WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J Garrett
Vice President, Engineering

June 30, 2006

ET 06-0026

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

Reference:
1) Letter ET 05-0001 dated April 18, 2005, from T. J. Garrett, WCNOC, to USNRC
2) Letter ET 05-0002 dated April 19, 2005, from T. J. Garrett, WCNOC, to USNRC
3) 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)"
4) Letter ET 06-0004 dated February 21, 2006, from T. J. Garrett, to USNRC

Subject: Docket No. 50-482: Revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) 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) Program," to incorporate changes in the steam generator inspection scope during Refueling Outage 15 and the subsequent operating cycle.

Reference 1 and Reference 2 proposed 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 and subsequent operating cycle. Reference 3 issued Amendment No. 162 revising TS 5.5.9 to add changes to the steam generator inspection scope for Refueling Outage 14 and the subsequent operating cycle. Specifically, Reference 3 modified the inspection requirements for portions of the steam generator tubes within the hot leg tubesheet region of the steam generators.

A001

Reference 4 proposed a permanent revision to TS 5.5.9 to exclude portions of the tube below the top of the tubesheet in the WCGS steam generators. On May 30, 2006, the Nuclear Regulatory Commission (NRC) Project Manager provided by electronic media a draft request for additional information. Based on subsequent discussions with NRC personnel, it was determined that sufficient time was not available for providing responses to the request for additional information and the NRC staff review of the responses to support approval of a permanent revision to TS 5.5.9 for Refueling Outage 15 (scheduled for October 2006). As such, WCNOC is proposing an additional one time change to TS 5.5.9 for Refueling Outage 15 and subsequent operating cycle, while the review of the permanent revision continues.

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

WCNOC requests the proposed change be approved by September 22, 2006, to support the preparations for Refueling Outage 15, which is scheduled to start in October 2006. Once approved, the amendment will be implemented prior to MODE 4 entry during startup from Refueling Outage 15.

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.

This amendment application was reviewed by the Plant 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.

If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Kevin Moles at (620) 364-4126.

Very truly yours,

Terry J. Garrett

TJG/rlt

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

cc: T. A. Conley (KDHE), w/a
J. N. Donohew (NRC), w/a
W. B. Jones (NRC), w/a
B. S. Mallett (NRC), w/a
Senior Resident Inspector (NRC), w/a

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 June, 2006.

OFFICIAL SEAL
RHONDA L. TIEMEYER
MY COMMISSION EXPIRES
January 11, 2010

Rhonda L. Tiemeyer
Notary Public

Expiration Date January 11, 2010

EVALUATION

1.0 DESCRIPTION

This amendment application proposes a one time revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Program," to incorporate changes in the steam generator inspection scope during Refueling Outage 15 and the subsequent operating cycle.

Wolf Creek Nuclear Operating Corporation (WCNOC) letter ET 05-0001 (Reference 1) and letter ET 05-0002 (Reference 2) proposed 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 and subsequent operating cycle. Amendment No. 162 (Reference 3) revised TS 5.5.9 to add changes to the steam generator inspection scope for Refueling Outage 14 and the subsequent operating cycle. Specifically, Reference 3 modified the inspection requirements for portions of the steam generator tubes within the hot leg tubesheet region of the steam generators.

WCNOC letter ET 06-0004 (Reference 4) proposed a permanent revision to TS 5.5.9 to exclude portions of the tube below the top of the tubesheet in the Wolf Creek Generating Station (WCGS) steam generators. On May 30, 2006, the Nuclear Regulatory Commission (NRC) Project Manager provided by electronic media a draft request for additional information. Based on subsequent discussions with NRC personnel, it was determined that sufficient time was not available for providing responses to the request for additional information and the NRC staff review of the responses to support approval of a permanent revision to TS 5.5.9 for Refueling Outage 15 (scheduled for October 2006). As such, WCNOC is proposing an additional one time change to TS 5.5.9 for Refueling Outage 15 and subsequent operating cycle, while the review of the permanent revision continues.

Application of the structural analysis and leak rate evaluation results to exclude portions of the tube from inspection and/or repair of tube indications is interpreted to constitute a redefinition of the primary-to-secondary pressure boundary. This change is supported by Westinghouse Electric Company LLC, LTR-CDME-05-82-P (Reference 5), "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at the Wolf Creek Generating Station," which was previously provided as Enclosure 1 to Reference 1.

2.0 PROPOSED CHANGE

- TS 5.5.9c.1. currently states:

  For Refueling Outage 14 and the subsequent operating cycle, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging. All tubes with degradation identified in the portion of 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.

This section would be revised to indicate Refueling Outage 15.

- TS 5.5.9d. currently states:

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

This section would be revised to indicate Refueling Outage 15.

3.0 BACKGROUND

WCGS is a four loop plant with Model F steam generators having 5626 tubes in each steam generator. A total of 181 tubes are plugged. 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.

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 6), "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 also fabricated with Alloy 600TT tubes. 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.

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.

The steam generator inspection scope is governed by TS 5.5.9, NEI 97-06 (Reference 7), Electric Power Research Institute (EPRI) Steam Generator Examination Guidelines (Reference 8), 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.

The most recent WCGS steam generator tube inspection was performed in the April 2005 refueling outage (Refueling Outage 14). The NRC granted on a one-time basis, an amendment (Reference 3) to exclude the portion of the tube below 17 inches from the top of the hot leg tubesheet for Refueling Outage 14 and the subsequent operating cycle. During Refueling Outage 14, WCNOC performed the following additional inspection requirements in steam generators "B" and "C" in order to use the limited hot leg tubesheet inspection methodology:

1. 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.
2. 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 19 inches below the top of the tubesheet. The inspection was performed using rotating pancake coil technology and focused on the area from the top of the hot leg tubesheet to 10 inches below the top of the tubesheet.

These inspections did not identify any indications of cracking.

Prior to each steam generator tube inspection, a degradation assessment, which includes a review of operating experience, is performed to identify degradation mechanisms that may be present. 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 from both WCGS and other plants, WCNOC has revised the steam generator tube inspection plan to include sampling of bulges and overexpansions within the tubesheet region. The sample is based on the guidance contained in EPRI Steam Generator Examination Guidelines and TS 5.5.9. This inspection plan is expanded according to industry guidelines if necessary due to confirmed degradation (i.e., a tube crack).

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 5), "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 9), "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 determination of 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 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 0.104 gpm (150 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 0.208 gpm, twice the normal operational limit. 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 intends to to use the limited hot leg tubesheet inspection methodology during Refueling Outage 15 and perform appropriate inspections as determined by the pre-outage Degradation Assessment. If cracking is identified, scope expansion will be based on the EPRI Steam Generator Examination Guidelines (Reference 8).

WCNOC is implementing the following plugging criteria and acceptance criteria for Refueling Outage 15 and the subsequent operating cycle.

- Degradation below 17 inches from the top of hot leg tubesheet is acceptable.

- Tube with 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 to Reference 1 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.

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 15.

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 7), "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 NEI 97-06, Revision 2, and 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 0.104 gpm (150 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 0.208 gpm, twice the normal operational leakage. 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 not 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 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.

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 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 NEI 97-06, Revision 2, 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:

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

2. 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.

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 REFERENCES

1. WCNOC letter ET 05-0001, "Exigent Request for Revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program,"" dated April 18, 2005,.

2. WCNOC letter ET 05-0002, "Supplement to Exigent Request for Revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program,"" dated April 19, 2005.

3. 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)."

4. WCNOC Letter ET 06-0004, Revision to Technical Specification 5.5.9, "Steam Generator Tube Surveillance Program," February 21, 2006.

5. Westinghouse Electric Company LTR-CDME-05-82-P, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Wolf Creek Generating Station," April 17, 2005.

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

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

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

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

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 may be applied as an alternative to the 40% depth-based criteria:

1. For Refueling Outage ~~14~~ 15 and the subsequent operating cycle, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging. All tubes with degradation identified in the portion of 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.

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 ~~14~~ 15 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. 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.

(continued)

ATTACHMENT III

RETYPED TECHNICAL SPECIFICATION PAGES

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program

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

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

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

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

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

(continued)

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 may be applied as an alternative to the 40% depth-based criteria:

1. For Refueling Outage 15 and the subsequent operating cycle, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging. All tubes with degradation identified in the portion of 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.

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 15 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. 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.

(continued)

LIST OF REGULATORY 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
The license amendment will be implemented prior to MODE 4 entry during startup from Refueling Outage 15	Prior to MODE 4 entry during startup from Refueling Outage 15