system (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 2.50 for WCGS, for a postulated SLB/FLB, has been calculated as shown in Revised Table 9-7 of Reference 15. 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 2.50 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.50 and compared to the observed operational leakage.

Reference 11 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 11 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.

WCAP-17071-P, section 9.8, provides a review of leak rate susceptibility due 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 Number 186, WCNOC committed to monitor for tube slippage as part of the steam generator tube inspection program. This commitment will remain in place with the approval of this amendment request.

As a condition for approving the WCGS Interim Alternate Repair Criterion (Reference 3), the NRC staff requested that WCNOC perform a validation of the tube expansion from the top of tubesheet to the beginning of expansion transition (BET) to determine if there are any significant deviations that would invalidate assumptions in WCAP-17071-P (Reference 11). WCNOC 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 P-Attachment, Rev. 1 (Reference 22), WCNOC did not identify any significant deviations from the top of tubesheet to the BET for WCGS.

4. REGULATORY EVALUATION

4.1 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 steam generator 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, established reactor siting criteria, with respect to the risk of public exposure to the release of radioactive fission products. Accidents involving leakage or tube burst of steam generator 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 steam generators as risk-significant components because they are relied upon to remain functional during and after design basis events. Steam generators are to be monitored under 10 CFR 50.65(a)(2) against industry established performance criteria. Meeting the performance criteria of Nuclear Energy Institute (NEI) 97-06, Revision 2, "Steam Generator Program Guidelines," provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining the reactor coolant pressure boundary. The NEI 97-06, Revision 2, steam generator performance criteria are:

- 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, cooldown 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 steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. Leakage is not to exceed 1 gpm per steam generator, except for specific types of degradation at specific locations when implementing alternate repair criteria as documented in the Steam Generator Program technical specifications.

- The Reactor Coolant System (RCS) operational primary to secondary leakage through any one steam generator shall be limited to 150 gallons per day.

The safety significant portion of the tube is the length of 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 steam generator operating conditions, including the most limiting accident conditions. The evaluation in this Attachment determined that degradation in tubing below the safety significant portion of the tube does not require plugging and serves as the bases for the tubesheet inspection program. As such, the WCGS inspection program provides a high level of confidence that the structural and leakage criteria are maintained during normal operating and accident conditions.

4.2 Significant Hazards Consideration

This amendment application proposes to revise Technical Specification (TS) 5.5.9, "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) 5.6.10, "Steam Generator Tube Inspection Report" to remove reference to previous interim alternate repair criteria and 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, Revision 2, "Steam Generator Program," performance criteria.

WCNOC 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) Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the steam generator inspection 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.

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

During the SGTR event, the required structural integrity margins of the steam generator 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 presence of the tubesheet and constraint provided by 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, from the differential pressure between the primary and secondary side, and tubesheet deflection. 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 steam generator 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 joint. 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 changes do not result in a significant increase in the consequences of a SGTR.

The consequences of a steam line break (SLB) are also not significantly affected by the proposed changes. During a SLB accident, the reduction in pressure above the tubesheet on the shell side of the steam generator creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet thereby restricting primary-to-secondary leakage below the midplane.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., an SLB) is limited by flow restrictions. These restrictions result from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications.

The leakage factor of 2.50 for WCGS, for a postulated SLB/FLB, has been calculated as shown in Revised Table 9-7 of Reference 15. 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 2.50 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.50 and compared to the observed operational leakage.

The probability of an SLB is unaffected by the potential failure of a steam generator tube as the failure of the tube is not an initiator for an SLB event. SLB leakage is limited by leakage flow restrictions resulting from the leakage path above potential cracks through the tube-to-tubesheet crevice. The leak rate during postulated accident conditions (including locked rotor) has been shown to remain within the accident analysis assumptions for all axial and or circumferentially orientated cracks occurring 15.2 inches below the top of the tubesheet. The accident induced leak rate limit for WCGS 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, accident leakage is approximately 10 times the allowable leakage, if only one steam generator is leaking. Using an SLB/FLB overall leakage factor of 2.50, accident induced leakage is approximately 0.5 gpm, if all 4 steam generators are leaking at 150 gpd at the beginning of the accident. Therefore, significant margin exists between the conservatively estimated accident induced leakage and the allowable accident leakage (1.0 gpm).

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change alters the steam generator inspection and reporting criteria. It 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 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 accident previously evaluated.

(3) Does the change involve a significant reduction in a margin of safety?

Response: No

The proposed change alters the steam generator inspection and reporting criteria. It maintains the required structural margins of the steam generator tubes for both normal and accident conditions. NEI 97-06, Revision 2, and RG 1.121, are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting 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. This RG 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 3, 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.

4.3 Conclusion

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 considerations above, 1) there is a 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.

5. ENVIRONMENTAL CONSIDERATION

WCNOC has evaluated the proposed amendment for environmental considerations. The review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, and would change an inspection or surveillance requirement. However, the proposed amendment does not involve (I) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set for in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6. REFERENCES

1. Letter dated April 28, 2005, from J. N. Donohew, USNRC, to R. A. Muench, WCNOC, "Wolf Creek Generating Station – Issuance of Exigent Amendment RE: Steam Generator (SG) Tube Surveillance Program (TAC NO. MC6757)."

2. Letter dated October 10, 2006, from J. N. Donohew, USNRC, to R. A. Muench, WCNOC, "Wolf Creek Generating Station – Issuance of Amendment RE: Steam Generator Tube Inspections Within The Tubesheet (TAC NO. MD2467)."

3. Letter dated April 4, 2008, from J. N. Donohew, USNRC, to R. A. Muench, WCNOC, "Wolf Creek Generating Station – Issuance of Amendment RE: Revision to Technical Specification 5.5.9 on the Steam Generator Program (TAC NO. MD8054)."

4. Letter dated October 19, 2009, from B. K. Singal, USNRC, to R. A. Muench, WCNOC, "Wolf Creek Generating Station – Issuance of Amendment RE: Revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10 "Steam Generator Tube Inspection Report," for Alternate Repair Criteria (TAC NO. ME1393)."

5. NEI 97-06, Rev. 2, "Steam Generator Program Guidelines," May 2005.

6. EPRI 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines."

7. EPRI 1019038; "Steam Generator Integrity Assessment Guidelines."

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

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

10. Westinghouse Electric Company LLC, WCAP-17071-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," April 2009.

11. WCNOC Letter ET 09-0016, "Revision to Technical Specification 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator Tube Inspection Report," for a Permanent Alternate Repair Criterion," June 2, 2009.

12. Letter dated August 11, 2009, from B. K. Singal, USNRC, to R. A. Muench, WCNOC, "Wolf Creek Generating Station – Request for Additional Information Regarding the Permanent Alternate Repair Criteria License Amendment Request (TAC NO. ME1393).

13. WCNOC Letter ET 09-0021, "Response to Request for Additional Information Related to License Amendment Request for a Permanent Alternate Repair Criterion to Technical Specification 5.5.9, "Steam Generator (SG) Program"," August 25, 2009.

14. WCNOC Letter ET 09-0023, "Response to Request for Additional Information Related to License Amendment Request for a Permanent Alternate Repair Criterion to Technical Specification 5.5.9, "Steam Generator (SG) Program"," September 3, 2009.

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

16. LTR-SGMP-09-109 P-Attachment, "Response to NRC Request For Additional Information on H*; RAI #4; Model F and Model D5 Steam Generators," August 25, 2009.

17. WCNOC Letter ET 09-0025, "Revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program", and TS 5.6.10, "Steam Generator Tube Inspection Report," September 15, 2009.

18. Letter dated December 9, 2009, from B. K. Singal, USNRC, to R. A. Muench, WCNOC, "Wolf Creek Generating Station – Transmittal of Unresolved Issues Regarding Permanent Alternate Repair Criteria for Steam Generators (TAC NO. ME1393)."

19. WCAP-17330-P, Rev. 0, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (Model F/Model D5)," November 2010.

20. LTR-NRC-10-68, "Submittal of LTR-SGMP-10-78 P-Attachment and LTR-SGMP-10-78 NP-Attachment, "Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and Their Relative Importance to H*," (Proprietary/Non-Proprietary) for Review and Approval," November 9, 2010.

21. LTR-NRC-10-70, Submittal of LTR-SGMP-10-33 P-Attachment and LTR-SGMP-10-33 NP-Attachment, LTR-SGMP-10-33 P-Attachment, "H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity," (Proprietary/Non-Proprietary) for Review and Approval," November 11, 2010.

22. LTR-NRC-10-69, Submittal of LTR-SGMP-09-111 P-Attachment, Rev. 1 and LTR-SGMP-09-111 NP-Attachment, Rev. 1, "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*," (Proprietary/Non-Proprietary) for Review and Approval," November 10, 2010.

ATTACHMENT II

Markup of Technical Specification pages

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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

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

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

1. For Refueling Outage ~~17~~ 18 and the subsequent operating cycle, tubes with service-induced flaws located greater than ~~13.1~~ 15.2 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to ~~13.1~~ 15.2 inches below the top of the tubesheet shall be plugged upon detection.

(continued)

5.5.9 Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Refueling Outage ~~17~~ 18 and the subsequent operating cycle, the portion of the tube below ~~13.1~~ 15.2 inches from the top of the tubesheet is excluded from this requirement. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. 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. 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. If crack indications are found in any portion of the SG tube not excluded above, 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.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

(continued)

5.6 Reporting Requirements

5.6.10 Steam Generator Tube Inspection Report

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

a. The scope of inspections performed on each SG;

b. 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;

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

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

i. Following completion of an inspection performed in Refueling Outage ~~17~~ 18 (and any inspections performed in the subsequent operating cycle) the calculated accident induced leakage rate from the portion of the tubes below ~~13.1~~ 15.2 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.50 times the maximum operational primary to secondary leak rate, the report should describe how it was determined; and

j. Following completion of an inspection performed in Refueling Outage ~~17~~ 18 (and any inspections performed in the subsequent operating cycle) the results of monitoring for the tube axial displacement (slippage). If slippage is discovered, the implications of discovery and corrective action shall be provided.

ATTACHMENT III

Markup of Technical Specification Bases pages (for information only)

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for an SGTR assumes the contaminated secondary fluid is released to the atmosphere via SG atmospheric relief valves and safety valves.

The analysis for design basis accidents and transients other than an SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

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In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. For Refueling Outage ~~17~~ and the subsequent operating cycle, a one-time alternate repair criterion for the portion of the tube below ~~13.1~~ inches from the top of the tubesheet is specified in TS 5.5.9c.1. (Ref. 7) The tube-to-tubesheet weld is not considered part of the tube. 15.2

ATTACHMENT IV

Retyped Technical Specification pages

5.5.9 Steam Generator (SG) Program (continued)

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

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

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

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

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. For Refueling Outage 18 and the subsequent operating cycle, the portion of the tube below 15.2 inches from the top of the tubesheet is excluded from this requirement. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. 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. 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. If crack indications are found in any portion of the SG tube not excluded above, 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.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

(continued)

5.6 Reporting Requirements

5.6.10 Steam Generator Tube Inspection Report

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

a. The scope of inspections performed on each SG;

b. 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;

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

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

i. Following completion of an inspection performed in Refueling Outage 18 (and any inspections performed in the subsequent operating cycle) the calculated accident induced leakage rate from the portion of the tubes below 15.2 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.50 times the maximum operational primary to secondary leak rate, the report should describe how it was determined; and

j. Following completion of an inspection performed in Refueling Outage 18 (and any inspections performed in the subsequent operating cycle) the results of monitoring for the tube axial displacement (slippage). If slippage is discovered, the implications of discovery and corrective action shall be provided.