WOLF CREEK
NUCLEAR OPERATING CORPORATION

Robert C. Hagan
Vice President Engineering

June 23, 1995

ET 95-0057

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Subject: Docket 50-482: Response to Generic Letter 95-03

Gentlemen:

Attached is Wolf Creek Nuclear Operating Corporation's response to Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes." As required by Generic Letter 95-03, a safety assessment justifying continued operation and a summary of the plans developed for the next steam generator tube inspection are provided in the attachment.

If you have any questions concerning this submittal, please contact me at (316) 364-8831, extension 4553, or Mr. Richard D. Flannigan at extension 4500.

Very truly yours,

Robert C. Hagan

Robert C. Hagan

RCH/jra

Attachment

cc: L. J. Callan (NRC), w/a
D. F. Kirsch (NRC), w/a
J. F. Ringwald (NRC), w/a
J. C. Stone (NRC), w/a

290017

A001

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Robert C. Hagan, 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 content thereof; that he has executed that 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.

LINDA M. OHMIE
Notary Public - State of Kansas
My Appt. Expires 8-31-98

By Robert C. Hagan
Robert C. Hagan
Vice President
Engineering

SUBSCRIBED and sworn to before me this 22 day of June, 1995.

Linda M. Ohmie
Notary Public

Expiration Date 8-31-98

1.0 INTRODUCTION

Wolf Creek Generating Station (WCGS) began commercial operation on September 3, 1985. WCGS is a four loop Westinghouse design plant with Model F steam generators. The steam generator tube material is Alloy 600 thermally treated (TT) and there are 5626 tubes per steam generator. The Model F steam generator has a full depth hydraulically expanded tube to tubesheet configuration with a stainless steel tube support plate with a broach quatrefoil design tube hole. WCGS has a hot leg inlet temperature of approximately 618 degrees Fahrenheit (°F). The secondary side chemistry program consisted of Ammonia and Hydrazine until March, 1994, at which time it was changed to Ethanolamine and Hydrazine. As of May 31, 1995, WCGS has 7.4 effective full power years of operation.

Recent examinations of steam generator tubing at Maine Yankee Atomic Power Station identified a large number of circumferential indications at the top of the steam generator tubesheet expansion region. These most recent inspection findings, coupled with previously documented inspection results regarding circumferential cracking, resulted in the issuance of Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes." Generic Letter 95-03 requested that all holders of operating licenses or construction permits for pressurized water reactors (PWRs): (1) evaluate recent operating experience with respect to the detection and sizing of circumferential indications to determine the applicability to their plants; (2) on the basis of the evaluation, past inspection scope and results, susceptibility to circumferential cracking, threshold of detection, expected or inferred crack growth rates, and other relevant factors, develop a safety assessment justifying continued operation until the next scheduled steam generator tube inspections are performed; and (3) develop plans for the next steam generator tube inspections as they pertain to the detection of circumferential cracking. The information detailed herein, will address the requested actions of Generic Letter 95-03 as they apply in general to Westinghouse designed steam generators and specifically to Wolf Creek Generating Station (WCGS) Model F Westinghouse steam generators. This response is supported by design and technical information provided by the Westinghouse Owners Group.

The most recent inspection findings concerning circumferential indications at the top of the tube sheet expansion region (Maine Yankee Atomic Power Station and Arkansas Nuclear One, Unit 2) appear to have impacted those steam generators utilizing the Combustion Engineering (C-E) EXPLANSION process more than steam generators utilizing other expansion processes. While there are similarities between the C-E EXPLANSION process and the Westinghouse WEXTEX process, the degree to which all Westinghouse units, regardless of tube expansion process, have been affected by circumferential cracking is significantly less than the most recent experience of the C-E units.

Successive inspection results using Motorized Rotating Pancake Coil (MRPC) probes for Westinghouse plants with hardrolled or explosively expanded tubes (WEXTEX) have indicated steadily declining numbers of new indications, declining angular extent and very low growth rates. The only occurrences of significant levels of circumferential cracking have been found when plants perform the first large scale MRPC inspection.

1.1 Historical Circumferential Degradation Locations

Available historical information indicates that for some Westinghouse plants, circumferential cracking has been detected at the top of the tubesheet expansion region, at the Row 1 and 2 U-bend tangent points, and at dented tube support plate intersections (one plant - two twin units). The main focus of this response is to address tubesheet region expansion transition cracking, since this was the primary reason for the issuance of the generic letter. Other circumferential crack initiation sites will be addressed in the following section due to the limited numbers of field indications detected and limited number of tubes which can be affected (specifically small radius U-bends and dented tube support plate intersections).

1.2 Circumferential Degradation Evaluation of Small Radius U-bends and Tube Support Plates

The incidence of circumferential indications at the Row 1 and 2 U-bend tangent points has not been significant, both in the numbers of indications and indicated MRPC angles. Some plants have administratively decided to preventively plug their Row 1 tubes, and some plants preventively plugged their Row 1 and 2 tubes. Other plants have applied U-bend heat treatment (UBHT) in this region and have effectively recovered tubes previously preventively plugged. This is not a concern at WCGS due to the use of Alloy 600 TT tubing and the wide U-bends at Rows 1 and 2.

A leakage event occurred at North Anna Unit 1 in 1987 which resulted in a steam generator tube rupture due to high cycle fatigue at a dented top tube support plate. In response to NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," all domestic Westinghouse steam generators with carbon steel tube support plates were analyzed for the potential to experience high cycle fatigue at this location using a methodology accepted by the NRC. In cases where the analysis indicated that fatigue usage could exceed 1.0, the tube was either plugged and stabilized or plugged using a leak limiting sentinel plug. Three conditions must be present for high cycle fatigue at the top tube support plate; denting, lack of antivibration bar (AVB) support, and locally elevated steam velocities due to non uniform AVB insertion depths. Steam generators with stainless steel support plates and broached (quatrefoil and trifoil) tube holes (like those utilized in WCGS's Model F steam generators) are not expected to experience this phenomenon.

An apparent inspection transient event occurred first in 1992 and involved the apparent identification of circumferentially-oriented indications at the top of tubesheet region in a plant with partial depth roll expansion. Degradation was detected using the bobbin probe, and denting was also associated with many of the indications. Many of the indications had large voltage distorted bobbin indications, which are uncharacteristic of circumferential degradation. Several tubes were pulled from the steam generator and destructively examined. The corrosion morphology was found to be closely spaced axial degradation and cellular degradation, as opposed to circumferential as suggested by MRPC. In a cellular morphology, closely spaced axial degradation and circumferentially-oriented degradation can link and form a patch-like structure. The significant axial components of the cellular morphology were attributed to the distorted bobbin indications. All tube pull specimens, which were burst tested, produced axial burst openings which suggests that the axial degradation dominated the morphology. The affected plant performed a 100 percent bobbin and 100 percent MRPC inspection of the top of tubesheet region of all steam generators in 1992 and also in 1994.

Circumferential cracking at dented tube support plate intersections has been detected at one plant (two twin units). The steam generators experiencing this phenomenon have been replaced. This phenomenon has not been detected at other units.

In many Westinghouse plants, an augmented top-of-tubesheet region inspection program is conducted on a cycle-to-cycle basis. Many Westinghouse plants have had all hot leg tubes inspected at the top-of-tubesheet region using the MRPC probe, and continue to do so on a cycle-to-cycle basis. Currently available probes, coupled with properly implemented calling criteria and techniques, have been demonstrated to be sufficient to identify circumferential indications at the top of the tubesheet region. Recognizing the potential susceptibility to cracking in the expansion transition regions, many Westinghouse units have implemented shotpeening or rotopeening of the expansion transitions to enhance the resistance of this region of the tube bundle to primary water stress corrosion cracking (PWSCC). This remedial measure, especially when implemented prior to commercial operation, can be effective in mitigating the effects of PWSCC. WCGS has determined that shotpeening is not necessary for the Alloy 600 TT steam generator tubes.

Collectively, the items discussed above and further detailed on the following pages provide justification for the continued operation of WCGS.

2.0 OPERATING EXPERIENCE FOR THE U.S. POPULATION OF WESTINGHOUSE STEAM GENERATORS SIMILAR TO THE MODEL F DESIGN

WCGS utilizes a Westinghouse Model F steam generator with a full depth hydraulically expanded top of tubesheet transition. The listing of other plants with steam generators using a similar expansion process are provided in Table 2. The steam generators at WCGS utilize Alloy 600 TT tubing. The nominal diameter of the tubing is 0.688 inches (11/16 inches) with a 0.040 inch nominal wall thickness.

TABLE 2

Hydraulically Expanded Plants Alloy 600 Thermally Treated (TT) Tubing				
Plant/ Steam Generator Model	Startup	First Time Circ. Cracking	Location	Tube Pull and Results
Braidwood Unit 2/ D5	1988	None	N/A	N/A
Byron Unit 2/ D5	1987	None	N/A	N/A
Callaway */ F	1984	None	N/A	N/A
Catawba Unit 2/ D5	1986	None	N/A	N/A
Commanche Peak 2/ D5	1993	None	N/A	N/A
Millstone Unit 3/ F	1986	None	N/A	N/A
Point Beach Unit 1 **/ 44	1984	None	N/A	N/A
Robinson Unit 2 **/ 44	1984	None	N/A	N/A
Seabrook/ F	1989	None	N/A	N/A
Surry Unit 1 **/ 51F	1980	None	N/A	Yes, no SCC found, Indication attributed to probe liftoff, tube geometric effects
Surry Unit 2 **/ 51F	1981	None	N/A	N/A
Turkey Point Unit 3 **/ 44F	1982	None	N/A	N/A
Turkey Point Unit 4 **/ 44F	1983	None	N/A	N/A
Vogtle Unit 1/ F	1987	None	N/A	N/A
Vogtle Unit 2/ F	1989	None	N/A	N/A
Wolf Creek/ F	1985	None	N/A	N/A

*: Rows 1 thru 10 are Alloy 600 TT only.

**: Replacement Steam Generators

3.0 SAFETY ASSESSMENT SUPPORT

3.1 Alloy 600 TT Tube Material Plants

Alloy 600 TT tubing represents an intermediate step in the evolution of progressively optimized corrosion resistant tubing materials. EPRI report NP-3501, "Optimization of Metallurgical Variables to Improve Corrosion Resistance on Inconel Alloy 600," shows the distinct advantages of Alloy 600 TT tubing over Alloy 600 mill annealed (MA) tubing. Data contained in the EPRI report shows minimal stress corrosion cracking (SCC) in Alloy 600 TT c-rings at 600° F in caustic solutions (10% NaOH). Crack depths were generally 2.5 to 4.5 times less than Alloy 600 MA at 600° F. Primary water SCC initiation times were also found to be greater for Alloy 600 TT small radius U-bends tested at 680° F. The EPRI report also showed a dependence upon residual stress level, crack growth rate and initiation times. In support of the EPRI report, which determined that Alloy 600 TT tubing is corrosion resistant, Westinghouse has also shown that the stress levels in hydraulically expanded tubing are less than the associated levels in either explosively or mechanically expanded tubes. The corrosive resistant nature of Alloy 600 TT tubing and the hydraulically expanded tubes are positive aspects of the WCGS steam generators.

3.2 Pulled Tube Examination Results

In 1990, two tubes, R10 C53 and R25 C57, were pulled from the Surry Unit 1 replacement steam generators (Alloy 600 TT tubing). Field nondestructive examinations (NDE) suggested the presence of circumferentially oriented degradation. Upon further review of tube R10 C53, it was concluded that the poorly defined MRPC signal was similar to that of a "ding" or mechanical deformation. Upon destructive examination of R10 C53, no corrosion, either inside diameter (ID) or outside diameter (OD) initiated, was detected. The source of the NDE indications was determined to be attributed to probe liftoff in the expansion transition and mechanical conditions in the tube resultant from the tube installation process. The pulled tube examination results for R25 C57 indicated that the maximum diameter occurred approximately 0.6 inches above the top of the tubesheet. A 70° "groove", mechanical in nature, was found on the tube OD and was attributed to the interaction of the tube with the edge of the tubesheet during the expansion process. It has been concluded that this tube was over extended above the top of the tubesheet. The hydraulic expansion process used was designed to locate the transition slightly below the top of the tubesheet.

3.3 End Of Cycle Structural Limit Crack Angle Calculations

WCGS uses 11/16 inch OD tubing in the steam generators. The following documents the conclusions from testing and analysis completed on 7/8 inch OD Alloy 600 TT tubing which is applicable to the 11/16 inch OD tubing.

No detectable degradation has been discovered at plants using Alloy 600 TT tube material. Based on the extended plant operational periods (1980 to date), it is unlikely that rapid tube degradation at the top of the tubesheet transition would occur prior to the next scheduled tube inspection or on a cycle to cycle basis in the near future. End Of Cycle (EOC) structural limits for 7/8 inch OD Alloy 600 TT tubing would be consistent with the available data developed initially for WEXTEX plants. Despite the fact that no circumferential degradation has been detected in plants using 7/8 inch OD Alloy 600 TT tubing, EOC structural limits are provided below for completeness.

To permit a rapid scoping assessment for tube burst capability of circumferential indications, a burst correlation was developed for throughwall circumferential indications. The burst correlation was then applied to define the structural limit on throughwall crack angles that satisfy the Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," burst margin for three times the normal operating pressure differential. If measured MRPC crack angles, after reduction for coil lead-in and lead-out affects (about 30°) for throughwall indications are less than the structural limit, it can be readily concluded that the indications satisfy burst margin guidelines. If the measured MRPC angles exceed the assumed throughwall structural limit, additional inspections (such as ultrasonic testing) or structural analysis are needed to assess structural integrity. This section describes the development of the throughwall crack angle structural limit.

Utilizing the burst correlations developed from Electrical Discharge Machining data and analytical models, the structural limits for throughwall circumferential indications were developed as given for the crack models in the following tables. The burst pressure data were adjusted to account for lower tolerance limit material properties.

7/8 Inch Tubing EOC Structural Limits for Circumferentially Oriented Degradation			
	Single Throughwall Crack Model	Single Throughwall Crack with 50% Degraded Ligament	Segmented Throughwall Crack Model
3ΔP = 4500 psi	210°	210°	264°
3ΔP = 4300 psi	226°	226°	269°
SLBΔP = 2560 psi	321°	283°	318°

11/16 inch Tubing, EOC Structural Limits for Circumferentially Oriented Degradation	
	Single TW Crack with 50% Degraded Ligament
3ΔP = 3750 psi	247°
SLBΔP = 2560 psi	283°

The single throughwall crack model is applicable to both ID or OD degradation. The segmented model is more typical of PWSCC. The throughwall plus 50 percent deep model was developed to represent 360° indications found for outer diameter stress corrosion cracking. Therefore, a single, uniform throughwall crack of 247°, with 50 percent deep OD degradation, existing over the remaining 113° tube arc, would satisfy the Regulatory Guide 1.121 three times delta pressure (3ΔP) burst recommendations. While a 283° throughwall crack with 50 percent deep OD degradation, existing over the remaining arc, would have a predicted burst pressure of 2560 pounds per square inch (psi) for 11/16 inch OD tubing (indicative of WCGS steam generators). Based on the industry accepted detection thresholds for circumferential cracking, this type of indication, if encountered in the field, would produce an MRPC crack angle of nearly 360°.

It is expected that throughwall cracks of this size would experience primary to secondary leakage, alerting the plant operators of the tube condition. Such was the case for the McGuire kinetically-welded sleeve leakage events. In both cases, the leakage was able to be trended and readily detectable. The crack morphology of these cracks was more that of a single uniform crack since the morphology was driven more by high residual stresses than by intergranular corrosion.

Based on the development work done for 7/8 inch tubing, the corresponding crack lengths for single uniform throughwall cracks without additional degradation in the remaining segment would be consistent at the 3ΔP condition and greater than 283° for 11/16 inch OD tubing at a postulated steam line break ΔP of 2560 psi. Based on the similarity of the limiting crack angles for the two tubing sizes, the limiting single throughwall crack angle which would support a burst capability of 2560 psi would be approximately 310 to 320°.

3.4 Inspection Methodologies and Adequateness

WCGS and the contract analysis personnel used during steam generator tube inspections are well aware of the tube degradation mechanisms that are being experienced by other utilities. All bobbin coil data is reviewed, looking for any and all tube degradation mechanisms and anomalies. All anomalies are further inspected and characterized with an MRPC probe. All data is independently reviewed by a secondary analysis team and all differences are resolved by the two lead analysts (one from each team) with overview performed by WCGS personnel. It is Wolf Creek Nuclear Operating Corporation's position (based on past operation and inspection results) that all four of the steam generators act similarly and the inspection results of one steam generator are indicative of the other three. Also, Surry 1 has been operating with their replacement steam generators (Model 51F, very similar to WCGS's Model F steam generators) since 1980 with no degradation present. This supports studies that indicate the resistive nature of Alloy 600 TT tubing to intergranular attack and/or stress corrosion cracking.

3.4.1 Top of Tubesheet Inspection Program

In addition to WCGS's bobbin coil inspection program, a 300 tube, top of tubesheet hot leg MRPC inspection program was performed in 1991 in steam generator "C" and a 626 tube, top of tubesheet hot leg MRPC program was performed in 1994 in Steam Generator "A." No abnormalities (circumferential and/or axial cracking) were noted during either inspection program. The MRPC inspection programs were performed to establish a baseline for future reference.

3.4.2 Small Radius U-Bend Inspections

To date, a small radius U-bend MRPC inspection program has not been performed at WCGS. However, each outage a sample of the Differential Flawlike Signals (DFS) at the U-bend tangents are further inspected and characterized with a MRPC. No abnormalities (circumferential and/or axial cracking) have been noted.

3.4.3 Dented Tube Support Plate (TSP) Intersections

The bobbin coil data indicates that denting at the TSPs has not occurred in the WCGS steam generators. All bobbin coil anomalies located within the TSP are further inspected and characterized with a MRPC probe. No abnormalities (circumferential and/or axial cracking) have been noted.

3.5 Tube Integrity Assessment Performed by Westinghouse

Due to the lack of circumferential indications in this type of tubesheet region expansion technique, Westinghouse has not performed any plant specific tube integrity evaluations for full depth hydraulically expanded Alloy 600 TT tubing.

3.6 WCGS Tube Integrity Assessments

WCGS has a strong program commitment to secondary side corrosion product removal. During each refueling outage, the top of the tubesheet is cleaned by high pressure water lancing. Also, in 1994, Pressure Pulse Cleaning (PPC) was implemented to assist in the cleaning of tube support plates and the support plate quatrefoils. Additionally, chemical cleaning is currently being planned for the next refueling outage (Spring 1996). The above mentioned actions are an attempt to prevent tube degradation (denting/cracking) due to corrosion products causing localized dry out and elevated temperatures. In addition to the mechanical cleaning processes (water lancing and PPC) performed, WCGS has an extensive secondary side visual inspection program which helps characterize the condition of the tubes, tube support plates and the top of the tubesheet.

The past inspection programs, in conjunction with the aggressive maintenance program have been adequate in size such that any structurally significant circumferential indications would have been found. There have been no reported domestic instances of degradation of any sort, axial or circumferential, in the expansion transitions of Alloy 600 TT tubing. Postulated EOC crack angles would be projected to be well below the EOC structural limits for single or single throughwall cracks with 50% degradation in the remaining ligament, as listed in Section 3.3 of this response. Since there is no domestic operating experiences of circumferential indications in full depth hydraulically expanded plants with Alloy 600 TT tubing (WCGS steam generators), it is reasonable to assume that growth rates of postulated circumferential indications are negligible, and it would be considered quite conservative to assume growth rates of any value, based on the currently available data. When considering these negligible growth rates and when factored with the industry accepted detection thresholds for throughwall circumferential degradation, no indications would be expected at WCGS that would challenge tube integrity at the end of the current operating cycle. Similarly, tube structural integrity would be expected to be maintained during all future operating cycles considering the historical performance of hydraulically expanded tubes, and also assuming that plant operating parameters will not be significantly altered from current conditions.

4.0 DEFENSE IN DEPTH ASSESSMENT POINTS

4.1 Alloy 600 TT Material

Plants with Alloy 600 TT tubing utilize full depth hydraulic expansion. The apparent lack of susceptibility to rapid degradation of hydraulically expanded Alloy 600 MA tubing is seen by the operating experience of a plant using Model E steam generators and another plant using Model F steam generators. Plants with Alloy 600 TT tubing have been operating since 1980 with no reports of corrosion degradation. There are no outside driving factors which suggest that rapid corrosion degradation of Alloy 600 TT tubing would be experienced, either up to the end of the current operating cycles for these units, or during any cycle in the near future.

4.2 Emergency Operating Procedures/Design

The emergency operating procedures are specifically designed to respond to single and multiple tube rupture scenarios. Also, the NRC has performed additional analysis efforts (outlined in Draft NUREG-1477 and NUREG-0844) which indicate that the refueling water storage tank would not become depleted during response to multiple tube rupture events.

5.0 FUTURE STEAM GENERATOR INSPECTION PLANS

WCGS's eighth refueling outage is scheduled for the Spring of 1996. WCGS will be performing a 100% bobbin coil inspection of steam generators "B" and "C". In addition, a 10 percent hot leg top of tubesheet inspection program (563 tubes per steam generator) will be performed in steam generators "B" and "C" to establish a baseline for future inspections. The majority of the sample will be in the sludge pile (kidney) region where history indicates that top of the tubesheet transition cracking should occur first, with the remaining sample throughout the tubesheet. The top of tubesheet inspection will be performed using a technique qualified to Appendix H of the EPRI PWR Steam Generator Examination Guideline. All data will be reviewed by two independent analysis teams with all the analysts qualified to Appendix G of the Examination Guideline.

In the event that top of the tubesheet circumferential cracking is detected in any one steam generator, the sample size will be expanded to 100 percent of all hot leg tubes in all steam generators.

6.0 SUMMARY

It is Wolf Creek Nuclear Operating Corporation's and Westinghouse's position that the structural integrity of the steam generators will not be compromised before the end of the current operating cycle (Spring 1996).

Future inspections will use a technique qualified to Appendix H and Analysts qualified to Appendix G of the EPRI PWR Steam Generator Examination Guidelines to ensure that the structural integrity of the steam generators are maintained.

In the Spring 1996 outage (eighth refueling outage), a 10 percent (563 tubes per steam generator) hot leg top of tubesheet inspection program in steam generators "B" and "C" will be performed to establish a baseline for future inspections.

If top of tubesheet circumferential cracking is detected in any one steam generator, the sample size will be expanded to 100 percent of all hot leg tubes in all steam generators.