INDIANA & MICHIGAN ELECTRIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT 2

STEAM GENERATOR TUBE INTEGRITY - APRIL 1987

An Assessment of the Next Operating Interval Length

Attachment 1 to AEP:NRC:0936J

Prepared by:

American Electric Power Service Corporation
One Riverside Plaza
Columbus, Ohio 43216

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LIST OF ABBREVIATIONS

ASME	American Society of Mechanical Engineers
DI	Distorted indication
EC, ECT	Eddy current, eddy current testing
EFPD	Effective full power day
EFPM	Effective full power month
gpd	Gallons per day
gpm	Gallons per minute
I&MECo	Indiana & Michigan Electric Company
IGA/SCC	Intergranular attack/stress corrosion cracking
kHz	Kilohertz
MWt	Megawatt, thermal
MWt-hr	Megawatt hours, thermal
NDD	No detectable degradation
R.G.	USNRC Regulatory Guide
RxxCxx	Row and column designation of steam generator tube
SG xx	Steam Generator No. 21, 22, 23, or 24
SLB/FLB	Steam line break or feed line break accident scenario
SQR	Squirrel
Su	Ultimate tensile stress
TW	Through-wall penetration
UDS	Undefined signal

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REFERENCE SUBMITTALS

1. AEP:NRC:0936G, letter to Mr. Harold R. Denton, NRR-USNRC, "Steam Generator Tube Leak and Return to Power Operation," dated April 9, 1987.

2. AEP:NRC:0936E, letter to Mr. Harold R. Denton, NRR-USNRC, "Steam Generator Tube Integrity," dated November 24, 1986; transmittal of WCAP-11329 (proprietary version) and WCAP-11330 (non-proprietary version).

3. AEP:NRC:0936C, letter to Mr. Harold R. Denton, NRR-USNRC, "Steam Generator Tube Integrity - Interim Status Report," dated February 7, 1986; transmittal of WCAP-11055 (Proprietary version) and WCAP-11056 (non-proprietary version).

4. AEP:NRC:0936A, letter to Mr. Harold R. Denton, NRR-USNRC, "Steam Generator Tube Plugging - Interim Status Report," dated October 10, 1985.

STEAM GENERATOR TUBE INTEGRITY - APRIL 1987

An Assessment of the Next Operating Interval Length

1.0 INTRODUCTION

1.1 Report Objectives

Reference Submittal 1 addressed the course of action taken as a result of the Donald C. Cook Nuclear Plant Unit 2 (Cook 2) steam generator tube leak which occurred on March 3, 1987. The letter documented the preliminary tube inspection results, addressed restoration of tube bundle integrity to the same level as at the beginning of the previous operating period, and presented qualitative justification for return to power and operation for a staff-recommended initial period of three months.

The purpose of this follow-up report is to provide a more complete evaluation of recent events, and to present quantitative justification for an operating interval in excess of the initial three months.

Historically, assessment of an operating interval between steam generator tube inspections has considered only the safety issues of USNRC Regulatory Guide 1.121. Such an assessment was presented for Cook 2 to justify operation through the entire current fuel cycle (Reference Submittal 2). However, even minor steam generator tube leakage, although not a safety issue and in fact allowable up to the limit set by the plant Technical Specifications, is undesirable from both regulatory and operating perspectives. In recognition of this fact, determination of the next operating interval will focus on minimizing the potential for a forced outage due to excessive steam generator leakage. The safety issues of R.G. 1.121 will of course be again met by this more conservative approach.

1.2 Operating Experience Overview

1.2.1 Background

Cook 2 incorporates a nuclear steam supply system manufactured by Westinghouse, and is licensed for 3411 MWt. Initial criticality occurred on March 10, 1978. The unit is

currently operating in its sixth fuel cycle; at the end of March 1987, about 5.7 effective full power years of operation have been accrued.

Cook 2 has four Westinghouse Series 51 steam generators. A description of significant features and a review of the types of tube degradation experienced prior to November 1983 are contained in Reference Submittal 3. All of the early tube degradation was unrelated to secondary side corrosion.

The first significant indication of secondary side tube corrosion in the Cook 2 steam generators occurred in November 1983, when the unit was removed from service due to steam generator tube leakage. Details of that and subsequent events have been discussed in two meetings with the staff (December 4, 1985 and September 16, 1986) and are documented in Reference Submittals 2, 3, and 4. For convenience, however, following is a chronology of significant steam generator events up to March 1987.

- o November 7, 1983 — <u>Forced Outage</u> - first steam generator tube leak due to secondary side corrosion.
  - Leak rate of 0.29 gpm
  - Leak identified in SG 21, Tube R16C40
  - ECT of 1225 tubes in two steam generators
  - Plugged three tubes, all due to indications of secondary side corrosion
  - Restarted on November 22, 1983

- o March 10, 1984 — <u>Refueling Outage</u>
  - 100 percent ECT in each steam generator
  - seven tube samples removed for analysis; confirmed intergranular corrosion in tubesheet region
  - Plugged 402 tubes, 68 of which were due to indications of secondary side corrosion
  - Restarted on July 7, 1984

- o July 15, 1985 — Forced Outage - steam generator tube leak
  - Leak rate of 0.22 gpm
  - Leak identified in SG 23, Tube R16C56
  - ECT of 25 tubes in SG 23
  - Plugged two tubes, both due to indications of secondary side corrosion
  - Attempted restart on August 2, 1985
- o August 2, 1985 — Forced Outage - steam generator tube leak during start-up
  - Leak rate measurements not possible
  - Leaks identified in SG 23, Tubes R7C28 and R14C70
  - ECT of 1500 tubes in SG 23
  - Plugged 35 tubes, all due to secondary side corrosion
  - Initiated boric acid treatment program (30 percent power soak and on-line addition)
  - Restarted on August 22, 1985
- o August 23, 1985 — Forced Outage - steam generator tube leak during 30 percent power soak
  - Leak rate of 0.2 gpm
  - Leaks identified in SG 22, Tube R14C41 and SG 24, Tube R19C52
  - 100 percent ECT in SGs 21, 22, and 24; ECT of all tubes in SG 23 not tested during August 2, outage
  - First EC indications noted at hot leg tube support plate intersections
  - Five tube samples removed for analysis; confirmed intergranular corrosion at tube support plate intersections

- Plugged 110 tubes, 104 of which were due to secondary side corrosion
- Decided to administratively limit unit power to about 80 percent
- Restarted on October 23, 1985

o December 4, 1985 — <u>Presentation to NRC Staff</u>

- Justified continued operation until scheduled refueling outage, approximately 90 effective full power days from the October 23 restart

o February 28, 1986 — <u>Refueling Outage</u>

- Minor steam generator leakage, about 0.04 gpm, at time of shutdown
- Leak identified in SG 22, R16C45
- ECT in accordance with Technical Specification surveillance requirements on initial sample of 550 tubes; classification of SGs 22 and 24 as C-3 required expansion of program to all tubes in each steam generator
- Plugged 151 tubes, 149 of which were due to secondary side corrosion
- Boric acid treatment program continued (crevice flushing, 30 percent power soak, and on-line addition)
- Restart on July 7, 1986
- Unit power again administratively limited to about 80 percent

o September 16, 1986 — <u>Presentation to NRC Staff</u>

- Justified continued operation through entire fuel cycle without shutdown for steam generator surveillance

- Informed staff of intent to replace Cook 2 steam generators

1.2.2 Most Recent Operating Period

Cook 2 began operation in Fuel Cycle 6 on July 7, 1986. Thermal power output throughout the cycle has again been administratively limited - typically to 80 percent, although there have been brief periods of operation at 90 percent in order to perform certain tests and to meet high system load demand during the summer peak period. Thermal generation through the end of March 1987 has been 14,990,974 MWt-hrs, or about 183 EFPDs.

One brief outage unrelated to steam generator tube degradation occurred early in Cycle 6. Following that, Cook 2 ran continuously for a period of 226 calendar days until being removed from service on March 3, 1987 due to an indicated primary-to-secondary leak in SG 22. The measured leak rate was 0.247 gpm, well below the Technical Specification leak rate limit of 500 gpd (0.347 gpm). The leaking tube in SG 22 was identified by hydrostatic testing as Tube R28C53, and was subsequently confirmed by eddy current testing to have a through-wall defect in the hot leg tubesheet crevice about 3.7 inches below the tubesheet surface. This defect is typical of the secondary side IGA/SCC previously identified in the Cook 2 steam generators.

To verify tube integrity prior to returning to service, an eddy current inspection program consistent with the requirements of Technical Specification 3/4.4.5 was performed. Testing results are presented and discussed in Section 2.0 of this report.

After restoring tube bundle integrity by plugging defective tubes, the unit returned to service on April 21, 1987. Before changing to Mode 3, crevice flushing with boric acid (1000-2000 ppm boron) was performed; a 32-hour soak at about 30 percent power with boric acid (50 ppm boron) was conducted; on-line addition of boric acid (5-10 ppm boron) will continue during power operation. Unit thermal power will again be administratively limited to about 80 percent, although brief periods of higher power operation may be necessary for testing or to meet system load demand.

2.0 CONDITION OF TUBE BUNDLES

2.1 Steam Generator Inspection and Tube Plugging - March 1987

Although not mandatory since the Technical Specification leak rate limit was not exceeded prior to shutdown, I&MECo elected to verify the condition of the Cook 2 steam generator tube bundles by performing an eddy current inspection consistent with the requirements of Technical Specification 3/4.4.5. Testing results of an initial sample of about six percent of the tubes in each of SGs 22 and 24 necessitated expanding the inspection to potentially affected areas of all tubes in each steam generator.

2.1.1 Eddy Current Analysis Criteria/Tube Plugging Criteria

The criteria used to analyze eddy current data during the March 1987 inspection were the same as those used during the May 1986 inspection. These criteria were developed from a correlation of field bobbin coil eddy current data with metallography results of tube samples removed in 1984 and 1985, and are discussed in detail in Reference Submittals 2 and 3. For convenience, following is a brief summary of pertinent eddy current signal classifications:

- Clear Indication (reported in percent through-wall penetration, or %TW) - A signal with an unequivocal phase angle measurable at 400 kHz, confirmed at 100 kHz; industry practice is to use a threshold voltage, usually about 1 volt, to discriminate between reportable and non-reportable clear indications; as a conservatism, however, all clear indications, regardless of voltage, were reported for disposition during the May 1986 and March 1987 inspections.

- Distorted Indication (DI) - A signal visible at 400 kHz believed by the interpreter to represent tube degradation, but with an unquantifiable phase angle; expected correlation in mixed frequencies or other single frequencies is not necessarily present.

- Squirrel (SQR) - A particular type of distorted indication in the tubesheet crevice region whose signal trace at 400 kHz is complex with an unquantifiable phase angle; these indications have historically been shown to compromise tube wall integrity.

- Undefined Signal (UDS) - An anomalous signal, not necessarily indicative of tube degradation, but which the interpreter believes should be noted for consideration and disposition.

- No Detectable Degradation (NDD) - A signal with no evidence of tube wall degradation; either there is no degradation or it is below the detection threshold.

Tube plugging criteria used during the March 1987 inspection were basically the same as those used during the May 1986 inspection, although an additional conservatism was incorporated for indications at tube support plate intersections, as noted below. Development and rationale for these criteria are contained in Reference Submittal 2. For convenience, following is a brief summary of the plugging criteria implemented for secondary side corrosion in each of the three areas of concern:

- Tubesheet crevice region, hot leg - All clear indications, DIs, SQRs, and UDSs in the tubesheet crevice region (from the tubesheet roll transition to the secondary face of the tubesheet) were considered pluggable, regardless of voltage or phase angle.

- Tubesheet surface region, hot leg - All clear indications, DIs, and UDSs in the tubesheet surface region (from the secondary face of the tubesheet up to about 6 inches into the free span of tubing) were considered pluggable, regardless of voltage or phase angle.

- Tube support plate intersection, hot leg - Clear indications meeting a threshold voltage of 1.75 volts and having an indicated through-wall penetration of $\geq$40 percent were considered pluggable. In addition, some indications not meeting the voltage threshold were plugged on phase angle alone based on recommendations of the data interpreter. This represents an added conservatism over the criteria used in May 1986.

The Technical Specification plugging criteria of $\geq$40 percent through-wall penetration was applied to all other areas of the steam generator tubing.

2.1.2 Eddy Current Inspection Results

Summaries of pertinent hot leg eddy current indications, by type and location, are given in Tables 1-A and 1-B.

Quantities in Table 1-A represent individual tubes; for tubes with multiple indications, only the indication deemed most severe is listed. Plugging criteria are illustrated by the boundary line drawn in the table. The 107 tubes inside the boundary were removed from service by plugging. In addition, three tubes were plugged due to reasons unrelated to secondary side corrosion (two because eddy current testing could not be performed and one as a precautionary measure due to a DI at the tubesheet roll transition).

In Table 1-B, all indications have been tabulated. The larger total compared to Table 1-A reflects the fact that some tubes have multiple indications, particularly at tube support plate intersections. This total population of indications is used in later degradation growth rate evaluations.

Figures 1, 2, 3, and 4 are tubesheet maps for each Cook 2 steam generator showing the location and extent of wall degradation in the steam generator tubing. Indications plotted are those contained in Table 1-A.

Figure 5 graphically depicts the data of Table 1-B for each steam generator. Figure 6 is a composite for all four steam generators, and gives a graphical comparison of total indications reported during the March 1987 inspection to the total indications left in service following the 1986 inspection. This provides an overview of tube degradation progression during the past operating period.

2.2 Tube Degradation Growth Rate Evaluation

The objectives of this section are to determine if the tube degradation observed during the most recent operating interval is consistent with average growth rates previously developed, and to attempt to identify characteristics of the statistical distribution of previous growth rate data which could be used in the evaluation of future operating intervals.

Three past operating intervals are of interest in this section, and for convenience are referred to as 84-85, 85-86, and 86-87. Pertinent factors in each interval are as follows:

Interval	Duration	EFPDs	Boric Acid?	Nominal Power
84-85	7/07/84 to 7/15/85	291.2	No	100%
85-86	10/23/85 to 2/28/86	85.0	Yes	80%
86-87	7/07/86 to 3/03/87	183.1	Yes	80%

The general, average growth rates in current use were developed after the 85-86 interval, and are based on applying identical analysis criteria to the 1985 and 1986 inspection data; specifics of this methodology are discussed in Section 2.2.2. The 85-86 interval provided a unique opportunity to develop a growth rate methodology, because many tubes left in service after the 1985 inspection would have been plugged had the later analysis and plugging criteria been in use at the time.

The validity of the existing general growth rate methodology will be examined in two ways. First, tube plugging history will be reviewed to see if the recent plugging is consistent with plugging experience during the earlier interval. This comparison can not demonstrate that the methodology is valid, but can be used to show that the methodology is not necessarily invalid. Second, the population of indications from the recent inspection will be evaluated to see if it statistically fits the distribution of the 85-86 interval growth rate data.

2.2.1 Tube Plugging Comparison

A broad indicator of tube degradation growth rate is the tube plugging required at the end of each operating interval, as shown in Item 1 of Table 2. An obvious fallacy with this gross comparison is that it does not account for different operating interval lengths or for changes in data analysis and plugging criteria. Accounting for operating intervals of 9.7, 2.8, and 6.1 EFPMs for the 84-85, 85-86, and 86-87 intervals, respectively, yields the comparison shown in Item 2 of Table 2. Further compensation for changes in analysis and plugging criteria results in the more meaningful comparison given in Item 3. This last comparison reflects the 107 tubes plugged at the end of the 85-86 interval which would have been plugged at the end of the 84-85 interval if the later criteria had been applied, and the 10 tubes plugged at the end of the 86-87 interval which need not have been plugged. (These latter 10 tubes

had support plate indications below the 1.75 volt threshold for plugging, but were plugged as an added conservatism based on recommendations of the data interpreter). Review of Table 2 shows that the compensated tube plugging rate during the 86-87 interval closely matches that of the 85-86 interval (15.9 vs. 14.8 tubes/EFPM), which indicates that the general tube degradation growth rate observed in the 86-87 interval is consistent with that observed in the 85-86 interval. This is an expected result since power level and chemistry parameters were consistent during each interval.

Since the growth rate methodology incorporates different general growth rates for the tubesheet crevice region, tubesheet surface region, and tube support plate intersections, a slightly more refined test is to evaluate tube plugging rates at each of these three areas. Table 3 provides a comparison of the compensated tube plugging rate for each area during the 85-86 and 86-87 intervals. Review of the table shows that the tube plugging rate in each area is fairly consistent for the two intervals, and further suggests that the growth rate metholodogy is valid.

From a review of tube plugging history, it can be concluded that the plugging required in March 1987 is consistent with the previous operating period. Therefore, the numerical degradation growth rate data developed during that prior period may be valid for assessing the next operating interval.

2.2.2 Growth Rate Determination

Quantitative general growth rates have been evaluated after past operating intervals for the three areas of interest. The determination of the average growth rate for each area has been made by comparing eddy current inspection results before and after an operating interval. Tubes without evidence of degradation or with very low, non-quantifiable degradation have been excluded from the calculations. Thus, the growth rates determined reflect the general, average degradation growth rate of tubes undergoing observable degradation - not the entire tube bundle.

Several methods for determining numerical growth rates have been used. The most objective and reliable is a direct comparison of clear indications from one interval to the next ("%TW - %TW" Method). The other methods are regarded as less dependable since they utilize assumptions on initial conditions. They are useful, however, because they allow a comparison to the "%TW - %TW" Method results and because they provide a larger sample size. As reported in prior

submittals, the alternate methods and the "%TW - %TW" Method yield consistent results.

As noted earlier, the 85-86 interval provided a unique opportunity to assess growth rates using the "%TW - %TW" Method. Because of new analysis and plugging criteria that evolved after the 1985 outage (from tube samples removed during the 1985 outage), a number of now-pluggable indications were left in service and given an opportunity to grow during the 85-86 operating interval. Comparison of the reevaluated 1985 data with the 1986 data resulted in development of the general growth rate methodology described in Reference Submittal 2, and summarized below for convenience:

<u>85-86 Interval IGA/SCC Growth Rates</u>

Location	Mean Growth Rate (percent/EFPM)	Sample Size
Tubesheet Crevice Region	1.60	19
Tubesheet Surface Region	0.82	18
Tube Support Plate Intersections	0.66	38

The ability to determine new growth rates during the 86-87 interval for the tubesheet crevice and tubesheet surface regions using the "%TW - %TW" Method has been effectively eliminated because of plugging criteria which removed all previous indications from service. Thus the population of %TWs from this operating interval represents the extreme in growth rate possibilities, i.e. tubes classified previously as NDD which grew to high %TWs. In essence, all that can be observed is the tail of the statistical distribution of growth rates. If the extremes of the population can be shown to fit the distribution of the previous growth rate data, the assumption can then be made that the distribution as a whole has not changed and a probabilistic growth rate model developed from the 85-86 interval data will be valid.

2.2.3 Probabilistic Model Verification

To evaluate the extremes in tube conditions observed in the most recent operating interval 1) a start-of-interval tube condition probability distribution was determined, 2) the growth rate probability distribution from the 85-86 interval

was assumed, 3) the two distributions were combined to define an hypothetical end-of-interval tube bundle condition, and 4) the hypothetical condition was then compared to the March 1987 inspection results to confirm the model.

The start-of-interval tube condition probability distribution was established from the 1986 inspection results and the probability of detection/non-detection for various indication sizes. Non-quantifiable indications (DIs, SQRs, and UDSs) were included in the population in an appropriate %TW size range based on detection threshold and other eddy current information independent of "sizing" parameters which were developed from correlation of previous tube sample analysis and eddy current data (see Reference Submittal 3, WCAP-11055, Figure 4.1 and Reference Submittal 2, WCAP-11329, Figure 2.2.4).

The end results of the above-described comparison are shown in Figure 7 for the tubesheet crevice region, Figure 8 for the tubesheet surface region, and Figures 9-A and 9-B for tube support plate intersections. In the tubesheet surface region and at tube support plate intersections, the model data was fit with a "best estimate" curve. In the tubesheet crevice region, the model data was fit with a more conservative "over-prediction" curve in recognition of the fact that crevice corrosion has been the limiting factor for continued operation.

Review of these figures shows very good agreement between the model's prediction and the actual inspection results. From this it is concluded that growth rate data from the 85-86 interval is valid for assessing the length of the next operating interval.

3.0 EVALUATION OF OPERATION THROUGH THE END OF FUEL CYCLE 6

3.1 USNRC Regulatory Guide 1.121 Basis

3.1.1 Minimum Allowable Wall Determination

Minimum wall requirements for the Cook 2 steam generator tubing were calculated in accordance with the criteria of R.G. 1.121, entitled "Bases for Plugging Degraded PWR Steam Generator Tubes". Confirmation that the recommendations of the guide are met in the Cook 2 steam generators was demonstrated in Reference Submittals 2 and 3, and is restated here for convenience.

The basic recommendations of R.G. 1.121 are outlined below.

I. Allowable minimum wall determination per the following:

1. For normal plant operation, primary tube stresses are limited such that a margin of safety of 3 is provided against exceeding the ultimate tensile stress of the tube material, and the yield strength of the material is not exceeded, considering normal and upset condition loadings.

2. For accident condition loadings, the requirements of paragraph NB-3225 of Section III of the ASME Code are to be met.

   In addition, it must be demonstrated that the applied loads are less than the burst strength of the tubes at operating temperature as determined by testing.

3. For all design transients, the cumulative fatigue usage factor must be less than unity.

II. Leak-Before-Break Verification, i.e., that a single through-wall crack with a specified leakage limit (Technical Specification leak rate limit) during normal operation would not propagate and result in tube rupture during postulated accident condition loadings.

In establishing the safe limiting condition of operation of a tube in terms of its remaining wall thickness, the effects of loadings during both normal operation and postulated accident conditions must be evaluated. Item I.3 is addressed in detail in both Reference Submittals 2 and 3. Briefly, from the viewpoint of fatigue and related implications of cracking, the causes of cracking are accounted for in the verification of leak-before-break.

In the calculation of tube minimum wall, three distinct areas of tube degradation within the Cook 2 steam generators were addressed: the tubesheet crevice region, the tubesheet surface region, and the tube support plate intersections.

Based on previous metallography, tube minimum wall determination for localized tube degradation occurring in the tubesheet crevice or at the top of the tubesheet assumed:

1. Tube degradation to be characterized as either multiple SCC or IGA/SCC (intergranular SCC combined with shallower, more widely spread IGA).

2. Tube wall degradation can be evaluated as equivalent thinning (as a result of IGA) with a superimposed crack.

3. The axial extent of the equivalent thinned length of tube degradation is 1.5 inches. Also, the IGA (equivalent thinning) was uniform around the tube circumference.

Likewise, the tube minimum wall determination for the localized tube degradation occurring at the tube support plate elevations assumed:

1. Tube degradation to be multiple SCC, with individual cracks 0.1 to 0.2 inch in axial extent.

2. Partial through-wall cracking can be evaluated as single and multiple cracks.

3. As tube support plate degradation was confined to the thickness of the tube support plate, the maximum macrocrack length is equal to the support plate thickness, or 0.75 inch.

4. Link-up of multiple SCC is improbable at postulated accident condition pressure differential as reflected in the tube specimen burst tests.

Results of these calculations are provided in Table 4 for each of the above areas of tube degradation. Moreover, Table 5 provides a summary of minimum wall determination for the three regions of localized tube degradation occurring in the D. C. Cook Unit 2 steam generators. In each case, the limiting criterion for determining the allowable wall reduction is the R.G. 1.121 criterion for normal operation that requires a margin of safety of 3 against exceeding the ultimate tensile stress of the material.

3.1.2 Leak-Before-Break Verification

The leak-before-break rationale is to limit the maximum allowable primary-to-secondary leak rate during normal operation such that the associated crack length through which Technical Specification leakage occurs is less than the critical crack length corresponding to tube burst at the

maximum postulated pressure condition loading (SLB/FLB). Again, Reference Submittals 2 and 3 show on the basis of normal operation that unstable crack growth in a tube is not expected to occur in the tubesheet crevice, top of the tubesheet, or tube support plate intersections of the Cook 2 steam generators in the unlikely event of a limiting accident. It is demonstrated that growth of partial through-wall cracks exhibit a limited aspect ratio. This characteristic results in crack extension through-wall prior to reaching the SLB/FLB critical crack length.

I&MECo's utilization of a primary-to-secondary leak monitoring policy which emphasizes both absolute leak rate measurement and rate of change, and which includes the initiation of action prior to reaching the Technical Specification limit, yields additional safety margin.

3.1.3 Eddy Current Testing Uncertainty

Comparison of in situ eddy current inspection results with laboratory destructive analysis of tube samples removed from the Cook 2 steam generators has provided a good basis for determining the eddy current testing uncertainty associated with the particular tube degradation experienced on Cook 2. For tube samples in which metallography revealed tube wall penetration to be at least 40 percent through-wall, the in situ eddy current tests yield a maximum under-prediction of 16 percent. As wall penetration gets deeper, the eddy current tests more closely predict the actual depth of penetration (see Reference Submittal 3, Figure 4-3). To be conservative, a 16 percent eddy current testing uncertainty is used to evaluate operating interval length.

3.2 Operating Interval Justification - Safety Assessment

The influence of the operating environment may affect some of the tubes in the steam generator and result in localized wall degradation. As part of a preventive program to detect tube degradation, in-service inspection using eddy current techniques was performed. Affected tubes with a remaining wall thickness greater than the minimum required wall thickness are acceptable for continued service, provided eddy current measurement uncertainty is accounted for and an operational allowance for continued degradation until the next scheduled inspection is considered. Table 6 summarizes the projected safety margins for locally degraded steam generator tubing, by tube elevation, upon completion of Cycle 6 operation of Cook 2 (about 240 EFPDs or 8.0 EFPMs from start-up on April 21, 1987). It is demonstrated from a safety perspective that operating interval margin exists at

all three tube areas in question with respect to tube minimum allowable wall. These margins are based on the maximum permissible wall loss calculated in accordance with R.G. 1.121 criteria, an eddy current testing uncertainty of 16 percent, and the general degradation growth rates described in Section 2.2.2.

While the above evaluation demonstrates that the recommendations of R.G. 1.121 are met for an operating interval of 8.0 EFPMs, the incidence of primary-to-secondary leakage during that interval is not precluded. I&MECo has conservatively chosen to establish an operating interval which minimizes the potential for forced outages due to steam generator tube leaks.

4.0 OPERATING INTERVAL DETERMINATION

4.1 Operational Considerations

As noted earlier, an operating interval between steam generator inspections will be selected such that the potential for a forced outage due to steam generator leakage is minimized. However, because of the high cost and high occupational radiation exposure associated with steam generator inspections, the operating interval should be as long as possible to minimize the number of intermediate inspections required prior to replacement of the Cook 2 steam generators. The selected interval should also be consistent with fuel cycle considerations, and should offer I&MECo some flexibility for scheduling based on system load requirements.

At start-up on April 21, 1987, Cook 2 had about 240 EFPDs of fuel remaining in Cycle 6. Since the recent tube leak occurred after only 183 EFPDs of operation, the need for an intermediate inspection is apparent. An obvious interval to look at would be the mid-point of the remaining fuel, or about 120 EFPDs. At 80 percent power, the earliest this could occur is mid-September 1987, which would not conflict with the scheduled Cook 1 refueling , and should be after the summer peak load period. However, choosing the exact mid-point of the remaining fuel affords I&MECo no flexibility as to when to remove the unit from service; a late shutdown would violate the justified interval and an early shutdown would make the second interval longer than justified. An allowance of about three weeks should be added to provide this needed scheduling flexibility. Therefore, an operating interval of 140 EFPDs, or 4.7 EFPMs,

is acceptable from an operational perspective. The potential for steam generator leakage during this interval is assessed in Section 4.2.

4.2 Tube Bundle Condition Projection

During the most recent operating interval, a steam generator tube leak of sufficient magnitude to initiate unit shutdown occurred sooner than expected based on the prior safety analysis which justified operation through Cycle 6. Although the leak was below Technical Specification limits and was well within operator control capabilities to prevent an off-site radiation release, the element of significant current interest is why the leak occurred in such a short time frame. In an effort to address this concern, several possibilities were identified. Each possibility, along with its associated response relative to selecting the next operating interval and an evaluation of its likelihood of being true, is outlined below:

- Possibility - General degradation growth rates are much higher than during previous periods.

  Response - Use the higher mean growth rates to adjust the operating interval to comply with safety analysis considerations.

  Evaluation - Little or no evidence could be found to support this possibility; as described in Section 2.2, growth rates are consistent with the 85-86 interval.

- Possibility - Leaking tube is an "outlier" beyond the general distributions of growth rates and initial conditions, and is therefore a random event.

  Response - Maintain the prior operating interval justification, and accommodate leakage from any additional "outliers" through leak rate monitoring and maintenance shutdowns as required.

  Evaluation - Some evidence supporting this possibility is found in the fact that there is a low number of very high level indications separated from the main distribution of indications.

- Possibility - Leaking tube is an expected result of combining the extremes of the general distributions of growth rates and initial conditions.

<u>Response</u> - Adjust the operating interval to reduce the potential for leakage by considering the statistical distribution of the growth rate data.

<u>Evaluation</u> - Evidence in support of this possibility was developed through a probabilistic model combining start-of-interval tube conditions and growth rates, as described in Section 2.2.3.

The results of the evaluation have largely eliminated the first possibility. While the second and third cases are still possible, the present information favors the third. Therefore, under the assumption that extreme degradation conditions are a function of operating interval and not a random occurrence, it seems prudent to adjust the operating interval to minimize the potential for leakage.

Consistent with the Section 4.1 discussion of reasonable operating interval lengths, an operating interval of 4.7 EFPMs was considered. To assess the reduction in potential for leakage, the probabilistic model described in Section 2.2.3 was applied in the same manner as used to assess growth rate. The analysis included new start-of-interval conditions resulting from the March 1987 inspection and plugging, and used the growth rate distribution derived from the 85-86 interval. The projected end-of-interval conditions for the tubesheet crevice region, tubesheet surface region, and tube support plate intersections are shown in Figures 10, 11, and 12. Since the end-of-interval projections show no appreciable number of tubes at extreme wall penetrations, such as might result in leakage, the 4.7 EFPM interval is considered appropriate.

5.0 CONCLUSIONS

The following conclusions have been drawn from review and evaluation of the March 1987 Cook 2 steam generator tube leak event and subsequent eddy current inspection results:

- The leak was typical of previous IGA/SCC degradation experienced in the Cook 2 steam generators. An adequate understanding of this degradation mechanism has been acquired through previous metallographic examination and burst testing of tube samples, so no further destructive testing is necessary.

- The recent overall eddy current inspection results and

the number of pluggable indications are consistent with experience in the prior operating interval, and can be used to show that the general, average degradation growth rate methodology developed from the 85-86 operating interval is still valid.

- A R.G. 1.121 safety evaluation based on tube structural limits for the Cook 2 steam generator tubing, general tube degradation growth rates, and a conservative eddy current uncertainty margin could be used to justify operation through the remaining 8.0 EFPMs of Cycle 6. However, there is a distinct probability of a tube leak occurring during that interval.

- A probabilistic growth rate model developed from the general growth rate data base can be used to predict extreme conditions of the tube bundles following a specified operating interval. Determination of an operating interval based on extreme rather than general tube conditions should greatly reduce the probability of a primary-to-secondary steam generator tube leak during that interval, although the possibility of a random (outlier) event cannot be precluded.

- Selection of a conservative operating interval based on extreme tube conditions should also include operational considerations to reasonably limit the economic penalties and increased personnel radiation exposure associated with more frequent steam generator inspections.

- An operating interval of about 4.7 EFPMs measured from the return-to-power in April 1987 appears most appropriate when considering both extreme tube conditions and remaining fuel in the current fuel cycle. I&MECo will remove Cook 2 from service within that interval to verify and restore as necessary the integrity of the steam generator tube bundles. The subsequent operating interval would end at the Cycle 7 refueling outage.

- Selection of operating intervals beyond Cycle 6 should consider operating experience during the next two intervals, the results of the next two tube inspection programs, length of the next fuel cycle, and scheduling of the steam generator replacement outage.

I&MECo recognizes that excessive steam generator tube leakage resulting in unscheduled shutdowns is not acceptable on a continuing basis, and has adopted a conservative.

approach to selecting the next operating interval which should greatly reduce the probability of a forced shutdown due to leakage. Previously instituted remedial measures (e.g. - better secondary water chemistry, boric acid treatment, and administrative power reduction) will be continued during the interval.

In the unlikely event that the incidence of extreme wall penetration is a random event and is not predicted by the foregoing probabilistic analysis, then I&MECo's leak rate monitoring program and the Technical Specification leak rate limit will ensure leak-before-break conditions and that an orderly shutdown can be affected. I&MECo's administrative policy of shutting down before reaching the actual leak rate limit adds additional margin to leak-before-break considerations.

Table 1

Indications of Hot Leg Secondary Side Corrosion - March 1987

A. Including only the most significant indication per tube, total for all 4 SGs.

Location	<40%	≥40%	DI	UDS	SQR	Total
Tubesheet Crevice	0	5	2	6	42	55
Tubesheet Surface	3	7	19	2	--	31
Tube Support Plates	15	21	594	0	--	630
Total	18	33	615	8	42	716

B. Including multiple indications per tube, total for all 4 SGs.

Location	<40%	≥40%	DI	UDS	SQR	Total
Tubesheet Crevice	0	5	2	6	42	55
Tubesheet Surface	3	7	19	2	--	31
Tube Support Plates	16	23	830	0	--	869
Total	19	35	851	8	42	955

Table 2

Tubes Plugged Due to IGA/SCC - General Comparison

		Operating Interval		
		84-85	85-86	86-87
1.	Tubes plugged due to secondary side IGA/SCC (total tubes)	141	149	107
2.	Tubes plugged due to secondary side IGA/SCC (tubes/EFPM)	14.5	52.7	17.5
3.	Tubes plugged due to secondary side IGA/SCC, compensated for changes in analysis and plugging criteria (tubes/EFPM)	25.5	14.8	15.9

Table 3

Tubes Plugged Due to IGA/SCC, Compensated for Changes in Analysis and Plugging Criteria - Comparison by Location

	Operating Interval	
Location	85-86	86-87
Tubesheet Crevice (tubes/EFPM)	10.9	9.0
Tubesheet Surface (tubes/EFPM)	2.8	5.1
Tube Support Plate Intersections (tubes/EFPM)	1.1	1.8
	14.8	15.9

r

e

e

p

c

Table 4

Cook 2 Steam Generator Tubing
Minimum Acceptable Wall Requirements

A. Tubesheet crevice and tubesheet surface regions.

Criteria	Condition	Minimum Wall (inches)
yield	normal	0.015
ASME Code	faulted	0.017
Su/3	normal	0.019

B. Tube support plate intersections.

Criteria	Condition	Minimum Wall (inches)
yield	normal	0.012
ASME Code	faulted	0.013
Su/3	normal	0.015

Table 5

Cook 2 Steam Generator Tubing
Allowable Wall Loss Determination

Location	Geometric Condition	Basis	Allowable Wall Loss (%)
Tubesheet Crevice Region	Axial extent >1.5 inches	Su/3	62
Tubesheet Surface Region	Axial extent >1.5 inches	Su/3	62
Tube Support Plate Intersections	Axial extent $\leq$0.75 inches	Su/3	70

Table 6

Operating Interval Justification
Remainder of Fuel Cycle 6 - R.G. 1.121 Basis

Item	Tubesheet Crevice	Tubesheet Surface	Tube Support Plates
Allowable tube wall loss (%)	62*	62	70*
ECT uncertainty (%)	16	16	16
Growth (%/EFPM)	1.6	0.82	0.66
Projected growth (%/8.0 EFPM)	12.8	6.6	5.3
Plugging level required (%)	33.2	39.4	48.7
Plugging level implemented (%)	All	All	40.0

* Tube burst within the tubesheet crevice region or at tube support plate intersections is considered to be incredible.

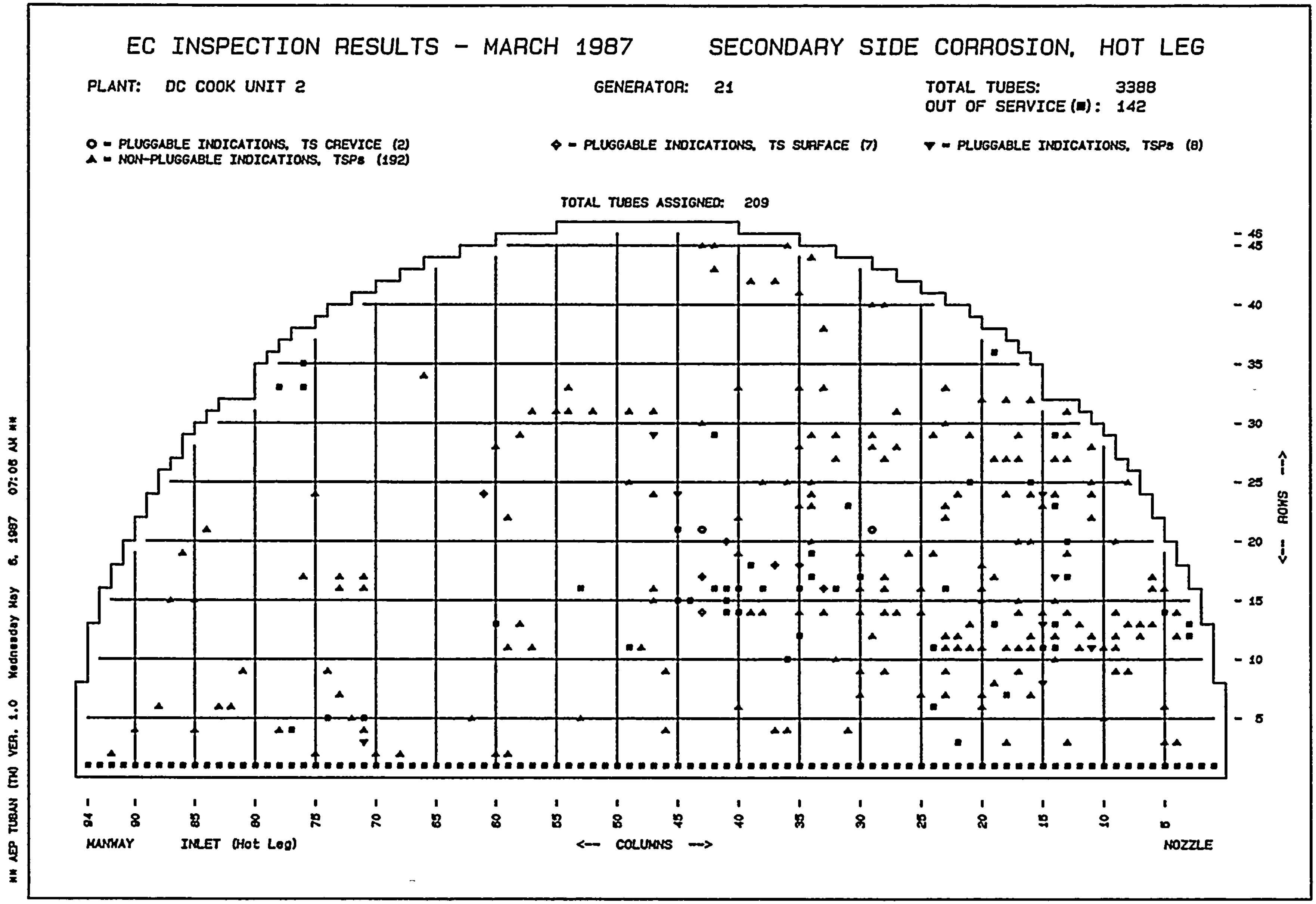

EC INSPECTION RESULTS - MARCH 1987 SECONDARY SIDE CORROSION, HOT LEG
PLANT: DC COOK UNIT 2
GENERATOR: 21
TOTAL TUBES: 3388
OUT OF SERVICE (■): 142
○ = PLUGGABLE INDICATIONS, TS CREVICE (2)
▲ = NON-PLUGGABLE INDICATIONS, TSPs (192)
◆ = PLUGGABLE INDICATIONS, TS SURFACE (7)
▼ = PLUGGABLE INDICATIONS, TSPs (8)
TOTAL TUBES ASSIGNED: 209
<-- ROWS -->
<-- COLUMNS -->
MANWAY
INLET (Hot Leg)
NOZZLE
** AEP TUBAN (TM) VER. 1.0 Wednesday May 6, 1987 07:08 AM **

Figure 1

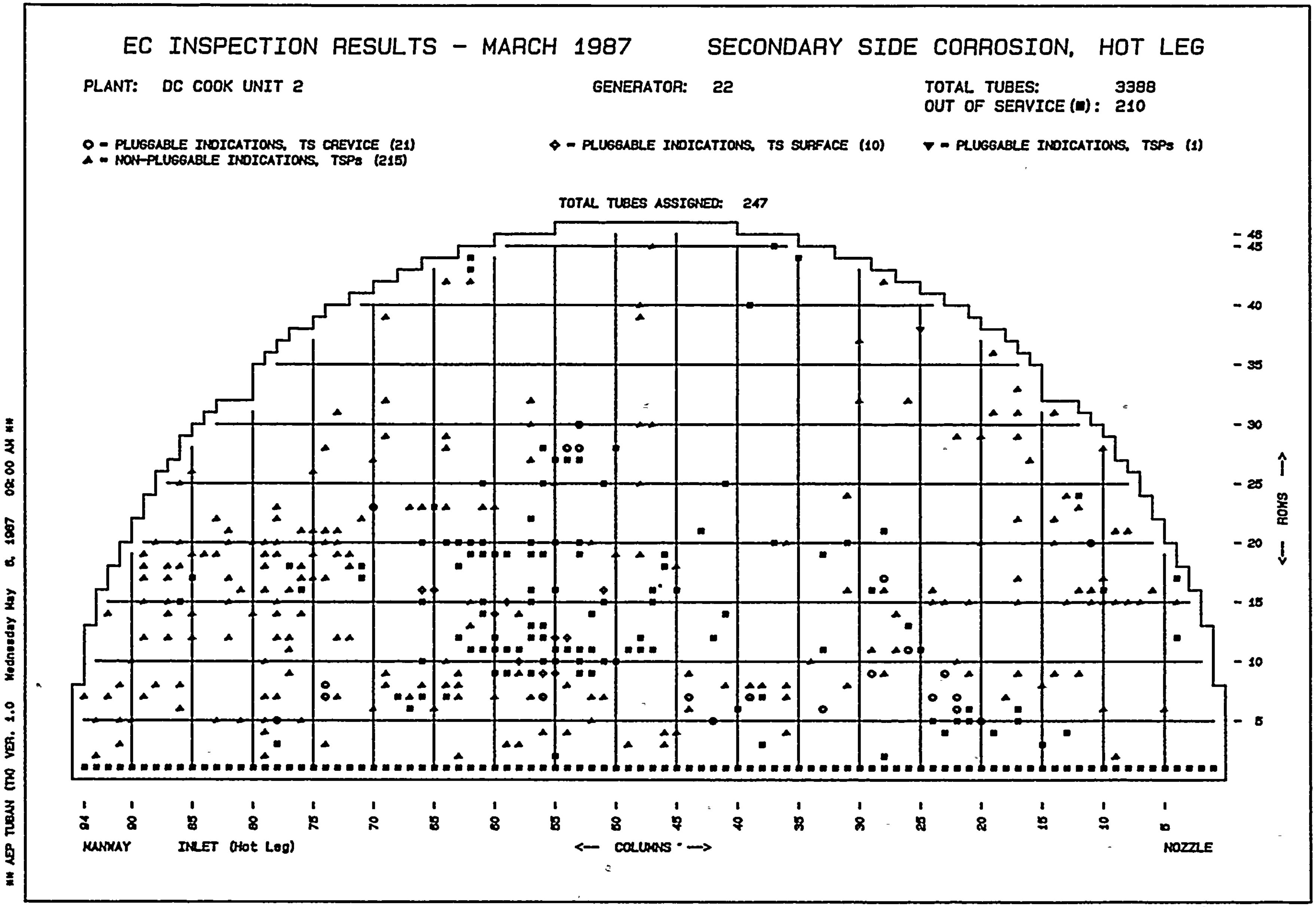

EC INSPECTION RESULTS - MARCH 1987 SECONDARY SIDE CORROSION, HOT LEG
PLANT: DC COOK UNIT 2
GENERATOR: 22
TOTAL TUBES: 3388
OUT OF SERVICE (■): 210
○ = PLUGGABLE INDICATIONS, TS CREVICE (21)
▲ = NON-PLUGGABLE INDICATIONS, TSPs (215)
◇ = PLUGGABLE INDICATIONS, TS SURFACE (10)
▼ = PLUGGABLE INDICATIONS, TSPs (1)
TOTAL TUBES ASSIGNED: 247
46
45
40
35
30
25
20
15
10
5
<-- ROWS -->
94
90
85
80
75
70
65
60
55
50
45
40
35
30
25
20
15
10
5
MANWAY
INLET (Hot Leg)
<-- COLUMNS -->
NOZZLE
** AEP TUBAN (TM) VER. 1.0 Wednesday May 6, 1987 09:00 AM **

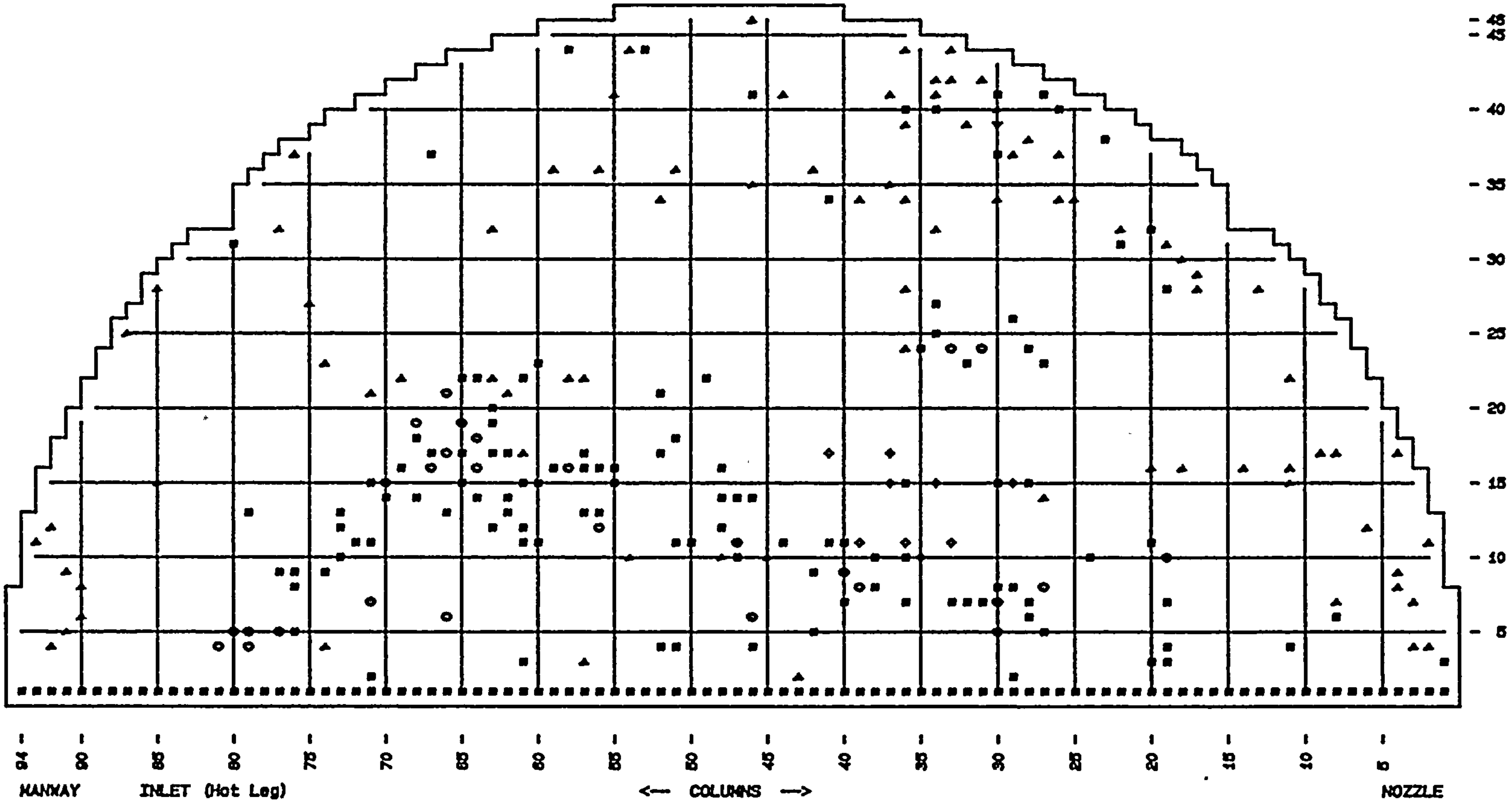
Figure 2

EC INSPECTION RESULTS - MARCH 1987 SECONDARY SIDE CORROSION, HOT LEG

PLANT: DC COOK UNIT 2 GENERATOR: 23 TOTAL TUBES: 3388
OUT OF SERVICE (■): 210

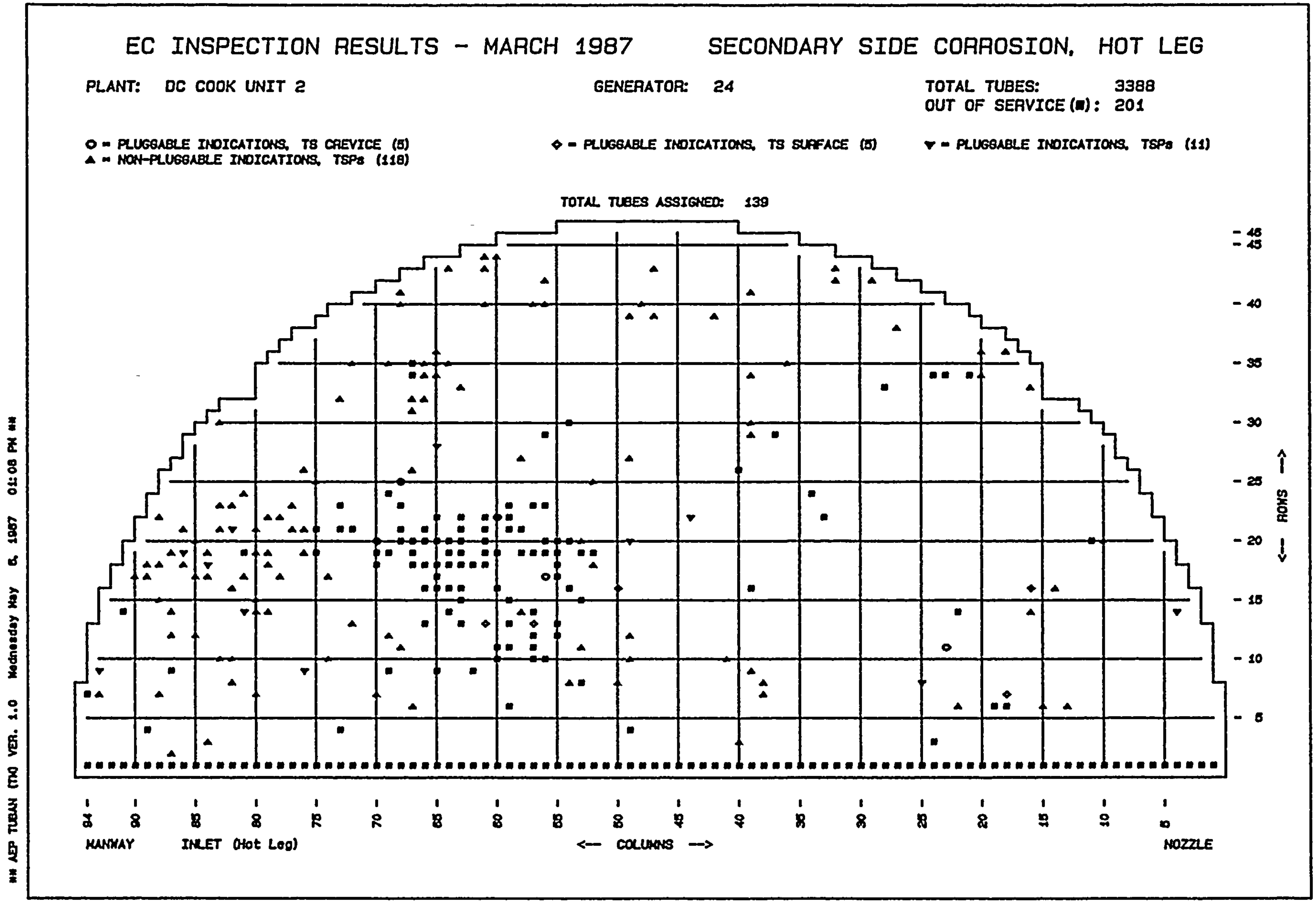
○ = PLUGGABLE INDICATIONS, TS CREVICE (27) ◇ = PLUGGABLE INDICATIONS, TS SURFACE (9) ▼ = PLUGGABLE INDICATIONS, TSPs (1)
▲ = NON-PLUGGABLE INDICATIONS, TSPs (84)

TOTAL TUBES ASSIGNED: 121



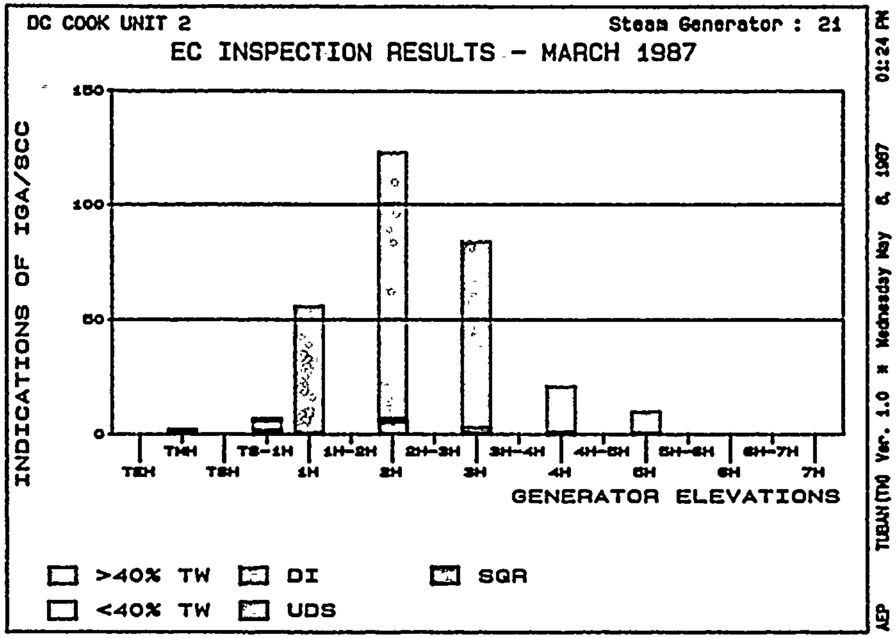
** AEP TUBAN (TM) VER. 1.0 Wednesday May 6, 1987 10:07 AM **

Figure 3



EC INSPECTION RESULTS - MARCH 1987 SECONDARY SIDE CORROSION, HOT LEG
PLANT: DC COOK UNIT 2
GENERATOR: 24
TOTAL TUBES: 3388
OUT OF SERVICE (■): 201
○ = PLUGGABLE INDICATIONS, TS CREVICE (5)
▲ = NON-PLUGGABLE INDICATIONS, TSPs (118)
◇ = PLUGGABLE INDICATIONS, TS SURFACE (5)
▼ = PLUGGABLE INDICATIONS, TSPs (11)
TOTAL TUBES ASSIGNED: 139
46
45
40
35
30
25
20
15
10
5
<-- ROWS -->
94 90 85 80 75 70 65 60 55 50 45 40 35 30 25 20 15 10 5
MANWAY
INLET (Hot Leg)
<-- COLUMNS -->
NOZZLE
** AEP TUBAN (TM) VER. 1.0 Wednesday May 6, 1987 01:08 PM **

Figure 4


DC COOK UNIT 2
Steam Generator : 21
EC INSPECTION RESULTS - MARCH 1987
INDICATIONS OF IGA/SCC
GENERATOR ELEVATIONS
>40% TW
<40% TW
DI
UDS
SQR
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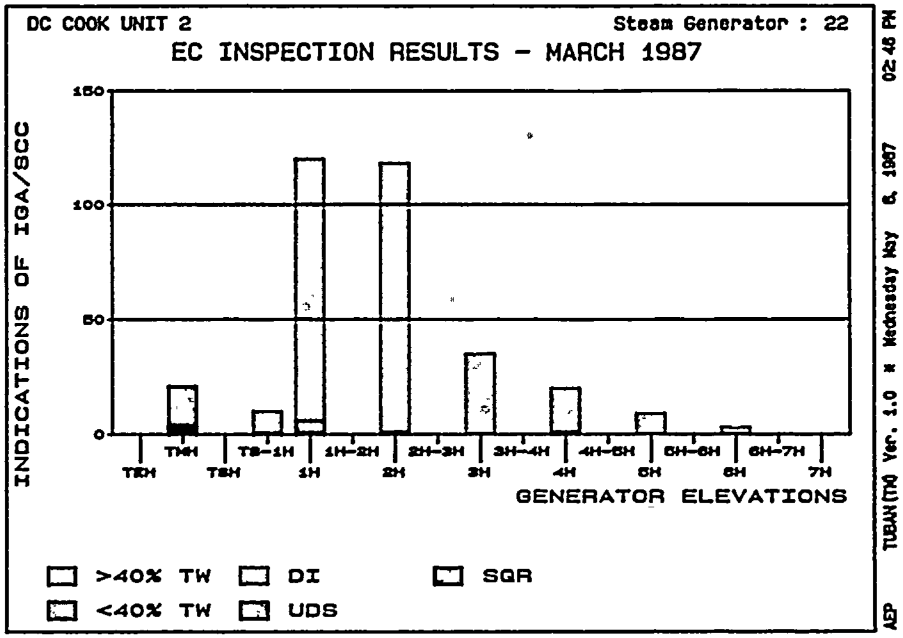
AEP

DC COOK UNIT 2
Steam Generator : 22
EC INSPECTION RESULTS - MARCH 1987
INDICATIONS OF IGA/SCC
GENERATOR ELEVATIONS
>40% TW
<40% TW
DI
UDS
SQR
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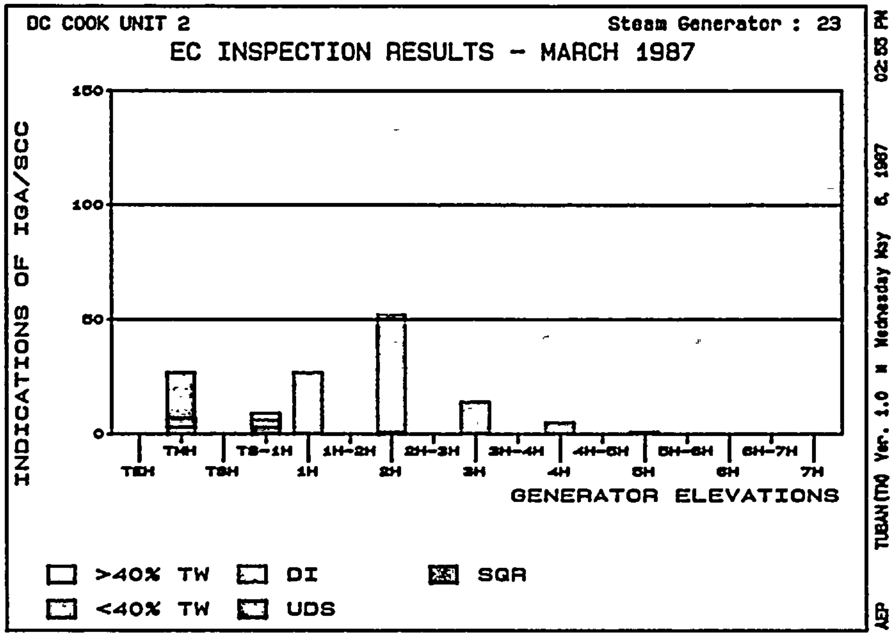
DC COOK UNIT 2
Steam Generator : 23
EC INSPECTION RESULTS - MARCH 1987
INDICATIONS OF IGA/SCC
GENERATOR ELEVATIONS
>40% TW
<40% TW
DI
UDS
SQR
TUBAN(TM) Ver. 1.0 * Wednesday May 6, 1987 02:55 PM
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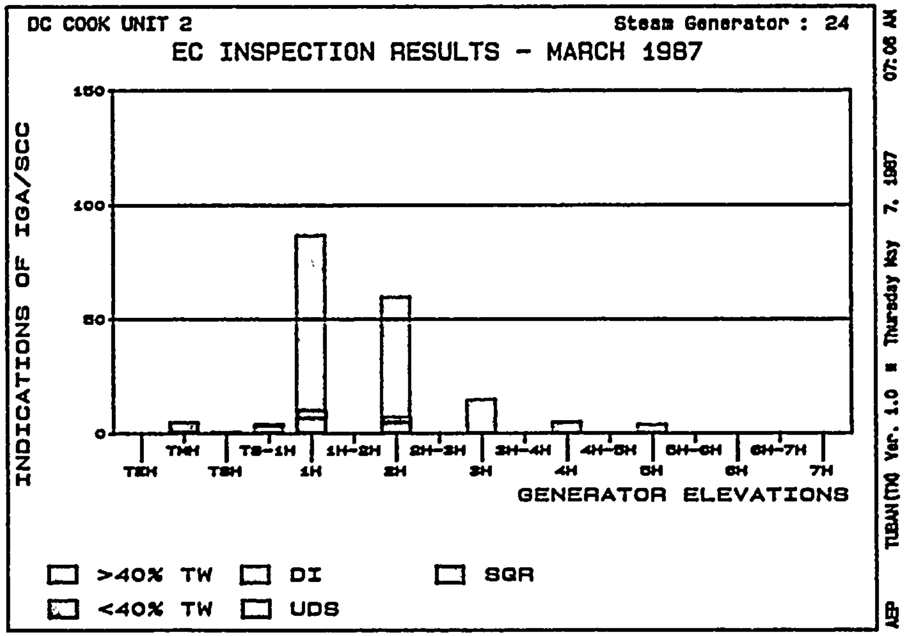
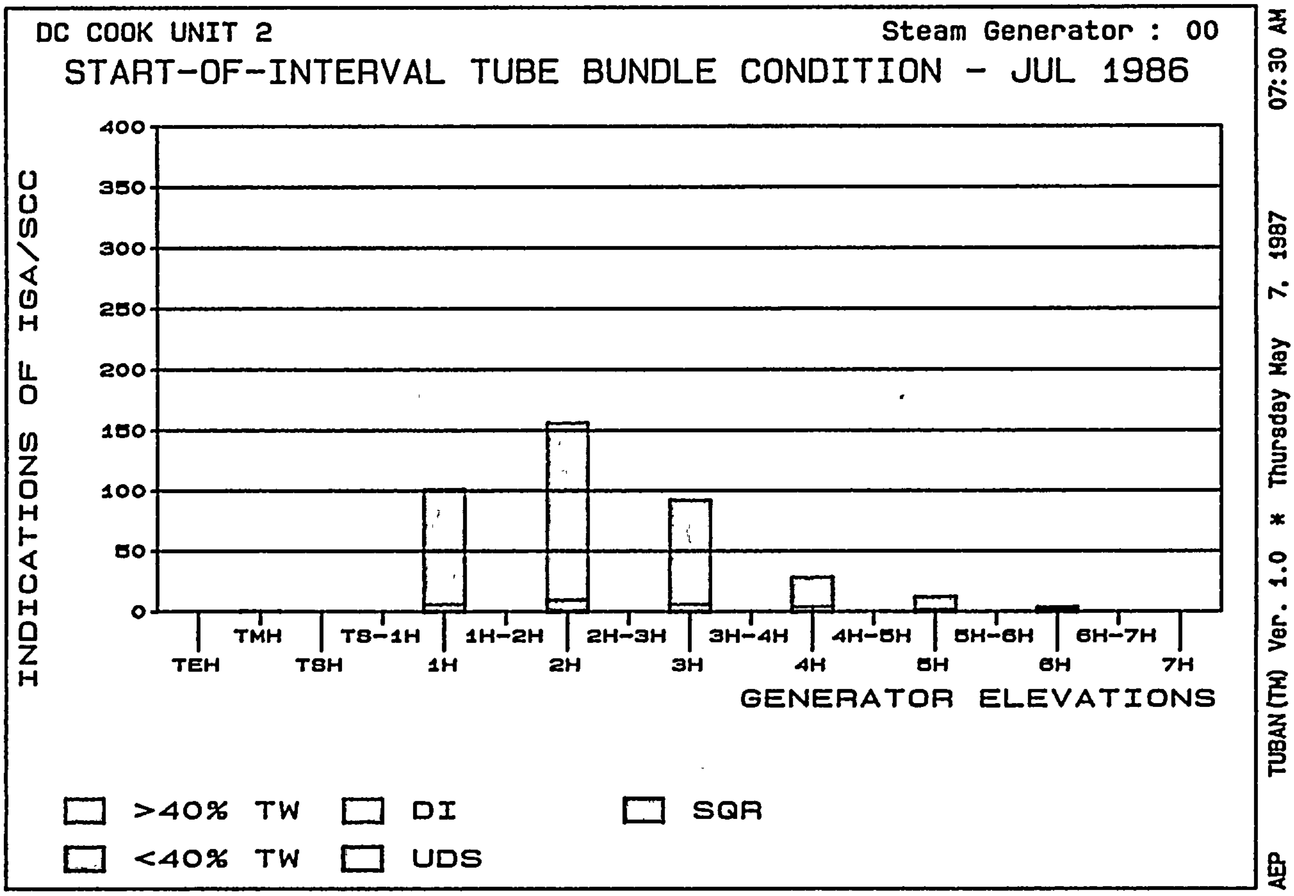

DC COOK UNIT 2
Steam Generator : 24
EC INSPECTION RESULTS - MARCH 1987
INDICATIONS OF IGA/SCC
GENERATOR ELEVATIONS
>40% TW
<40% TW
DI
UDS
SQR
TUBAN(TM) Ver. 1.0 * Thursday May 7, 1987 07:06 AM
AEP

Figure 5


DC COOK UNIT 2
Steam Generator : 00
START-OF-INTERVAL TUBE BUNDLE CONDITION - JUL 1986
INDICATIONS OF IGA/SCC
400
350
300
250
200
150
100
50
0
TMH
TS-1H
1H-2H
2H-3H
3H-4H
4H-5H
5H-6H
6H-7H
TEH
TSH
1H
2H
3H
4H
5H
6H
7H
GENERATOR ELEVATIONS
>40% TW
DI
SQR
<40% TW
UDS
AEP TUBAN(TM) Ver. 1.0 * Thursday May 7, 1987 07:30 AM

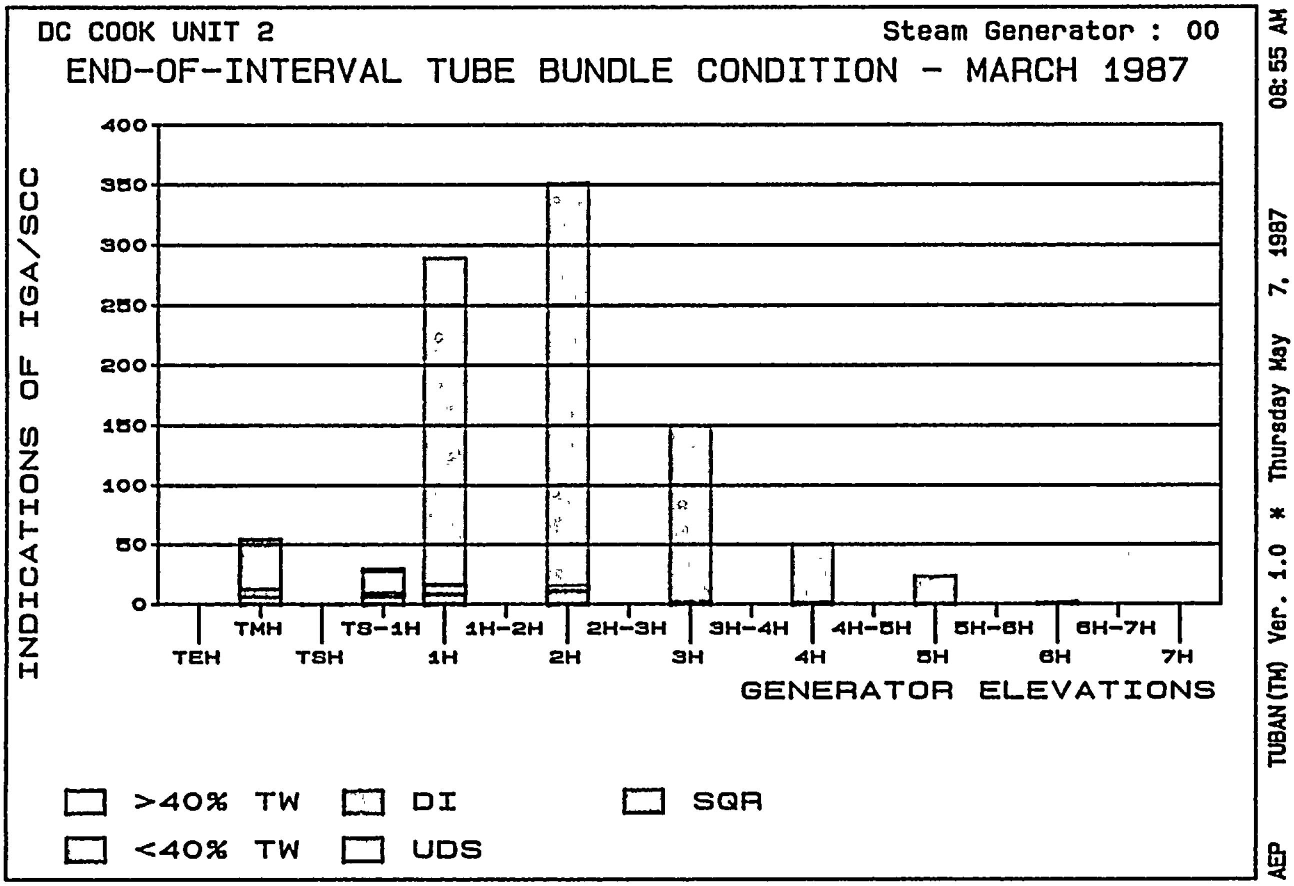

DC COOK UNIT 2
Steam Generator : 00
END-OF-INTERVAL TUBE BUNDLE CONDITION - MARCH 1987
INDICATIONS OF IGA/SCC
400
350
300
250
200
150
100
50
0
TMH
TS-1H
1H-2H
2H-3H
3H-4H
4H-5H
5H-6H
6H-7H
TEH
TSH
1H
2H
3H
4H
5H
6H
7H
GENERATOR ELEVATIONS
>40% TW
DI
SQR
<40% TW
UDS
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Figure 6

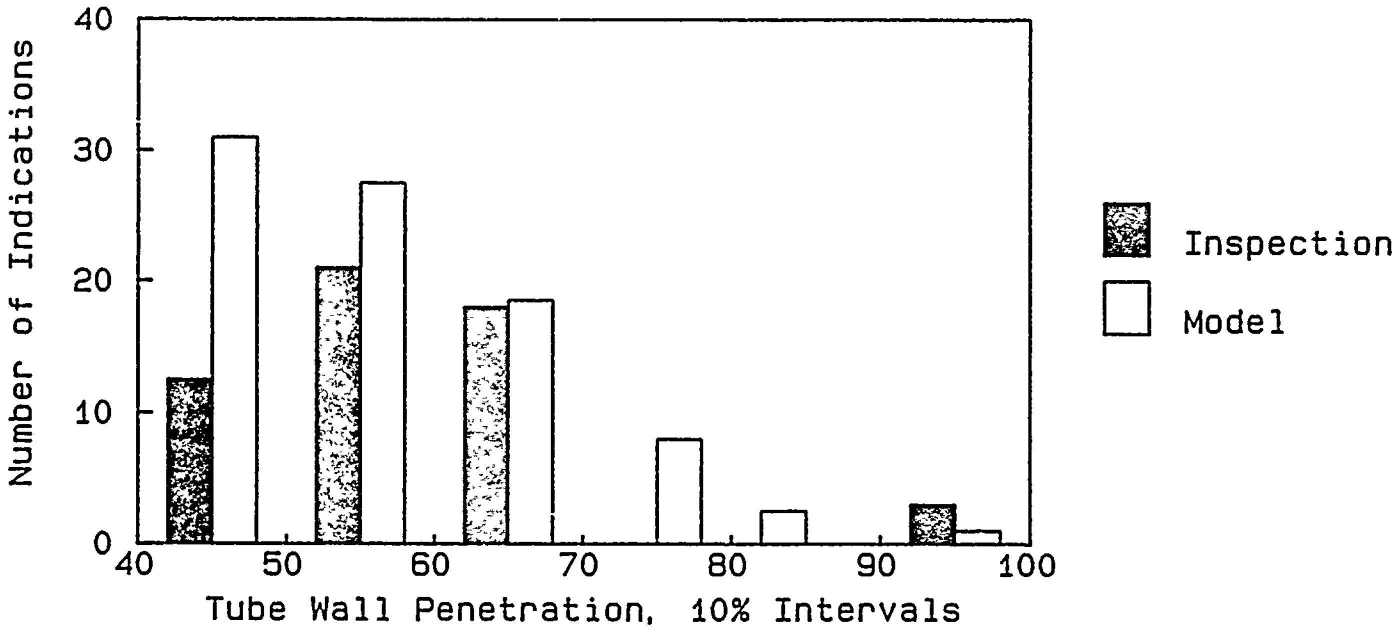


Figure 7 Comparison of Model Prediction to
Actual EC Inspection Results,
Tubesheet Crevice Region

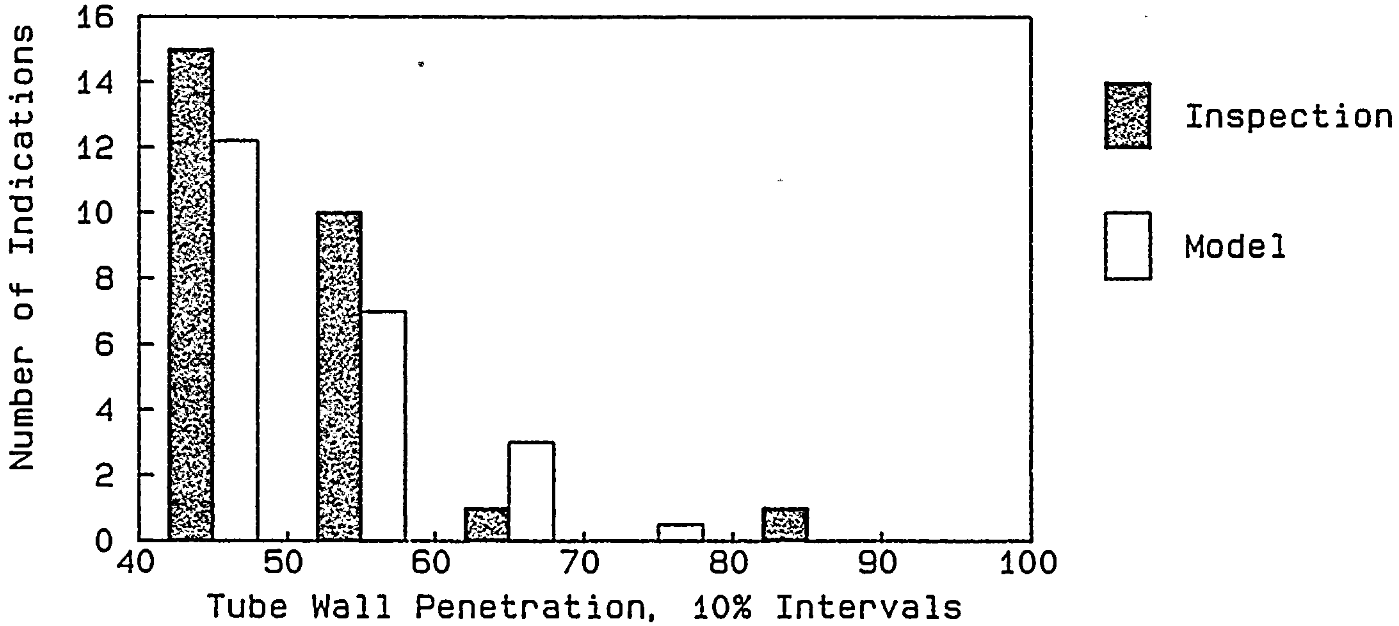


Figure 8 Comparison of Model Prediction to Actual EC Inspection Results, Tubesheet Surface Region

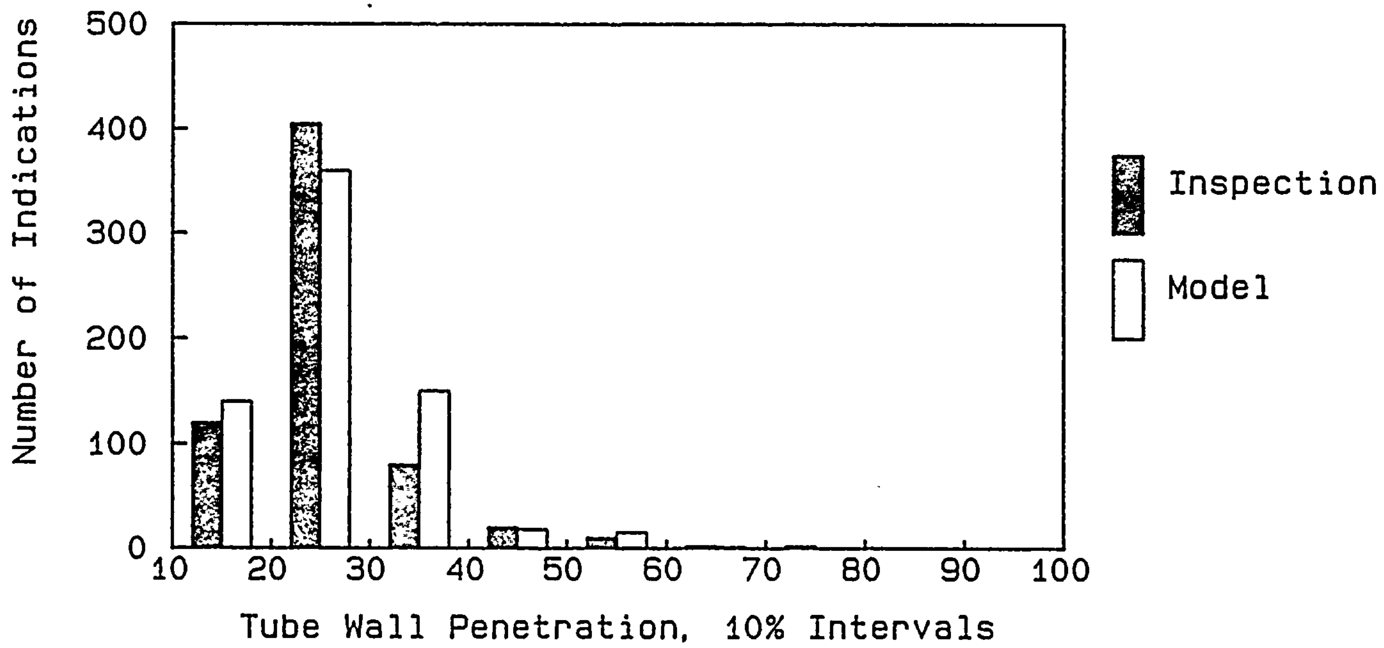


Figure 9-A Comparison of Model Prediction to Actual EC Inspection Results, Tube Support Plate Intersections Full Range, 0 to 100% TW

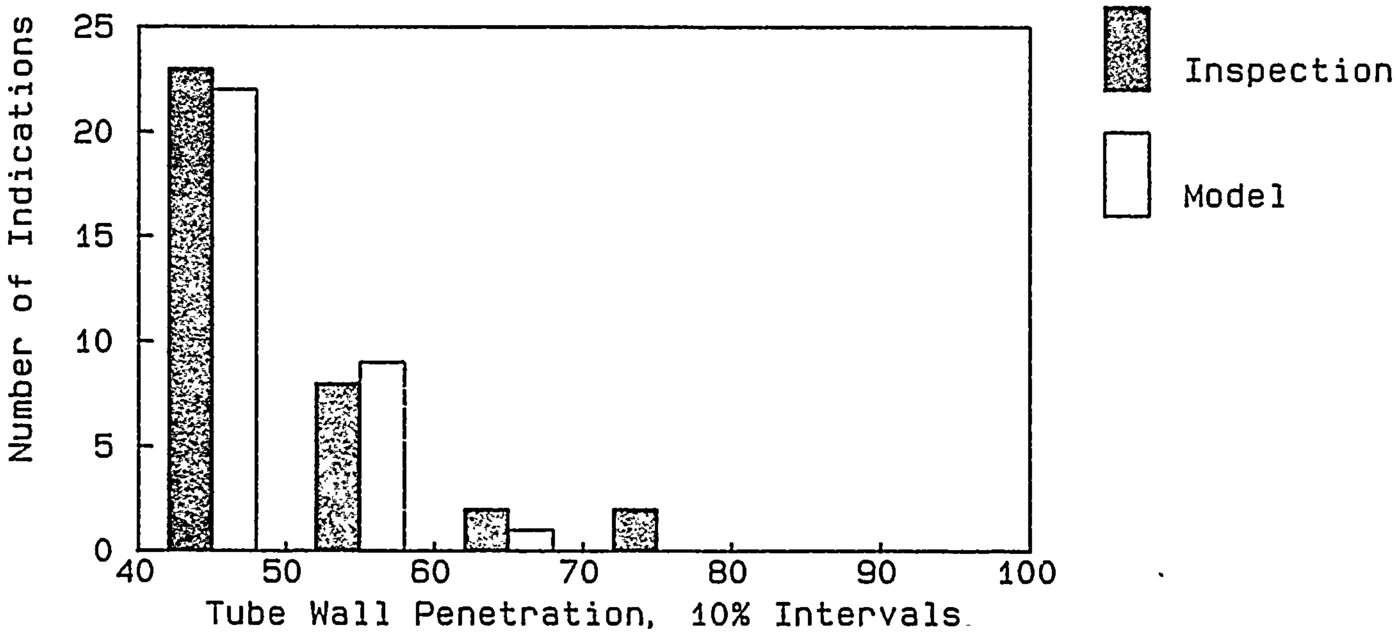


Figure 9-B Comparison of Model Prediction to Actual EC Inspection Results, Tube Support Plate Intersections Blowup of 40 to 100% TW Range

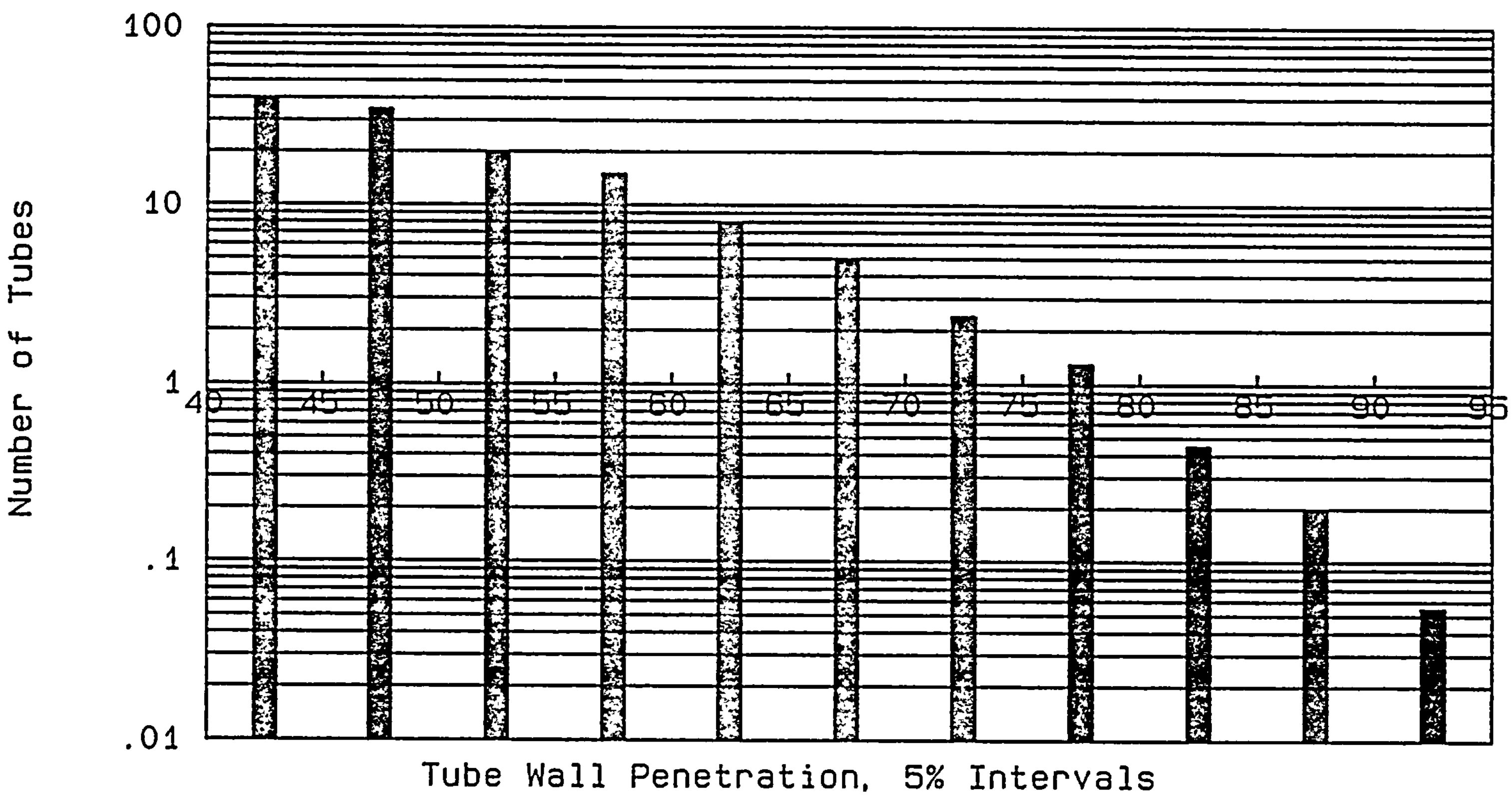


Figure 10 Predicted End-of-Interval Condition,
Tubesheet Crevice Region

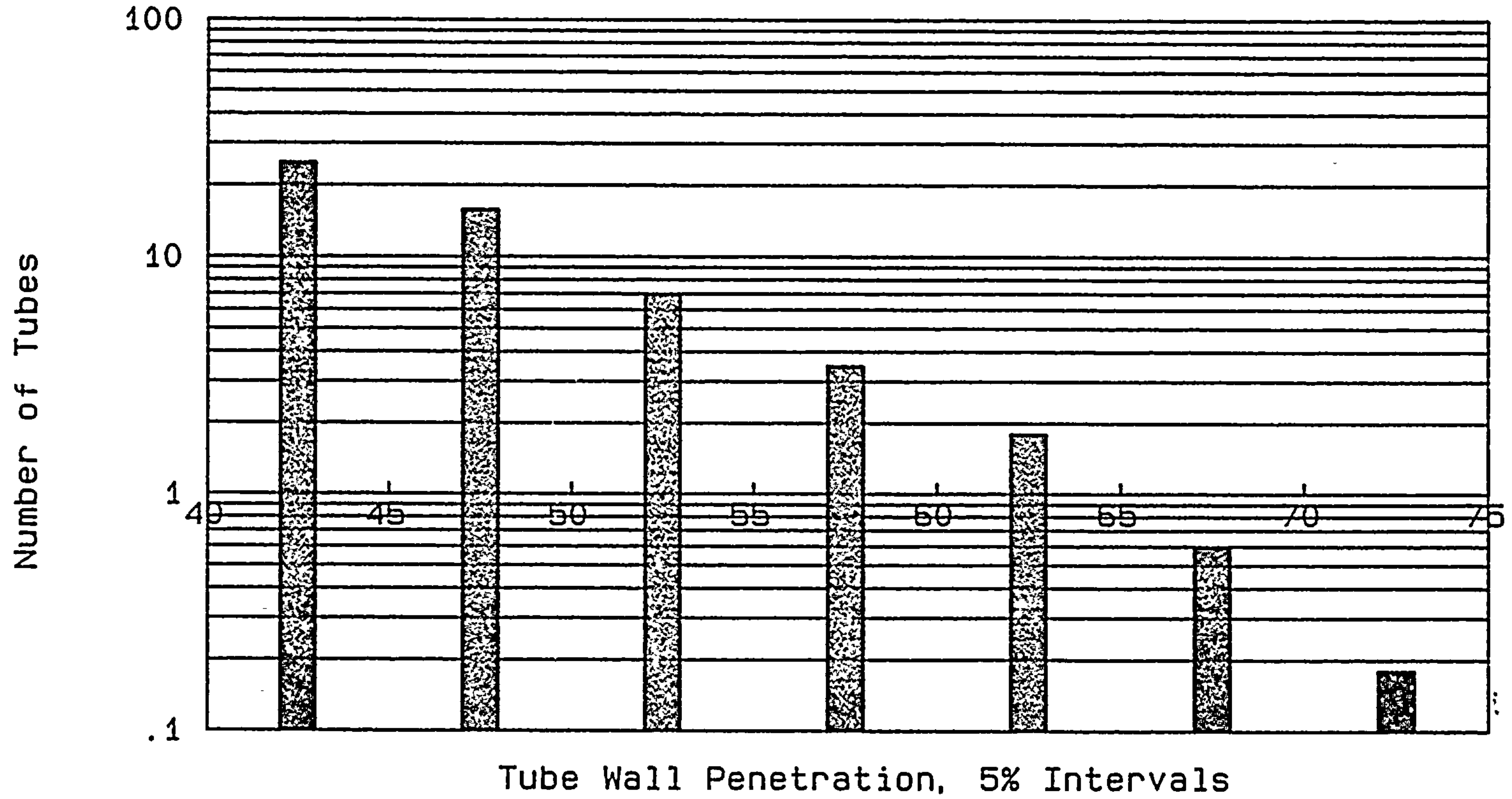


Figure 11 Predicted End-of-Interval Condition,

Tubesheet Surface Region

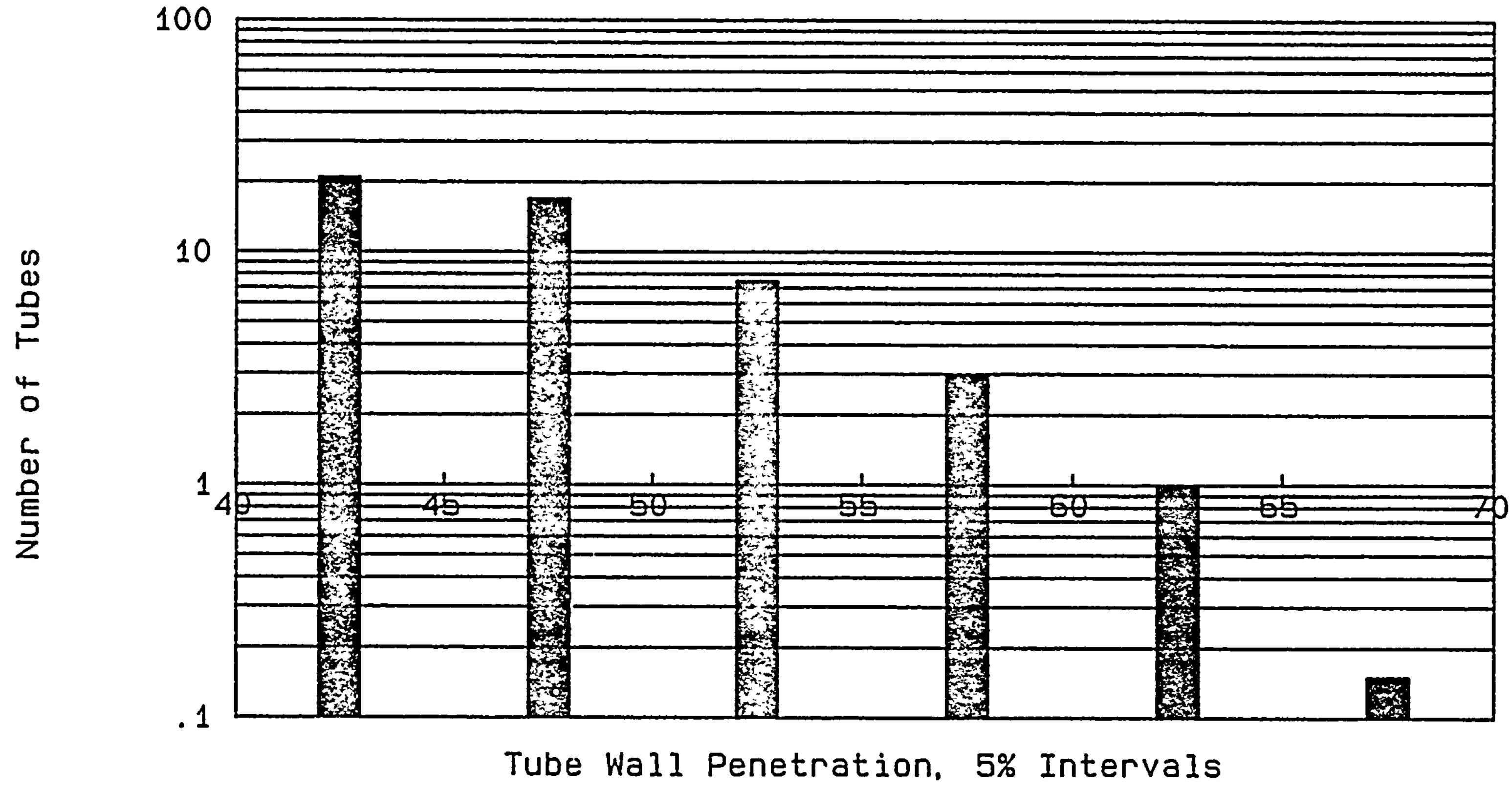


Figure 12 Predicted End-of-Interval Condition,

Tube Support Plate Intersections

Attachment 2 to AEP:NRC:0936J

Steam Generator Manway Cover Closure Repairs
March-April 1987

STEAM GENERATOR MANWAY COVER CLOSURE REPAIRS

March-April 1987

Each steam generator channel head half (hot leg and cold leg) has a 16-inch manway; design of the bolted closure is shown in Figure 1. When opening the manways to perform tube inspections following the March 1987 steam generator tube leak, difficulty in removing the bolts on both legs of SGs 22 and 23 was encountered. There was evidence of galling under the bolt head at some locations, and an observation was made that insufficient thread lubricant may have been used during the previous installation. Five bolts on SG 23 could not be removed by de-torquing and were drilled out. The bolts on SGs 21 and 24 were removed without difficulty.

Actions taken by I&MECo as a result of the bolt removal problem included:

- A design change (RFC) to allow use of hardened steel washers under the bolt heads was approved. This change is intended to provide a more uniform friction factor under the head, and therefore introduce more uniform bolt tension.

- The newly-approved washers and new manway cover bolts were procured for use in re-installing the manway covers.

- Westinghouse was hired to inspect and gauge the bolt holes. A "go/not go" gauge was used to determine the acceptability of the hole pitch diameter. The gauge tolerances were those of a <u>new</u> hole and were therefore very conservative.

Results of the bolt hole gauging program on SGs 22 and 23 were as follows:

- <u>SG 22</u> - Five holes on the hot leg and five holes on the cold leg had oversize pitch diameters and required repair.

- <u>SG 23</u> - Thirteen holes on the hot leg and eight holes on the cold leg had oversize pitch diameters and required repair.

Even though no difficulty was experienced on SGs 21 and 24, the manway cover bolt holes on those two steam generators were gauged as an added precaution. Results of that inspection are as follows:

- o SG 21 - All sixteen holes on both hot and cold legs were slightly oversize and could not be dispositioned by Westinghouse. In all likelihood, the holes were acceptable and a complete analysis would have allowed disposition of them in the "as-found" condition. However, due to the inherent difficulty in measuring in situ female thread parameters (e.g. - thread form, thread angle, and actual pitch diameter), sufficient data to do a complete analysis could not be readily acquired, so it was decided to repair these also.

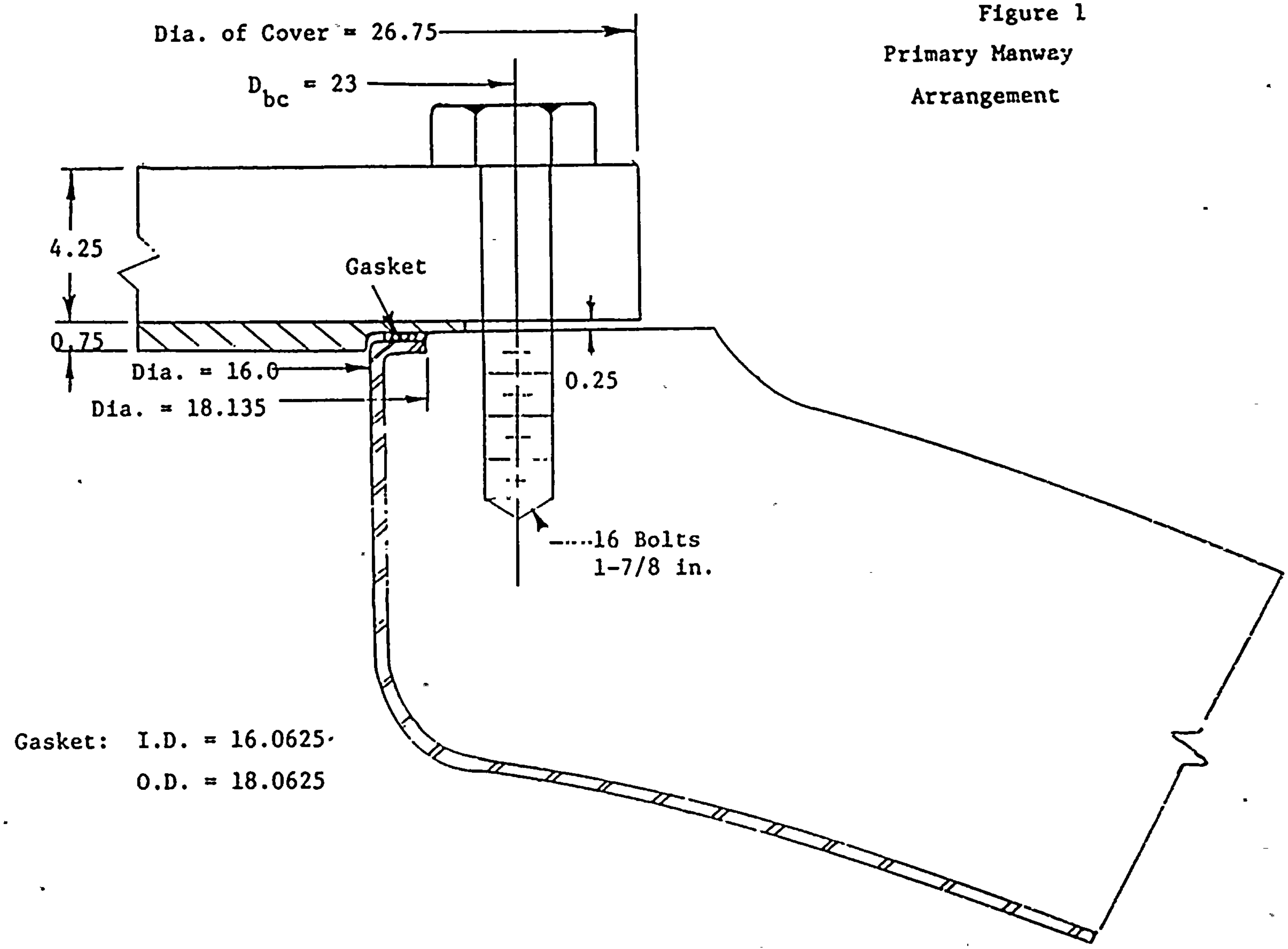
- o SG 24 - All holes were acceptable.

Two methods of female thread repair are in common use: replacement of the existing threads with a Heli-coil and installation of a threaded insert. The Heli-coil method was selected for the Unit 2 repairs, with the threaded insert method held as a back-up in the event the Heli-coil technique was unsuccessful on a particular hole. Westinghouse provided a safety evaluation and installation procedure for each method; an RFC to allow the use of either was approved. However, use of threaded inserts was not necessary.

The Heli-coil repair technique consists of drilling the existing bolt hole about 1/8 inch over the nominal size to remove the old thread, threading the resultant hole with an appropriate sized thread tap, and then screwing in a stainless steel Heli-coil (trade name for a helical thread whose outer surface mates with the newly-tapped hole threads and whose inner surface forms female threads for the bolt hole). The new hole accepts the same sized bolt as before, and actually has "better" threads (closer tolerances, more exact thread form, and - in this case - better material). Heli-coils are considered a permanent repair.

The Heli-coil repairs were made to all affected bolt holes as noted above, and the manway covers were put in place - using washers and new bolts - without further incident. We are evaluating the cause of this problem and we will inform the NRC of the results of this evaluation when it is completed

Figure 1
Primary Manway
Arrangement



Attachment 3 to AEP:NRC:0936J

Westinghouse Nuclear Safety Evaluation
of Loose Mechanical Plug
in Steam Generator 22

SECL-87-229
Customer Reference No(s).

Westinghouse Reference No(s).

NS-RCSCL-C/L-87-450

WESTINGHOUSE
NUCLEAR SAFETY EVALUATION CHECK LIST

1) NUCLEAR PLANT(S) D. C. COOK UNIT 2

2) CHECK LIST APPLICABLE TO: LOOSE MECHANICAL PLUG STEAM GENERATOR #22
(Subject of Change)

3) The safety evaluation of the revised procedure, design change or modification required by 10CFR50.59 has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A

(3.1) Yes___ No_X_ A change to the plant as described in the FSAR?
(3.2) Yes___ No_X_ A change to procedures as described in the FSAR?
(3.3) Yes___ No_X_ A test or experiment not described in the FSAR?
(3.4) Yes___ No_X_ A change to the plant technical specifications (Appendix A to the Operating License)?

4) CHECK LIST - PART B (Justification for Part B answers must be included on page 2.)

(4.1) Yes___ No_X_ Will the probability of an accident previously evaluated in the FSAR be increased?
(4.2) Yes___ No_X_ Will the consequences of an accident previously evaluated in the FSAR be increased?
(4.3) Yes___ No_X_ May the possibility of an accident which is different than any already evaluated in the FSAR be created?
(4.4) Yes___ No_X_ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
(4.5) Yes___ No_X_ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
(4.6) Yes___ No_X_ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
(4.7) Yes___ No_X_ Will the margin of safety as defined in the bases to any technical specification be reduced?

If the answers to any of the above questions are unknown, indicate under 5) REMARKS and explain below.

If the answer to any of the above questions in 4) cannot be answered in the negative, based on written safety evaluation, the change cannot be approved without an application for license amendment submitted to the NRC pursuant to 10CFR50.59.

5) REMARKS:

None

The following summarizes the justification upon the written safety evaluation, (*) for answers given in Part B of the Safety Evaluation Check List:

See attached Safety Evaluation.

(*) Reference to document(s) containing written safety evaluation:______

FOR FSAR UPDATE

Section:________ Pages:_________ Tables:__________ Figures:__________

Reason for / Description of Change: None

Prepared by (Nuclear Safety):MATTHEWS R.J. Matthews Date: 5-18-87

Coordinated with Engineer(s):NELSON L.A. Nelson Date: 5/18/87

Coordinated Group Manager(s):KEATING RF Keating Date: 5/18/87

Nuclear Safety Group Manager:HIRST C. W. Hirst Date: 5/18/87

D. C. COOK UNIT 2
LOOSE MECHANICAL PLUG STEAM GENERATOR #22
SAFETY EVALUATION

INTRODUCTION

This evaluation is provided to address the safety impact of an object found lodged in a tube on the hot leg side of steam generator #22 of D. C. Cook Unit 2. The item has been identified as a mechanical plug originally installed in the hot leg tube end of another tube in the same steam generator. This evaluation considers the effect of disengagement of the plug from the tube in which it was originally installed, the effect of the plug on the tube in which it became lodged and the impact of the plug on the hot leg channel head components while the plug was mobile and not lodged in any tube.

BACKGROUND

During the recent 100% eddy current program at D. C. Cook Unit #2, a foreign object was reported to be lodged in the hot leg of steam generator #22. The object was located approximately 0.75 inches above the tube end of Row 3 Column 5. The foreign object was reported to be round and it appeared to closely fill the tube inner diameter (ID).

After preliminary attempts were made to dislodge and remove the foreign object, an attempt was made by site personnel to drive the object further into the tube. This was intended to allow enough access to install a mechanical plug behind it. Finally the foreign object was successfully removed by initially drilling a pilot hole, followed by drilling a 3/8 inch access hole through the material, inserting a slide hammer and then pulling it free from the tube ID. Once removed, the foreign object was identified as a Westinghouse mechanical plug that had lodged in the tube end in an inverted position.

A thorough review of video tapes of the tubesheet in the hot leg of steam generator #22 showed that the tube end at Row 40 Column 39 was missing a mechanical plug. This tube end was documented as having been plugged in the April, 1986 outage, was determined to be open and was the apparent source of the mechanical plug found in R3-C5.

To investigate the possible cause of the plug moving from the tube end into which it had been installed, the removed mechanical plug and the tube end at R40-C39 were visually and mechanically inspected including the expanded diameters and the expander translation. Visual examination of the plug by experienced mechanical plugging and quality assurance personnel revealed that the plug exhibited scratches on the surface as well as the plug lands had been rounded

off. The tube ID (R40-C39) in the elevation range where the plug is designed to seal, was measured at 0.5 inch intervals at two azimuths. The recorded diameters are consistent with the nominal roll expanded diameters for steam generators with 7/8 inch diameter tubing. The tube end was visually examined to check for any circumferential indentations that are occasionally left in the tube ID after a successful installation and subsequent removal and none were evident.

The tube end in which the plug became lodged (R3-C5) was inspected in accordance with the proper acceptance criteria as specified in the procedure for mechanical plugging of steam generator tubes. It was evaluated as acceptable for mechanical plugging. The hot leg tube end at R40-C39 was also evaluated as acceptable for installation of a new plug.

Both ends of the tube in which the plug had become lodged (R3-C5) were mechanically plugged and the tube removed from service as a precautionary measure. The hot leg tube end of R40-C39, that was missing the mechanical plug was also mechanically plugged. The process parameters for these plugging operations were witnessed, verified and recorded.

EVALUATION

The condition for which the R40-C39 tube had been plugged in the April 1986 outage was an eddy current indication termed a squirrel. Such an indication is a signal in the tubesheet region whose trace at 400 KHz is complex and phase angle unclear, but whose presence represents change. These indications have been historically proven to compromise tube wall integrity if the tube remains in service and thus have been classified as tube degradation. In the D. C. Cook Unit 2 steam generators these indications are associated with degradation on the outside surface of the tube in the tube to tubesheet crevice.

The corrosion resistance of a steam generator tube plugged on the cold leg only was evaluated. General forms of corrosion are typically environmentally and/or materially controlled. Most secondary side initiated tubing corrosion found in recirculating steam generators has occurred in localized regions (most commonly crevices) of a steam generator tube in which dissolved chemical species can be concentrated to levels far greater than those in the bulk primary or secondary fluid. Heat transfer is necessary such that the available superheat (local wall temperature minus fluid saturation temperature) is increased compared to values associated with conventional nucleate boiling processes as they exist on the tube surface. The elevated temperatures provide the driving force for promoting chemical concentration i.e., the potential for the formation of a locally concentrated solution can be correlated with the expected available superheat within the region. As the primary fluid within a tube plugged on the cold leg only would be at

approximately secondary side bulk fluid saturation temperature and in a subcooled state, no heat transfer would be expected across the tube surface and any localized tube degradation including continuing degradation at the site of the previously located eddy current signal would be expected to be minimal.

The safety impact of operation of tube R40-C39 with what is normally a pluggable indication is mitigated by the geometry of the region. The tube to tubesheet crevice is the space between the tubesheet and the unexpanded tubes and is on the order of a few mills. Tube plugging limits are established in part based on predicted performance of a degraded tube under postulated faulted conditions, specifically steam line break conditions. For indications in the tubesheet crevice region, tube rupture is not possible due to the presence of the tubesheet around the tube which would contain the movement of the tube wall required to effect a burst tube condition. Therefore, in the event of a postulated steam line break with the mechanical plug missing from one end of the R40-C39 tube and the previously observed eddy current indication would not be expected to result in primary to secondary leakage in excess of that used for accident analyses.

The effect of plant operation on plug integrity for up to one year with the steam generator tube plugged on only the cold leg side has been evaluated. The mechanical plug was designed to accommodate the design conditions specified for the steam generator. The design conditions envelop the approximate 10 psi pressure differential which occurs across the channel head in a tube which has been plugged on the cold leg only but not on the hot leg. The design verification program simulated the steam generator service conditions of temperature and pressure as well as thermal cycling associated with the various plant conditions. The design verification program for the expanded mechanical plug demonstrated pressure boundary integrity under simulated faulted condition loadings in addition to other plant operating conditions.

The design of the Series 51 steam generators at D. C. Cook Unit 2 includes a small extension of the tube end past the bottom of the tubesheet surface. A foreign object removed from the channel head during a previous outage had resulted in some deformation of the tube ends. None of the tube ends of the other tubes had a restriction that would prevent insertion of an eddy current probe and the tube ends had no apparent additional damage due to the loose plug. The tube to tubesheet welds are partially shielded from impact of an object of the size of a mechanical plug and the welds had no apparent damage. The cladding of the channel head and the tubesheet also showed no apparent damage. The tubes, channel head and tubesheet cladding, weld metal and the mechanical plug are all composed of very ductile material. Repeated impact of the plug on the cladding, tube ends, and tube to tubesheet weld would not be expected to cause

cracking or small pieces to break loose from the surfaces impacted by the loose plug. Evaluations of more significant deformation of tube ends in other steam generators of similar design have shown that deformation of the tube end will not significantly degrade the structural integrity of the tube or the tube to tubesheet weld or cause a significant increase in the restriction to flow through the steam generator

The mechanical plug land outer diameters approximate the tube inner diameters in the seating area of R40-C39. In order for the mechanical plug to have proper sealing, the plug should have been larger than the tube ID to allow for an interference fit.

There was no visually discernible evidence on the ID of the tube at R40-C39 that the plug had a positive interference fit with the tube, although it is not mandatory to have this for a properly installed plug. In some tube ends that are approximately as hard as the plug lands, however, there are no interference marks and plugs are successfully installed.

The estimate of the actual translation of the expander in the removed mechanical plug would indicate that insufficient expansion had occurred. The estimated expander translation distance did not meet the procedure installation minimum requirement.

The possibile anomalies in the tube-to-tubesheet joint contributing to the disengagement of the mechanical plug were reviewed. The diameter measurements and visual inspection showed negligible ovality, no taper nor any other problem (such as a lack of roll expansion in the plug sealing area), which would indicate that the configuration of the tube joint contributed to the as installed condition of the mechanical plug.

Based on the findings of the investigation outlined above it has been concluded that successful installation parameters for mechanical plug were not achieved and it was eventually displaced from the tube end during the operating period preceding the discovery of the misplaced plug.

Relevant Westinghouse log books, data sheets, notes and procedures were reviewed in detail from the April, 1986 outage in an attempt to identify a potential area to account for the as installed condition of the plug. The job site coordinator, shift supervisors and other key personnel were queried to attempt to identify a causative factor. In all cases there was nothing identified. Based on prior Westinghouse experience of virtually 100% successful installations over an eight year period of over 25,000 previous mechanical plug installations, coupled with other installation data collected on surveillance reports from a large percentage of mechanical plug

installations during the April, 1986 program, the judgment has been made that the probability that the other mechanical plugs installed at D. C. Cook Unit #2 during the April, 1986 outage were installed correctly approximates 100% .

CONCLUSIONS

On the basis of the investigation and evaluation as outlined above, it has been concluded that the mechanical plug lodged in the hot leg of steam generator #22 in R3-C5 is the same mechanical plug that was originally installed in the R40-C39 in the same leg of the same steam generator during the 4/86 outage. Due to the conditions of the fluid in the partially plugged tube, significant additional or continuing corrosion would not be expected to occur. Operation of the steam generator with one plug in the R40-C39 tube is not expected to have resulted in a condition which would have caused primary to secondary leakage in the event of a postulated steam line break in excess of that assumed for accident analyses. The integrity of the plug on the cold leg of the tube from which the hot leg side plug was displaced, was maintained under normal operating and postulated accident condition loadings. The impact of the loose plug prior to becoming lodged in tube R3-C5 caused no apparent damage to the tube ends or other surfaces in the channel head. The apparent cause of the as installed condition of the subject mechanical plug is the termination of the installation process prior to reaching successful installation parameters.

Therefore the displacement of a mechanical plug from the hot leg end of tube R40-C39, the impact of the loose plug on the channel head surfaces, and the subsequent lodging of the plug in tube R3-C5 did not result in the possibility of a previously unanalyzed accident or increase the probability of a previously analyzed accident. The margin of safety was not reduced. Based on the information outlined above, the loose plug in the hot leg of D. C. Cook steam generator #22 did not result in an unreviewed safety question as defined in the criteria of 10CFR 50.59.