WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
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April 18, 2005

ET 05-0001

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

Subject: Docket No. 50-482: Exigent Request for Revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program"

Gentlemen:

Pursuant to 10 CFR 50.90, Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS). This amendment application proposes a one time revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to incorporate changes in the steam generator inspection scope during Refueling Outage 14.

WCGS commenced a refueling outage on April 9, 2005 with steam generator tube inspections to be performed on steam generators "B" and "C." Prior to each tube inspection, a degradation assessment, which includes operating experience, is performed to identify degradation mechanisms that may be present, and a validation assessment is performed to verify that the eddy current techniques are capable of detecting those flaw types identified in the degradation assessment. Based on recent operating experience at Catawba Nuclear Station, Unit 2, and Vogtle Electric Generating Plant, Unit 1, WCNOC has revised the steam generator tube inspection plan to include a sampling of bulges and overexpansions within the tubesheet region. The sample is based on the guidance in Electric Power Research Institute (EPRI) TR-107569, "Steam Generator Examination Guidelines." This inspection plan will be expanded according to industry guidelines if necessary due to confirmed degradation (i.e., a tube crack).

The proposed change defines the region of the tube that must be examined. A justification has been developed by Westinghouse Electric Company to identify the specific rotating pancake coil probe inspection depth below which any type of axial or circumferential primary water stress corrosion cracking can be shown to meet Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines" performance criteria.

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Attachments I through III provide the evaluation, markup of technical specification pages, and retyped technical specification pages respectively, in support of this amendment request. Attachment IV contains a list of commitments.

Enclosure I provides the proprietary Westinghouse Electric Company LTR-CDME-05-82-P, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Wolf Creek Generating Station." As Enclosure I contains information proprietary to Westinghouse Electric Company, it is supported by an affidavit signed by Westinghouse Electric Company, 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 2.790 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 10 CFR 2.790 of the Commission's regulations. This affidavit, along with a Westinghouse authorization letter, CAW-05-1983, Application for Withholding Proprietary Information from Public Disclosure, is contained in Enclosure III.

Enclosure II provides non-proprietary Westinghouse Electric Company LTR-CDME-05-82-NP, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Wolf Creek Generating Station."

WCNOC requests the proposed change be approved by April 27, 2005 to support the completion of the steam generator tube inspections during Refueling Outage 14. We request that this proposed change be considered under exigent circumstances as described in 10 CFR 50.91, "Notice for public comment, State consultation," paragraph (a)(6), in that failure to act quickly could result in an extended outage for WCGS. A statement of the exigent circumstances surrounding this request, as required by 10 CFR 50.91(a)(6), is included in Attachment I.

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

Please contact me at (620) 364-4084 or Mr. Kevin Moles at (620) 364-4126 for any questions you may have regarding this application.

Very truly yours,

Terry J. Garrett

TJG/rlg

Attachments: I - Evaluation
II - Markup of Technical Specification pages
III - Retyped Technical Specification pages
IV - List of Commitments

Enclosures: I - Westinghouse Electric Company LTR-CDME-05-82-P, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Wolf Creek Generating Station" (Proprietary).
II - Westinghouse Electric Company LTR-CDME-05-82-NP, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Wolf Creek Generating Station" (non-Proprietary).
III - Westinghouse Electric Company LTR CAW-05-1983, Application for Withholding Proprietary Information from Public Disclosure

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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 18th day of April, 2005.

NOTARY PUBLIC STATE OF KANSAS CINDY NOVINGER My Appt. Exp. 7/8/06

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Notary Public

Expiration Date 7/8/06

# EVALUATION

## 1.0 DESCRIPTION

This amendment application proposes a one time revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to incorporate changes in the steam generator inspection scope during Refueling Outage 14.

WCGS commenced a refueling outage on April 9, 2005 with steam generator tube inspections to be performed on steam generators "B" and "C." Prior to each tube inspection, a degradation assessment, which includes operating experience, is performed to identify degradation mechanisms that may be present, and a validation assessment is performed to verify that the eddy current techniques are capable of detecting those flaw types identified in the degradation assessment. Based on recent operating experience at Catawba Nuclear Station Unit 2 and Vogtle Electric Generating Plant Unit 1, WCNOC has revised the steam generator tube inspection plan to include a sampling of bulges and overexpansions within the tubesheet region, the sample is based on the guidance in Electric Power Research Institute (EPRI) TR-107569, "Steam Generator Examination Guidelines." This inspection plan will be expanded according to industry guidelines if necessary due to confirmed degradation (i.e., a tube crack). The proposed change modifies the inspection requirements for portions of the steam generator tubes within the hot leg tubesheet region of the steam generators.

The proposed change defines the region of the tube that must be examined. A justification has been developed by Westinghouse Electric Company to identify the specific rotating pancake coil probe inspection depth below which any type of axial or circumferential primary water stress corrosion cracking can be shown to meet Nuclear Energy Institute (NEI) 97-06 (Reference 2), "Steam Generator Program Guidelines" performance criteria.

## 2.0 PROPOSED CHANGE

- TS 5.5.9b, "<u>Steam Generator Tube Sample Selection and Inspection</u>," is revised to add the following new requirement:

  "4. For Refueling Outage 14, a sample of the SG B and C inservice tubes from the top of the hot leg tubesheet to 10 inches below the top of the tubesheet shall be inspected by rotating probe. This sample shall include a 20% minimum sample of the total population of bulges and overexpansions within the SG from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet."

- TS 5.5.9d.1.f) currently states:

  "Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;"

  This acceptance criteria would be revised as follows:

  "Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness. During Refueling Outage 14, this criterion does not apply to degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet. Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging. During Refueling Outage 14 and the subsequent operating cycle, all tubes with degradation identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be removed from service;"

- TS 5.5.9d.1.h) currently states:

  "Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg; and"

  This acceptance criteria would be revised as follows:

  "Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg. During Refueling Outage 14 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded;"

- TS 5.5.9d., "Acceptance Criteria," is revised to define bulge and overexpansion:

  "j) During Refueling Outage 14 and the subsequent operating cycle:

  Bulge refers to a tube diameter deviation within the tubesheet of 18 volts or greater as measured by bobbin probe; and

  Overexpansion refers to a tube diameter deviation within the tubesheet of 1.5 mils or greater as measured by bobbin probe."

The proposed changes delineate the scope of the steam generator tube inspection required in the tubesheet region of the WCGS steam generators during Refueling Outage 14.

## 3.0 BACKGROUND

WCGS is a four loop plant with Model F steam generators having 5626 tubes in each steam generator. A total of 173 tubes are plugged. WCGS recently completed Cycle 14 operation. The design of the steam generators includes Alloy 600 thermally treated tubing, full-depth hydraulically expanded tubesheet joints, and broached hole quatrefoil tube support plates constructed of stainless steel. To date, the only tube degradation identified in the steam generators is related to tube wear (loose part or anti-vibration bar). No corrosion-related tube degradation mechanisms have been detected.

The most recent WCGS steam generator tube inspection was performed in the October 2003 refueling outage. The steam generator inspection scope is governed by TS 5.5.9, NEI 97-06, EPRI Steam Generator Examination Guidelines, WCGS procedure AP 29A-003, "Steam Generator Management," and the results of the WCGS steam generator degradation assessment. Criterion IX, "Control of Special Processes" of 10 CFR Part 50, Appendix B, requires in part that nondestructive testing is to be accomplished by qualified personnel using qualified procedures in accordance with the applicable criteria. The inspection techniques and equipment were capable of reliably detecting the known and potential specific degradation mechanisms applicable to WCGS. The inspection techniques, essential variables and equipment were qualified to Appendix H, "Performance Demonstration for Eddy Current Examination," of the EPRI Steam Generator Guidelines.

Subsequent to the most recent WCGS steam generator tube inspection, indications of cracking were reported at Catawba Nuclear Station, Unit 2, based on the results from the nondestructive, eddy current examination of the steam generator tubes during the fall 2004 outage, as described in NRC Information Notice 2005-09 (Reference 3), "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds." Tube indications were reported approximately seven inches from 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. Finally, indications were also reported in the tube-end welds, also known as tube-to-tubesheet welds, joining the tube to the tubesheet.

Catawba Nuclear Station, Unit 2, has Westinghouse designed Model D5 steam generators. Model D5 steam generators were fabricated with Alloy 600TT (i.e., thermally treated) tubes. The WCGS Model F steam generators were fabricated with Alloy 600TT. Thus, there is a potential for tube indications similar to those reported at Catawba Nuclear Station, Unit 2, within the hot leg tubesheet region to be identified in the WCGS steam generators if similar inspections were to be performed during the Refueling Outage 14 steam generator inspection.

Potential inspection plans for the tubes and the welds underwent intensive industry discussions in March 2005. The findings in the Catawba Nuclear Station, Unit 2, 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 the adjacent tube material.

## 4.0 TECHNICAL ANALYSIS

In order 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 hot leg tubesheet necessary to maintain structural and leakage integrity for both normal operating and accident conditions. Tube inspections will be limited to identifying and plugging degradation in this portion of the tubes. The technical justification for the inspection and repair methodology is provided in Westinghouse Electric Company LTR-CDME-05-82-P (Reference 4), "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Wolf Creek Generating Station." The limited hot leg tubesheet inspection criteria were developed for the hot leg tubesheet region of Model F steam generators considering the most stringent loads associated with plant operation, including transients and postulated accident conditions. The limited hot leg 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 steam line break (SLB) leakage limits are not exceeded. LTR-CDME-05-82-P provides technical justification for allowing tubes with indications that are below 17 inches from the top of the hot leg tubesheet (i.e., within approximately four inches of the tube end) to remain in-service.

Constraint provided by the hot leg 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 (Reference 5), "Bases for Plugging Degraded PWR Steam Generator Tubes," are satisfied due to the constraint provided by the tubesheet. Through application of the limited hot leg tubesheet inspection scope described herein, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur during a postulated steam line break (SLB) event.

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 determined that 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.

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 LTR-CDME-05-82-P. The tube-to-tubesheet radial contact pressure provides resistance to tube pull-out and resistance to leakage during plant operation and transients. Temperature effects and upward bending of the tubesheet due to primary and secondary differential pressure during normal and transient conditions, result in the tube-to-tubesheet contact pressure increasing with distance from the top of the tubesheet. Due to these effects, the tubesheet bore tends to dilate near the top of the tubesheet and constricts the tube near the bottom of the tubesheet. Testing and analyses have shown that tube-to-tubesheet engagement lengths of approximately 3.3 inches to 8.5 inches were sufficient to maintain structural integrity (i.e., resist tube pull-out resulting from loading considering differential pressures of three times the normal operating pressure difference and 1.4 times the limiting accident pressure difference). The variation of the required engagement length is a function of the radial tube location within the tube bundle. WCNOC has decided to add additional conservatism to the minimum structural distances of 3.3 inches to 8.5 inches by performing an evaluation to depths of 17 inches (actual inspection sample is biased to 10 inches) below the top of the hot leg tubesheet. The increase in contact pressure at this depth significantly increases the tube structural strength and resistance to leakage.

Since the proposed 17-inch tube inspection depth traverses below the mid-plane of the hot leg tubesheet, the tube-to-tubesheet contact pressure significantly aids in restricting primary-to-secondary leakage as differential pressure increases. Based on engineering judgment, given that there is no significant primary-to-secondary leakage during normal operation, there will be no significant leakage during postulated accident conditions from indications located below the mid-plane of the tubesheet (i.e., greater than approximately 10.5 inches below the top of the tubesheet). The rationale for this conclusion is based upon the interaction of temperature and tubesheet bending effects that increases the contact pressure between the tube and the tubesheet.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., a SLB) is limited by flow restrictions resulting 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 primary-to-secondary leak rate during postulated SLB accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet (i.e., including indications in the tube-end welds). This conclusion is based on the observation that while the driving pressure causing leakage increases by approximately a factor of two, the flow resistance associated with an increase in the tube-to-tubesheet contact pressure, during a SLB, increases by approximately a factor of 6. While such a leakage decrease is logically expected, the postulated accident leak rate could be conservatively bounded by twice the normal operating leak rate if the increase in contact pressure is ignored. Since normal operating leakage is limited to less than 0.347 gpm (500 gpd) per TS 3.4.13, "RCS Operational LEAKAGE," the associated accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by 0.694 gpm. This value is well within the assumed accident leakage rate of 1.0 gpm discussed in WCGS Updated Safety Analysis Report, Table 15.1-3, "Parameters Used in Evaluating the Radiological Consequences of a Main Steam Line Break." Hence it is reasonable to omit any consideration of inspection of the tube, tube-end weld, bulges/overexpansions or other anomalies below 17 inches from the top of the hot leg tubesheet.

Degradation found in the portion of the tube below 17 inches from the top of the hot leg does not require plugging as shown in LTR-CDME-05-82-P.

WCNOC is performing the following inspection requirements in steam generators "B" and "C" in order to use the limited hot leg tubesheet inspection methodology:

1. Perform a 55% minimum inspection of the hot leg side tubes using rotating pancake coil probe technology from three inches above the top of the hot leg tubesheet to three inches below the top of the tubesheet. Expand to 100% of the affected steam generator and 20% of the unaffected steam generators in this region only if cracking is found that is not associated with a bulge or overexpansion as described below.
2. Perform an inspection of sufficient hot leg side tubes to include a minimum 20% sample of the total bulges and overexpansion population between the top of the hot leg tubesheet and 17 inches below the top of the tubesheet. The inspection will be performed using rotating pancake coil technology and will be focused in an area from the top of the hot leg tubesheet to 10 inches below the top of the tubesheet. Prior analysis

concludes that the upper half of the tubesheet is the region which principally contributes to leakage during accident conditions and, thus, represents the most safety significant area for examination.

3. If cracking is found in the sample population of bulges or overexpansions, the inspection scope will be increased to 100% of the bulges and overexpansions population for the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet in the affected steam generator and 20% of the bulges and overexpansions population in the unaffected steam generators from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet.
4. If cracking is reported at one or more tube locations not designated as either a top of the tubesheet expansion transition, a bulge or an overexpansion, an engineering evaluation will be performed. This evaluation will determine the cause for the signal, e.g., some other tube anomaly, in order to identify a critical area for the expansion of the inspection. This expanded inspection will be limited to the identified critical area within 17 inches from the top of the hot leg tubesheet.

WCNOC is implementing the following plugging criteria and acceptance criteria.

- Degradation below 17 inches from the top of hot leg tubesheet is acceptable.
- Degradation within 17 inches from the top of hot leg tubesheet must be plugged.

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 Enclosure I 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 inspection program at WCGS provides a high level of confidence that the structural and leakage criteria are maintained during normal operating and accident conditions. On April 14, 2005, the proposed scope of the steam generator tube inspections for Refueling Outage 14 was discussed with NRC staff which included personnel from NRC Region IV and the Office of Nuclear Reactor Regulation.

## 5.0 REGULATORY ANALYSIS

### 5.1 No Significant Hazards Consideration

This amendment application proposes a one time revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to incorporate changes in the steam generator inspection scope during Refueling Outage 14.

The proposed change defines the region of the tube that must be examined. A justification has been developed by Westinghouse Electric Company to identify the specific rotating pancake coil probe inspection depth below which any type of axial or circumferential primary water stress corrosion cracking can be shown to meet Nuclear Energy Institute (NEI) 97-06 (Reference 2), "Steam Generator Program Guidelines" 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 do 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 changes to the steam generator tube inspection criteria, are the steam generator tube rupture (SGTR) event and the steam line break (SLB) accident.

During the SGTR event, the required structural integrity margins of the steam generator tubes will be maintained by the presence of the steam generator tubesheet. Steam generator tubes are hydraulically expanded in the tubesheet area. Tube rupture in tubes with cracks in the tubesheet is precluded by the constraint provided by the tubesheet. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. Based on this design, the structural margins against burst, discussed in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," are maintained for both normal and postulated accident conditions.

The proposed change does not affect other systems, structures, components or operational features. Therefore, the proposed changes result in no significant increase in the probability of the occurrence of a SGTR accident.

At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below the proposed limited inspection depth is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the hydraulic expansion by precluding tube deformation beyond its initial hydraulically expanded outside diameter.

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

The consequences of a SLB are also not significantly affected by the proposed change. 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., a SLB) is limited by flow restrictions resulting 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 primary-to-secondary leak rate during postulated SLB accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet (i.e., including indications in the tube-end welds). This conclusion is based on the observation that while the driving pressure causing leakage increases by approximately a factor of two, the flow resistance associated with an increase in the tube-to-tubesheet contact pressure, during a SLB, increases by approximately a factor of 6. While such a leakage decrease is logically expected, the postulated accident leak rate could be conservatively bounded by twice the normal operating leak rate if the increase in contact pressure is ignored. Since normal operating leakage is limited to less than 0.347 gpm (500 gpd) per TS 3.4.13, "RCS Operational LEAKAGE," the associated accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by 0.694 gpm. This value is well within the assumed accident leakage rate of 1.0 gpm discussed in WCGS Updated Safety Analysis Report, Table 15.1-3, "Parameters Used in Evaluating the Radiological Consequences of a Main Steam Line Break." Hence it is reasonable to omit any consideration of inspection of the tube, tube-end weld, bulges/overexpansions or other anomalies below 17 inches from the top of the hot leg tubesheet. Therefore, the consequences of a SLB accident remain unaffected.

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

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

Response: No

The proposed change 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 changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No

The proposed changes maintain the required structural margins of the steam generator tubes for both normal and accident conditions. Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," and RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," are used as the bases in the development of the limited hot leg 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 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. 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, Westinghouse letter LTR-CDME-05-82-P, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Wolf Creek Generating Station," 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 leg tubesheet inspection depth 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 hot leg tubesheet inspection depth criteria.

Therefore, the proposed changes do no involve a significant reduction in any margin to safety.

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

5.2 <u>Applicable Regulatory Requirements/Criteria</u>

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.

General design criterion (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 applies 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 NEI 97-06, Revision 1, 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 1 steam generator performance criteria are:

1. Steam generator tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and 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 and a safety factor of 1.4 against burst under the limiting design basis accident. Any additional loading combinations shall be included as required by the existing design and licensing basis.

2. The primary-to-secondary accident induced leakage rate for the limiting 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 leak rate for an individual steam generator.

3. The 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 Enclosure I 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 inspection program at WCGS 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 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 need be prepared in connection with the proposed amendment.

## 7.0 STATEMENT OF EXIGENT CIRCUMSTANCES

10 CFR 50.91, "Notice for public comment; State consultation," paragraph (a)(6), states that whenever an exigent condition exists, a licensee requesting an amendment must explain why this exigent situation occurred and why it could not be avoided.

On December 17, 2004, the industry was notified that tube degradation had been detected in the tubesheet region in Catawba Nuclear Station, Unit 2, Westinghouse Model D5 steam generators. Additional details of the findings at Catawba Nuclear Station, Unit 2, are contained in NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," issued on April 7, 2005. The Catawba Nuclear Station, Unit 2, steam generator tubes are similar to the WCGS Model F steam generators in that they both contain Alloy 600 thermally treated tubes. This information was considered in the WCGS Refueling Outage 14 degradation assessment. The degradation assessment concluded that tube degradation could potentially occur in the WCGS steam generator tubesheet region. Specific potential degradation locations are the top of tubesheet expansion transition region, tube bulge or overexpansion locations, tack expansion region and degradation propagating from the tube-end weld into the tube. In preparation for the inspection of these regions, a method for dispositioning potential indications within the tube-end weld was initiated. An evaluation was performed to justify a limited tubesheet inspection in the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet. The evaluation concluded that degradation contained within the tube-end weld could not be properly addressed by American Society of Mechanical Engineers (ASME) code analysis methods. The evaluation also determined that the bottom portion of the tube is not a critical portion of the tube necessary to maintain structural and leakage integrity.

In January 2005, WCNOC requested a conference call with the NRC staff to discuss the proposed steam generator tube inspections planned for Refueling Outage 14 and the NRC staff determined that a conference call was not necessary. WCNOC proceeded with the planning of the tube inspections based in part on the how the Alloy 600 mill-annealed licensees responded to Generic Letter 2004-01 (Reference 6), "Requirements for Steam Generator Tube Inspections," and indications from the NRC on the acceptability of the WCNOC response to Generic Letter 2004-01. Subsequently, the NRC staff requested a conference call on April 14, 2005 to discuss the steam generator tube inspections particularly in the tubesheet region. It was concluded that application of a limited tube inspection in the tubesheet region in areas where degradation potential could occur required a change to TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program."

Due to the short time interval between identification of the need for a TS change to allow a limited steam generator tube inspection scope based on the guidance provided in Information Notice 2005-09, and the actual performance of the WCGS steam generator inspection in the current refueling outage, insufficient time remains for normal NRC processing and notification. Therefore, WCNOC requests that this proposed TS change be considered under exigent circumstances as described in 10 CFR 50.91(a)(6).

## 8.0 REFERENCES

1. EPRI TR-107569, "Steam Generator Examination Guidelines," Revision 6.

2. NEI 97-06, Revision 1, "Steam Generator Program Guidelines," January 2001.

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

4. Westinghouse Electric Company LTR-CDME-05-82-P, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at WCGS," April 17, 2005.

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

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

# ATTACHMENT II
# MARKUP OF TECHNICAL SPECIFICATION PAGES

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

adjacent tube shall be selected and subjected to a tube inspection.

3. The tubes selected as the second and third samples (if required by Table 5.5.9-2 during each inservice inspection may be subjected to a partial tube inspection provided:

a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

b) The inspections include those portions of the tubes where imperfections were previously found.

INSERT A

The results of each sample inspection shall be classified into one of the following three categories:

| Category | Inspection Results |
|---|---|
| C-1 | Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective. |
| C-2 | One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes. |
| C-3 | More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective. |
| Note: | In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations. |

c. Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

1. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months

(continued)

INSERT A

4. For Refueling Outage 14, a sample of the SG B and C inservice tubes from the top of the hot leg tubesheet to 10 inches below the top of the tubesheet shall be inspected by rotating probe. This sample shall include a 20% minimum sample of the total population of bulges and overexpansions within the SG from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

e) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness; INSERT B

g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of a Double Design Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.c.3.c, above;

h) Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg; ~~and~~ INSERT C

i) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial Power Operation using the equipment and techniques expected to be used during subsequent inservice inspections. ; and

INSERT D

(continued)

INSERT B

During Refueling Outage 14, this criterion does not apply to degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet. Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging. During Refueling Outage 14 and the subsequent operating cycle, all tubes with degradation identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be removed from service;

INSERT C

During Refueling Outage 14 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded;

INSERT D

j) During Refueling Outage 14 and the subsequent operating cycle:

Bulge refers to a tube diameter deviation within the tubesheet of 18 volts or greater as measured by bobbin probe; and

Overexpansion refers to a tube diameter deviation within the tubesheet of 1.5 mils or greater as measured by bobbin probe.

**ATTACHMENT III**
**RETYPED TECHNICAL SPECIFICATION PAGES**

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

adjacent tube shall be selected and subjected to a tube inspection.

3. The tubes selected as the second and third samples (if required by Table 5.5.9-2 during each inservice inspection may be subjected to a partial tube inspection provided:

   a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

   b) The inspections include those portions of the tubes where imperfections were previously found.

4. For Refueling Outage 14, a sample of the SG B and C inservice tubes from the top of the hot leg tubesheet to 10 inches below the top of the tubesheet shall be inspected by rotating probe. This sample shall include a 20% minimum sample of the total population of bulges and overexpansions within the SG from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet.

The results of each sample inspection shall be classified into one of the following three categories:

| Category | Inspection Results |
|---|---|
| C-1 | Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective. |
| C-2 | One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes. |
| C-3 | More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective. |
| Note: | In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations. |

c. Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

(continued)

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

1. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

2. If the results of the inservice inspection of a steam generator conducted in accordance with Table 5.5.9-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.c.1. The interval may then be extended to a maximum of once per 40 months; and

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:

   a) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13; or

   b) A seismic occurrence greater than the Double Design Earthquake, or

   c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   d) A main steam line or feedwater line break.

d. Acceptance Criteria

1. As used in this Specification:

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

e) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness. During Refueling Outage 14, this criterion does not apply to degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet. Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging. During Refueling Outage 14 and the subsequent operating cycle, all tubes with degradation identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be removed from service;

g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of a Double Design Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.c.3.c, above;

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

h) Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg. During Refueling Outage 14 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded;

i) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial Power Operation using the equipment and techniques expected to be used during subsequent inservice inspections; and

j) During Refueling Outage 14 and the subsequent operating cycle:

Bulge refers to a tube diameter deviation within the tubesheet of 18 volts or greater as measured by bobbin probe; and

Overexpansion refers to a tube diameter deviation within the tubesheet of 1.5 mils or greater as measured by bobbin probe.

2. Steam generator tube integrity shall be determined after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5.9-2.

e. Reports

The contents and frequency of reports concerning the steam generator tube surveillance program shall be in accordance with Specification 5.6.10.

(continued)

## LIST OF COMMITMENTS

The following table identifies those actions committed to by WCNOC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Kevin Moles at (620) 364-4126.

| COMMITMENT | Due Date/Event |
|---|---|
| If cracking is found in the sample population of bulges or overexpansions, the inspection scope will be increased to 100% of the bulges and overexpansions population for the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet in the affected steam generator and 20% of the bulges and overexpansions population in the unaffected steam generators from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet. | Prior to startup from Refueling Outage 14 |
| If cracking is reported at one or more tube locations not designated as either a top of the tubesheet expansion transition, a bulge or an overexpansion, an engineering evaluation will be performed. This evaluation will determine the cause for the signal, e.g., some other tube anomaly, in order to identify a critical area for the expansion of the inspection. This expanded inspection will be limited to the identified critical area within 17 inches from the top of the hot leg tubesheet. | Prior to startup from Refueling Outage 14 |

Wolf Creek Nuclear Operating Corporation

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC:

Enclosed is:

1. 1 copy of LTR-CDME-05-82-P, "Limited Inspection of the Steam Generator Tube Portion within the Tubesheet at Wolf Creek Generating Station," dated April 2005 (Proprietary).
2. 1 copy of LTR-CDME-05-82-NP, "Limited Inspection of the Steam Generator Tube Portion within the Tubesheet at Wolf Creek Generating Station", (Nonproprietary) dated April 2005.

Also enclosed is Westinghouse authorization letter CAW-05-1983 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 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 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 Section 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 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-05-1983 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.

## PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

## COPYRIGHT NOTICE