ATTACHMENT 2 TO AEP:NRC:1295

FRAMATOME ASSESSMENT OF STEAM GENERATOR DEGRADATION
DURING UNIT 1 EXTENDED SHUT DOWN

FRAMATOME TECHNOLOGIES

ENGINEERING INFORMATION RECORD

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Title Cook Unit 1 Steam Generator Operability Re-review

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Remarks:

This document provides a review of the steam generator assessment report written subsequent to the last eddy current tube inspection of the Cook Unit 1 steam generators, with respect to the impact of the accumulated (cycle 16) runtime coupled with an extended shutdown period. An assessment of steam generator overall degradation potential during the extended shutdown period is evaluated and concluded not to be a concern with respect to a need to perform a steam generator inspection prior to returning the steam generators to service.

The conclusions of this evaluation are expected to remain valid regardless of the length of the shutdown, given that steam generator lay-up conditions are maintained within acceptable limits. In the event the lay-up conditions change from those characterized by the plant chemistry program, the condition should be evaluated to assess its impact on overall steam generator integrity.

FTI Non-Proprietary

Record of Revision

00	Original Release
01	Incorporated Comments from Cook Plant including, removal of specific shutdown time period, discussion on what actions need to be performed when chemistry parameters are not kept within specified tolerances.

1.0 Introduction

The Cook Unit 1 steam generators are Westinghouse model 51's which were placed in service in 1975. Key design features include alloy 600 mill annealed tubing, a partial depth hardroll expansion at the tube-to-tubesheet joint, and drilled carbon steel support plates. The nominal tubing OD is 0.875 inch with a nominal wall thickness of 0.050 inch. These units were last inspected in the spring of 1997 (U1R97).

The Unit 1 steam generators were inspected with various types of eddy current testing (ET) methods during the end of cycle (EOC) 15 U1R97 scheduled outage. The inspection was the most thorough ever performed at Cook Unit 1, and included the following (applicable to all four steam generators unless otherwise noted):

- 100% full length bobbin coil
- 100% full depth hot leg tubesheets from tube end hot to tube sheet hot + 3"with a rotating pancake coil (RPC)
- 20% full depth cold leg tubesheets from tube end cold to tube sheet cold + 3" with RPC (one SG)
- 100% U-bend exam of rows 1 and 2 (also row 3 in one SG) with RPC
- Inspection of bobbin coil I-codes with RPC
- Inspection of all dents with bobbin voltage > 5.0 volts
- Inspection of all support plate residuals that could mask a bobbin signal
- 100% plus point inspection of all in service sleeves

In addition to the tube examinations, in-situ pressure testing was performed on select tubes and pre- and post-repair secondary side pressure testing was performed on each tube bundle. The in-situ testing provided information used to assess and insure tube integrity while the bundle pressure testing provided assurance of the adequacy of repair operations.

Unit 1 re-started on May 1, 1997 and operated for 3059.5 effective full power hours (EFPH) before being taken offline on September 8, 1997. Unit 1 currently remains offline pending the resolution of various design bases issues. As a precursor to unit startup, it was felt that the impact of the accumulated runtime plus an extended shutdown period on overall steam generator integrity should be examined. For the purpose of this study, the extended shutdown period is considered to have an unlimited duration, provided the controlling chemistry parameters remain within acceptable limits. It is recognized that short-term fluctuations may occur in the plant chemistry parameters. However, the significance of any parameter excursion is reviewed by plant chemistry staff in order to address its significance and the need for any corrective actions. Previous EOC 15 and 16 integrity assessments are documented in Reference 1.

2.0 Plant Technical Specification Requirements

The plant operating license contains surveillance requirements that govern the inspection of the steam generator tubes in order to maintain adequate margin of safety against burst and leakage concerns during postulated accident conditions. Section 4.4.5.3 of the Technical Specifications defines the frequency of required steam generator tube inspections.

Subsection 4.4.5.3.a requires that no more than 24 calendar months pass between inspections. If the results of two consecutive inspections following service under AVT (all volatile treatment), fall into the C-1 category, or if two consecutive inspections indicate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval can be increased to 40 calendar months. However, if the results of inspection performed at a 40-month interval fall into the C-3 category, the inspection frequency shall be increased to at least once per 20-months.

At EOC 15, the ET inspection results placed Cook Unit 1 in the C-3 category; however, Unit 1 has never been on a 40-month inspection interval. The Cook Plant has taken the position that based upon the requirements defined in the Technical Specifications, an inspection of the tubes would be required in April of 1999, due to the 24 calendar month maximum inspection interval. The Cook plant has determined that the 25 percent surveillance grace period offered under Technical Specification 4.0.2 is not applicable in the case of the steam generators.

3.0 Chemistry Controls During Shutdown Period

3.1 Secondary Side Chemistry

The secondary side of the steam generators has been maintained in wet lay-up conditions since September 24, 1997. Cook Plant procedure 12 THP 6020 CHM.205 provides the guidance for placing the secondary side of the steam generators in wet lay-up conditions and has been followed as the administrative control during the lay-up period. Wet lay-up conditions typically consist of the entire tube bundle being covered with the secondary water. The water contains oxygen scavenging, pH control chemicals and a nitrogen cover gas, which is kept above the water level to prevent oxygen from coming in contact with the steam generator tubes. The requirements of this plant procedure are being followed during the shutdown conditions and as such, the secondary side environment is being kept in good chemical balance. The procedure contains specific precautions concerning low hydrazine/carbohydrazine concentrations, which allow oxygen levels to rise in the water and reduce the oxidation of metals, which in turn may lead to corrosion damage of the tubes.

The specifications for control of pH, sodium, chloride, sulfate, boron, hydrazine, carbohydrazide, and dissolved oxygen outlined in the Cook plant procedure are within the ranges specified in Reference 7.5. Based upon a review of the shutdown chemistry reports from October 1997 through February 1998, the chemistry, water level and temperature parameters have remained within defined limits. The pH was not maintained at ≥ 9.8 as specified in Reference 7.5 due to the use of carbohydrazide. However, lower pH values, between 9.2 - 10 are acceptable when using carbohydrazide for the control of oxygen levels at shutdown temperatures. Temperature conditions are low and considered for chemistry conditions only and should not cause any accelerated corrosion.

The secondary side environment as prescribed by EPRI and the Cook procedure does not provide a mechanism for continued or accelerated steam generator tube degradation. This environment is expected to remain benign to degradation during the shutdown period, as long as the secondary side conditions are maintained. Drain down periods are

not expected to adversely affect these conditions, because of the minute amount of time involved that the tubes are uncovered and subsequently re-filled.

3.2 Primary Side Chemistry – Reactor Coolant System (RCS) Shutdown Chemistry Conditions

Since the unit shutdown, the reactor coolant chemistry has been maintained within the specifications set forth in Cook Plant Procedure 12 THP 6020 CHM.110. The chemistry parameter goal and limit values specified in CHM.110 are equivalent to the Technical Specifications or UFSAR, or values recommended in INPO Guidelines for Chemistry, and EPRI PWR Primary Water Chemistry Guidelines, whichever is most restrictive.

During shutdown conditions, the primary side of the RCS is borated in order to aid in controlling core reactivity, and the lithium concentration is lowered. The water level in the steam generators is maintained full throughout the tubes. Upon shutdown, the reactor coolant system is borated to approximately 2500 ppm boron, with a residual hydrogen concentration. A cation and HOH mixed bed demineralizer is used to reduce Lithium concentrations in the RCS and enter an acid-reducing phase. This reduction phase increases the solubility of corrosion products on RCS piping and on the fuel. The resultant "crud burst" is then cleaned up via letdown demineralization. After RCS hydrogen de-gas and the RCS temperature is reduced to <200 F, hydrogen peroxide is added to create a oxidant induced soluabilization, or "larger crud burst", which is subsequently removed via Chemical Volume and Control System demineralization.

Unit 1 has been in MODE 5 for essentially the entire shutdown period. During this period the RCS lithium concentration has been maintained less than 1 ppm, and sulfur (as sulfate) less than a maximum of 12 ppb (typically, < 3 ppb). Thus, the steam generator tubes have not been subjected to an aggressive chemical environment as a result of exposure to the reactor coolant. These conditions are expected to continue throughout the extended shutdown period given that primary side chemistry is maintained within specification. In the event of an out of specification condition, the effect on chemistry excursion on tube degradation should be reevaluated. Also, RCS chloride and fluoride concentrations are kept well below the Technical Specification 150 ppb limit. These levels are usually less than 5 ppb under both operating and shutdown conditions, thus minimizing chloride induced stress corrosion cracking on the primary side. Draining the RCS to half loop should not impact the initiation of and/or crack growth rate of alloy 600, since this activity does not introduce additional contaminants into the reactor coolant other than oxygen. Oxygen dissolved in the reactor coolant, if not reduced to within the specified concentration prior to heating up above 250 degrees F, could result in an increase in the general corrosion rate of alloy 600, but not necessarily SCC. However, oxygen control in the RCS prior to heat-up is generally not a concern at Unit 1.

4.0 Assessment of Damage Mechanisms

During the EOC-15 (U1R97) inspection of the Cook unit 1 steam generators, a number of different tube degradation mechanisms were identified and characterized. No new types of tube degradation were detected during the inspection. No indications were of such severity that the steam generators would not have maintained tube integrity during cycle 15. This is based upon a rigorous program of in-situ pressure testing performed during U1R97.

The evaluation for tube integrity mainly focused on tube rupture potential and leakage. Based upon the results of the in-situ testing performed during U1R97, the steam generator tubes maintained adequate margin against tube rupture under bounding conditions and also against the allowable leakage under postulated accident conditions for cycle 15 operation.

4.1 Crack Growth Rates - General

Per Reference 7.5, the growth rates of stress corrosion cracking (SCC) and intergranular attack (IGA) in 600 mill annealed tubing has been found to be generically affected by a combination of the following variables, at a minimum:

- Metallurgical structure of the material itself
- Presence of cold work
- Stress and stress intensity
- Temperature (elevated)
- pH
- Electrochemical Potential
- Chemical contaminants

Growth rates have been extensively studied by EPRI as part of the water chemistry programs. The results of the studies indicated that 600 mill annealed tubing is susceptible to cracking in certain environments. The testing was performed at elevated temperature (>500 F), where the steam generator tubing would be most susceptible to degradation.

The alloy 600 mill annealed tubing was subjected to a wide range of chemical environments and solutions, all of which caused some type of tube degradation except for the high temperature solutions with organic acids as the main pollutants. In summary, the model boiler tests indicated that:

- Concentrated caustics are the most aggressive environment that form under heat transfer conditions, and these environments can lead to through wall cracking much more rapidly than in operating plants.

- Use of on-line boric acid addition largely prevents initiation of caustic attack and strongly inhibits propagation. Resin ingress can lead to both denting and IGA/SCC, but the rate is not as rapid as with pure caustics, and support plate corrosion causes denting which leads to high tube stress and potential PWSCC initiation sites.

- Organics plus sulfates and lead doped sludge can cause IGA/SCC as rapidly as seen in operating plants, but at much less severe rates than caused by pure caustics.

Another important trait of crack growth rate that was discovered during these tests that simulated operational environmental conditions, is the fact that once a crack is initiated, it can continue to grow in an environment that is not sufficiently severe to initiate other new cracks.

4.2 Shutdown Effects on Cook 1 Crack Growth Rates

The results of the last ET inspection of the Cook unit 1 steam generators, identified the most serious tube degradation as primary water stress corrosion cracking (PWSCC) and outside diameter stress corrosion cracking (ODSCC). The PWSCC occurred mostly in the U-bend region of rows 1 and 2 and in the original equipment manufacturer (OEM) roll transitions in the hot leg tubesheet. These two particular areas contain the stresses required to produce SCC as described in Reference 7.5. The crevice of the tubesheet region was the area most affected by ODSCC, with a small population of the support plates being affected, as well. A large contributing factor to the initiation and growth of these types of degradation is exposure to high temperature. The time maintained at shutdown conditions should not adversely affect the initiation and growth rate of these types of indications due to the absence of elevated temperatures.

4.2.1 PWSCC

Reference 7.7 states that the environmental factors that affect PWSCC of alloy 600 tubing are temperature, hydrogen and lithium concentrations, and electrochemical potential. Temperature is the factor that most significantly affects the initiation of PWSCC. Industry experience substantiates this by the mere number of occurrences in the hot leg expansion transitions, compared to that of the cold leg and other regions of the generator. The temperature affects are believed to be in accordance with the activation energy model for thermally controlled processes, $e^{-Q/RT}$. Operating plants that have reduced T_{hot}, such as Cook unit 1, have experienced a small reduction in the degradation of tubes due to PWSCC. However, the total elimination of PWSCC in the rolled region of the steam generators is likely not obtainable, even with significant lower hot leg temperatures. This is due to the increased stresses on the ID of the tube wall that were generated from the rolling (cold working) of the metal. The time to cracking has been hypothesized by a logarithmic fit of experimental data from various strain level tests, and is represented by:

$$t = [\ln(\frac{1}{x})]^m$$

where x is a function of the applied stress, the threshold stress and the ultimate strength of the tubing material. The rate of progression of PWSCC can vary widely between plants as shown by field experience. Even plants with very similar tubing, ET techniques, fabrication methods, and operating procedures and temperatures, vary considerably for the time to first detection of PWSCC, as well as the rate of degradation, once detected.

Additional testing has shown that the amount of dissolved hydrogen, as well as the concentration of lithium in the primary water have adverse affects for the initiation of PWSCC in alloy 600, in the typical steam generator operating temperature range. EPRI has concluded that cracks may continue to progress in an environment that is not severe enough to initiate new cracks. This point may provide some explanation as the numbers of tubes affected by PWSCC in the roll transitions. The earlier operating cycles, prior to reducing hot leg temperature, most likely initiated the majority of the indications. Once initiated, the indications grew in the reduced temperature environment until the point at which they were detectable by ET. It should be noted however, that the primary side chemistry affects are considered as secondary to residual stress and material affects, with respect to the initiation and propagation of PWSCC in alloy 600.

The primary chemistry parameters identified in various source documents as causing or accelerating PWSCC of mill annealed alloy 600 steam generator tubing are sustained power operation with the lithium concentration above 3.5 ppm, and high (>150 ppb) mode 5 concentrations of sulfur bearing species in the RCS. Accordingly, the maximum concentrations for these constituents in the RCS defined by the referenced chemistry source documents are well below those needed for the initiation and acceleration of PWSCC in alloy 600.

While at power, lithium hydroxide is used to maintain RCS pH between 6.9 and 7.2. Lithium concentration is maintained less than 3.5 ppm, due to concern over the potential effects of prolonged exposure to 3.5 ppm lithium on primary water stress corrosion cracks. This concern is relative to at power conditions, and not during shutdown conditions [7.9]. Dissolved oxygen is minimized at power by the use of hydrogen, thus minimizing oxidizing conditions and minimizing both SCC and general corrosion in the RCS.

4.2.2 ODSCC

The numerous ODSCC indications were detected in the crevice region of the hot leg tubesheet, near the secondary face. These indications were axial in nature and were typical of crevice corrosion attack of the OD surface. Cook unit 1 has previously removed tubes for this type of degradation and confirmed both its orientation and characteristics. The environment in the crevice region at Cook unit 1 is typical of that described in Reference 7.5 for the initiation and growth of IGA/SCC due to a caustic environment. The presence of sludge in this region has long been known to be detrimental to the tube OD surface. Localized chemical attack due to concentration of impurities, insulation of the tube (increased tube wall temperatures), local electrochemical potential are all contributing factors leading to the initiation and growth of SCC in this region. However, these factors are all influenced by the normal operating conditions of the steam generator. A hideout return analysis was performed at the time of shutdown to aid in determining the crevice chemistry. The data indicates that the crevices (tubesheet and TSP) are near neutral in the range for prediction of benign crevice pH (6-9). MULTEQ pH predictions are within the lowest at-temperature for IGA growth rates in alloy 600. This is not to say that secondary side attack of the tubes cannot take place during shutdown conditions. However, with the proper chemistry controls on the secondary side environment, the OD degradation to the tubes while in a shutdown mode, is not expected to continue or accelerate, since the stresses of operation

are more severe and temperature conditions are essentially at ambient. Additionally, the rate of heat transfer is drastically reduced and the influx of chemical species and iron transport is not occurring at the same rate, as well.

4.3 Shutdown Effects on Cook 1 Steam Generator Internals

The support plates, anti-vibration bars, wrapper, and other steam generator internals are not primary-to-secondary leak paths, but are important to overall steam generator integrity and the ability of the generator to perform its safety functions following an accident event. The steam generator internals are exposed to the same physical environment as the OD surface of the tubes, with exception of the tubesheet and support plate crevice regions.

Degradation of the steam generator internals in 51 series Westinghouse steam generators has been documented in various NRC Information Notices and Generic Letter 97-06. The primary instigator for these concerns was foreign utility experiences associated with misapplication of a chemical cleaning process, inadequate clearance for differential thermal expansion, severe cooling transients and erosion-corrosion of an unknown origin. For the most part these instances have been traceable to a specific operating event or inadequate design parameter.

In support of concerns over internals degradation and the aforementioned NRC notices, Westinghouse has reviewed industry experience and steam generator design factors to identify susceptible areas of the model 51 steam generators that require periodic review. Various activities were conducted during the Cook U1R97 steam generator inspection to comply with the resultant Westinghouse secondary side inspection guidelines.

During the U1R97 ET inspection of the steam generators, the low frequency response of the bobbin coils was used to screen for potential cracked or missing support plate ligaments. The results of this examination identified some potential indications that were further examined with rotating coil techniques. The rotating techniques did not confirm the presence of any missing tube support plate ligaments, but did confirm that anomalies did exist. Further evaluation of the fabrication records confirmed that these anomalous signals were due to the patch plate welds that were used to re-attach an area of the TSPs that were cut out to allow the tubing of the steam generator and then replaced. Additional visual inspections were performed by FTI using standard Welch Allen equipment to inspect the annulus, inner bundle, divider lane, and the wrapper barrel. The inspections on the tubesheet were performed to evaluate the effects of the sludge lancing. The wrapper was inspected to verify that no wrapper drop had occurred. The first TSP was also inspected in the area of the tie rods and a sample of periphery tubes to look for any signs of tube support plate degradation or cracking. The results of the inspections concluded that no tube support plate degradation or wrapper drop had occurred in the steam generators.

No abnormalities were identified during the course of the described inspections. Instances of internals degradation are most commonly associated with severe initiating events. The absence of such an event during the current cycle, coupled with exposure to the shutdown lay-up conditions designed to mitigate corrosion of carbon steel components, provides assurance that steam generator internals integrity is being

adequately maintained.

5.0 Cycle 16 Operability Re-Review

5.1 Assessment of Tube Rupture at EOC-16

The U1R97 in-situ pressure testing demonstrated tube burst under bounding worst case conditions at EOC-16 is not a concern. Additionally, the types and characteristics of the indications detected during the EOC-15 inspection will likely bound those indications found in future inspections at Cook unit 1. Even though cycle 16 duration may exceed that of the previous cycle, the flaw growth rates are not a concern. The majority of all large flaws (most of which are not a concern for tube burst due to location) were classified as "two-cycle" flaws, based upon review of historical data from previous inspections. The review included a two-outage re-analysis of past data. The U-bend indications were larger than expected primarily because the steam generators in which they were located had not been previously inspected with a rotating technique in previous inspections. Therefore, the growth rate of the most limiting indications is based upon two cycles of operation and is not considered a concern for one cycle of operation.

The probability of detection has also increased due to the use of the enhanced ET techniques and analyst site specific testing/training, both of which contribute to decreasing the likelihood of returning a significant indication to service for cycle 16. Due to these enhanced techniques, and limitations of the ET analyses techniques used in the previous inspection (U1R95), the indications detected at EOC-15 are expected to bound any indications found in future inspections at Cook Unit 1.

The bounding indications for tube rupture (U-bend PWSCC and hot leg top of tubesheet ODSCC) were in-situ pressure tested to room temperature equivalent bounding pressure differentials and did not rupture. Therefore, as lay-up conditions are not expected to impact this finding, the likelihood of tube rupture under bounding RG 1.121 pressures (3 x NOP), is not a concern for these types of indications at EOC-16, and future inspections.

5.2 Assessment of Projected Leakage at EOC-16

Based upon the results from the in-situ pressure testing, an evaluation was performed for an estimated leakage under postulated accident conditions on the last day of cycle 16. The leakage assessment uses the information obtained at EOC-15 to conservatively bound the leakage. The leak rates that were reported during the in-situ pressure testing were not adjusted for at temperature conditions, therefore the estimated leak rates at assumed operating conditions for EOC-15 and 16 are considered conservative by almost a factor of 3, due to the differences in the fluid densities.

Additionally, for each area of degradation that had an associated leak rate from an in-situ pressure test, the same number of indications that were detected at EOC-15 was assumed to repeat at EOC-16 and have the same contribution to steam generator leakage. This is another conservatism since U1R97 was the most extensive ET

examination of the Cook Unit 1 steam generator tubing, and all flaws that were detected were repaired accordingly.

Tables 5-1 through 5-4 provides the overall summary of the cumulative estimated leak rates for each steam generator at EOC-16, during main steam line break (MSLB) conditions. The estimates show that S/G 11 is the bounding steam generator. Results show that the estimated cumulative leak rates for each steam generator are well below the primary-to-secondary Technical Specification limit (8.4 gpm at operating conditions, in a faulted loop during a potential steam line break event), even without correcting the leak rates downward for fluid property differences.

Table 5-1
S/G 11 EOC-16
Cumulative Leakage Estimation

CATEGORY	Number of Locations	Leak Rate (gpd)	Leak Rate Total per Type	Category Total
Tubes with HEJ sleeves • Inservice Sleeves	817	0.0046	3.76	3.76
Tubes with Reroll Repair • Inservice Rerolls (Westinghouse + FTI) • Estimated New Indications in Reroll RT	331 22	0.001081 4.20	0.36 92.4	92.76
PWSCC in OEM RT • Projected Indications	75	0.00	0.00	0.00
ODSCC at Tube Support Plates	575	Calculated per GL 95-05		1386.86
U-bend PWSCC • Circ. Indications Expected • Axial Indications Expected	4 2	0.80 44.80	3.20 89.60	92.80
TTS ODSCC • Circ. Oriented Indications Expected • Axial Indications > 3 volts Expected • Axial Indications <= 3 volts Expected	0 5 258	0.00 7.50 0.00	0.00 37.50 0.00	37.50
Steam Generator Total (gpd)				1613.68
Steam Generator Total (gpm)				1.12

Table 5-2 S/G-12 EOC-16 Cumulative Leakage Estimation				
CATEGORY	Number of Locations	Leak Rate (gpd)	Leak Rate Total per Type	Category Total
Tubes with HEJ sleeves • Inservice Sleeves	173	0.0046	0.80	0.80
Tubes with Reroll Repair • Inservice Rerolls (Westinghouse + FTI) • Estimated New Indications in Reroll RT	224 8	0.001081 4.20	0.24 33.60	33.84
PWSCC in OEM RT • Projected Indications	75	0.00	0.00	0.00
ODSCC at Tube Support Plates	259	Calculated per GL 95-05		681.11
U-bend PWSCC • Circ. Indications Expected • Axial Indications Expected	1 9	0.80 44.80	0.80 403.20	404.00
TTS ODSCC • Circ. Oriented Indications Expected • Axial Indications > 3 volts Expected • Axial Indications <= 3 volts Expected	0 0 95	0.00 7.50 0.00	0.00 0.00 0.00	0.00
Steam Generator Total (gpd)				1119.75
Steam Generator Total (gpm)				0.78

Table 5-3 S/G 13 EOC 16 Cumulative Leakage Estimation				
CATEGORY	Number of Locations	Leak Rate (gpd)	Leak Rate Total per Type	Category Total
Tubes with HEJ sleeves • Inservice Sleeves	449	0.0046	2.07	2.07
Tubes with Reroll Repair • Inservice Rerolls (Westinghouse + FTI) • Estimated New Indications in Reroll RT	596 35	0.001081 4.20	0.64 147.00	147.64
PWSCC in OEM RT • Projected Indications	75	0.00	0.00	0.00
ODSCC at Tube Support Plates	247	Calculated per GL 95-05		345.24
U-bend PWSCC • Circ. Indications Expected • Axial Indications Expected	7 2	0.80 44.80	5.60 89.60	95.20
TTS ODSCC • Circ. Oriented Indications Expected • Axial Indications > 3 volts Expected • Axial Indications <= 3 volts Expected	0 3 227	0.00 7.50 0.00	0.00 22.50 0.00	22.50
Steam Generator Total (gpd)				612.65
Steam Generator Total (gpm)				0.43

Table 5-4 S/G 14 EOC-16 Cumulative Leakage Estimation				
CATEGORY	Number of Locations	Leak Rate (gpd)	Leak Rate Total per Type	Category Total
Tubes with HEJ sleeves • Inservice Sleeves	374	0.0046	1.72	1.72
Tubes with Reroll Repair • Inservice Rerolls (Westinghouse + FTI) • Estimated New Indications in Reroll RT	146 8	0.001081 4.20	0.16 33.60	33.78
PWSCC in OEM RT • Projected Indications	75	0.00	0.00	0.00
ODSCC at Tube Support Plates	526	Calculated per GL 95-05		1243.63
U-bend PWSCC • Circ. Indications Expected • Axial Indications Expected	5 4	0.80 44.80	4.00 179.20	183.20
TTS ODSCC • Circ. Oriented Indications Expected • Axial Indications > 3 volts Expected • Axial Indications <= 3 volts Expected	0 9 291	0.00 7.50 0.00	0.00 67.50 0.00	67.50
Steam Generator Total (gpd)				1529.83
Steam Generator Total (gpm)				1.06

6.0 Conclusions with Respect to Inspection Interval and Overall SG Degradation Rate

The information presented in this report indicates that the structural integrity and the expected degradation rates of the active damage mechanisms at Cook Unit 1 should not be adversely affected by the extended shutdown period, provided that the shutdown chemistry remains within acceptable limits. Chemistry conditions will be reviewed prior to unit start-up to ensure no chemistry conditions have occurred, which could adversely impact this assessment. If such conditions are identified, their impact on tube corrosion and growth rate should be evaluated. Normal short-term chemistry excursions are evaluated at the time of the excursion and corrective actions are taken when necessary in order to return the conditions within acceptable limits. Any parameter excursion is reviewed by plant chemistry staff in order to address its significance and the need for any corrective actions. Evaluations of any significant excursions and their affect on steam generator integrity should be performed by engineering and chemistry, to ensure the conclusions presented herein remain valid.

The following conclusions were identified:

- Flaw growth rates during shutdown conditions are not expected to continue or accelerate, based upon a review of the procedures and the actual conditions reported since October, 1997, assuming that a large chemistry excursion does not take place prior to startup, during startup, or the remaining operating period.

- The previous tube integrity assessment (Reference 7.1) is not adversely impacted by the accumulated unit runtime and extended shutdown period (assuming current steam generator lay-up conditions are continued).

- Continued tube integrity provides justification to support a NRC relief request from the current Technical Specification calendar based inspection frequency. Because time period at elevated temperatures is a major catalyst for tube degradation, a re-alignment of the inspection frequency based upon effective full power months (EFPM) is warranted and consistent with industry guidance contained in Reference 7.6. Additionally:

  - ❖ The growth rate of the flaws is not severe, as the results of the Reference 1 study document that the large flaws were in-service for more than one cycle of operation.

  - ❖ None of the largest flaws failed in-situ pressure testing with respect to tube rupture.

  - ❖ A very conservative estimate of the total steam generator leakage (all flaws types, all repair types) did not exceed that of the Technical Specifications limit.

  - ❖ The time period that the unit is in shutdown is not equivalent to the same time period at plant operating conditions. When considering the effects of significant flaws that may remain in operation for two cycles and still maintaining adequate tube integrity, the likelihood of approaching the same conditions found during the U1R97 inspection after being subjected to wet lay-up conditions plus one cycle is not likely.

- ❖ Degradation of the steam generator internals (shell, tube support plates, AVBs, welds, wrapper, lugs) is not anticipated during the shutdown conditions that exist in the secondary side.

7.0 References

7.1 FTI Document 51-1264432-01, Cook Unit 1 Steam Generator Evaluation Report EOC-15 and EOC-16, April, 1998.

7.2 Cook Unit 1 Plant Technical Specifications, Section 3/4.4 Reactor Coolant System, Steam Generators.

7.3 Cook Nuclear Plant Steam Generator Wet Lay-Up Procedure, 12 THP 6020 CHM.205, Revision 2.

7.4 Cook Plant Secondary Chemistry Overview Reports, October-December 1997, January-February 1998.

7.5 EPRI Report TR-102134, PWR Secondary Water Chemistry Guidelines – Revision 4, November, 1996.

7.6 EPRI Report TR-107569-V1R5, PWR Steam Generator Examination Guidelines: Revision 5, Volume 1: Requirements

7.7 EPRI Report TR-103824, Steam Generator Reference Book, Revision 1, Volume 1, December 1994.

7.8 FTI Report 02-1268279-00, AEP DC Cook Unit 1 Secondary Side Visual Inspection Summary Report, March 1997.

7.9 EPRI Report TR-105714, Revision 3, PWR Primary Water Chemistry Guidelines, November, 1995.