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June 27, 1995

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U. S. Nuclear Regulatory Commission
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Gentlemen:

Donald C. Cook Nuclear Plant Units 1 and 2
GENERIC LETTER 95-03 RESPONSE CIRCUMFERENTIAL
CRACKING OF STEAM GENERATOR TUBES

The purpose of this letter is to respond to Generic Letter 95-03. The attachment outlines the operating history of the steam generator tubes for Donald C. Cook Nuclear Power Plant units 1 and 2 and provides a safety analysis to justify continued operation. The attachment also provides the next refueling outage schedule for both units.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 27th DAY OF June 1995

Notary Public

My Commission Expires: 6-28-99

plt

Attachment



A001

cc: A. A. Blind
G. Charnoff
J. B. Martin
NFEM Section Chief
NRC Resident Inspector - Bridgman
J. R. Padgett

# ATTACHMENT TO AEP:NRC:1166T

# GENERIC LETTER 95-03 RESPONSE

# CIRCUMFERENTIAL CRACKING OF STEAM GENERATOR TUBES

## 1.0 Introduction

A recent examination of steam generator tubing at Maine Yankee Nuclear Plant identified a large number of circumferential indications at the top of the tubesheet region. These findings, coupled with previously documented inspection results regarding circumferential cracking, resulted in the issuance of Generic Letter (GL) 95-03, "Circumferential Cracking of Steam Generator Tubes." The information detailed herein addresses the requested actions of GL 95-03 as they pertain to the Cook Nuclear Plant units 1 and 2 steam generators.

## 2.0 Background and Operating Experience

Available historical information shows that some Westinghouse plants have reported circumferential cracking at the tube expansion transition in the tubesheet region, at the U-bend tangent point of rows 1 and 2, and at dented tube support plate (TSP) intersections.

Operating experience for domestic Westinghouse steam generators similar in design and tube material to the Cook Nuclear Plant units 1 and 2 steam generators is provided below. Due to distinct differences in the design of the two types of steam generators in service, each unit is addressed separately.

### 2.1 Cook Nuclear Plant Unit 1

Unit 1 has four Westinghouse Series 51 steam generators which were placed in service in 1975. Key design features include mill annealed Alloy 600 tubing, a partial depth hardroll expansion at the tube-to-tubesheet joint, and drilled hole carbon steel support plates. The nominal tube outside diameter (OD) is 0.875 inch with a nominal wall thickness of 0.050 inch.

Table 2.1-1 lists all domestic plants with Westinghouse steam generators that have partial depth hardroll expansions in the tube-to-tubesheet joint. A summary of their experience with circumferential cracking is also included in the table.

Table 2.1-2 shows domestic Westinghouse plants with mill annealed Alloy 600 tubing and drilled hole carbon steel support plates that have reported circumferential cracking at either low row U-bends or TSP intersections. The majority of plants with these design features have not reported circumferential cracking at these locations.

In 1992, Westinghouse hybrid expansion joint (HEJ) sleeves were used to repair the hot leg tube end of 1840 tubes in unit 1. Plants using a similar sleeving process are listed in Table 2.1-3.

### 2.2 Cook Nuclear Plant Unit 2

In 1989, the four original Westinghouse Series 51 steam generators in unit 2 were replaced with Westinghouse Model 54F steam generators. Key design features of the new steam generators include thermally treated Alloy 690 (Alloy 690 TT) tubing, a full

depth hydraulic expansion at the tube-to-tubesheet joint, quatrefoil stainless steel support plates, increased row 1 U-bend radius, and heat treated low row U-bends. The nominal tube OD is 0.875 inch with a nominal wall thickness of 0.050 inch. A listing of domestic plants with steam generators having similar design features is provided in Table 2.2-1. No circumferential cracking of any type has been reported in steam generators similar to unit 2.

Table 2.1-1
Partial Depth Hardroll Expansion Plants
Alloy 600 Mill Annealed Tubing

| Plant/SG Model | Startup | First Circ Cracking | Location | Tube Pull Results |
|---|---|---|---|---|
| Connecticut Yankee/27 | 1968 | None | N/A | None |
| Cook 1/51 | 1975 | 8/92 | Top-of-tubesheet dent | Axial dominated cellular bands of ODSCC |
| Ginna/44 | 1970 | Unknown | Roll transition | Unknown |
| Indian Point 2/44 | 1973 | 3/95 | Roll transition | None |
| Kewaunee/51 | 1974 | None | N/A | No roll transition circ SCC, axial SCC in crevice |
| Point Beach 2/44 | 1972 | None | N/A | N/A |
| Prairie Island 1/51 | 1974 | None | N/A | N/A |
| Prairie Island 2/51 | 1976 | None | N/A | N/A |
| Zion 1/51 | 1973 | None | N/A | N/A |
| Zion 2/51 | 1974 | None | N/A | N/A |

| Table 2.1-2<br>TSP Intersections and U-Bend Cracking<br>Alloy 600 Mill Annealed Tubing | | | | |
|---|---|---|---|---|
| Plant/SG Model | Startup | First Circ Cracking | Location | Tube Pull Results |
| Diablo Canyon 1/51 | 1984 | 10/92 | Row 1 | None |
| Farley 1/51 | 1977 | 3/91 | Row 1 | None |
| Sequoyah 2/51 | 1981 | 4/88 | Row 1 | None |
| North Anna 1/51 | 1978 | 1987 | Dented TSP intersection | Circ axial ODSCC |
| North Anna 2/51 | 1980 | 1988 | Dented TSP intersection | None |

| Table 2.1-3<br>Westinghouse HEJ Sleeved Tubes | | | | |
|---|---|---|---|---|
| Plant/SG Model | Sleeve Date | First Circ Cracking | Location | Tube Pull Results |
| Kewaunee (HEJ)/51 | 1987, 1988, 1991 | 4/94 | Parent tube upper HEJ lower roll transition [a] | No results to date |
| Point Beach 2 (HEJ)/44 | 1983, 1984 | 09/94 | Parent tube upper HEJ lower roll transition [a] | None |
| Cook 1 (HEJ)/51 | 1992 | None (no RPC) | N/A | N/A |
| Zion 2 (HEJ)/51 | 1988 | None (no RPC) | N/A | N/A |

Note a: Nearly all indications were detected at HEJ hardroll lower transition. A few indications were detected at the upper and lower hydraulic transitions.

| Table 2.2-1<br>Hydraulically Expanded Plants<br>Alloy 690 TT Tubing | | | | |
|---|---|---|---|---|
| Plant/SG Model | Startup | First Circ Cracking | Location | Tube Pull Results |
| Cook 2/54F[a] | 1989 | None | N/A | N/A |
| Indian Point 3/44F[a] | 1989 | None | N/A | N/A |
| North Anna 1/54F[a] | 1993 | None | N/A | N/A |
| North Anna 2/54F[a] | mid 1995 | None | N/A | N/A |
| V.C. Summer/ Delta 75[a] | 1994 | None | N/A | N/A |

Note a: Replacement steam generators

## 3.0 Safety Assessment Report

Collectively, the experience presented in Section 2 and the information detailed in this section provide justification for continued operation of the units 1 and 2 steam generators until the next scheduled in-service inspections.

### 3.1 Cook Nuclear Plant Unit 1

Plants with partial depth hardroll expansion at the tube-to-tubesheet joints and other design features similar to unit 1 represent the earliest units to come on-line. Some of these units have over twenty years of operational experience and all but one have at least twenty calendar years of operation. On a population basis, the incidence of circumferential cracking in these plants has been extremely low.

<u>Tubesheet Region</u>

Many Westinghouse plants conduct an augmented tubesheet region inspection program using rotating pancake coil (RPC) or other enhanced eddy current inspection techniques during each refueling outage. In general, all hot leg tubes are included in the inspections. Currently available probes, coupled with properly implemented data evaluation criteria and techniques, have demonstrated the ability to identify circumferential indications in the tubesheet region.

Relevant industry experience with circumferentially oriented indications in the tubesheet region is as follows.

- In 1990 two tubes were removed from Kewaunee to determine the nature of eddy current indications found at the top-of-tubesheet region and within the non-expanded tube lengths within the tubesheet region. These indications were confirmed to be axially oriented outside diameter stress corrosion cracking (ODSCC). Insignificant OD degradation (2-3 percent deep) was detected in the partial depth hardroll transition during the destructive examination; no primary side stress corrosion cracking (PWSCC) was detected at the roll transition.

- In 1992 indications of either a circumferential or cellular nature, as determined by RPC eddy current inspection, were reported at the top-of-tubesheet region in our unit 1 steam generators. Samples from four tubes were removed to determine the nature of tube degradation in this region. One tube circumferentially separated at the top-of-tubesheet elevation during the tube pull. Further investigation of this tube and the remaining three showed the cracking pattern to be OD-initiated cellular corrosion. (In cellular corrosion, axial cracking dominates the morphology. Short, circumferential crack components of lesser depth than the dominant axial cracking can cause linkage between the axial cracks.) The remaining three top-of-tubesheet elevations were burst tested. Burst pressures were 5625 psi, 7400 psi, and 8100 psi, respectively, with all specimens developing axial burst openings. In situ, indications of similar morphology would be expected to burst at even higher pressures due to the proximity of the tubesheet. Consistent with the nature of cellular corrosion, the burst macrocracks were all comprised of multiple microcracks. The extent of the burst cracks for all samples was from about 0.2 inch above to about 1 inch below the top-of-tubesheet. This degradation in non-expanded tubing was attributed to localized denting.

- In March 1995 circumferentially oriented indications in the partial depth expansion transition region were reported at Indian Point 2, but tube samples were not removed for verification. These indications are believed to be bands of closely spaced axial cracks as opposed to circumferential cracking.

Although the unit 1 steam generators have experienced tube degradation at the partial depth hardroll expansion transition, the indications have not been circumferentially oriented. In 1994 RPC inspections conducted as part of a program to repair tubes by re-rolling at the hardroll region (using F* criteria) confirmed indications at the expansion transition to be axial in nature.

Due to the minimal, if any, circumferentially oriented indications in the tubesheet region of partial depth roll expanded plants, Westinghouse has not performed any plant specific tube integrity evaluations for this region. Despite this apparent lack of

circumferential indications in partial depth roll expanded plants, the following end-of-cycle (EOC) structural limit values are provided for information. Such limits would apply not only at expansion transitions but also at other regions of the tube.

- In partial depth roll expanded plants, tube burst is essentially precluded by the presence of the tubesheet. Even under the assumption that a circumferential indication might propagate to separation, the thin gap between the tube OD and tubesheet hole inside diameter (ID) would limit the amount of leakage encountered. The tubesheet hole to tube OD gap represents approximately 0.018 square inch of flow area, and further hydraulic resistance would be provided by the thin gap length of approximately 18 inches from a postulated separation location to the top of the tubesheet. As a comparison, in the case of the North Anna plug top release event, estimated primary-to-secondary leak rate was less than 80 gpm. The flow area through the plug expander is approximately 0.092 square inch. Based on this flow rate and area, a reasonable assumption is that leakage from a postulated circumferentially separated partial depth roll expanded tube, which is separated at the top of the roll transition, would be much less than the make-up capacity of the plant. Another reasonable assumption is that tube rupture is not a potential during the subsequent operating cycle regardless of the extent of circumferentially oriented indications at EOC. In any event, further flow restriction would be provided by tubesheet bow effects and sludge accumulation in the tubesheet crevice.

  Despite these considerations, EOC structural limits for circumferential degradation are provided in Table 3.1-1. The data used in this table were developed for 7/8 inch OD WEXTEX expanded tubes and is considered a conservative application to 7/8 inch OD partial depth roll expanded tubing. In this program, the crack simulation was performed by slitting tube samples using an electrical discharge machining (EDM) process. A sealing bladder with thin reinforcing foil was used to prevent premature bladder extrusion through the EDM slit. EDM crack simulations and subsequent burst testing were performed for single throughwall cracks, segmented crack networks, and complex crack networks (50 percent OD degradation in the non-throughwall areas).

  Lateral support provided by the tubesheet restrains bending of steam generator tubes during pressurization and provides additional margins to these data, which were developed using modeling of nominal TSP gap lateral restraint. Additional burst pressure capability would be provided if the tube was axially constrained by the TSP due to corrosion product buildup for plants with drilled hole carbon steel TSPs or by axial constraint provided in the tubesheet region by corrosion product accumulation in this area. The burst pressure

correlations were developed based on burst tests with lateral but not axial restraint.

Utilizing the burst correlations developed from EDM data and analytical models, structural limits for throughwall circumferential indications were developed as given for the crack models in Table 3.1-1. The burst pressure data were adjusted to account for lower tolerance limit material properties.

The single throughwall crack model is applicable to both ID or OD degradation. The segmented throughwall crack model is more typical of PWSCC. The single throughwall crack with 50 percent degraded ligament model was developed to represent 360° indications typical of ODSCC.

Table 3.1-1
7/8 Inch Tubing
EOC Structural Limits for Circumferentially Oriented Degradation

| | Single Throughwall Crack Model | Single TW Crack with 50 Percent Degraded Ligament | Segmented Throughwall Crack Model |
|---|---|---|---|
| 3 delta P = 4500 psi | 210° | 210° | 264° |
| 3 delta P = 4300 psi | 226° | 226° | 269° |
| SLB delta P = 2560 psi | 321° | 283° | 318° |

Unit 1 steam generator inspections conducted in 1992 and 1994 were performed in accordance with the plant technical specifications and utilized the guidance of EPRI Report NP-6201, Revision 3, November 1993, "PWR Steam Generator Examination Guidelines." An independent data review was conducted for both examinations; data analysts received plant specific training. In 1994 a plant specific examination was administered to all data analysts, and the requirement to have an EPRI qualified data analyst was implemented.

The 1992 program included the following examinations in each steam generator: 100 percent bobbin coil probe testing, full length, tube end to tube end; 100 percent RPC probe testing of the hot leg top-of-tubesheet region; 100 percent crosswound probe testing of all hot leg HEJ tubesheet sleeves; RPC probe examination of all indications >1.0 volt at TSP intersections; RPC probe examination of all dent signals >5 volts at TSP intersections; and RPC testing of 100 randomly selected TSP intersections.

The 1994 program included the same examinations that were performed in·1992, but HEJ sleeve testing was limited to a sample of 68 HEJ sleeves spread among three steam generators and a 39 tube RPC program was added to qualify repair of tubes by re-rolling to F* criteria.

The validity of unit 1 eddy current test results in the tubesheet region is enhanced by the very low accumulation of sludge due to good secondary water chemistry and frequent tubesheet cleaning and crevice flushing. This is contrary to the experience at Maine Yankee, where the reported sludge pile height of 18 inches may have influenced detectability of indications at the tubesheet surface.

The unit 1 steam generator tube inspection programs of 1992 and 1994 ensured that any structurally significant circumferential indications in the tubesheet region would have been identified.

Tube inspections of unit 1 have not indicated circumferential indications in the partial depth roll expansion region. Considering this lack of relevant indications, a growth rate for this region is difficult to establish. Because experiences of circumferential indications in partial depth roll expanded plants is limited, it is reasonable to assume that growth rates for this region are negligible. A conservative assumption is that postulated growth rates for these tubing would be on the order of the detection threshold (approximately 50° as a bounding value). When considering these growth rates and factoring in the industry accepted detection thresholds for throughwall circumferential degradation, no indications would be expected in unit 1 that would challenge tube integrity at the end of the current operating cycle. Tube EOC structural integrity limits for partial depth roll plants are provided for information only. As previously discussed, the geometric configuration of partial depth roll expansions essentially precludes large leakage events from expansion transition regions in partial depth roll expanded plants. Bounding leakage from postulated large circumferential indications in partial depth roll expansions would be expected to be less than the normal make-up capacity, even for a postulated circumferentially separated tube.

Furthermore, future inspection plans for unit 1 will continue to address EPRI guidelines and recommendations. Following this inspection program and considering the lack of field data regarding circumferential indications for partial depth roll expanded plants, EOC throughwall circumferential indications of lengths and depths that would challenge tube integrity would not be expected.

Rows 1 and 2 U-Bends

The domestic incidence of circumferential indications at tight radius U-bend tangent points has generally not been significant in terms of either numbers of indications or in angular extent as measured by RPC eddy current inspection. U-bend cracking in rows

1 and 2 has occurred in Westinghouse Series 51 units similar to unit 1; however, this phenomenon has not been an issue in the unit 1 steam generators. Stringent chemistry control since 1983 and reduced temperature and pressure operation since 1985 contribute to this good operating history.

The original Cook Nuclear Plant unit 2 steam generators, which were replaced in 1989, did experience row 1 U-bend cracking which led to minor tube leakage. The morphology of the cracking was not confirmed by tube analysis. All row 1 tubes were subsequently plugged. Bobbin coil eddy current inspection, which successfully identified the early defects in unit 2, has also been employed during each unit 1 steam generator inspection without evidence of U-bend degradation.

Tube Support Plate Intersections

Pursuant to NRC Bulletin 88-02, which addressed the 1987 North Anna tube rupture due to high cycle fatigue at a dented top TSP, 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 potentially affected tube was either plugged and stabilized or plugged using a leak limiting sentinel plug. The aforementioned analysis and remedial actions were taken for the unit 1 steam generators to preclude this type of tube failure. Based on analysis conducted at the flow regime corresponding to the currently licensed plugging limit, the lone affected tube was repaired in 1994.

Circumferential cracking at dented TSP intersections has been detected at one plant with twin units. The steam generators experiencing this phenomenon have since been replaced. That plant also operated at higher temperatures than most other units. Similar degradation has not been detected at other units.

Analysis of unit 1 TSP intersection tube samples, performed as part of the justification for a voltage-based interim plugging criteria in 1992, confirmed that cracking at TSP intersections is axial in nature. RPC eddy current testing during the 1994 inspection verified that the morphology has not changed.

HEJ Sleeved Tubes

Throughout the industry, numerous tubes have been sleeved using the HEJ sleeving process. At one time, approximately 25,000 HEJ sleeved tubes were in service. The hybrid expansion joint is formed by first expanding the tube and sleeve to achieve a specified tube OD expansion. Then the hardroll joint is produced within the hydraulically expanded region. The intent of the initial hydraulic expansion is to lessen residual stress levels in the joint area.

In 1995 three HEJ sleeved tubes were removed from Kewaunee for examination of parent tube indications in the lower hardroll transition region of the upper (free span) HEJ joint. At this time, final destructive examination results are not available.

In 1994 a development program aimed at defining an acceptable circumferential crack angle for parent tube indications in HEJ sleeved tubes was undertaken. This program would support continued operation and included both structural (tensile load capability) and leak testing aspects.

- Based on conservative adjustments to the leak test data at a delta P of 2600 psi, a bounding leak rate of 0.033 gpm was established for a 240° throughwall crack.

- Structural integrity testing of a sleeved tube was performed by tensile loading. The tube was affixed to one end of the tensile machine while the sleeve was affixed at the opposite end. Specimens were slit prior to sleeve installation. For most specimens, structural integrity testing indicated that application of an axial load in the slit tube specimens resulted in deflection of the tube in the direction of the degradation. This deflection caused the tube and sleeve to essentially "lock-up," resulting in failure of the sleeve in tension at approximately 8,000 lb. load. Additional specimens were configured with the tube hardrolled into a collar at the primary side tube entry point. These specimens were tested to determine the structural overload capacity of the non-degraded tube ligament. For these specimens, the structural overload capacity of the non-degraded ligament was approximately twice the calculated overload capacity. The additional load bearing capacity was attributed to the "lock-up" effect previously discussed. After the ligament failed, the frictional loads required to pull the tube over the expanded sleeves was approximately equal to the RG 1.121 3 delta P loading.

  A beginning-of-cycle, single throughwall circumferential crack with an angular extent of 179° was shown to meet RG 1.121 burst recommendations. Integrity was based solely on tensile overload capacity of the non-degraded ligament and conservatively neglected the effects of off-axis lockup and friction after joint slippage.

The 1840 HEJ sleeves installed in unit 1 have not been inspected with the enhanced eddy current techniques used to detect circumferential cracking in the parent tubes at Kewaunee and Point Beach 2. However, similar degradation in unit 1, if present at all, is expected to be less pronounced due to the lower operating temperature and significantly shorter in-service time of the HEJ sleeves.

### 3.2 Cook Nuclear Plant Unit 2

The Alloy 690 TT tubing used in the unit 2 steam generators is state-of-the-art in the evolution of progressively optimized corrosion resistant tubing materials. Testing programs have indicated that Alloy 690 TT provides significant PWSCC resistance and increases in ODSCC resistance, compared to Alloy 600 TT.

Alloy 690 TT tubing has been in service since 1989 in unit 2 with no reported instances of localized tube wall degradation. Alloy 690 TT sleeves have been in service in the industry since 1983 with no reported degradation.

The inherent corrosion resistance of the unit 2 steam generator tubing coupled with the low design stress levels in the full depth hydraulically expanded tube-to-tubesheet joint effectively preclude concerns about circumferential cracking in the tubesheet region. Similarly, the potential for U-bend cracking was addressed in the design of the steam generators by increasing the minimum bend radius and heat treating the U-bends after bending. Potential problems associated with dented TSP intersections were eliminated by using quatrefoil stainless steel support plates in unit 2.

Unit 2 tube examinations were conducted in accordance with plant technical specifications and utilized the guidance of EPRI Report NP-6201, Revision 3, November 1992, "PWR Steam Generator Examination Guidelines." The program consisted of a 6 percent full length random sample bobbin coil probe examination conducted in two steam generators during each of the two most recent outages.

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 690 TT tubing.

The past two inspection programs for unit 2 have been performed in accordance with plant technical specification sample requirements. EPRI and industry guidelines have also been followed regarding calling criteria such that any structurally significant circumferential indications would have been identified. No reported domestic instances of degradation of any sort, axial or circumferential, in the expansion transitions of thermally treated tubing have been reported. For the sake of completeness, postulated EOC circumferential crack angles will be considered. Postulated EOC crack angles would be projected to be well below the EOC structural limits for single or single throughwall cracks with 50 percent degradation in the remaining ligament (see Section 3.1 of this response). Because there are no domestic operating experiences of circumferential indications in full depth hydraulically expanded plants with Alloy 600 TT or Alloy 690 TT tubing, a reasonable assumption is that growth rates of postulated circumferential indications are negligible. Based on the currently available data, growth rates of any value are unlikely to occur.

When considering these negligible growth rates and factoring in the industry accepted detection thresholds for throughwall circumferential degradation, no indications would be expected in unit 2 steam generator tubes that would challenge tube integrity at the end of the current operating cycle. Similarly, considering the historical performance of this material and joint configuration and assuming that plant operating parameters will not be significantly altered from current conditions, tube structural integrity would be expected to be maintained during all future operating cycles.

## 4.0 Defense-in-Depth Assessment Points

### 4.1 Cook Nuclear Plant Unit 1

Tubesheet Region

Partial depth roll expanded plants are different from full depth expanded plants should postulated severe circumferentially oriented degradation occur in the expansion transition. The elevation of the postulated degradation is approximately 18 inches below the top of the tubesheet and, as such, this distance would prevent horizontal displacement of the tube ends such that full steam generator tube rupture release rates would not be anticipated. Sludge accumulation in the tubesheet crevice region would also act to restrict any potential leakage. Excluding the presence of sludge in the crevice, the gap between the postulated separated tube and tubesheet is limited. For a postulated tube separation with a 17 inch crevice (1 inch of vertical displacement is assumed), the expected primary-to-secondary leak rates for primary-to-secondary pressure differentials of 1500 and 2600 psi would be expected to be less than the normal make-up capacity of the plant. Also, tubesheet bow effects would act to close any available gap between the tube and tubesheet tube hole.

HEJ Sleeved Tubes

An evaluation of the HEJ sleeve installation process indicates that tube mean residual stresses above the hardrolled area would be compressive in nature while tube mean stresses below the hardrolled area would be tensile, which is supported by the available inspection results. Over 99 percent of the indications reported in the industry are located at the lower hardroll transition of the upper HEJ. Testing results indicate that, even for postulated circumferentially separated parent tubes, frictional forces between the tube and sleeve would exceed the RG 1.121 burst requirements. Tube rupture in this configuration can only occur if the tube is axially displaced by more than one inch.

### 4.2 Cook Nuclear Plant Unit 2

Plants with hydraulically expanded Alloy 600 TT have operated for many years with no reports of relevant corrosion degradation.

Alloy 690 TT tubing has been shown by extensive testing programs to have even better corrosion resistance than Alloy 600 TT. No factors suggest that rapid corrosion degradation of the Alloy 690 TT tubing in unit 2 would be experienced before the end of the current operating cycle or during any cycle in the near future.

### 4.3 Emergency Operating Procedures (EOPs)

EOPs are specifically intended to respond to single and multiple tube rupture scenarios. The NRC has performed additional analysis efforts (outlined in draft NUREG-1477 and NUREG-0844) indicating that the refueling water storage tank RWST would not become depleted during response to a multiple tube rupture event.

## 5.0 Inspection Plans and Schedule

Inspection plans for both the unit 1 and unit 2 steam generators will be conducted in accordance with the plant technical specifications and will be consistent with applicable EPRI recommended practices for sample selection, NDE techniques, and data analysis.

### 5.1 Cook Nuclear Plant Unit 1

Current inspection plans for each steam generator include a 100 percent tube end-to-tube end bobbin coil inspection in each steam generator, 100 percent Cecco-5 inspection from the hot leg tube ends to the top of tubesheet, 100 percent Cecco-5 inspection of all sleeved tubes and include the required RPC samples per the voltage-based TSP interim plugging criteria. In addition, a sample RPC inspection of the rows 1 and 2 tubes in one steam generator will be conducted.

The next inspection of the unit 1 steam generator tubing will occur during the refueling outage currently scheduled to commence in mid-September 1995.

### 5.2 Cook Nuclear Plant Unit 2

Current inspection plans include a full length 20 percent bobbin coil random sample inspection in one steam generator. This inspection will be augmented by a 20 percent Cecco-5 random sample inspection of tubes at the hot leg expansion transition regions in the same steam generator.

The unit 2 steam generators are on a 40-month inspection cycle per the plant technical specifications. The next in-service inspection is due to be performed at the end of fuel cycle 12, which is currently expected to occur in late 1997.