ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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License No.:	NPF-30
Report No.:	50-483/96-10
Licensee:	Union Electric Company
Facility:	Callaway Plant
Location:	Junction Hwy. CC and Hwy. O Fulton, Missouri
Dates:	October 21 through December 9, 1996
Inspectors:	I. Barnes, Technical Assistant C. Beardslee, Materials Engineer D. Graves, Senior Project Engineer W. McNeill, Reactor Inspector P. Rush, Materials Engineer C. Dodd, NRC Consultant
Approved By:	Kenneth E. Brockman, Deputy Director Division of Reactor Safety

ATTACHMENTS

Attachment 1:	Supplemental Information
Attachment 2:	Steam Generator Tube Integrity Review
Attachment 3:	December 17, 1996, Facsimile From Licensee
Attachment 4:	November 26 and December 19, 1996, Facsimiles From Licensee

EXECUTIVE SUMMARY

Callaway Plant
NRC Inspection Report 50-483/96-10

Operations

- The off-normal and emergency operating procedures provided appropriate guidance for the operators in the event of a steam generator tube rupture (Section O3.1).

- The licensee's training program was comprehensive with respect to steam generator tube ruptures and focused on integrated plant response (Section O5.1).

Maintenance

- The inservice inspection program for repair/replacement was adequately implemented (Section M1.1).

- An error was identified by the inspectors in the Anser software used by Westinghouse which resulted in different voltage measurements by primary and secondary analysts for the same eddy current indications. The inspectors agreed with the licensee's conclusion that the error did not affect the analyst's capability to detect indications of steam generator tube degradation. The majority of the indications reviewed were detectable only with the plus point coil which suggested to the inspectors that they were small in size. The resolution process for addressing differences in calls between primary and secondary analysts appeared to have been effectively implemented (Section M1.2).

- The inspectors concluded that, in general, the defined laser welded steam generator tube sleeve installation process requirements were consistent with the criteria approved by the NRC and were satisfactorily implemented. An unresolved item was identified, however, in regard to the adequacy of monitoring of laser weld essential variables during field welding of sleeves (Section M1.3).

Engineering

- The inspectors concluded that the April 1992 shot peening should improve resistance to initiation of new primary water stress corrosion cracks, but would be expected to be of minimal benefit with respect to delaying propagation of cracks that were present at the time of peening (Section E1.1).

- The licensee has appropriately responded to emerging degradation modes with use of sample sizes and eddy current examination methods that are consistent with regulatory requirements and industry guidance (Section E1.2).

- Areas of weakness were noted in the licensee data analysis guidelines which included absence of criteria for establishing when eddy current data noise was unacceptable, a lack of restrictions on assignment of through-wall depths from bobbin coil data, and absence of specific requirements for disposition of potential manufacturer's burnish mark indications. A violation was identified with respect to the failure to establish requirements to assure use of Appendix H qualified span and phase rotation settings during analysis of plus point coil data. The controls used to ensure use of correct inspection probes during acquisition of eddy current examination data were found to be effective (Section E1.3).

- The programmatic requirement for analysts to be certified as Qualified Data Analysts was considered a program strength. The reliance on only Callaway eddy current data for the site-specific performance demonstration test resulted in limited testing of the ability of analysts to identify stress corrosion cracking and precluded verification of their capability to analyze plus point probe data. The lack of site-specific training for analysts was viewed as an eddy current program weakness (Section E1.4).

- The licensee had performed appropriate testing to demonstrate maintenance of structural and leakage integrity (Section E1.5).

- The licensee performed surveillances of primary eddy current analysis activities at the Westinghouse Waltz Mill facility in the last two refueling outages. The documentation of the surveillances reflected reviews of programmatic subjects such as qualifications, certifications, and testing of analysts. No information was included, however, in regard to any specific actions taken to assess the quality of eddy current analyst performance. Similarly, the Nuclear Procurement Issues Committee audit report that was used by the licensee as a basis for supplier approval was programmatic in approach, with no information provided in regard to assessment of data analysis performance (Section E7.1).

Plant Support

- The licensee has maintained excellent historical program conformance to the analytical limits and actions stipulated in the Electric Power Research Institute secondary water chemistry guidelines. The chemistry initiatives adopted since 1992 were considered reflective of strong management support for actions that may inhibit or reduce steam generator tube degradation (Section R1.1).

- The licensee's monitoring capability for secondary system parameters was good and provided for early indication of secondary water chemistry problems. Procedures were sufficient to ensure that appropriate actions were initiated upon indication of secondary system chemistry problems (Section R2.1).

- The licensee has installed equipment with the capability to detect primary-to-secondary leakage at a very early stage. Procedures incorporated industry guidance and were sufficient to quantify the leakage and respond appropriately. The inspectors concluded that the monitoring system alarm setpoints were sufficiently low to alert operators of increasing primary-to-secondary leakage at an early stage (Section R2.2).

- The data reviewed indicated that the licensee had appropriately responded to identified secondary water chemistry issues and was successful in progressively improving water chemistry quality (Section R3.1).

- Off-normal water chemistry early in plant life was considered a possible contributor to tube degradation. Licensee actions have significantly improved chemistry controls from Cycle 3 onwards (Section R3.2).

- The scope and depth of audits and surveillances of chemistry activities were appropriate, with the review effort documented in Surveillance Report SP96-015 considered commendable (Section R7.1).

Report Details

The primary purpose of this inspection was to establish comprehensive baseline information in regard to: (1) history and material condition of steam generator tubing; (2) the effectiveness of licensee programs in detection and analysis of degraded tubing, repair of defects, and correction of conditions contributing to tube degradation; and (3) the effectiveness of licensee programs and training in regard to detection of and response to steam generator primary-to-secondary tube leakage. In addition to the report details described below, more comprehensive documentation of the inspection was prepared similar to that used for prior inspections of this type. This was done both for consistency and to provide a reference vehicle for steam generator tube integrity history and scope and status of licensee programs. The more comprehensive documentation is contained in Attachment 2 to this inspection report. A review of inservice inspection activities was also conducted during this inspection period, which focused on replacement of a portion of the Loop 4 reactor coolant system excess letdown line.

I. Operations

O3 Operations Procedures and Documentation

O3.1 Inspection Scope (50002)

The inspectors reviewed the emergency operating procedures related to steam generator tube ruptures.

a. Observations and Findings

The emergency operating procedures were found to be consistent with the plant configuration. Guidance was provided to continuously monitor radiation levels and grab sample results. This approach ensures entry into steam generator tube rupture procedures if radiological conditions indicate occurrence of a steam generator tube rupture. Additionally, OTO-BB-00001, the off-normal procedure for steam generator tube leaks, provided a caution that if the leak was of such a magnitude that a safety injection was not required, then the attachment to the off-normal procedure must be completed after exiting the emergency operating procedures. This caution ensured isolation of the leaking steam generator, even if the leak was not large enough to require isolation during performance of the emergency operating procedures.

b. Conclusion

The off-normal and emergency operating procedures provided appropriate guidance for the operators in the event of a steam generator tube rupture.

O5 Operator Training and Qualification

O5.1 Inspection Scope (50002)

The inspectors reviewed the training provided to operators with regard to steam generator tube ruptures.

a. Observations and Findings

The licensee's training program contained, as a required element, training on steam generator tube leaks and ruptures. During the previous 2 years (ten requalification training cycles), in addition to classroom training, simulator training had contained three steam generator tube leak scenarios and eight steam generator tube rupture scenarios. All crews would have been exposed to each scenario during the 2-year cycle. Additionally, the tube rupture training conducted during the 95-4 requalification cycle included personnel from the chemistry, health physics, and rad waste departments in an attempt to provide more realistic, integrated response to the event.

The simulator was out of service for repairs during this inspection, so no observation of simulated plant response could be performed. However, the training staff was in the process of updating the simulator model, which included radiation monitor response. Discussions with training department personnel indicated that the values currently utilized in the simulator for radiation monitor steady state values were based on actual plant information, but as a result of no significant steam generator tube leakage history at the Callaway Plant, the transient information was based on calculated values and data retrieved from other plants.

b. Conclusion

The licensee's training program was comprehensive with respect to steam generator tube ruptures and focused on integrated plant response.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Letdown Branch Line Replacement

a. Inspection Scope (73753)

The inspectors observed activities associated with the repair/replacement of a portion of the reactor coolant Loop 4 excess letdown branch line (Line BB-74-BCA-2"). This line developed a leak in October 1995, as a result of the

development of a crack in an elbow. An interim repair using weld overlay was performed, with the permanent repair accomplished during Refueling Outage RF8 by the witnessed replacement. The piping subassembly used for this replacement was fabricated on site.

b. Observations and Findings

The inspectors reviewed the welding and liquid penetrant examination procedures, and the repair/replacement work packages and associated records. In review of the certified material test reports applicable to the materials used for the piping subassembly, the inspectors noted that the Combustion Engineering certified material test report for the 2-inch pipe (Heat B6844) that was used in the manufacture of the subassembly did not clearly identify whether the material had been manufactured in accordance with applicable ASME Code Section III quality assurance program requirements. Specifically, the document copy that was microfilmed for the licensee permanent records was made in such a manner that the bottom of page 2 of the document was not copied. This error eliminated the majority of the quality assurance program statement that had been typed at this location. The inspectors were able to finally conclude, after comparison of the remaining text with that on other Combustion Engineering certified material test reports, that appropriate origin material appeared to have been used.

Based on observation of welding parameters such as voltage, amperage, travel speed, electrodes, setup, and gas flows, the inspectors found the welding of field Welds 1 and 2 on Work Request W175752 to be well controlled and in accordance with work instructions. The installation welding was performed by a vendor. The inspectors also witnessed liquid penetrant examination of field Welds 1 and 2 and noted that correct materials, durations, and technique were used by licensee nondestructive examination personnel.

c. Conclusion

The inservice inspection program for repair/replacement was adequately implemented.

M1.2 Review of Tube Examination Data

a. Inspection Scope (50002)

The inspectors performed a limited independent assessment of eddy current data which included discrepancies in calls between the primary and secondary data analysts and an evaluation of the signal characteristics of indications confirmed by the resolution analysts.

b. Observations and Findings

The inspectors reviewed the eddy current data for expansion transitions in a total of 29 tubes and ascertained from the sample that a majority of the indications reviewed were detectable only with the plus point coil. It was also observed that voltage measurements for all the indications reported by the primary data analysts were approximately three times larger than those reported by the secondary analysts. These measurements should have been approximately equal, because the voltage calibration procedures were similar between the two parties. The inspectors questioned the licensee as to the source of the voltage anomaly identified during the inspection. After the NRC identification of the anomaly, the licensee contacted Westinghouse who investigated the issue and concluded that the ANSER analysis software contained an apparent error in saving the voltage calibration to the other channels. Because the software error only affected the voltage measurement and not the appearance of a signal on the analysis screen, the licensee indicated that this would not diminish an analyst's capability to detect indications of steam generator tube degradation. The inspectors agreed with the licensee's conclusion. The secondary analysis performed by Framatome-Rockridge used Eddynet95 software and was, thus, not subject to the Anser software error.

The inspectors reviewed rotating probe eddy current data for a sample of tubes where the primary and secondary analysts reported different calls. In such situations, the indication is forwarded to the resolution analysts for final disposition. Based on review of some of the calls forwarded to the resolution analysts, the inspectors noted that the primary analysts called more possible indications than the secondary analysts. However, for the sample of tubes reviewed, these tubes with potential indications of degradation were resolved as containing no detectable degradation following resolution. Based on an independent review of discrepant calls between the primary and secondary data analysts, the inspectors agreed with the final resolution of all the potential tube indications included in the sample.

The inspectors noted two types of indication which appeared to lead to differing calls between analysts. One was when a ferromagnetic signal was present and the other was a type of indication which did not appear to be associated with a known tube degradation mechanism. The inspectors agreed with the resolution of both types of indication. Specific information on these indications is included in Section 3.3 of Attachment 2 to the inspection report.

The inspectors also completed an independent data review of a sample of tubes resolved as containing defects based on the inspection data. Although several of the indications were detectable using all the inspection coils, the inspectors qualitatively concluded that the majority of indications reviewed in the sample

appeared to be small. This was evidenced by the fact that these smaller indications were detectable with the plus point coil, but did not exhibit detectable degradation with the pancake coils. No problems were noted with the calls made on the sample of tubes. Additional information on the sample reviewed is included in Section 3.3 of Attachment 2 to the inspection report.

c. Conclusions

An error was identified by the inspectors in the Anser software used by Westinghouse which resulted in increased voltage measurements for indications. The inspectors agreed with the licensee's conclusion that the error did not affect the analyst's capability to detect indications of steam generator tube degradation. The majority of the indications reviewed were detectable only with the plus point coil which suggested to the inspectors that they were small in size. The resolution process for addressing differences in calls between primary and secondary analysts appeared to have been effectively implemented.

M1.3 Use of Laser Welded Sleeves for Steam Generator Tube Repair

a. Inspection Scope (50002)

Inspection of this activity was restricted to documentation review as a result of sleeve installations being completed at a time when the inspectors were not on site. Documentation reviewed by the inspectors included procedures, work control documents, technical reports, topical reports, and correspondence between the licensee and the NRC related to the NRC's review and approval of the sleeving process.

b. Observations and Findings

The primary document used to control the laser welded sleeving activities was ascertained by the inspectors to be Procedure ETP-BB-01336, "Callaway Unit #1, Laser Welded Sleeving, 0.688" OD x 0.040" Wall Steam Generator Tubing, Model F," Revision 0, dated September 4, 1996. The inspectors confirmed that the Westinghouse developed procedure and field change requests had been reviewed and approved for use at Callaway in accordance with site procedures. The inspectors verified from review of Steam Generator C process control records that: (1) tube cleaning attributes were controlled in accordance with procedural requirements, (2) sleeve insertion height was monitored through periodic hardstop verifications, and (3) system expansion pressures were calibrated and monitored during implementation of the expansion process. The inspectors also confirmed that all laser welds were located above the sludge pile, laser weld locations were consistent with the specified minimum distance from the indications, and that supporting documentation existed for all Steam Generator C sleeves which indicated satisfactory welding and nondestructive examination results.

The inspectors noted from review of the Callaway Technical Specifications that Section 4.4.5.4.a 10) requires that tube repairs be performed by laser welded sleeving as described in Westinghouse Technical Report WCAP-14596-P, "Laser Welded Elevated Tube Sheet Sleeves for Westinghouse Model F Steam Generators," March 1996 (W (Proprietary). The inspectors ascertained that Section 6.3 of WCAP-14596-P states, in part, "The welding parameters, qualified to the rules of the ASME Code, are computer controlled at the weld operator station. The essential variables per Code Case N-395 are monitored and documented for field weld acceptance."

The inspectors reviewed the applicable Westinghouse welding procedure specification (WPS 74376, Revision 0) and supporting procedure qualification records (PQRs 512, 513, 514, 515, 516, and 517) that were used for laser welding of sleeves during Refueling Outage RF8, and evaluated the definition and qualification of the essential variables with respect to the requirements stipulated in ASME Code Sections IX and XI, and ASME Code Case N-395. The inspectors determined that overall conformance to ASME Code requirements was good. One area of concern was noted pertaining to an electrical characteristic essential variable. Code Case N-395 lists as an essential variable a change in the wattage of more than ± 2 percent from that qualified. During review of the sleeve welding records for Refueling Outage RF8, the inspectors noted cases where the documented beam power was outside of a ± 2 percent range from the qualified wattage value. The inspectors were subsequently informed by Westinghouse personnel that the recorded wattage values for the laser welds were obtained from a non-calibrated instrument which would not be expected to provide accurate monitoring of wattage. The inspectors ascertained that a calibrated device was used by the vendor prior to commencement of a series of sleeve welds (and also after mirror and equipment change out) to verify beam power was within the specified wattage range. The vendor did not, however, require that the calibrated device be used, after completing a scheduled series of welds, in order to verify that beam power was still within the permissible ± 2 percent range from that qualified. The inspectors accordingly questioned whether this practice conformed to the requirement in Section 6.3 of WCAP-14596-P to monitor and document Code Case N-395 essential variables.

Additional information on this subject was provided by telephone and facsimile subsequent to the conclusion of the onsite inspection. The facsimile information has been included as Attachment 4 to the inspection report. During a final telephone exit meeting on December 9, 1996, the licensee was informed that the adequacy of monitoring of essential variables during sleeve welding was considered an unresolved item pending receipt of additional information regarding the infra-red feedback process. Attachment 4 contains a December 19, 1996, facsimile that documents the questions posed to the licensee and the responses obtained by the

licensee from Westinghouse. Monitoring and documentation of laser weld essential variables are considered to remain an unresolved item pending review of the Westinghouse supporting information for the December 19, 1996, facsimile, including the infra-red data for the Callaway sleeve welds (50-483/9610-02).

c. Conclusions

The inspectors concluded that, in general, the defined sleeve installation process requirements were consistent with the criteria approved by the NRC and were satisfactorily implemented. An unresolved item was identified, however, in regard to the adequacy of monitoring of laser weld essential variables during field welding of sleeves.

III. Engineering

E1 Conduct of Engineering

E1.1 Licensee Actions to Increase Resistance of Tubing to Stress Corrosion Cracking

a. Inspection Scope (50002)

The inspectors reviewed licensee actions that had been taken to increase the inherent resistance of the Inconel 600 steam generator tubing material to stress corrosion cracking.

b. Observations and Findings

The inspectors were informed by licensee personnel that onsite shot peening (of the tube sheet region through the expansion transition) was performed during Refueling Outage RF5 (April 1992) on the hot-leg side of all steam generator tubes. This activity was performed to induce surface compressive stresses on the tube inside diameter surface and, thus, increase resistance to primary water stress corrosion cracking. The inspectors considered this information in the context that the plant had begun commercial service in December 1984, and had, thus, operated for an extended period prior to performance of shot peening. It appeared to the inspectors that the compressive stresses induced by this activity should increase the resistance to initiation of new primary water stress corrosion cracks. The inspectors considered that the shallow depth of compressively stressed material would, however, have minimal effect in altering stress conditions at the tips of cracks that existed prior to performance of shot peening and, thus, the activity was not expected to delay propagation of cracks that had initiated prior to Refueling Outage RF5.

c. Conclusion

The inspectors concluded that the peening should improve resistance to initiation of new primary water stress corrosion cracks, but would be expected to be of minimal benefit with respect to delaying propagation of cracks that were present at the time of peening.

E1.2 Review of Tube Examination History

a. Inspection Scope (50002)

The inspectors reviewed the historical tube examination scope and methods used with respect to Technical Specification requirements, industry guidance, and as a result of emerging degradation modes. Response to generic communications is addressed in Section E8.

b. Observations and Findings

The inspectors ascertained that full-length bobbin coil examinations were performed in Refueling Outages RF1 through RF6 using sample sizes that always exceeded Technical Specification requirements. The licensee approach examined steam generators every other outage which was permitted by the Technical Specifications. In the April 1987 maintenance outage, full-length bobbin coil examinations were performed using sample sizes that satisfied the current 20 percent sample size recommendation contained in Electric Power Research Institute Document, "PWR Steam Generator Examination Guidelines." Following the initial identification of anti-vibration bar wear in April 1987, the licensee responded by increasing the sample size for full-length bobbin coil examinations to 62 percent in Refueling Outage RF2 and 100 percent in Refueling Outage RF3. A 100-percent bobbin coil sample size has been used by the licensee in subsequent refueling outages.

Limited hot-leg side motorized rotating pancake coil examinations were also performed during Refueling Outage RF3 for the first time at tangent areas and at the top of tube sheet in the sludge pile region (Rows 11-20, Columns 51-75). The licensee restricted the examinations to the two steam generators selected for bobbin coil examination. The motorized rotating pancake coil examinations found no evidence of significant degradation. In Refueling Outages RF4 through RF6, the licensee performed a motorized rotating pancake coil examination of hot-leg side expansion transitions using a sample size of approximately 11 percent of the tubes. The examinations during each of these outages were restricted to the two steam generators selected for bobbin coil examination. No evidence of significant degradation was found during these examinations. The inspectors viewed the licensee actions as an appropriate response to industry notifications and generic communications pertaining to discovery of stress corrosion cracking and intergranular attack at the top of tube sheet.

During Refueling Outage RF7, the initial planned inspection was to examine Steam Generators B and C with the same scope as was used in Refueling Outage RF5. The licensee ultimately expanded the scope of motorized rotating pancake coil examinations to 100 percent of the mill annealed tubes at the top of tube sheet in all steam generators, as a result of the initial detection of an apparent crack in Steam Generator C at this location. This detection was the first identification of apparent stress corrosion cracking in the Callaway steam generators.

In Refueling Outage RF8, the licensee performed a full-length bobbin coil examination of 100 percent of the tubes in Steam Generators A and D. In addition, a plus point coil examination was performed of 100 percent of the expansion transitions on the hot-leg side of each steam generator and the low radius bends in Steam Generator C. The inspectors viewed the adoption of new eddy current technology for examination of all the hot-leg side expansion transitions to be both proactive and indicative of management support for steam generator tube integrity initiatives.

c. Conclusions

The licensee has historically selected sample sizes and eddy current examination methods that are consistent with regulatory requirements and industry guidance, and are an appropriate response to emerging degradation modes.

E1.3 Review of Steam Generator Tube Examination Program Requirements

a. Inspection Scope (50002)

The inspectors compared the steam generator eddy current examination program requirements for Refueling Outage RF8 against regulatory requirements, industry guidelines and qualification criteria, and specific commitments made in response to Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes." The inspectors also observed vendor activities associated with data acquisition.

b. Observations and Findings

The inspectors noted areas of weakness during review of the ETP-BB-01309 data analysis guidelines. Examples noted included an absence of criteria for establishing when eddy data noise was unacceptable and a lack of restrictions on the data analysts with respect to assignment of percent through-wall depth calls from bobbin coil data based on the phase angle response of the eddy current signal. Although there is a theoretical basis for estimating flaw depth from the phase angle of an indication, actual experience has demonstrated that such a practice can result in a significant degree of error in depth measurements. Nevertheless, Section 5.3.5 specified that "[a]ll percent through-wall calls should be recorded with their as measured percent through wall." The inspectors also noted other sections of the data analysis guidelines which addressed the practice of assigning depths to

indications without regard to the mode of degradation. The inspectors considered the absence of program limitations in regard to assignment of through-wall depths from phase angles of indications to have the potential for potentially defective tubes being classified as degraded during an inspection.

Another area of concern noted by the inspectors during review of ETP-BB-01309 pertained to the lack of explicit guidance for disposition of indications that were perceived by analysts from review of bobbin coil data to be probably manufacturer's burnish marks. Because the bobbin coil probe cannot adequately differentiate between stress corrosion cracking and volumetric indications such as manufacturer's burnish marks, additional inspections and/or a review of prior inspection data is necessary prior to disposition of such signals. Although the discussion in Attachment 4 of ETP-BB-01309 briefly mentioned performing historical data reviews or additional inspections for the disposition of manufacturer's burnish mark-type indications, the inspectors noted that the analysis guidelines did not specifically require such actions.

The inspectors reviewed the acquisition technique sheets for both the Callaway tube examinations and the qualification testing, and verified that the acquisition system used at Callaway was qualified in accordance with Electric Power Research Institute (EPRI) NP-6201, "PWR Steam Generator Examination Guidelines." (Note: This document will be subsequently referred to in the inspection report as "EPRI Guidelines").

The inspectors noted an inconsistency between the analysis guidelines and the acquisition technique sheets for the plus point qualification testing. Specific directions for establishing the span and rotational (phase) settings for analysis of plus point probe data were not included in the ETP-BB-01309 data analysis guidelines. Section 5.1.3.6 of the ETP-BB-01309 data analysis guidelines did provide span and rotational instructions for analysis of rotating pancake coil data. These directions differed, however, from the settings utilized during the qualification of the plus point probe. ETP-BB-01309 required span settings for the rotating pancake coil be set such that a 40 percent notch is detectable, and rotation adjusted so that probe motion is horizontal. The Appendix H plus point probe qualification (Westinghouse Report DDM-96-009, "Documentation of Appendix H Compliance and Equivalency,") was performed; however, with the 40 percent notch response set at one-half screen height, and rotation set to 20 degrees for a 100 percent axial through-wall notch.

In that the span and phase settings for each channel are of primary importance with regard to the ability of an analyst to detect indications of tube degradation, the inspectors questioned the licensee as to how the primary and secondary analysts (i.e., Westinghouse and Framatome-Rockridge, respectively) actually established these settings during the inspection. Westinghouse personnel subsequently identified that specific direction was not provided to the primary analysts for the settings to be used in analysis of plus point data. The vendor further identified

that a review had been performed of data from the inspection which indicated that plus point data analysis settings had been established based on current industry practice. However, no information was provided that would indicate the settings were established in accordance with appropriate documented instructions. The inspectors concluded that the settings used during primary analysis could have varied between different data analysts. The inspectors additionally ascertained that the lead analyst for secondary data analysis had outlined the settings to be used by the secondary data analysts prior to the Callaway steam generator tube examinations. The inspectors reviewed these instructions and concluded that the set up would provide an adequate level of detection capability for the data analysis software. The contract criteria established by the licensee for the secondary analysis did not require, however, vendor submittal of the set up instructions to the licensee for approval (i.e., the contract was placed as nonsafety related for secondary analysis). The failure to establish requirements to assure use of Appendix H qualified span and phase rotation settings for analysis of plus point coil data is a violation of Criterion IX of Appendix B to 10 CFR Part 50 (50-483/9610-01).

During the course of the tube examinations, the inspectors observed activities involving data acquisition. Copies of data acquisition guidelines, ETP-22-01300, "Eddy Current Inspection of Preservice and Inservice Heat Exchanger Tubing," Revision 8, were observed by the inspectors to be located at the acquisition stations outside containment, and discussions with the acquisition personnel indicated they were familiar with the content of the procedures. The inspectors observed the vendor's process for replacing eddy current probes during the inspection, in order to verify that proper controls were established to ensure use of the correct inspection probes during examinations. The vendor was ascertained to have established a process which included two independent verifications of the probe type prior to installing the probe on the probe pusher outside the steam generator. The inspectors noted no problems with the verification methodology and found the controls to be effective.

c. Conclusions

Areas of weakness were noted in the licensee data analysis guidelines which included absence of criteria for establishing when eddy current data noise was unacceptable, a lack of restrictions on assignment of through-wall depths from bobbin coil data, and absence of specific requirements for disposition of potential manufacturer's burnish mark indications. A violation was identified with respect to the failure to establish and implement appropriate documented instructions for calibration of plus point probes. The controls used to ensure use of correct inspection probes during acquisition of eddy current examination data were found to be effective.

E1.4 Requirements for Training and Testing of Data Analysts

a. Inspection Scope (50002)

The inspectors reviewed the training and testing requirements for data analysts that were established for Refueling Outage RF8.

b. Observations and Findings

The inspectors ascertained that the data analysis guidelines required that data analysts be certified as Qualified Data Analysts in accordance with Appendix G of the EPRI Guidelines, be certified to Level IIA in accordance with American Society of Nondestructive Testing Recommended Practice SNT-TC-1A-1984, and must successfully complete a site-specific performance demonstration. The inspectors considered the requirement for analysts to be certified as Qualified Data Analysts to be a program strength. The inspectors confirmed that several Refueling Outage RF8 analysts had been certified as Qualified Data Analysts and reviewed documentation to verify that all analysts had successfully passed the site-specific performance demonstration test.

As part of the assessment of analyst testing, the inspectors reviewed the set up and content of the site-specific performance demonstration test. The data base of indications used for the test was developed using previous Callaway bobbin coil and rotating pancake coil probe inspection data. As a result, it did not include data obtained using plus point probes, the rotating probe used during the Refueling Outage RF8 examinations and, thus, did not test analysts on their capability to analyze data obtained with this probe. In addition, the inspectors observed that the indications data base included only a small number of indications of stress corrosion cracking relative to the overall number of defect indications. The inspectors ascertained that random selection from this type of data base had resulted in some analysts passing the test while being required to analyze only a small number of stress corrosion cracking indications.

The inspectors ascertained that the licensee had not required or offered structured training to the eddy current data analysts in regard to the data analysis guidelines and the potential modes of degradation that could be anticipated during the Refueling Outage RF8 tube examinations. The inspectors viewed this lack of formal site-specific analyst training as an eddy current program weakness.

c. Conclusions

The programmatic requirement for analysts to be certified as Qualified Data Analysts was considered a program strength. The reliance on only Callaway eddy current data for the site-specific performance demonstration test resulted in limited testing of the ability of analysts to identify stress corrosion cracking and precluded verification of their capability to analyze plus point probe data. The lack of site-specific training for analysts was viewed as an eddy current program weakness.

E1.5 In-Situ Pressure Testing of Steam Generator Tubes

a. Inspection Scope

The inspectors reviewed the results of the in-situ pressure tests and the tube selection criteria used to demonstrate compliance with leakage limits and Regulatory Guide 1.121 structural integrity requirements.

b. Observations and Findings

The inspectors noted from review of the in-situ pressure test data that none of the tubes burst during the in-situ pressure tests, thus, meeting structural integrity requirements. Two tubes leaked at approximately 0.36 gallons per day (gpd) when tested at a pressure differential of 3100 psi. This leakrate was not adjusted for temperature, which would result in an even lower leakrate. Based on the limited leakage seen in only two of the most significant indications, the licensee determined that leakage under postulated accident conditions would have been within allowable limits. The inspectors reviewed the eddy current data reports for all indications identified during the current outage and determined that the ten tubes selected for in-situ testing met the licensee's tube selection criteria.

c. Conclusions

The licensee had performed appropriate testing to demonstrate maintenance of structural and leakage integrity.

E2 Engineering Support of Facilities and Equipment

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Safety Analysis Report description highlighted the need for a special focused review that compares plant practices, procedures, and/or parameters to the Updated Safety Analysis Report description. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the Updated Safety Analysis Report that related to the areas inspected.

One observation made by the inspectors was that the main steam line N-16 monitoring system, which was made functional October 10, 1996, was not addressed in the Updated Safety Analysis Report. The licensee informed the inspectors that, although the system was functional, not all applicable paperwork associated with the design modification package had been closed out and reviewed. When questioned by the inspectors as to whether the Updated Safety Analysis Report would be updated to include the N-16 monitoring system, the licensee indicated that they had not planned to do so and did not believe that it was necessary. They also indicated that they would review that decision in subsequent updates to the Updated Safety Analysis Report.

E7 Quality Assurance in Engineering Activities

E7.1 Eddy Current Program Oversight

a. Inspection Scope (50002)

The inspectors reviewed recent records pertaining to licensee oversight of eddy current examination activities.

b. Observations and Findings

The inspectors requested to see available records pertaining to licensee oversight of eddy current activities. Licensee personnel provided three quality assurance surveillance reports (SP95-036, SG96-06, and SP96-015) in response to this request. Surveillance Reports SP95-036 and SG96-06 documented licensee surveillances of Westinghouse eddy current examination activities at its Waltz Mill facility near Pittsburgh, Pennsylvania, during Refueling Outages RF7 and RF8. Surveillance Report SP96-015 addressed conformance of the steam generator examination program and steam generator chemistry controls to EPRI guidelines.

The inspectors ascertained, from review of Surveillance Report SP95-036, that the text focused primarily on the qualification and certification of data analysts that were used by Westinghouse for the Refueling Outage RF7 examinations. It was pointed out, however, that a documented training plan was not used by Westinghouse and data interpretation and reviews (that were stated by Westinghouse personnel to have been performed as part of the training activity for analysts) were not obvious from the training records. No specific information, other than a statement that activities were found to be acceptable, was included in the report in regard to analyst performance during evaluation of Refueling Outage RF7 data. Surveillance Report SG96-06 documented review of the translation of Callaway contractual requirements into Westinghouse documents, the site-specific performance demonstration testing that was conducted at Waltz Mill during Refueling Outage RF8, the qualification records for subcontractors that were used by Westinghouse for data analysis, the most recent internal audit of the Westinghouse eddy current examination program, and observation of turnovers

between shift crews. Other than a summary statement that Westinghouse appeared to be performing data analysis in an acceptable fashion, no specific information was included in regard to assessment of data analyst performance during review of Refueling Outage RF8 examination data.

Surveillance Report SP96-015, which was issued on March 12, 1996, was noted by the inspectors to comprehensively address the conformance of steam generator examination program and chemistry controls to the respective EPRI guideline documents. The inspectors observed that the surveillance report discussed two issues which involved organizational knowledge of eddy current examination technology. The auditors identified that there was no current documented policy regarding personnel training needs in eddy current examination technology and offered a view that certification to SNT-TC-1A, Level IIA would be beneficial. The auditors also recommended management consideration, as a result of their identification that EPRI guideline items were not included in ETP-BB-01309, that to address some of the items may require the expertise of a Level III eddy current technician. SNT-TC-1A eddy current certifications were not currently held by licensee staff. Based on the program observations made during the inspection, the inspectors considered the auditors' views pertinent in the context of recognition that there was a current lack of organizational knowledge in the area of eddy current examination technology.

The inspectors requested information from licensee quality assurance staff regarding the method used to support the current approval of the Westinghouse Waltz Mill Service Center as a supplier. In response, licensee personnel furnished Nuclear Procurement Issues Committee Audit 6-94-11. A review was performed of the special processes section of the audit checklist in order to ascertain the type of assessment that was performed in this area by the multi-utility audit team. The inspectors found that the section was strictly programmatic in approach, with reviews performed of personnel certifications, the SNT-TC-1A written practice, and procedures pertinent to training, qualification, and certification. No information was provided regarding review of data analysis performance.

c. Conclusions

The licensee performed surveillances of primary eddy current analysis activities at the Westinghouse Waltz Mill facility in the last two refueling outages. The documentation of the surveillances reflected reviews of programmatic subjects such as qualifications, certifications, and testing of analysts. No information was included, however, in regard to any specific actions taken to assess the quality of eddy current analyst performance. Similarly, the Nuclear Procurement Issues Committee audit report that was used by the licensee as a basis for supplier approval was programmatic in approach, with no information provided in regard to assessment of data analysis performance.

E8 Miscellaneous Engineering Issues

The inspectors completed a limited review of licensee actions taken in response to generic communications issued pertaining to steam generator tube degradation. The scope of the review included an assessment of licensee actions taken concerning Information Notice 94-88, "Inservice Inspection Deficiencies Result in Severely Degraded Steam Generator Tubes," and Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes." As part of the review, the inspectors verified that the licensee implemented the proposed actions in its steam generator inservice inspection program requirements for Refueling Outage RF8.

The inspectors noted, with respect to Information Notice 94-88, that the licensee evaluation did not appropriately address all of the issues (see Section 3.2.3 in Attachment 2 for further details). The licensee has, however, since taken a proactive approach toward addressing potential circumferential cracking of the Callaway steam generator tubes, as evidenced by its commitments made in response to Generic Letter 95-03 (i.e., examination of expansion transitions on the hot-leg side using a technique qualified in accordance with Appendix H of the EPRI Guidelines and use of Qualified Data Analysts) and selection of the plus point coil for examination of the expansion transitions. This approach indicated to the inspectors the licensee's commitment to address circumferential cracking at the earliest possible stages.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Secondary Water Chemistry Controls

a. Inspection Scope (50002)

The inspectors performed a review of the evolution of the licensee's secondary water chemistry control program and its historical conformance to the EPRI secondary water chemistry guidelines.

b. Observations and Findings

The inspectors were informed that the early historical chemistry program requirements were contained in Procedure APA-ZZ-00630, "Secondary Chemistry Program," Revisions 0 through 4. This procedure was superseded on January 23, 1987, by Procedure APA-ZZ-01021, "Secondary Chemistry Program," Revision 0. The current revision of the procedure, Revision 9, was approved on January 6, 1994. The inspectors compared the evolution of program requirements against the criteria contained in the EPRI, "PWR Secondary Water Chemistry Guidelines." These guidelines were initially issued as EPRI NP-2704-SR in October 1982, with a different document number assigned for each issued revision (i.e., Revision 1,

EPRI NP-5056-SR; Revision 2, EPRI NP-6239; and Revision 3, EPRI TR-102134). The inspectors found from the review that the licensee had been responsive to the industry secondary water chemistry initiatives, with chemistry program requirements consistently revised in a timely manner to incorporate changes in the guidelines.

Other licensee secondary water chemistry initiatives that were considered of note by the inspectors included the start in 1992 of the use of elevated hydrazine levels (> 100 ppb) to promote the most favorable electrochemical potential in the steam generators; the installation in 1993 of a Micromax Display and Control System for computer monitoring and trending of chemistry parameters; the implementation in 1993 of the use of the alternative amine, ethanolamine, to reduce iron transport to the steam generators; and the initiation in 1995 of titanium dioxide additions for their potential value as an inhibitor of stress corrosion cracking and intergranular attack.

c. Conclusions

The licensee has maintained excellent historical program conformance to the analytical limits and actions stipulated in the EPRI secondary water chemistry guidelines. The chemistry initiatives adopted since 1992 were considered reflective of strong management support for actions that may inhibit or reduce steam generator tube degradation.

R2 Status of Radiological Protection and Chemistry Facilities and Equipment

R2.1 Secondary Chemistry

a. Inspection Scope (50002)

The inspectors toured the secondary chemistry laboratory, reviewed the capabilities for on-line monitoring of secondary water chemistry, the procedures for directing action following detection of a secondary chemistry anomaly, the capability to track out-of-specification chemistry conditions, and the general knowledge of personnel regarding detection and repair of condenser tube leaks.

b. Observations and Findings

The licensee utilized an on-line monitor (MicroMax) for continuous monitoring of chemistry parameters in the condenser hotwells, heater drains, condensate pump discharge, condensate polisher effluent, low pressure feedwater heater influent, main feedwater pump discharge, and steam generators. Parameters monitored included pH, sodium, cation conductivity, total conductivity, and oxygen. The equipment appeared to be in good physical condition.

Chemistry results were manually entered into a separate data base that contained information dating back to 1985. Out-of-specification values were highlighted for easy retrieval and recognition. Chemistry values that exceed Technical Specification or NPDES limits were required to be entered into the licensee's corrective action system by Procedure CDP-ZZ-00200, "Chemistry Schedule and Water Specs."

The alarm setpoints for sodium concentration were ascertained by the inspectors to be 0.3 ppb for the condensate pump discharge, and 1.0 ppb for the condenser hotwells. It, thus, appeared to the inspectors that the monitoring system would provide early indication of a condenser tube leak, with the first observable increase in sodium observed in either the condenser hotwell (depending on the MicroMax sample sequence) or condensate pump discharge (continuously monitored). Procedure CDP-ZZ-00200, "Chemistry Schedule and Water Specs," provided the limits and specifications for operation and action levels. Procedure APA-ZZ-01021, "Secondary Chemistry Program," provided guidance for actions based on exceeding chemistry action level limits.

The inspectors held discussions with a secondary chemistry supervisor to ascertain the individual's familiarity with condenser tube leak indications, procedures to be utilized, and the actions to be taken in response including general methodology for the tube repair. The inspectors concluded that the supervisor was appropriately conversant with identification and response actions.

c. Conclusions

The licensee's monitoring capability for secondary system parameters was good and provided for early indication of secondary water chemistry problems. Procedures were sufficient to ensure that appropriate actions were initiated upon indication of secondary system chemistry problems.

R2.2 Primary-to-Secondary Leakage Monitoring

a. Inspection Scope (50002)

The inspectors reviewed the effectiveness of the licensee's procedures, equipment, and practices for monitoring and responding to primary-to-secondary leakage. In particular, the inspectors reviewed the capability of monitoring systems to provide early detection of primary-to-secondary leakage and the rationale for the monitor setpoints utilized for detection and mitigation of primary-to-secondary leakage.

b. Observations and Findings

The inspectors reviewed the installed radiation monitors which could be utilized to alert operators to a steam generator tube leak or rupture. Additionally, the inspectors reviewed the procedures associated with determining the primary-to-secondary leak rate and the guidance contained in Electrical Power Research Institute Report TR-104788, "Primary-to-Secondary Leak Guidelines," dated May 1995.

By the end of Cycle 8 in mid-October 1996, indicated primary-to-secondary leakage was on the order of approximately 1 gpd, based on analysis of grab samples. The licensee was trending the output of all radiation monitors as well as the results of grab samples, and a very slight positive trend was observable on secondary activity that the licensee believed was due to steam generator tube leakage. In addition to routine grab samples, the licensee utilized radiation monitoring of the condenser off-gas, steam generator continuous blowdown, and main steam line flow as the primary monitors that would provide early indication of steam generator tube leakage. The addition of the main steam line N-16 monitoring was a significant enhancement in the ability to provide early indication of a primary-to-secondary leak. The radiation monitor trends were printed out daily as a routine shift duty and maintained in the count room.

The procedures regarding identification of, and response to, steam generator tube leakage were reviewed and found to be good. The equations used to quantify leakage based on grab sample results were found to be the same as those in industry guidelines (EPRI TR-104788) and had been appropriately incorporated into the procedures. Sufficient guidance was provided to chemistry and operations personnel to ensure that appropriate actions were taken to identify, quantify, and mitigate the consequences of a steam generator tube leak. The guidance provided by TR-104788 was found to be appropriately incorporated into the procedures. The inspectors noted one deficiency in Procedure OTO-BB-00001, "Steam Generator Tube Leak," Revision 5, in that Attachment 3, which provided guidance on how to isolate a leaking steam generator, did not direct the operators to close or verify closed the steam generator low point drains, as did the emergency operating procedures. A procedure revision was subsequently initiated to include the step in Attachment 3.

The setpoints for the condenser off-gas monitor and the steam generator blowdown monitors were established at 1 decade above the minimum detectable activity for the "Alert" level and 2 decades above the minimum detectable activity for the "High" alarm. Based on the values actually recorded and the results of grab samples, the values utilized should provide for early detection of steam generator tube leakage. The main steam line N-16 monitors had the design capability to detect a 1 gpd leak when utilized above approximately 40 percent power, but based on the observations of increased secondary activity made at the end of Cycle 8, the licensee believes that useful information can be gained from the system at leakage

rates of less than 1 gpd depending on the location of the leak in the tube. The alarm setpoint for the N-16 monitors was 5 gpd which was also the entry point for Procedure APA-ZZ-01023, "Steam Generator Tube Leak Contingency Guidelines." The annunciator response procedure, OTA-SP-RM011, for the radiation monitoring system appropriately refers operators to Off-Normal Procedure OTO-BB-0001, "Steam Generator Tube Leak," upon receipt of one of the subject alarms and subsequent confirmation of elevated activity in the secondary system.

c. Conclusions

The licensee has installed equipment with the capability to detect primary-to-secondary leakage at a very early stage. Procedures incorporated industry guidance and were sufficient to quantify the leakage and respond appropriately. The inspectors concluded that the monitoring system alarm setpoints were sufficiently low to alert operators of increasing primary-to-secondary leakage at an early stage.

R3 Radiological Protection and Chemistry Procedures and Documentation

R3.1 Secondary Side Chemistry History

a. Inspection Scope (50002)

The inspectors reviewed available chemistry history for the Callaway steam generators with respect to conformance to the EPRI secondary water chemistry guidelines and chemistry initiatives.

b. Observations and Findings

The inspectors noted that a high mean blowdown sulfate level (i.e., 30 ppb) occurred during Cycle 1, which was attributed by the licensee to an organic sulfate problem. The problem was initially addressed in November 1985, by changing the types of resin used in the condensate polishers. Review of Cycle 2 data showed, however, that monthly mean blowdown sulfate values continued to range from 20 ppb to 30 ppb between April and August 1986. After replacement of polisher cation resin, a significant ongoing improvement in monthly blowdown mean sulfate concentration values was noted by the inspectors. High mean condensate dissolved oxygen values also occurred in Cycles 1 and 2 (i.e, 10 ppb and 8.5 ppb, respectively). To minimize oxygen levels, the licensee installed a nitrogen sparger system in 1988 on the condensate storage tank and, in 1989, installed permanent piping and flowmeters for nitrogen sparging of individual condenser hotwells. The mean condensate dissolved oxygen value declined to 5 ppb in Cycle 3 as a result of the modifications and has typically been 3 ppb since Cycle 5.

The licensee installed corrosion transport samplers in 1990 during Cycle 4 to provide monitoring and trending capability of iron and copper contents of feedwater. The mean iron content of the feedwater was found from the initial Cycle 4 data to be relatively high (i.e., 14.8 ppb). Utilization of morpholine as an alternative amine during Cycle 5 was successful in significantly reducing iron transport (i.e., 6.8 ppb mean value), but concurrently increased copper transport to the steam generators from a mean value of 0.07 ppb to 0.32 ppb. Trials with ethanolamine as an alternative amine during Cycle 6 were successful in reducing iron transport without an accompanying increase in copper transport to the steam generators. The licensee accordingly replaced ammonium hydroxide with ethanolamine for pH control in September 1993. Iron transport reduced to a mean value of 3.2 ppb following conversion to ethanolamine and has subsequently remained below the current 5 ppb EPRI limit. The inspectors noted that, with the adoption of ethanolamine as an alternative amine, the licensee was successful in significantly reducing both iron and copper transport during Cycles 6, 7, and 8.

c. Conclusions

Overall, the inspectors considered that the data reviewed indicated that the licensee had appropriately responded to identified secondary water chemistry issues and was successful in progressively improving water chemistry quality.

R3.2 Off-Normal Secondary Chemistry History

a. Inspection Scope (50002)

The inspectors reviewed the off-normal secondary water chemistry history for commercial operation in order to ascertain whether chemistry conditions had occurred which could contribute to steam generator tube degradation. The source document used for the review was the history contained in Surveillance Report SP96-015.

b. Observations and Findings

The chemistry history results indicated overall good chemistry performance subsequent to Cycle 2, with no significant out-of-specification conditions noted that would be expected to contribute to tube degradation.

The total hours during Cycles 1 and 2, when water chemistry parameters exceeded current EPRI secondary water chemistry guideline, Action Level 1 values were as follows: Cycle 1 - sodium-1188, sulfate-3938, chloride-268, cation conductivity-1809; Cycle 2 - sodium-33, sulfate-931, chloride-26, cation conductivity-496. The results showed that the number of hours was significant, particularly in Cycle 1. The inspectors ascertained from review of Surveillance Report SP96-015 that the tabulated Action Level 1 hours also included time periods when blowdown chemistry values exceeded current Action Level 2 and 3 values.

For sodium, the hours included, respectively, 70 hours and 14 hours above the Action Level 2 value during Cycles 1 and 2. Eight of the 14 hours in Cycle 2 that were above Action Level 2 pertained to time above the Action Level 3 limit. The auditors ascertained during their review of history that the plant did not follow EPRI secondary water guideline recommendations when significant off-normal sodium conditions occurred (i.e., power was typically held for Action Level 2 values versus the recommended power reduction, and power was reduced to 30 percent for the Action Level 3 occurrence versus the recommended shutdown). With respect to cation conductivity, the Action Level 2 value was exceeded for a total of 154 hours in Cycle 1 and 56 hours in Cycle 2. The Cycle 2 hours included 4 hours when the Action Level 3 value was exceeded. The plant actions were the same as noted for sodium, which was assumed by the inspectors to be because the elevation of sodium and cation conductivity occurred concurrently. This assumption was not specifically validated by the inspectors.

Level 2 chloride limits were exceeded for 3 hours during Cycle 2, with the highest level reached being 272 ppb. In this case, the licensee did reduce power. The sulfate hours included 186 hours in Cycle 1 and 59 hours in Cycle 2 above the current Action Level 2 value. An average sulfate concentration of 326 ppb occurred during the Cycle 2 time above Action Level 2, with a peak concentration recorded of 898 ppb. It should be noted that Action Level values were not established by the EPRI for sulfate in the time frame of Cycles 1 and 2. The absence of specific program requirements, thus, provides an explanation of why the accrued hours reached the levels noted. The inspectors considered the sulfate information from Cycles 1 and 2 to be germane because of the known causal effects of sulfates with respect to stress corrosion cracking and intergranular attack.

c. Conclusions

Overall, the inspectors considered the early off-normal water chemistry as a possible contributor to tube degradation. Licensee actions have significantly improved chemistry controls from Cycle 3 onwards.

R7 Quality Assurance in Radiological Protection and Chemistry Activities

R7.1 Self-Assessment of Secondary Water Chemistry

a. Inspection Scope (50002)

The inspectors reviewed licensee audit and surveillance reports pertaining to the secondary water chemistry program which covered a period from 1992 to 1996, in order to assess both the effectiveness of oversight activities and whether any conditions were identified that could be expected to contribute to steam generator tube degradation.

b. Observations and Findings

In general, the inspectors found the scope of the audits and surveillances to be appropriate for evaluation of chemistry performance. No specific findings were noted, with the exception of the observations made in Surveillance SP96-015, that the inspectors considered could contribute to steam generator tube degradation. The inspectors found Surveillance Report SP96-015 to be of particular value in assessment of historical secondary water chemistry historical performance, as a result of the extensive reviews of chemistry data that were performed. The comprehensive evaluation of steam generator programs that was accomplished in this surveillance was considered by the inspectors to be a commendable effort.

c. Conclusions

The scope and depth of audits and surveillances of chemistry activities were appropriate, with the review effort documented in Surveillance Report SP96-015 considered commendable.

R8 Miscellaneous Radiological Protection and Chemistry Issues

The inspectors reviewed the licensee's evaluation of Bulletin 88-02; Information Notices 88-99, 91-43, 93-56, and 94-43; and Significant Operating Experience Report 93-1. The evaluations were found to adequately address the issues identified in the subject generic communication. There was no record of how one recommendation was dispositioned in the evaluation of Information Notice 91-43. The recommendation was to review the alarm setpoints for the radiation monitors utilized for primary-to-secondary leak detection to ensure they were set at the appropriate value. A review of the alarm setpoints was, however, performed as part of the Significant Operating Experience Report 93-1 evaluation and response.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on October 25 and November 22, 1996, at the conclusion of two onsite inspection weeks. Following additional in-office inspection, a final telephonic exit meeting was held with your staff on December 9, 1996. The licensee provided additional information to the NRC on two areas of concern on November 26, December 17, and December 19, 1996, which have been included as Attachments 3 and 4 to the inspection report. The licensee acknowledged the findings presented. Westinghouse documents pertaining to laser welded sleeves were reviewed during the inspection which were identified as containing proprietary information. No information was included in the inspection report that was considered proprietary.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Affolter, Manager, Callaway Plant
F. Bianco, Secondary Chemistry Supervisor
H. Bono, Supervising Engineer, Licensing Fuels and Site Licensing
J. Dampf, Training Supervisor
T. DeVicentis, Design Engineer
M. Evans, Superintendent, Health Physics
F. Forck, Quality Assurance Scientist
G. Hamilton, Supervising Engineer, Quality Assurance
T. Hermann, Supervising Engineer, NSSS and HVAC Design
D. Hollabaugh, Supervising Engineer, Technical Support
B. Huhmann, Design Engineer
L. Kanuckel, Supervising Engineer, Quality Assurance
J. Kerrigan, Count Room Supervisor
D. Martin, Quality Assurance Engineer
C. Naslund, Manager, Nuclear Engineering
B. Newton, Welding Engineer
E. Olson, Supervisor, Chemistry
T. Pettus, Steam Generator Engineer
G. Randolph, Vice President, Nuclear Operations
M. Reidmeyer, Engineer, Quality Assurance
M. Rice, Inservice Inspection Engineer
R. Roselius, Superintendent, Chemistry and Radwaste
D. Schnell, Senior Vice President, Nuclear
G. Schultz, Chemical Engineer
E. Stewart, Training Supervisor
E. Thornton, Quality Assurance Engineering Evaluator
W. Witt, Superintendent, Technical Support Engineering

NRC

F. Brush, Resident Inspector
D. Passehl, Senior Resident Inspector

INSPECTION PROCEDURES USED

IP 50002	Steam Generators
IP 73753	Inservice Inspection

ITEMS OPENED

Opened

50-483/9610-01	VIO	Failure to Follow Procedure (Section E1.X)
50-483/9610-02	URI	Monitoring of power used in laser welding of sleeves in steam generator tubes

LIST OF DOCUMENTS REVIEWED

Emergency Operating Procedures

E-0	Reactor Trip or Safety Injection	Revision 1B2
E-3	Steam Generator Tube Rupture	Revision 1B2
ES-0.1	Reactor Trip Response	Revision 1B1
ES-1.1	SI Termination	Revision 1B2

Other Procedures/Documents

OTO-BB-00001, "Steam Generator Tube Leak," Revision 5

OTA-SP-RM011, "Alarm Response Procedure Radiation Monitor Control Panel RM-11," Revision 14

APA-ZZ-01023, "Steam Generator Tube Leak Contingency Guidelines," Revision 1

CDP-ZZ-00200, "Chemistry Schedule and Water Specs," Revision 51

CTP-ZZ-02590, "Steam Generator Tube Leak Rate Determination," Revision 12

HPCI-84-03, "Health Physics Department Calculation Sheet," Revision 0

Licensee Reviews of NRC Bulletin 88-02 and NRC Information Notices 88-99, 91-43, 93-56, 94-43, and Significant Operating Experience Report 93-01

EPRI TR-104788, "PWR Primary-to-Secondary Leak Guidelines," May 1995

APA-ZZ-00662, "ASME Section XI Repair/Replacement Program," Revision 9

ASME Section XI Repair/Replacement Plan for Work Request W175752

Work Requests W175752, A175752A, and A175752B

MTW-BB-WP001, "Welding Services Inc. Procedures WPS-A0820 and WPS-M8d," Revision 0

MTW-ZZ-WP514, "Welding of P-8 Materials," Revision 10

QCP-ZZ-05000, "Liquid Penetrant Examination," Revision 9 with Temporary Change Notice 95-0789

Welding Services Inc., Welder Performance Qualification Reports for Welders DGA2658, KJB7575, WWC7520, RDH3010, CAP2339, RP4532, and WWS6714

Welder Qualification Reports for Welders BFI 24002 and MRM 23756

Letter UEL-96-004, Reedy Associates to Union Electric, April 4, 1996

Design Input, "Leak on BB-74-BCA-2," Revision 0

Report SOS 95-1891

Letter SCP-96-126, Westinghouse to Union Electric, June 30, 1996

Purchase Order 94052, August 6, 1996

Request for Resolution 16737, "Provide Options for RF8 Excess Letdown Replacement," Revision A

Certified Material Test Reports for Heats B6844, 04930 and 692882

Letter NSD-TAP-3150, Westinghouse to Union Electric, "Callaway SG Recommendations on Tubes for In-Situ Testing," dated November 15, 1996.

Letter, Union Electric to NRC, "Request for License Amendment on Laser Welding," dated April 12, 1996

Letters, Union Electric to NRC dated August 2 and 19, 1996, and September 5, 1996, regarding additional information on the laser welded sleeving process relevant to the NRC's ongoing License Amendment Request review

Memorandum, Westinghouse to Union Electric, "Amplitude Discrepancy Between ANSWER & Eddynet95," dated November 19, 1996

Union Electric Procedure APA-ZZ-00500, "Corrective Action Program," Revision 27, August 2, 1996

ETP-BB-01336, "Callaway Unit #1 LWS, 0.688" OD x 0.040" Wall SG Tubing Model 'F'," Revision 0, dated September 4, 1996

Memorandum, Westinghouse to Union Electric, "Sound Material Requirements for Performing Laser Welds & Lower Hardroll Process on 12" Elevated Tubesheet Sleeves for Callaway Unit 1"

ATTACHMENT 2

STEAM GENERATOR TUBE INTEGRITY REVIEW (50002)

1 OBJECTIVES

The objectives of this inspection were: (a) to ascertain the history and material condition of steam generator tubing including, when applicable, the type, magnitude, and location(s) of active degradation mechanisms; (b) to assess the effectiveness of licensee programs in detection and analysis of degraded tubing, repair of defects, and correction of conditions contributing to tube degradation; and (c) to assess the effectiveness of licensee programs and training in regard to detection of, and response to, steam generator primary-to-secondary tube leakage.

2 STEAM GENERATOR FABRICATION AND DEGRADATION HISTORY

2.1 Steam Generator Description

Callaway Plant is a Westinghouse-designed 1236 MWe pressurized water reactor which commenced commercial operation on December 19, 1984. A standardized design approach was used for the nuclear block at Callaway Plant (and also Wolf Creek Generating Station), which has been named the Standardized Nuclear Unit Power Plant System. The Standardized Nuclear Unit Power Plant System unit design utilizes four Westinghouse Model F vertical recirculating steam generators. This model of steam generator contains 5626 Inconel 600 (ASME Material Specification SB-163) U-tubes, with a nominal diameter and wall thickness, respectively, of 0.688 inches and 0.040 inches. The ends of the tubes are inserted into drilled holes in a tube sheet forging, followed by full-length hydraulic expansion of the tubes against the hole surfaces and welding to compatible composition weld cladding on the primary side surface of the tube sheet. Secondary side tube support structures consist, in sequential order from the tube sheet, of a horizontal flow distribution baffle, seven horizontal tube support plates, and three sets of Inconel 600 anti-vibration bars in the upper tube bundle. The flow distribution baffle and tube support plates are fabricated from Type 405 ferritic stainless steel, with a quatrefoil tube hole configuration used in the tube support plates and drilled holes in the flow distribution baffle. The inspectors considered that the selection of a ferritic stainless steel for the tube supports and the use of a quatrefoil hole configuration should minimize long-term tube denting at these locations, due to the avoidance of magnetite formation and more limited entrainment of corrosion products in the interstices between the tubes and supports.

2.2 Hot-Leg Temperature

The inspectors noted during review of the licensee response to Generic Letter 95-03 that the licensee had included T_{Hot} data for the individual steam generators which exhibited a 4°F range (i.e., Steam Generator A, 616.6°F; Steam Generator B, 613.8°F; Steam Generator C, 617.8°F; and Steam Generator D, 615.5°F). These temperature values appeared to correlate with the current incidence of degradation in the individual steam generators (i.e., Table 2 shows the incidence of inservice repairs increased with increasing steam generator inlet temperature). Later licensee data (i.e., November 18, 1996) showed the individual T_{Hot} values for Steam Generators A, B, C, and D were, respectively, 618.3°F,

615.6°F, 617.5°F, and 616.5°F. Although Steam Generator A now showed a slightly higher T_{Hot} value than Steam Generator C, the inspectors considered the data was continuing to exhibit a good correlation between individual steam generator T_{Hot} value and incidence of degradation.

2.3 Tubing Material

The inspectors were informed by licensee personnel that an additional thermal treatment (to the annealing cycle that the finished tubing received) was performed on the formed U-tubes that were used in the inner ten rows of the steam generator tube bundle. The remaining U-tubes in the tube bundle did not receive the additional thermal treatment. The inspectors requested to see the procurement requirements for the Callaway Plant steam generator tubing that had been imposed by Westinghouse on its tubing vendor. In response to the licensee's request for this information, Westinghouse provided Westinghouse Material Specifications 2655A65, "Material-Special Thermal Treated Inconel Tubing (High Yield Strength in accordance with C.C. 1484)," Revision 4, and 2656A84, "Material-Nickel-Chromium-Iron Tubing (High Yield Strength in accordance with Code Case 1484)." The inspectors ascertained from review of Material Specification 2655A65 that ASME Material Specification SB-163 (i.e., Inconel 600) tubing, in conformance with ASME Code Case 1484, was required to be furnished in the annealed and thermally treated condition, with test requirements including ultrasonic examination and eddy current examination. Lower radius U-bends were also specified to receive a stress relief heat treatment following tube bending. Details of the thermal treatment criteria and stress relief heat treatment have not been included in the inspection report, as a result of the specification being considered by Westinghouse to contain proprietary information. The inspectors noted from review of data contained in EPRI Report NP-1354, "Optimization of Metallurgical Variables to Improve the Stress Corrosion Resistance of Inconel 600," dated March 1980, that the specified heat treatment parameters should improve the resistance to both primary water and caustic stress corrosion cracking. Material Specification 2656A84 was noted to contain similar technical requirements to Material Specification 2655A65, with the primary difference being the absence of any requirement to perform an additional thermal treatment on the tubing. The inspectors concluded from the review of the two material specifications that the first ten rows of tubes in the tube bundle would be expected to demonstrate a significantly better resistance to initiation of stress corrosion cracking than the remainder of the tube bundle.

The inspectors noted that the material specification did not directly identify the annealing temperature to be used for the tubing (i.e., the specification required the vendor, Westinghouse Specialty Metals Division, to submit the annealing heat treatment procedure for approval, but did not identify an applicable annealing temperature range). The certified material test reports furnished by the vendor also did not indicate the actual annealing temperature used. The inspectors calculated mean and standard deviation values for chemical composition and mechanical properties of the Callaway Plant steam generator tubing. The source document used for the calculations was a licensee data base which had been prepared from the original certified material test reports. The results obtained are listed below in Table 1. The mean and standard deviation composition values were noted

by the inspectors to be very similar for both the thermally treated and non-thermally treated material, with the reduction in mean 0.2 percent yield strength and increase in mean tensile strength for the thermally treated material considered attributable to the effects of the thermal treatment. The inspectors additionally noted during the review of mechanical property data that the reported strength properties were fully consistent with ASME Code Case 1484 requirements of 40,000 to 65,000 psi for 0.2 percent yield strength and 80,000 psi minimum ultimate tensile strength.

Table 1

Element (% by wt.)	Mean Composition (%)		Standard Deviation (%)	
	Non-TT[1]	TT[2]	Non-TT[1]	TT[2]
Carbon	0.026	0.026	0.007	0.007
Iron	8.94	8.96	0.54	0.52
Sulfur	0.003	0.003	0.002	0.003
Silicon	0.13	0.13	0.05	0.06
Copper	0.34	0.35	0.06	0.06
Nickel	75.10	75.11	0.81	0.76
Chromium	15.25	15.22	0.59	0.50
Aluminum	0.25	0.25	0.08	0.08
Titanium	0.21	0.21	0.04	0.04
Cobalt	0.04	0.04	0.01	0.01
Manganese	0.21	0.21	0.05	0.05
Phosphorus	0.009	0.001	0.002	0.002

Mechanical Property	Mean Value		Standard Deviation	
	Non-TT[1]	TT[2]	Non-TT[1]	TT[2]
0.2% Yield Strength (psi)	53,120	47,636	3,700	3,289
Tensile Stength (psi)	101,720	104,126	3,680	3,623
% Elongation	38.4	38.4	1.9	1.7
Hardness (Rockwell B)	82.4	81.9	3.3	3.5

1 - Non-thermally treated; 2 - Thermally treated

2.4 Tube-to-Tube Sheet Expansion

The inspectors requested the licensee to obtain the applicable tube-to-tube sheet expansion procedure that was used in the manufacture of the Callaway Plant steam generators. Westinghouse furnished Process Specification 81013, "Hydraulic Tube Expansion," Issue 1, in response to the licensee's request. The inspectors noted that the process specification contained detailed inspection verification requirements with respect to performance of the tube-to-tube sheet expansion activities. The inspectors also were informed that additional inspection of tube-to-tube sheet expansions was performed by Westinghouse prior to commercial service. Licensee personnel provided the inspectors a copy of Field Deficiency Report SCPM-10255 which identified that Westinghouse hydraulically re-expanded a total of 49 tubes at the Callaway Plant in May 1983. The inspectors concluded that the scope of inspection of the tube-to-tube sheet expansions prior to unit service should have resulted in a minimal number of tube sheet crevices being present and, accordingly, the number of favorable sites for initiation of secondary side stress corrosion cracking should have been reduced.

During the inspection, the inspectors reviewed EPRI Report EPRI TR-100865, "Characterization of the Resistance to PWSCC of Hydraulic Tube-Tubesheet Expansions," dated July 1992. The inspectors concluded that the report data indicated that: (a) thermally treated tubing should have a significantly reduced susceptibility to stress corrosion cracking compared with mill annealed tubing, and (b) the use of hydraulic expansion for tubes should at least extend time for initiation of stress corrosion cracking.

2.5 Licensee Actions to Increase Resistance of Tubing to Stress Corrosion Cracking

The inspectors were informed by licensee personnel that onsite shot peening (of the tube sheet region through the expansion transition) was performed during Refueling Outage RF5 (April 1992) on the hot-leg side of all steam generator tubes. This activity was performed to induce surface compressive stresses on the tube inside diameter surface and, thus, increase resistance to primary water stress corrosion cracking. It appeared to the inspectors that the compressive stresses induced by this activity should increase the resistance to initiation of new primary water stress corrosion cracks. The inspectors considered that the shallow depth of compressively stressed material would, however, have minimal effect in altering stress conditions at the tips of cracks that existed prior to performance of shot peening and, thus, the activity was not expected to delay propagation of cracks that had initiated prior to Refueling Outage RF5.

2.6 Steam Generator Tube Degradation History

Prior to operational service, the Callaway Plant steam generators contained a total of 25 plugged tubes (i.e., Steam Generator A, 8; Steam Generator B, 3; Steam Generator C, 7; and Steam Generator D, 7). Table 2 below provides the tube repair history for the four steam generators as a function of effective full-power years of operation at the time of repair. Table 3 below provides a breakdown of the tube repair history in terms of active degradation mechanisms. The references in Table 3 to ID and OD axial and circumferential

indications are considered by the inspectors to indicate the presence, respectively, of primary and secondary side axial and circumferential stress corrosion cracking. It should be noted, however, that laboratory examinations have not been performed, to date, of pulled tube samples from the Callaway steam generators, thus, precluding specific confirmation of the degradation mechanisms that have been detected by eddy current examination.

Initial identification of tube wear at anti-vibration bar intersections was made during the maintenance outage in May 1987, with 5 tubes preventively plugged. Detected anti-vibration bar wear continued to increase through Refueling Outage RF6 (104 tubes total during service, 31 tubes in Refueling Outage RF6), and then declined progressively to 7 tubes in Refueling Outage RF7 and 3 tubes in Refueling Outage RF8. The inspectors considered the inspection results from Refueling Outages RF7 and RF8 to be an indicator that the number of active anti-vibration bar wear sites was diminishing.

In Refueling Outage RF7, the licensee identified the first evidence of probable stress corrosion cracking and intergranular attack at the top of tube sheet on the hot-leg side of Steam Generators A, C, and D (i.e., Steam Generator A, 11 tubes; Steam Generator B, 0 tubes; Steam Generator C, 15 tubes; and Steam Generator D, 3 tubes). The respective types and numbers of indications found by eddy current examination in steam generator tubes were: Steam Generator A - 6 circumferential, 5 axial; Steam Generator C - 3 circumferential, 7 axial, 5 volumetric; Steam Generator D - 1 circumferential, 1 axial, 1 volumetric. The inspectors considered the most likely explanation for the volumetric indications was the presence of intergranular attack. All of the degradation detected at the top of tube sheet during Refueling Outage RF7 was noted by the inspectors to have occurred in the mill-annealed tubing in the outer rows of the steam generator tube bundles.

During Refueling Outage RF8, the number of tubes found by eddy current examination to exhibit degradation at the top of tube sheet increased in Steam Generators A, C, and D from the previous outage, with limited degradation also found for the first time in Steam Generator B. The respective tube repairs for degradation at this location were: Steam Generator A - 48 (31 circumferential, 12 axial, 5 volumetric); Steam Generator B - 8 (3 circumferential, 3 axial, 2 volumetric); Steam Generator C - 44 (28 circumferential, 11 axial, 5 volumetric); Steam Generator D - 22 (15 circumferential, 7 axial, 2 volumetric). The discrepancy between the 22 total and indicated 24 (for summed individual degradation types) for top of tube sheet degradation in Steam Generator D pertains to 2 tubes being counted twice due to the presence of both circumferential and axial defect indications. Three tubes were also repaired in Steam Generator D due to wear at anti-vibration bars, which accounts for the total repair number of 25 tubes indicated by Table 2.

The inspectors noted during review of the degradation data that a limited amount of the hot-leg side tube degradation at the top of tube sheet had occurred in the first ten rows of the tube bundle (i.e., in tubes which were reported to have been thermally treated). A total of three tubes were identified as containing single circumferention indications (i.e., Steam Generator A - Row 8, Column 115; Steam Generator B - Row 1, Column 100; Steam Generator C - Row 10, Column 70). A single axial indication was also found in one

tube (Row 2, Column 6) in Steam Generator C. Volumetric indications were detected in an additional four tubes (i.e., Steam Generator B - Row 1, Column 119; Steam Generator C - Row 10, Column 3 and Row 10, Column 48; Steam Generator D - Row 7, Column 102). The inspectors considered the detection of this type of degradation at the current accrued effective full-power years of operation to be somewhat surprising, based both on: (1) being unaware of any stress corrosion cracking that had been previously identified in domestic steam generators which had used hydraulically expanded thermally treated Inconel 600 tubing, and (2) the information noted in Section 2.4 above regarding review of EPRI Report EPRI TR-100865.

Table 2

STEAM GENERATORS (SGs) A, B, C, AND D TUBE REPAIR HISTORY					
Time of Repair Refueling Outage (RF)	Effective Full Power Years of Operation	SG A Repairs-Number of Tubes Repaired	SG B Repairs-Number of Tubes Repaired	SG C Repairs-Number of Tubes Repaired	SG D Repairs-Number of Tubes Repaired
Preservice	0.0	8	3	7	7
RF1 (5/1986)	*	0	**	**	0
M*** (4/1987)	1.9	**	2	3	**
RF2 (9/1987)	2.2	3	**	**	5
RF3 (4/1989)	3.5	**	4	9	**
RF4 (10/1990)	4.7	6	**	**	16
RF5 (4/1992)	6.0	**	15	14	2****
RF6 (10/1993)	7.3	19	**	**	18
RF7 (4/1995)	8.6	11	5	20	4
RF8 (10/1996)	9.97	48	8	44	25
Total Repairs		95	37	97	77
% Repairs (Inservice)		1.55	0.60	1.60	1.24

* - Not provided
** - Steam generator tubes not examined
*** - Maintenance outage
**** - Steam generator tubes not examined. Two tubes preventively plugged based on Refueling Outage RF4 examination results.

3 REVIEW OF TUBE EXAMINATION HISTORY, PROGRAM REQUIREMENTS, AND DATA

3.1 Review of Tube Examination History

In Refueling Outage RF1, full-length bobbin coil examinations were performed of 7 and 20 percent, respectively, of the tubes in Steam Generators A and D. During the April 1987 maintenance outage, full-length bobbin coil examinations were performed in Steam Generators B and C, using respective sample sizes of 20 and 21 percent of the tubes. In Refueling Outage RF2 (September 1987), the licensee performed full-length bobbin coil examinations of Steam Generators A and D, using a sample size (62 percent) in each steam generator which significantly exceeded the 20 percent sample size recommendation contained in EPRI Document, "PWR Steam Generator Examination Guidelines."

In Refueling Outage RF3, the licensee continued to restrict bobbin coil examinations to two steam generators (Steam Generators B and C), but also increased the sample size to 100 percent. Limited hot-leg side motorized rotating pancake coil examinations were also performed for the first time at tangent areas and at the top of tube sheet in the sludge pile region (Rows 11-20, Columns 51-75). The motorized rotating pancake coil examinations found no evidence of significant degradation.

During Refueling Outage RF4, the licensee performed a full-length bobbin coil examination of 100 percent of the tubes in Steam Generators A and D and also performed a motorized rotating pancake coil examination of a 11 percent sample of hot-leg side tube expansion transitions. Wear at anti-vibration bars, which was initially identified in the April 1987 maintenance outage, was noted to have increased to the point where 138 tubes had some degree of wear (with 49 of the 138 tubes showing wear greater than 20 percent through wall). No evidence of stress corrosion cracking was identified at the tube expansion transitions regions.

In Refueling Outage RF5, the licensee performed a full-length bobbin coil examination of 100 percent of the tubes in Steam Generators B and C. In addition, a 10.7 percent sample was examined by the motorized rotating pancake coil at hot-leg side expansion transitions. Some degree of anti-vibration bar wear was found in 188 tubes, with 113 of these tubes containing locations where wear was 20 percent or greater through wall. No cracking was found at expansion transitions.

In Refueling Outage RF6, the licensee performed the same scope of examination of Steam Generators A and D as was performed during Refueling Outage RF5 for Steam Generators B and C. Some degree of anti-vibration bar wear was found in 199 tubes, with 104 of these containing locations where wear was 20 percent or greater through wall. No evidence of cracking was found at expansion transitions.

Table 3

CALLAWAY UNIT INSERVICE REPAIR HISTORY BY DEGRADATION MECHANISM										
Tube Degradation Mechanism[5]	Number of Tubes Repaired by Refueling Outage (RF)									
	RF1	4/87[1]	RF2	RF3	RF4	RF5	RF6	RF7	RF8	Total
AVB Wear[2]		5	5	12	22	29	31	7	3	114
OD (Circumferential)								6[4]	18	24
ID (Circumferential)								4	47	51
Unknown[3] (Circumferential)									12	12
OD (Axial)								4	13	17
ID (Axial)								8	13	21
Unknown[3] (Axial)								3	5	8
OD (Volumetric)								6	3	9
ID (Volumetric)									7	7
Unknown[3] (Volumetric)									3	3
Other			3	1		2	5	4	1	16
Total	0	5	8	13	22	29	36	40	125	282

1 - Maintenance outage;
2 - Wear at anti-vibration bar locations;
3 - Initiation surface for degradation could not be determined;
4 - Two tubes with circumferential secondary side cracking at top of tube sheet also had axial cracks at this location.
5 - All crack-like and volumetric indications detected through Refueling Outage RF8 were located at the top of tube sheet.

During Refueling Outage RF7, the initial planned inspection was to examine Steam Generators B and C with the same scope as was used in Refueling Outage RF5. The licensee ultimately expanded the scope of motorized rotating pancake coil examinations to 100 percent of the mill annealed tubes at the top of tube sheet, as a result of the initial detection of an apparent crack in Steam Generator C at this location. This detection was the first identification of apparent stress corrosion cracking in the Callaway steam generators. The scope of degradation detected at the top of tube sheet is discussed in Section 2.6 above.

In Refueling Outage RF8, the licensee performed a full-length bobbin coil examination of 100 percent of the tubes in Steam Generators A and D. In addition, a plus point coil examination was performed of 100 percent of the expansion transitions on the hot-leg side of each steam generator, and a rotating pancake coil examination was performed of the low radius tubes in Steam Generator C. The scope of degradation detected is discussed in Section 2.6 above.

3.2 Review of Examination Program Requirements

The inspectors ascertained that the overall steam generator eddy current test program requirements for Refueling Outage RF8 were defined by Union Electric Purchase Order 092702. Applicable documents included Specification S-1032(Q), Revision 5, dated February 14, 1994; Westinghouse Proposal 9432050 (including clarification letters) and Attachment C, "Steam Generator Primary Side and Secondary Side Services for Refuel 8," dated April 11, 1994. The licensee also imposed additional requirements in its eddy current guideline document, ETP-BB-01309, "Steam Generator Eddy Current Testing Acquisition and Analysis Guidelines," Revision 8, which was issued October 19, 1996.

In its response to Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes," dated June 27, 1995, the licensee committed to examination of the hot-leg side expansion transitions during the eighth refueling outage using techniques qualified in accordance with Appendix H of EPRI NP-6201, "PWR Steam Generator Examination Guidelines." (Note: This document will be subsequently referred to in the inspection report as "EPRI Guidelines"). In addition, the licensee committed to a bobbin coil examination of 100 percent of the tubes in two steam generators, and that all data analysts used in the refueling outage would hold EPRI certifications as Qualified Data Analysts.

3.2.1 Current Program

The inspectors reviewed the documents identified above and compared the requirements of ETP-BB-01309 against the EPRI Guidelines. Methods and equipment used for data acquisition were also reviewed to verify consistency with licensee commitments stated in response to Generic Letter 95-03. In addition, the inspectors observed vendor activities associated with data acquisition.

The inspectors noted areas of weakness during review of the ETP-BB-01309 data analysis guidelines. Examples noted included an absence of criteria for establishing when eddy data noise was unacceptable and a lack of restrictions on the data analysts with respect to assignment of percent through-wall depth calls from bobbin coil data based on the phase angle response of the eddy current signal. Although there is a theoretical basis for estimating flaw depth from the phase angle of an indication, actual experience has demonstrated that such a practice can result in a significant degree of error in depth measurements. Nevertheless, Section 5.3.5 specified that "[a]ll percent through-wall calls should be recorded with their as measured percent through wall." The inspectors also noted other sections of the data analysis guidelines which addressed the practice of assigning depths to indications without regard to the mode of degradation. The inspectors considered the absence of program limitations in regard to assignment of through-wall depths from phase angles of indications to have the potential for potentially defective tubes being classified as degraded during an inspection.

Another area of concern noted by the inspectors during review of ETP-BB-01309 pertained to the lack of explicit guidance for disposition of indications that were perceived by analysts from review of bobbin coil data to be probably manufacturer's burnish marks. Manufacturer's burnish marks originate from polishing or grinding operations performed on tubing during manufacture (i.e., prior to steam generator service). Data analysts will frequently encounter eddy current signals that are characteristic of manufacturer's burnish marks. However, because the bobbin coil probe cannot adequately differentiate between stress corrosion cracking and volumetric indications such as manufacturer's burnish marks, additional inspections and/or a review of prior inspection data is necessary prior to disposition of such signals. Although the discussion in Attachment 4 of ETP-BB-01309 briefly mentioned performing historical data reviews or additional inspections for the disposition of manufacturer's burnish mark type indications, the inspectors noted that the analysis guidelines did not specifically require such actions.

The inspectors reviewed the acquisition technique sheets for both the Callaway tube examinations and the qualification testing, and verified that the acquisition system used at Callaway was qualified in accordance with the EPRI Guidelines. The inspectors noted an inconsistency between the analysis guidelines and the acquisition technique sheets for the plus point qualification testing. Specific directions for establishing the span and rotational (phase) settings for analysis of plus point probe data were not included in the ETP-BB-01309 data analysis guidelines. Section 5.1.3.6 of the ETP-BB-01309 data analysis guidelines did provide span and rotational instructions for analysis of rotating pancake coil data. These directions differed, however, from the settings utilized during the qualification of the plus point probe. ETP-BB-01309 required span settings for the rotating pancake coil be set such that a 40 percent notch is detectable, and rotation adjusted so that probe motion is horizontal. The Appendix H plus point probe qualification (Westinghouse Report DDM-96-009, "Documentation of Appendix H Compliance and Equivalency") was performed, however, with the 40 percent notch response set at one-half screen height and rotation set to 20 degrees for a 100 percent axial through-wall notch.

The inspectors considered that implementation of the data analysis guideline requirement (for the rotating pancake coil) to adjust rotation so that probe motion is horizontal would be inappropriate for a plus point probe. This view was based on the fact that plus point probes are designed to be virtually insensitive to probe motion, and which would result in the ability of the analyst being severely affected in establishing the phase consistent with the qualification testing.

In that the span and phase settings for each channel are of primary importance with regard to the ability of an analyst to detect indications of tube degradation, the inspectors questioned the licensee as to how the primary and secondary analysts (i.e., Westinghouse and Framatome-Rockridge, respectively) actually established these settings during the inspection. During a phone call between the NRC, the licensee, and Westinghouse on December 2, 1996, a vendor representative stated that specific direction was not provided to the primary analysts for the settings to be used in analysis of plus point data. The vendor further identified that a review had been performed of data from the inspection which indicated that plus point data analysis settings had been established based on current industry practice. However, no information was provided that would indicate the settings were established in accordance with appropriate documented instructions. The inspectors concluded that the settings used during primary analysis could have varied between different data analysts. The inspectors additionally ascertained that the lead analyst for secondary data analysis had outlined the settings to be used by the secondary data analysts prior to the Callaway steam generator tube examinations. The inspectors reviewed these instructions and concluded that the set up would provide an adequate level of detection capability for the data analysis software. The contract criteria established by the licensee for the secondary analysis did not require, however, vendor submittal of the set up instructions to the licensee for approval (i.e., the contract was placed as nonsafety related for secondary analysis). The failure to establish requirements to assure use of Appendix H qualified span and phase rotation settings for analysis of plus point coil data is a violation of Criterion IX of Appendix B to 10 CFR Part 50 (50-483/9610-01).

The licensee's bobbin coil inspection scope that was established for Refueling Outage RF8 included a 100 percent examination of the tubes in two of the four steam generators. Section 1.4.1 in ETP-BB-01309 discussed how the licensee's inspection plans were conservative with respect to the guidance specified in the EPRI Guidelines. The EPRI Guidelines recommend an inspection of a fraction of the tubes (generally 20 percent) within each steam generator every outage, such that all tubes are examined within an interval of 60 effective full power months. Based on the fact that anti-vibration bar wear has been the dominant mode of tube degradation at Callaway, the licensee concluded that its bobbin coil inspection scope was conservative with respect to the EPRI Guidelines since 50 percent of the unit tubes would be inspected during each refueling outage. As a result, tubes would undergo an inspection within an interval of approximately 36 effective full power months. The inspectors did not fully agree with the licensee's conclusion regarding the conservatism of its bobbin coil examinations, because thorough inspection in two steam generators may not be the most efficient means of detecting an emerging mode of tube degradation such as stress corrosion cracking. This mode of degradation can differ significantly between steam generators and, thus, a sample inspection of all steam generators at each outage

may improve the ability for detection. However, the inspectors noted that tube cracking detected through Refueling Outage RF8 has been limited to tube expansion transitions at the tube sheet. Accordingly, the inspectors considered the likelihood to be small that a significant stress corrosion cracking problem could exist within the two steam generators not examined by the bobbin coil method during the Refueling Outage RF8 inspections.

During the course of the tube examinations, the inspectors observed activities involving data acquisition. Copies of data acquisition guidelines, ETP-22-01300, "Eddy Current Inspection of Preservice and Inservice Heat Exchanger Tubing," Revision 8, were observed by the inspectors to be located at the acquisition stations outside containment, and discussions with the acquisition personnel indicated they were familiar with the content of the procedures. The inspectors observed the vendor's process for replacing eddy current probes during the inspection, in order to verify that proper controls were established to ensure use of the correct inspection probes during examinations. The vendor was ascertained to have established a process which included two independent verifications of the probe type prior to installing the probe on the probe pusher outside the steam generator. The inspectors noted no problems with the verification methodology and found the controls to be effective.

3.2.2 Training and Testing of Data Analysts

The inspectors ascertained that the Callaway eddy current inspection guidelines required that data analysts: (1) be certified as Qualified Data Analysts in accordance with Appendix G of the EPRI Guidelines, (2) be certified to Level IIA in accordance with American Society of Nondestructive Testing Recommended Practice SNT-TC-1A-1984, and (3) must successfully complete a site-specific performance demonstration. The inspectors considered the requirement for analysts to be certified as Qualified Data Analysts to be a program strength. The inspectors reviewed the qualification records for several data analysts and confirmed that they had been certified as Qualified Data Analysts. In addition, the licensee provided documentation which indicated that all analysts had successfully passed the site-specific performance demonstration test. However, the inspectors were unable to interview any of the primary or secondary data analysts, because analysis activities were being performed at offsite locations remote from the site.

As part of the assessment of analyst testing, the inspectors reviewed the set up and content of the site-specific performance demonstration test. The computer-based test was developed for the licensee by the EPRI and required each analyst to correctly disposition a number of indications that were contained within an eddy current inspection data base. The data base of indications was developed using previous Callaway bobbin coil and rotating pancake coil probe inspection data. As a result, it did not include data obtained using plus point probes, the rotating probe used during the Refueling Outage RF8 examinations. The inspectors noted that, because of the differences in the response from the plus point probe as compared to that from a rotating pancake coil probe, analysts would not be tested on their capability to analyze data obtained with plus point probes.

In addition, the inspectors observed that the indications data base included only a small number of indications of stress corrosion cracking relative to the overall number of defect indications. Thus, when the computer randomly selected the data for each test, an analyst may not have been required to demonstrate the ability to detect crack-like indications in order to successfully complete the test. The inspectors reviewed test scores for several analysts that had successfully completed the site-specific performance demonstration and confirmed that some had passed the test while analyzing only a small number of stress corrosion cracking indications.

The inspectors concluded from review of the overall number and type of indications in the site-specific performance demonstration data base that the licensee had inappropriately included a large number of manufacturer's burnish mark indications. This conclusion was based on the fact that manufacturer's burnish mark indications are not normally considered a type of degradation which would jeopardize the structural and leakage integrity of a steam generator tube, and the site-specific performance demonstration test did not include the necessary provisions for fully dispositioning manufacturer's burnish mark indications (i.e., availability of historical and/or motorized rotating pancake coil examination data to allow verification that the indication is not a stress corrosion crack).

The inspectors ascertained that the licensee had not required or offered structured training to the eddy current data analysts in regard to the data analysis guidelines and the potential modes of degradation that could be anticipated during the Refueling Outage RF8 tube examinations. The inspectors viewed this lack of formal site-specific analyst training as an eddy current program weakness and, when considered in conjunction with the identified site-specific performance demonstration test program deficiencies, suggested the preparation of the analysts for the Refueling Outage RF8 examinations was somewhat lacking.

3.2.3 Response to Generic Communications

The inspectors completed a limited review of licensee actions taken in response to generic communications issued by the NRC. The scope of the review included an assessment of licensee actions taken concerning Information Notice 94-88, "Inservice Inspection Deficiencies Result in Severely Degraded Steam Generator Tubes," and Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes." As part of the review, the inspectors verified that the licensee implemented the proposed actions in its steam generator inservice inspection program requirements for Refueling Outage RF8.

The NRC issued Information Notice 94-88 on December 23, 1994, to inform all pressurized water reactor licensees of lessons learned from one utility's experience with inspecting tube sheet expansion transitions for the presence of circumferential cracking. The notice emphasized several issues such as: (1) the importance of reviewing C-scan plots (i.e., terrain maps) of rotating probe data, (2) the limitations of inspection technology and optimization of inspection techniques, and (3) the need to develop a sound resolution process to address potential interfering signals that could be experienced during inspections. The licensee's Nuclear Engineering group documented its assessment of

Information Notice 94-88 in Memorandum NED-95-073, dated March 3, 1995. The document outlined the steps the licensee had taken to "attempt to identify any cracking at an early stage." The inspectors noted that some of the items listed were responsive to issues outlined in the information notice, however, not all the deficiencies were addressed and the core concern expressed by the NRC appeared to have misinterpreted for some of those that were addressed. For example, the licensee stated that "all RPC probes are the superior 3-coil probes described in the information notice." However, the information notice emphasized the limitations of relying on three-coil probes rather than the strength in using such technology. The document also stated that data is to be analyzed in accordance with the site (Callaway) specific guidelines. Although the inspectors could not directly observe the conformance of primary and secondary data analysts to the licensee's data analysis guidelines during the Refueling Outage RF8 inspections, due to the analysis being performed offsite at remote locations, it appeared from the assessment of the guidelines (discussed in Section 3.2.1 of this report) that the analysts were not working strictly in accordance with the procedural requirements in ETP-BB-01309, Revision 8.

Generic Letter 95-03 requested, in part, that all pressurized water reactor licensees evaluate recent operating experience with steam generator tube circumferential cracking and provide the NRC with inspection plans developed for the next planned inspection. The licensee submitted its response to Generic Letter 95-03 on June 27, 1995, and a supplement on January 12, 1996. In its initial response to Generic Letter 95-03, the licensee indicated that 100 percent of the hot-leg side tube expansion transitions would be inspected in Refueling Outage RF8 with techniques qualified in accordance with Appendix H of the EPRI Guidelines. As discussed previously in Section 3.2.1 of this report, the inspectors reviewed the licensee use during Refueling Outage RF8 of techniques qualified in accordance with the EPRI Guidelines. The submittal dated January 12, 1996, to the NRC regarding Generic Letter 95-03 described the measures the licensee has taken to address circumferential cracking at locations other than at the tube sheet expansion transitions. Although only limited tube cracking has been identified in the Callaway steam generators, the licensee completed a sample inspection of the U-bend region in low radius tubes within Steam Generator C during Refueling Outage RF8. The inspectors considered that the licensee was appropriately addressing the potential problem of tube circumferential cracking.

Despite the apparent deficiencies identified in the licensee's assessment of issues discussed in Information Notice 94-88, the licensee has since developed a more proactive approach toward addressing potential circumferential cracking of the Callaway steam generator tubes. This is evidenced by the response provided to Generic Letter 95-03 and the scope of specialized inspections (i.e., rotating probe) that were scheduled prior to Refueling Outage RF8. Although only limited cracking was identified in the Refueling Outage RF7 inspections, the completion during Refueling Outage RF8 of a plus point examination of 100 percent of hot-leg side tube sheet expansion transitions indicated to the inspectors the licensee's commitment to address circumferential cracking at the earliest possible stages.

3.2.4 Eddy Current Program Oversight

The inspectors requested to see available records pertaining to licensee oversight of eddy current contractor activities. Licensee personnel provided three quality assurance surveillance reports (SP95-036, SG96-06, and SP96-015) in response to this request. Surveillance Reports SP95-036 and SG96-06 documented licensee surveillances of Westinghouse eddy current examination activities at its Waltz Mill facility near Pittsburgh, Pennsylvania, during Refueling Outages RF7 and RF8. Surveillance Report SP96-015 addressed conformance of the steam generator examination program and steam generator chemistry controls to EPRI guidelines.

The inspectors ascertained from review of Surveillance Report SP95-036 that the text focused primarily on the qualification and certification of data analysts that were used by Westinghouse for the Refueling Outage RF7 examinations. The report noted that the analysts had reviewed the current licensee data analysis guidelines, ETP-BB-01309, and also eddy current data from previous Callaway steam generator examinations. It was pointed out, however, that a documented training plan was not used by Westinghouse and data interpretation and reviews (that were stated by Westinghouse personnel to have been performed as part of the training activity for analysts) were not obvious from the training records. No specific information, other than a statement that activities were found to be acceptable, was included in the report in regard to analyst performance during evaluation of Refueling Outage RF7 data. Surveillance Report SG96-06 documented review of the translation of Callaway contractual requirements into Westinghouse documents, the site-specific performance demonstration testing that was conducted at Waltz Mill during Refueling Outage RF8, the qualification records for subcontractors that were used by Westinghouse for data analysis, the most recent internal audit of the Westinghouse eddy current examination program, and observation of turnovers between shift crews. Other than a summary statement that Westinghouse appeared to be performing data analysis in an acceptable fashion, no specific information was included in regard to assessment of data analyst performance during review of Refueling Outage RF8 examination data.

Surveillance Report SP96-015, which was issued on March 12, 1996, was noted by the inspectors to comprehensively address the conformance of steam generator examination program and chemistry controls to the respective EPRI guideline documents. As noted in Section 7.3 below, the inspectors found the historical chemistry data that had been assembled during the surveillance to be of particular value in the assessment of chemistry performance during commercial operations. The inspectors observed that the surveillance report discussed two issues which involved organizational knowledge of eddy current examination technology. The auditors identified that there was no current documented policy regarding personnel training needs in eddy current examination technology, and offered a view (based on guideline recommendations regarding training and qualification of engineering personnel in steam generator nondestructive examination) that certification to SNT-TC-1A, Level IIA would be beneficial for involved engineering personnel. The surveillance report also recommended management consideration, as a result of the identification by the auditors of EPRI guideline items that were not included in ETP-BB-01309, that to address some of the items may require the expertise of a Level III

eddy current technician. SNT-TC-1A eddy current certifications were not held by current licensee employees. The inspectors considered the deficiencies discussed in Sections 3.2.1 and 3.2.2 above to be related, in part, to personnel with appropriate eddy current skills not being utilized for finalization of eddy current program requirements and independent oversight of contractor performance. Accordingly, the inspectors considered the auditors' views to be pertinent, in that they at least indicated an understanding that there was a current lack of available staff (or contracted) knowledge in the area of eddy current examination technology.

The inspectors requested information from licensee quality assurance staff regarding the method used to support the current approval of the Westinghouse Waltz Mill Service Center as a supplier. In response, licensee personnel furnished Nuclear Procurement Issues Committee Audit 6-94-11. A review was performed of the special processes section of the audit checklist, in order to ascertain the type of assessment that was performed in this area by the multi-utility audit team. The inspectors found that the section was strictly programmatic in approach, with reviews performed of personnel certifications, the SNT-TC-1A written practice, and procedures pertinent to training, qualification, and certification. No information was provided regarding review of data analysis performance.

3.3 Review of Tube Examination Data

As part of the inspection of the steam generator tube inspection activities at the Callaway plant, the inspectors completed a limited review of selected eddy current inspection data. The objective of the data review was to independently assess discrepant calls between the primary and secondary data analysts and to evaluate the signal characteristics of indications confirmed by the resolution analysts. The inspectors reviewed the eddy current data for expansion transitions in a total of 29 tubes.

For the rotating probe inspections, the licensee's eddy current data analysis guidelines instructed the data analysts to review the strip chart and C-scan data from the 300 kHz channels for the plus point, the 0.115-inch pancake, and the 0.080-inch pancake coils. Because the primary and secondary data analysis activities were completed at Westinghouse and Rockridge facilities, the inspectors could not verify whether all channels were reviewed in accordance with the guidelines. The inspectors ascertained from the independent review of the data from the selected tube sample that a majority of the indications reviewed were detectable only with the plus point coil. It was also observed that voltage measurements for all the indications reported by the primary data analysts were approximately three times larger than those reported by the secondary analysts. These measurements should have been approximately equal, because the voltage calibration procedures were similar between the two parties. Specifically, ETP-BB-01309 stated that the voltage for the 0.115-inch pancake coil probe is to be set to 20 volts on the

100-percent through-wall axial notch in the calibration standard. This measurement is then stored to the remaining channels. This set up was used by both the primary and secondary data analysts; therefore, the reported voltages from all analysts should be consistent for measurements made using the same channels. The inspectors questioned the licensee as to the source of the voltage anomaly identified during the inspection.

The primary data analysis at Callaway was completed by Westinghouse analysts using ANSER software. The secondary analysts from Rockridge reviewed the eddy current data with Eddynet95 software. After the NRC identification of the anomaly, the licensee contacted Westinghouse, who investigated the issue and concluded that the ANSER analysis software contained an apparent error in saving the voltage calibration to the other channels. Because the software error only affected the voltage measurement and not the appearance of a signal on the analysis screen, the licensee indicated that this would not diminish an analyst's capability to detect indications of steam generator tube degradation. The inspectors agreed with the licensee's conclusion.

Although the ability to detect indications was not affected by the anomaly in the ANSER analysis software, the inspectors questioned the licensee as to whether the selection of the most significantly degraded tubes for in-situ pressure testing was adversely affected. Licensees often select tubes for in-situ pressure testing based, in part, on eddy current signal voltage amplitudes. If voltage measurements were shown to be in error, in-situ pressure testing could potentially be performed on sample tubes with indications of lesser significance than others. The licensee identified, however, that tubes were selected for testing based on voltage measurements and depth profiles from eddy current phase angle depth sizing estimates. The licensee stated that depth profiles would not be affected by the voltage error. The inspectors agreed with this conclusion. The voltage measurements for tube selection were completed by analysts at Rockridge using Eddynet95 analysis software. Because this software package is not affected by the identified problem, the voltage measurements should be accurate. The licensee stated that the data for the tubes selected for testing were re-reviewed to confirm voltage measurements. The vendor concluded that the tube selection for testing was not affected by the voltage measurement error. The inspectors concluded that the selection of tubes for in-situ pressure testing was unaffected by the ANSER analysis software error.

The inspectors reviewed rotating probe eddy current data for a sample of tubes where the primary and secondary analysts reported different calls. In such situations, the indication is forwarded to the resolution analysts for final disposition. Based on review of some of the calls forwarded to the resolution analysts, the inspectors noted that the primary analysts called more possible indications than the secondary analysts. However, for the sample of tubes reviewed, these tubes with potential indications of degradation were resolved as containing no detectable degradation following resolution. Based on an independent review of discrepant calls between the primary and secondary data analysts, the inspectors agreed with the final resolution of all the potential tube indications included in the sample.

The inspectors noted that one type indication which appeared to lead to differing calls between analysts was when a ferromagnetic signal was present, such as in the tube located at Row 55, Column 50 in Steam Generator C. Figure 1 shows what appears to be a circumferential defect on the 300 kHz plus point channel. However, the inspectors concluded that the phase on lower frequencies did not rotate properly, and the signal from the locator channel (i.e., 20 kHz frequency) shown in Figure 2 displayed what appeared to be a ferromagnetic signal. The inspectors concurred with the licensee classification of the indication as a possible loose part and the actions taken to assure the structural integrity of abutting tubes.

The inspectors identified another type of indication which did not appear to be associated with a known tube degradation mechanism and caused some disagreement between the two sets of analysts. The tube located at Row 9, Column 14 in Steam Generator A showed what appeared to be a small axial indication located at the top of the tube sheet (Figure 3). However, as seen in Figure 4, this indication is associated with a "ripple" on the locator channel. The resolution analyst postulated that this indication was either a rolling mark from steam generator fabrication, or it was due to cross talk between another probe acquiring data in the adjacent tube. This tube was initially called as containing a defect by the primary analyst, but it was eventually declared as no detectable degradation following resolution. The inspectors agreed with the resolution of this tube.

The inspectors also completed an independent data review of a sample of tubes resolved as containing defects based on the inspection data. Although several of the indications were detectable using all the inspection coils, the inspectors qualitatively concluded that the majority of indications reviewed in the sample appeared to be small. This was evidenced by the fact that these smaller indications were detectable with the plus point coil, but did not exhibit detectable degradation with the pancake coils. For example, the tube located at Row 21, Column 48 in Steam Generator A was called as a single circumferential indication below the top of the tube sheet. As shown in Figure 5, the phase of the indication is consistent with defects for the plus point at 300 kHz, 200 kHz, and 100 kHz. However, this indication was not detectable on the 300 kHz frequencies for the 0.115-inch pancake and the 0.080-inch pancake coils. The licensee elected to sleeve the tube.

One of the larger indications reviewed during the data assessment was in the Steam Generator A tube located at Row 17, Column 57. Figure 6 shows data obtained from this tube which was called as containing multiple circumferential indications at the top of the tube sheet. As shown in Figure 7, the phase of the signal is consistent with defects for the plus point at 300 kHz, 200 kHz, and 100 kHz. Although the indications was one of the larger signals seen with the plus point coil, this indication was only marginally detectable on the 300 kHz frequencies for the 0.115-inch pancake and the 0.080-inch pancake coils. The licensee declared this tube defective, and elected to sleeve the tube.

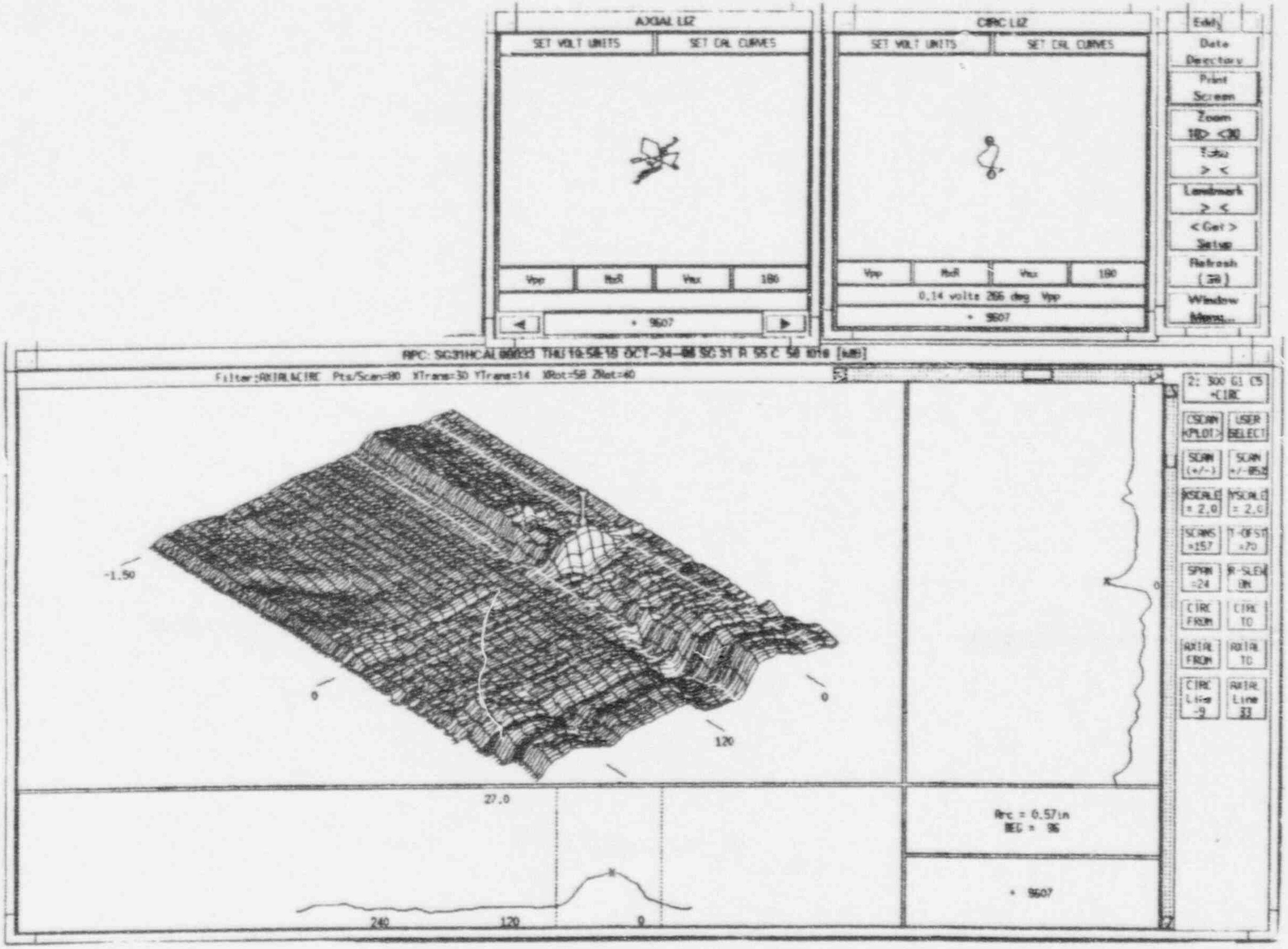

Figure 1: Apparent circumferential defect on the 300 kHz plus point channel

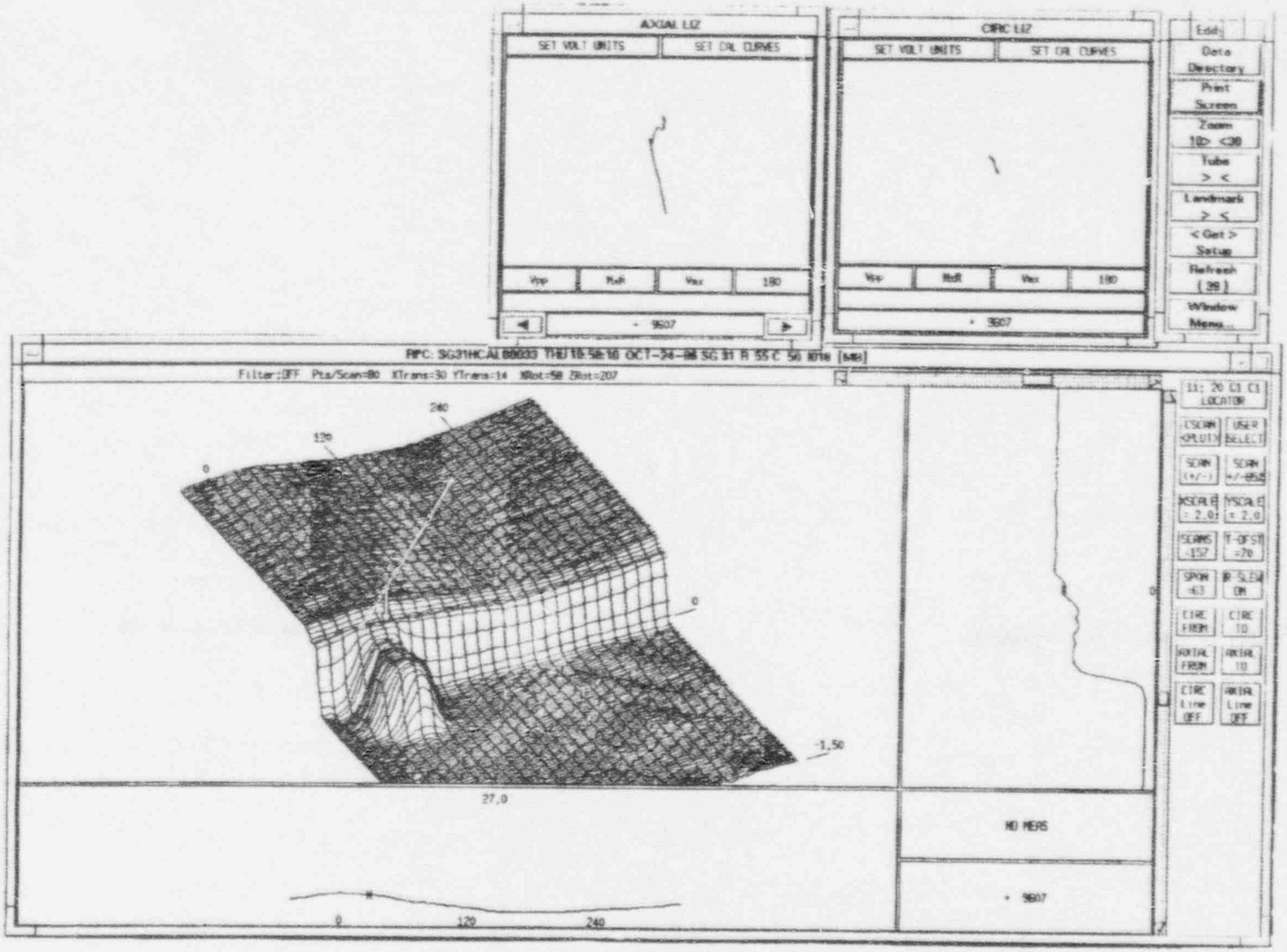

Figure 2: Ferromagnetic indication at the tubesheet secondary face on the locator channel

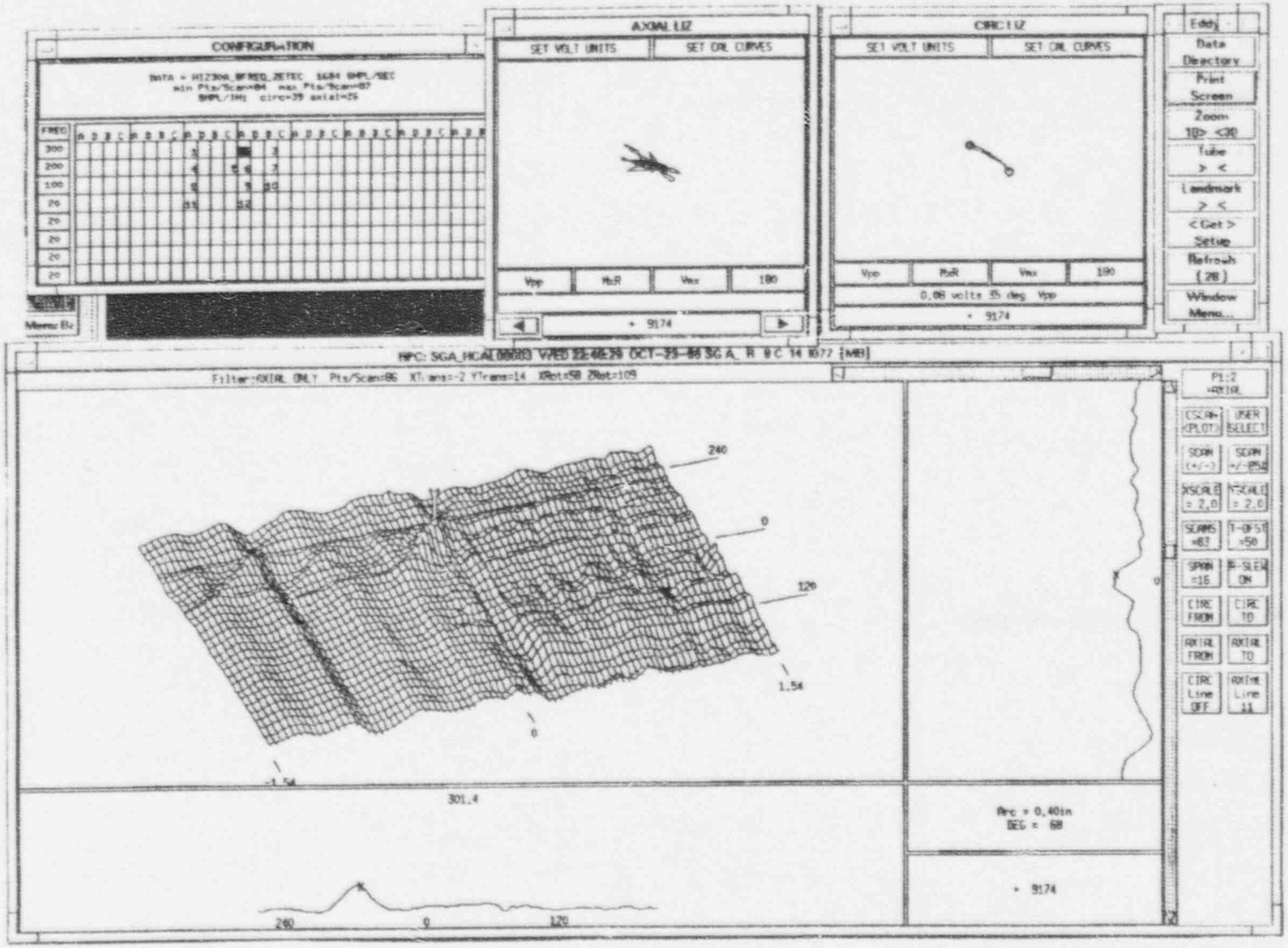

Figure 3: Small axial indication at the top of the tubesheet in tube R9/C14 (SG-A)

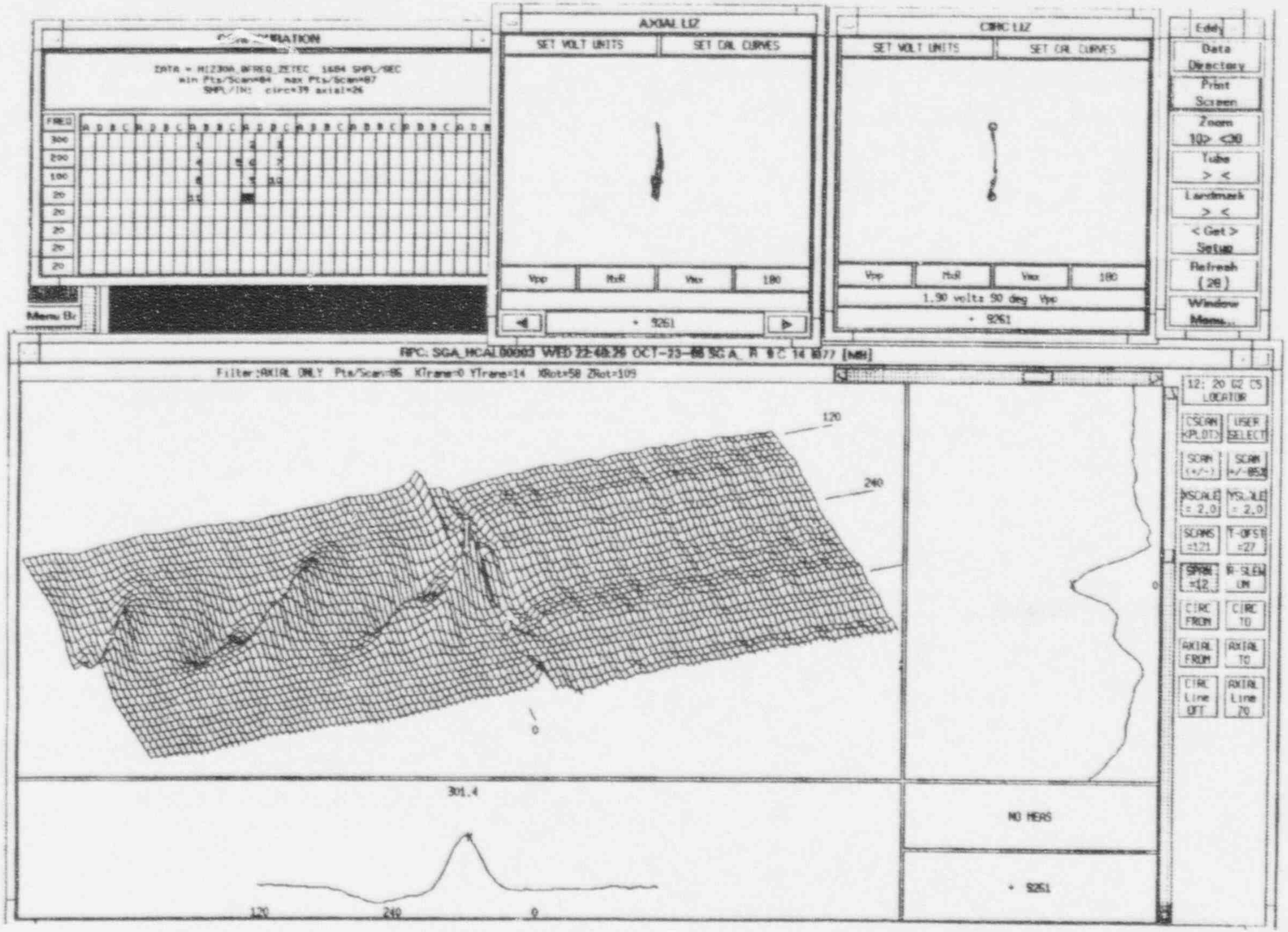

Figure 4: "Ripple" associated with indication in Figure 3 as seen on the locator channel

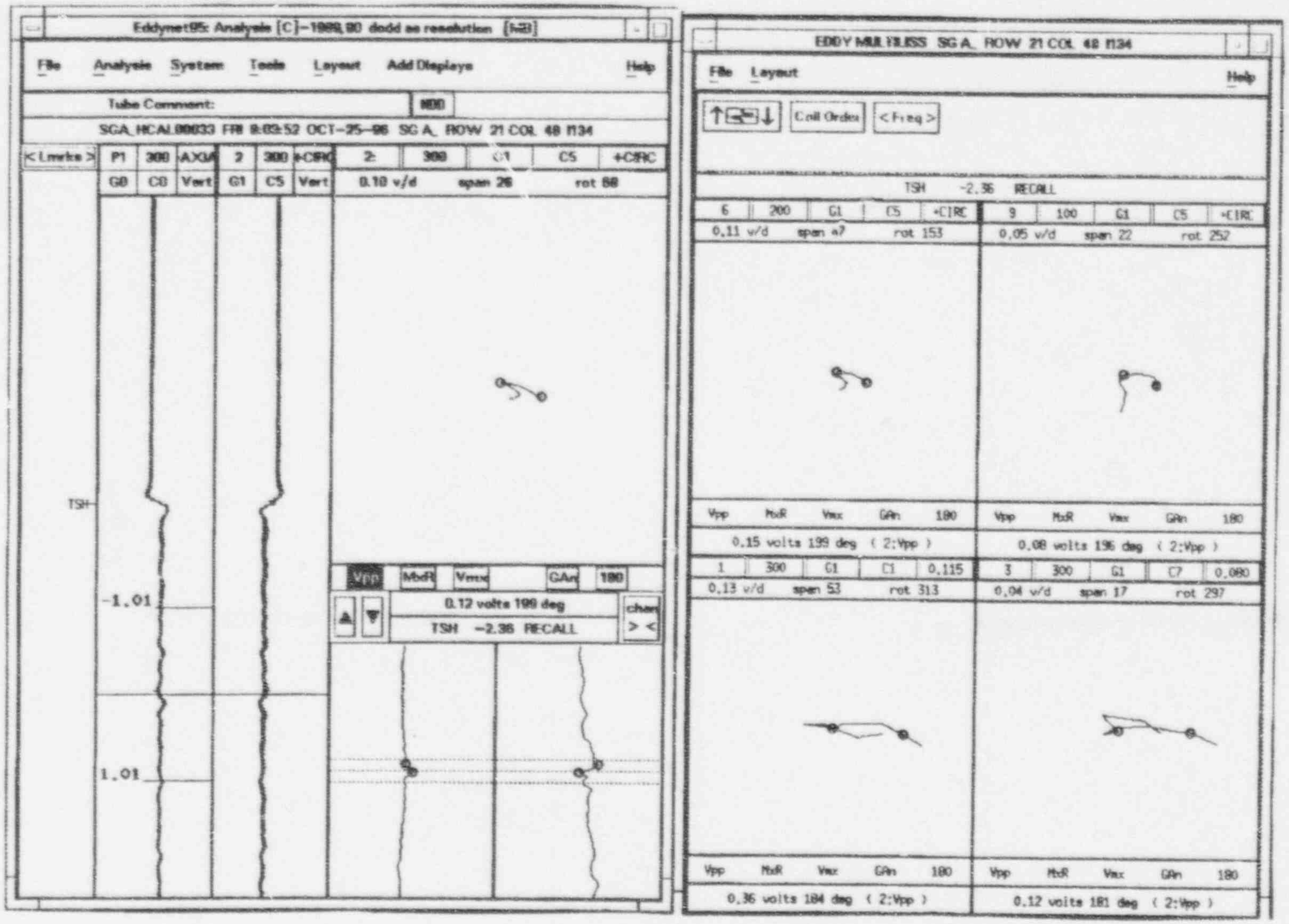

Figure 5: Lissajous signals for tubesheet expansion circumferential indication in tube R21/C48 (SG-A)

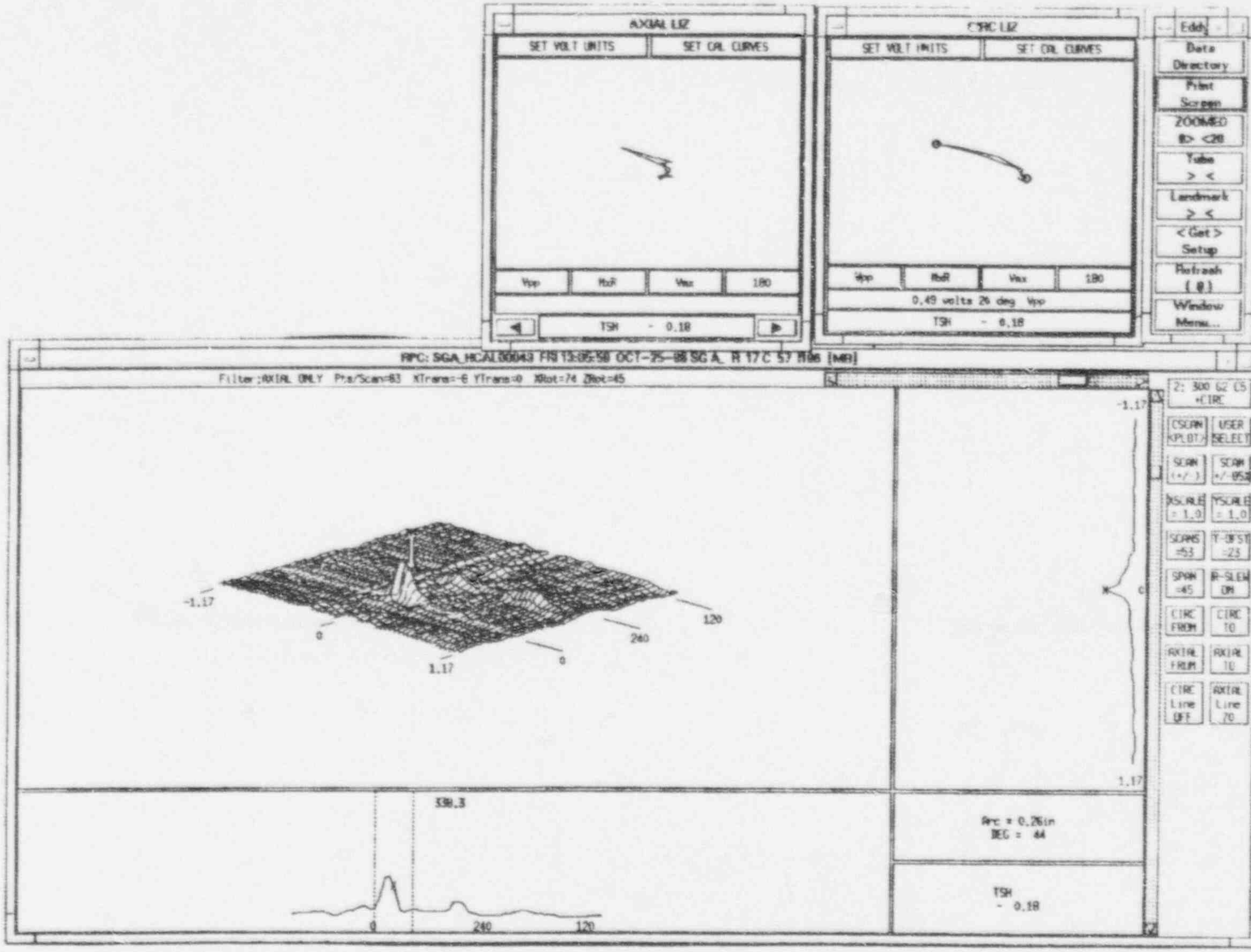

Figure 6: Multiple circumferential indication at the top of the tubesheet in tube R17/C57 (SG-A)

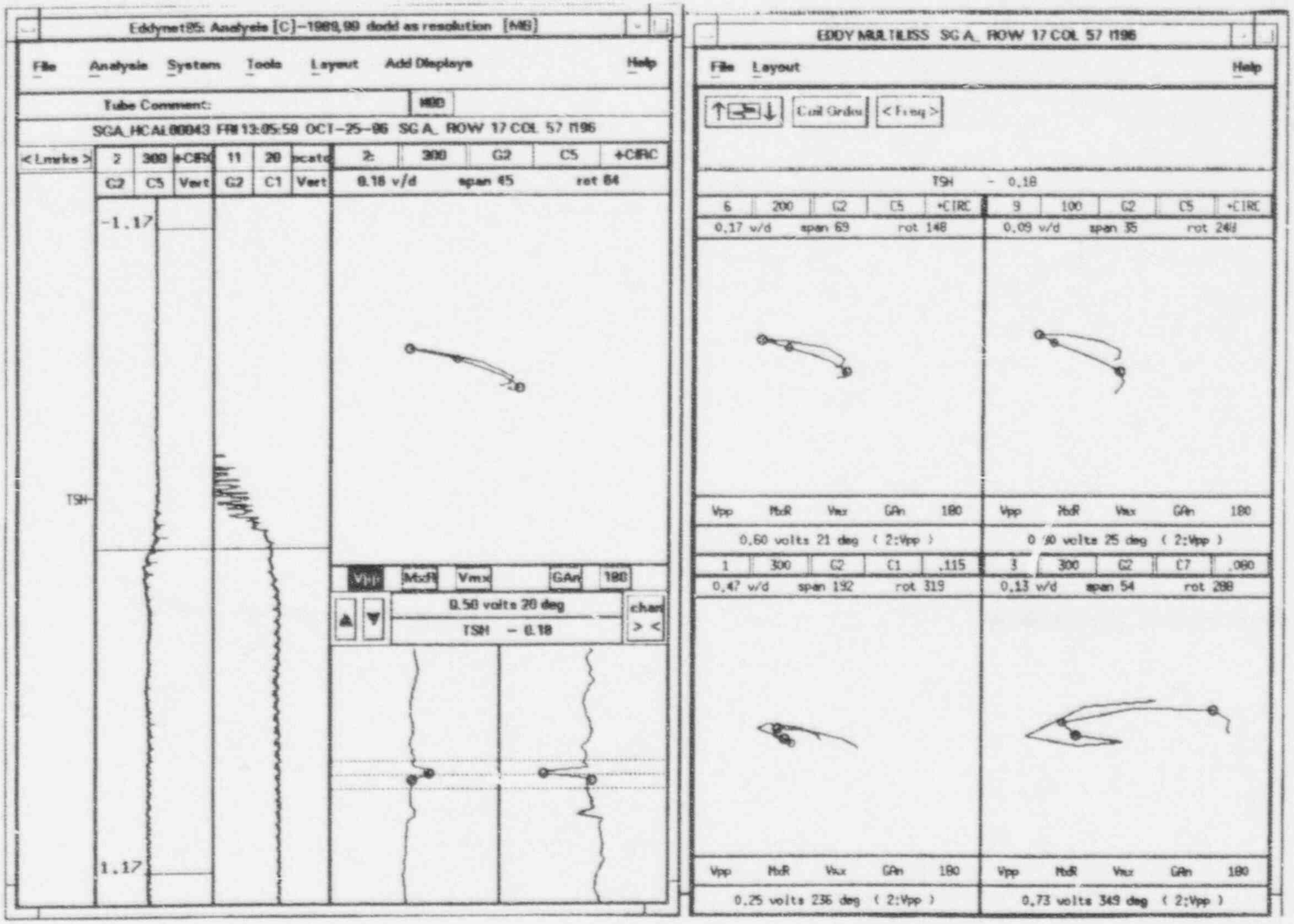

Figure 7: Lissajous signals for circumferential indication in tube R17/C57 (SG-A)

4 IN-SITU PRESSURE TESTING OF STEAM GENERATOR TUBES

The licensee performed in-situ pressure testing on ten steam generator tubes to demonstrate tube integrity. The licensee's approach for tube selection was to identify from the eddy current data the largest and/or deepest cracks that could potentially challenge structural integrity in accordance with NRC Regulatory Guide 1.121, or that could result in significant leakage potentially challenging allowable leakage limits. None of the tubes burst, thereby meeting structural integrity requirements. Two tubes leaked at approximately 0.36 gallons per day when tested at a pressure differential of 3100 psi (this leakrate was not adjusted for temperature which would result in an even lower leakrate). The licensee had not completed the formal structural and leakage integrity assessments at the time of this inspection, but based on the limited leakage seen in only two of the most significant indications, the licensee determined that leakage under postulated accident conditions would have been within allowable limits.

The inspectors reviewed the eddy current data reports for all indications identified during the current outage and determined that the ten tubes selected for in-situ testing met the licensee's tube selection criteria. As noted in Section 3.3 above, the inspectors concluded after review of eddy current data and discussions with the licensee that the tubes selection process was satisfactory.

The inspectors determined, based on the initial ten tube selection and the results of the in-situ pressure testing, that the licensee had bounded the worst-case indications and performed appropriate testing to demonstrate maintenance of structural and leakage integrity.

5 USE OF LASER WELDED SLEEVES FOR TUBE REPAIR

During Refueling Outage RF8, the first site use was made of laser welded sleeves to span tube degradation at the top of tube sheet and form a new reactor coolant pressure boundary. The 12-inch long Inconel 690 sleeves were installed by Westinghouse, with a total of 44 tubes in Steam Generator A and 36 tubes in Steam Generator C repaired by this method. The sleeves were installed using the following sequence of operations: tube cleaning, sleeve insertion into the steam generator tube and expansion, laser welding of the upper end of the sleeve to the steam generator tube, ultrasonic inspection of the weld, post-weld heat treatment, hard rolling the lower end of the sleeve, and eddy current examination of the entire sleeve. Sleeve installations were completed prior to the time of this inspection, thus, precluding observation by the inspectors of sleeve installation work activities. Documentation reviewed by the inspectors included procedures, work control documents, technical reports, topical reports, and correspondence between the licensee and the NRC related to the NRC's review and approval of the sleeving process.

5.1 Overall Process Control

The primary document used to control the laser welded sleeving activities was ascertained by the inspectors to be Procedure ETP-BB-01336, "Callaway Unit #1, Laser Welded Sleeving, 0.688" OD x 0.040" Wall Steam Generator Tubing, Model F," Revision 0, dated September 4, 1996. The inspectors confirmed that the procedure, which was developed by Westinghouse, was reviewed and approved for use at Callaway in accordance with site procedures. The inspectors also determined that all field change requests to the procedure had been formally reviewed and approved by both the vendor and the licensee. The inspectors reviewed the controls and requirements for tube cleaning, sleeve insertion and expansion of the upper sleeve end which were contained in Section 2 of Procedure ETP-BB-01336. The inspectors then reviewed the "Tube Cleaning Data Sheet" (Process Control Sheet 1.1) for the tubes in Steam Generator C and determined that brush insertion height, brush rotational speed, and brush life were monitored and adequately controlled in accordance with procedural requirements. In addition, the inspectors reviewed Process Control Sheets 2.1, 2.2, and 2.4 to verify that sleeve insertion height was monitored through periodic hardstop verifications, and system expansion pressures were calibrated and monitored during implementation of the expansion process.

The inspectors reviewed the technical criteria contained in Westinghouse Topical Report WCAP-14596, "Laser Welded Elevated Tube Sheet Sleeves for Westinghouse Model F Steam Generators, Generic Sleeving Report," dated March 1996, and correspondence between the licensee and the NRC. The inspectors determined from eddy current measurements of sludge pile heights that all laser welds were located above the sludge pile. In addition, the inspectors reviewed all indication locations (i.e. distance above the top of tube sheet) for tubes selected for sleeving, and verified that the laser weld location was consistent with the specified minimum distance from the indications. The inspectors noted that the vendor had identified by memorandum to the licensee that some tubes were not eligible for sleeving because indications were located too far above the top of tube sheet to meet the stated acceptance criteria. The inspectors also verified that the procedural controls for laser weld placement, as delineated in ETP-BB-01336, would assure correct weld placement and were appropriately controlled as documented in Process Control Sheet 3.3.

The inspectors verified that documentation existed to indicate that every sleeve installed in Steam Generator C was satisfactorily welded, ultrasonically inspected and eddy current tested. The inspectors also confirmed that the ultrasonic and eddy current equipment, inspection and analysis procedures, and acceptance criteria were in accordance with the vendor's topical report requirements.

The inspectors concluded that, with the exception discussed in Section 5.2 below, the defined sleeve installation process requirements were consistent with the criteria approved by the NRC and were satisfactorily implemented.

5.2 Control of Welding

The inspectors noted from review of the Callaway Technical Specifications that Section 4.4.5.4.a 10) requires that tube repairs (i.e., a process that restores tube serviceability) be performed by laser welded sleeving as described in Westinghouse Technical Report WCAP-14596-P, "Laser Welded Elevated Tube Sheet Sleeves for Westinghouse Model F Steam Generators," March 1996 (W (Proprietary). The inspectors ascertained from review of this report that Section 2.3.1 identifies that the essential welding variables defined in ASME Code Section IX, ASME Code Case N-395, and ASME Code Section XI, IWB-4300 were used to develop the weld process. The inspectors additionally noted that Section 6.3 of WCAP-14596-P states, in part, "The welding parameters, qualified to the rules of the ASME Code, are computer controlled at the weld operator station. The essential variables in accordance with Code Case N-395 are monitored and documented for field weld acceptance."

The inspectors reviewed the applicable Westinghouse welding procedure specification (WPS 74376, Revision 0) and supporting procedure qualification records (PQRs 512, 513, 514, 515, 516, and 517) that were used for laser welding of sleeves during Refueling Outage RF8, and evaluated the definition and qualification of the essential variables with respect to the requirements stipulated in ASME Code Sections IX and XI and ASME Code Case N-395. The inspectors determined that overall conformance to ASME Code requirements was good. One area of concern was noted pertaining to an electrical characteristic essential variable. Code Case N-395 lists as an essential variable a change in the wattage of more than ± 2 percent from that qualified. During review of the sleeve welding records for Refueling Outage RF8, the inspectors noted cases where the documented beam power was outside of a ± 2 percent range from the qualified wattage value. The inspectors were subsequently informed by Westinghouse personnel that the recorded wattage values for the laser welds were obtained from a non-calibrated instrument which would not be expected to provide accurate monitoring of wattage. The inspectors ascertained that a calibrated device was used by the vendor prior to commencement of a series of sleeve welds (and also after mirror and equipment change out) to verify beam power was within the specified wattage range. The vendor did not, however, require that the calibrated device be used, after completing a scheduled series of welds, in order to verify that beam power was still within the permissible ± 2 percent range from that qualified. The inspectors accordingly questioned whether this practice conformed to the requirement in Section 6.3 of WCAP-14596-P to monitor and document Code Case N-395 essential variables.

In response to this question, Westinghouse personnel informed the inspectors regarding the use of: (1) ultrasonic examination to verify the presence of adequate weld width at the sleeve/tube interface, and (2) infra-red feedback during weld performance to provide in-process assurance of acceptable weld quality. A Westinghouse November 26, 1996, letter on this subject was forwarded by the licensee using facsimile. The contents of the facsimile have been included in Attachment 4 to this inspection report. During a final telephone exit meeting on December 9, 1996, the licensee was informed that the adequacy of monitoring of essential variables during sleeve welding was considered an unresolved

item pending receipt of additional information regarding the infra-red feedback process. The questions posed to the licensee and the responses obtained by the licensee from Westinghouse are documented in a December 19, 1996, licensee facsimile which has been included in Attachment 4 to this inspection report. Monitoring and documentation of laser weld essential variables are considered to remain an unresolved item pending review of the Westinghouse supporting information for the December 19, 1996, facsimile, including the infra-red data for the Callaway sleeve welds (50-483/9610-02).

6 PRIMARY-TO-SECONDARY LEAKAGE MONITORING AND RESPONSE

The inspectors reviewed the effectiveness of the licensee's procedures, equipments, and practices for monitoring and responding to primary-to-secondary leakage. Specifically, the inspectors reviewed licensee responses to generic communications, the capability of monitoring systems to provide early detection of primary-to-secondary leakage, and the adequacy of operating procedures and training related to response to steam generator tube ruptures.

6.1 Licensee Response to Generic Communications

The inspectors reviewed the licensee's evaluation of NRC Bulletin 88-02; NRC Information Notices 88-99, 91-43, 93-56, and 94-43; and Significant Operating Experience Report 93-1. In general, the evaluations were found to satisfactorily address the issues identified in the subject generic communication. The inspectors did note that there was no record of how one recommendation from the evaluation of Information Notice 91-43 was dispositioned. The recommendation pertained to reviewing the alarm setpoints for the radiation monitors that were utilized for primary-to-secondary leak detection in order to ensure appropriate setpoints were used. A review of the alarm setpoints was, however, performed as part of the Significant Operating Experience Report 93-1 evaluation and response.

6.2 Procedures and Equipment Adequacy for Leak Rate Information

The inspectors reviewed the installed radiation monitors which could be utilized to alert operators to a steam generator tube leak or rupture. Additionally, the inspectors reviewed the procedures associated with determining the primary-to-secondary leak rate and the guidance contained in Electrical Power Research Institute Report TR-104788, "Primary-to-Secondary Leak Guidelines," dated May 1995.

By the end of Cycle 8 in mid-October 1996, there had been indication of primary-to-secondary leakage on the order of approximately 1 gallon per day (gpd) based on analysis of grab samples. The licensee was trending the output of all radiation monitors as well as the results of grab samples, and a very slight positive trend was observable on secondary activity that the licensee believed to be due to steam generator tube leakage. In addition to routine grab samples, the licensee utilized radiation monitoring of the condenser off-gas, steam generator continuous blowdown, and main steam line flow for early

indication of steam generator tube leakage. The addition of the main steam line N-16 monitoring was a significant enhancement in the ability to provide early indication of a primary-to-secondary leak. The radiation monitor trends were printed out daily as a matter of routine shift duties and maintained in the count room.

The procedures regarding identification of and response to steam generator tube leakage were reviewed and found to be good. The inspectors verified that the equations used to quantify leakage from grab sample results were accurate and were appropriately incorporated into the procedures. Sufficient guidance was provided to chemistry and operations personnel to ensure that appropriate actions were taken to identify, quantify, and mitigate the consequences of a steam generator tube leak. The guidance provided by EPRI Report TR-104788 was found to have been appropriately incorporated into the procedures. The inspectors noted one deficiency in Attachment 3 to Procedure OTO-BB-00001, "Steam Generator Tube Leak," Revision 5. This attachment, which provided guidance on how to isolate a leaking steam generator, did not direct the operators to close or verify closed the steam generator low point drains, as did the emergency operating procedures. A procedure revision was subsequently initiated to include the step in Attachment 3.

Overall, the inspectors ascertained that the licensee had provided the necessary equipment to detect primary-to-secondary leakage at a very early stage. Procedures were appropriate for quantification of leakage and response actions.

6.3 Alarm Setpoints on Radiation Monitors

The inspectors reviewed the rationale and documentation for the monitor setpoints utilized for detection and mitigation of primary-to-secondary leakage.

The inspectors noted that the setpoints for the condenser off-gas monitor and the steam generator blowdown monitors were established at one decade above the minimum detectable activity for the "alert" level and two decades above the minimum detectable activity for the "high" alarm. Based on the values actually recorded and the results of grab samples, the values utilized should provide for early detection of steam generator tube leakage. The main steam line N-16 monitors were ascertained by the inspectors to have a design capability to detect a 1 gpd leak at above approximately 40 percent power, but the licensee believes (based on the observations of increased secondary activity made at the end of operating Cycle 8) that useful information can be gained from the system at leakage rates of less than 1 gpd depending on the location of the leak in the steam generator tube. The alarm setpoint for the N-16 monitors was 5 gpd, which was also the entry point for Procedure APA-ZZ-01023, "Steam Generator Tube Leak Contingency Guidelines." The annunciator response procedure, OTA-SP-RM011, for the radiation monitoring system was ascertained by the inspectors to appropriately refer operators to Off-Normal Procedure OTO-BB-00001, "Steam Generator Tube Leak," upon receipt of one of the subject alarms and subsequent confirmation of elevated activity in the secondary system.

Overall, the inspectors concluded that the monitoring system alarm setpoints were sufficiently low to alert operators of increasing primary-to-secondary leakage at an early stage.

6.4 Adequacy of Emergency Operating Procedures and Operator Training

The inspectors reviewed the emergency operating procedures, as well as the training provided, with respect to steam generator tube ruptures.

The inspectors found that the emergency operating procedures were consistent with the plant configuration. Guidance was provided to continuously monitor radiation levels and grab sample results. This approach ensures entry into steam generator tube rupture procedures if radiological conditions indicate occurrence of a steam generator tube rupture. Additionally, the off-normal procedure for steam generator tube leaks, OTO-BB-00001, provides a caution that if the leak is of such a magnitude that a safety injection is not required, then the attachment to the procedure must be completed after exiting the emergency operating procedures. This caution would ensure that the leaking steam generator was isolated, even if the leak was not large enough that performance of the emergency operating procedures would require it.

The licensee's training program contained, as a required element, training on steam generator tube leaks and ruptures. During the previous 2 years (i.e., ten requalification training cycles), in addition to classroom training, simulator training had contained three steam generator tube leak scenarios and eight steam generator tube rupture scenarios. All crews would have been exposed to each scenario during the 2-year cycle. Additionally, the tube rupture training conducted during the 95-4 requalification cycle included not only operators, but also included personnel from the chemistry, health physics, and rad waste departments in an attempt to provide a more realistic, integrated response to the event.

The simulator was out of service for repairs during this part of the inspection, so no observation of simulated plant response could be observed. However, the training staff was in the process of updating the simulator model, which included radiation monitor response. Discussions with training department personnel indicated that the values currently utilized in the simulator for radiation monitor steady state values were based on actual plant information, but with no steam generator tube leakage history at the Callaway Plant, the transient information was based on calculated values and data retrieved from other plants.

Overall, the licensee's procedures and training were considered adequate to mitigate a steam generator tube rupture.

6.5 Review of Updated Safety Analysis Report Commitments

A recent discovery of a licensee operating its facility in a manner contrary to the Updated Safety Analysis Report description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the Updated Safety Analysis Report description. While performing the inspections discussed in this report, the inspectors reviewed the portions of the Updated Safety Analysis Report that were applicable to the areas inspected.

During these reviews, the inspectors observed that the main steam line N-16 monitoring system, which was made functional on October 10, 1996, was not addressed in the Updated Safety Analysis Report. The licensee informed the inspectors that, although the system was functional, not all applicable paperwork associated with the design modification package had been closed out and reviewed. When questioned by the inspectors as to whether the Updated Safety Analysis Report would be updated to include the N-16 monitoring system, the licensee indicated that they had not planned to update the Updated Safety Analysis Report to include discussion of the newly installed system and did not believe that it was necessary. Licensee personnel also indicated that they would review that decision in subsequent updates to the Updated Safety Analysis Report.

7 REVIEW OF SECONDARY WATER CHEMISTRY CONTROLS AND HISTORY

Many impurities that enter the secondary side of steam generators can contribute to corrosion of steam generator tubes and support plates. While the concentration of impurities needed to cause corrosion problems is normally much higher than that present in steam generator bulk water, concentration of impurities to aggressive levels is possible in occluded areas where dryout occurs. Typical areas where dryout and resulting concentration of impurities can occur are tube sheet crevices, tube support plate crevices, and sludge piles. Impurities known to contribute to tube denting (i.e., squeezing of tubes at tube supports or tube sheets as a result of the pressure of corrosion products) are chlorides, sulfates, and copper and its oxides. Pitting of steam generator tubes has been attributed to the presence of copper and concentrated chlorides. Concentrated sulfates and sodium hydroxide are believed to be major causes of intergranular stress corrosion cracking and intergranular attack in steam generator tubes. Iron oxide deposits and sludge promote local boiling and concentration of impurities leading to these damage mechanisms.

7.1 Program Evolution

The inspectors performed a review of the licensee's secondary water chemistry control program. The early historical chemistry program requirements were contained in Procedure APA-ZZ-00630, "Secondary Chemistry Program," Revisions 0 through 4. This procedure was superseded on January 23, 1987, by Procedure APA-ZZ-01021, "Secondary Chemistry Program," Revision 0. The current revision of the procedure, Revision 9, was approved on January 6, 1994. The inspectors compared the evolution of program requirements against the criteria contained in the EPRI "PWR Secondary Water Chemistry Guidelines." These guidelines were initially issued as EPRI NP-2704-SR in October 1982,

with a different document number assigned for each issued revision (i.e., Revision 1, EPRI NP-5056-SR; Revision 2, EPRI NP-6239; and Revision 3, EPRI TR-102134). The inspectors found from the review that the licensee had been responsive to industry secondary water chemistry initiatives, with chemistry program requirements consistently revised in a timely manner to incorporate changes in the guidelines.

Other licensee secondary water chemistry initiatives that were considered of note by the inspectors included: (1) the start in 1992 of the use of elevated hydrazine levels (> 100 ppb) to promote the most favorable electrochemical potential in the steam generators; (2) the installation in 1993 of a Micromax Display and Control System (see Section 7.4 below) for computer monitoring and trending of chemistry parameters; (3) the implementation in 1993 of the use of the alternative amine, ethanolamine (see Section 7.2 below), to reduce iron transport to the steam generators; and (4) the initiation in 1995 of titanium dioxide additions for their potential value as an inhibitor of stress corrosion cracking and intergranular attack.

7.2 Secondary Side Chemistry History

The inspectors reviewed available history for the Callaway steam generators with respect to significant chemistry events and conformance to the EPRI secondary water chemistry guidelines. Details on off-normal chemistry are discussed below in Section 7.5. As part of this review, the inspectors requested available historical information from the licensee for cycle chemistry history. The information provided in response by the licensee is listed below in Tables 4 and 5. The licensee identified in the information provided that the calculations of the mean values excluded hideout returns that occurred during power reductions and plant trips. The licensee also noted that increased impurity concentrations in the steam generator blowdown during Cycle 8 were expected, because of the removal of tube scale, an impurity hideout site, by the steam generator chemical cleaning that was performed in Refueling Outage RF7. The inspectors agreed with the licensee's position that an increase in blowdown impurities could be expected, particularly in the case of sulfates which tend to adsorb on tube surfaces rather than in crevices.

The inspectors noted that a high mean blowdown sulfate level occurred during Cycle 1, which was attributed by the licensee to an organic sulfate problem. The problem was initially addressed in November 1985, by changing the types of resin used in the condensate polishers. Review of Cycle 2 data showed, however, that monthly mean blowdown sulfate values continued to range from 20 ppb to 30 ppb between April 1986 and August 1986. After replacement of polisher cation resin (which was indicated by licensee data as being necessitated by an inability to rinse down organics and sulfate ions), a significant ongoing improvement in monthly blowdown mean sulfate concentration values was noted by the inspectors. High mean condensate dissolved oxygen values also occurred in Cycles 1 and 2. To minimize oxygen levels, the licensee installed a nitrogen sparger system in 1988 on the condensate storage tank and, in 1989, installed permanent piping and flowmeters for nitrogen sparging of individual condenser hotwells. As noted in Table 4, mean condensate dissolved oxygen values showed a significant reduction following completion of the licensee modifications.

Table 4

MEAN CYCLE VALUES FOR SECONDARY WATER CHEMISTRY						
Cycle	Sodium[2] ppb	Chloride[2] ppb	Sulfate[2] ppb	Silica[2] ppb	Feedwater pH	DO[3] ppb
1	6	3	30	150	8.9	10
2	4.5	3	10	50	8.85	8.5
3	4.5	1	4	40	8.85	5
4	4	1	2	60	9.0	4.5
5	1	< 1	2	25	9.0	3
6	0.4	0.8	0.5	13	8.95	2.5
7	0.3	0.5	0.7	10	9.3	3
8	0.6	0.7	4	280	9.1	3
Current Limit[1]	≤ 20	≤ 20	≤ 20	No limit	Note 4	≤ 10

1 - Electric Power Research Institute Report TR-102134, Revision 3, dated May 1993.
2 - Blowdown chemistry.
3 - Dissolved oxygen at condensate pump discharge.
4 - Per station pH program.

The licensee installed corrosion transport samplers in 1990 during Cycle 4 to provide monitoring and trending capability of iron and copper contents of feedwater. As noted in Table 5, the mean iron content of the feedwater was found from the initial Cycle 4 data to be relatively high. Utilization of morpholine as an alternative amine during Cycle 5 was successful in significantly reducing iron transport, but concurrently increased copper transport to the steam generators. Trials with ethanolamine as an alternative amine during Cycle 6 were successful in reducing iron transport without an accompanying increase in copper transport to the steam generators. The licensee accordingly replaced ammonium hydroxide with ethanolamine for pH control in September 1993. Iron transport has remained below the current 5 ppb EPRI limit since conversion to ethanolamine. The inspectors were informed that the main condenser was tubed with primarily cupronickel tubing (i.e., 90 percent copper, 10 percent nickel). The presence of this alloy historically restricted the licensee in the maximum pH that could be utilized in condensate as a result of the solubility of copper alloys in ammonium hydroxide. As a result, use of higher pH values

could not be used as a strategy to reduce iron transport. The inspectors noted that, with the adoption of ethanolamine as an alternative amine, the licensee was successful in significantly reducing both iron and copper transport during Cycles 6, 7, and 8. Overall, the inspectors considered that the data in Tables 4 and 5 indicated that the licensee had appropriately responded to identified secondary water chemistry issues and was successful in progressively improving water chemistry quality.

Table 5

STEAM GENERATOR FEEDWATER IRON AND COPPER TRANSPORT HISTORY					
Mean Value ppb	Cycle 4	Cycle 5	Cycle 6	Cycle 7	Cycle 8[1]
Iron	14.8	10.0/6.8[2]	7.2/3.2[3]	2.7	4.7
Copper	0.24	0.07/0.32[2]	0.06/0.02[3]	0.03	0.03
EPRI TR-102134 Limit: Iron ≤ 5 ppb; Copper ≤ 1 ppb					

1 - Data through September 1995 utilized for determination of Cycle 8 mean iron and copper values.
2 - Second value obtained during testing of morpholine as an alternative amine in March 1992.
3 - Second value applicable to cycle period following conversion from ammonium hydroxide to ethanolamine for pH control in September 1993.

The inspectors requested historical information from the licensee regarding the weight of sludge that had been removed from the steam generator tube sheets by sludge lancing during refueling outages. The information provided by the licensee is listed below in Table 6. The data showed progressive increases in removal weight during successive refueling outages, with the addition of pressure pulse cleaning in Refueling Outage RF6 appearing to have moderately increased the weight of sludge removed. The performance of chemical cleaning of the steam generators during Refueling Outage RF7, in addition to sludge lancing of the tube sheets, resulted in a total of 17,620 lbs of material being removed from the steam generators. The inspectors considered that this quantity was illustrative of the magnitude of deposition that can occur on tube bundle surfaces and in tube support plate interstices.

The only sludge chemical composition data seen by the inspectors during the inspection was documented in Surveillance Report SP96-015. The report indicated that the sludge samples were taken during Refueling Outage RF5. Iron (65 percent by weight) was, as expected, the predominant constituent, with some nickel (0.78 percent by weight) and copper (0.14 percent by weight) present. Condenser tubes appear to be the most likely origin of the copper and nickel. Lead, for which there is some evidence that it is a contributor to stress corrosion cracking, was present only in a minor amount (i.e., 0.001 percent by weight).

Table 6

STEAM GENERATOR SLUDGE LANCING		
Outage	Number of Steam Generators Cleaned	Total Sludge Removed lbs
RF1 (5/86)	All	46.5
M (4/87)	All	47
RF2 (9/87)	None	0
RF3 (4/89)	All	69
RF4 (10/90)	All	192
RF5 (4/92)	All	265
RF6 (10/93)	All	812*
RF7 (4/95)	All	17,620**
RF8 (10/96)	All	68
Total Weight (lbs)		19,119.5

* - Includes pressure pulse cleaning in addition to sludge lancing.
** - Includes pressure pulse chemical cleaning in addition to sludge lancing.

7.3 Self Assessment of Secondary Water Chemistry

The inspectors reviewed licensee audit and surveillance reports pertaining to the secondary water chemistry program which covered a period from 1992 to 1996. In general, the inspectors found the scope of the audits and surveillances to be appropriate for evaluation of chemistry performance. As previously noted in Section 3.2.4 above, the inspectors found Surveillance Report SP96-015 to be of particular value in assessment of historical secondary water chemistry performance, as a result of the extensive reviews of chemistry data that were performed in the course of this surveillance. The comprehensive evaluation of steam generator programs that was accomplished in this surveillance was considered by the inspectors to be a commendable effort. Data from this surveillance report was used to prepare the off-normal history information that is contained in Section 7.5 below.

7.4 Chemistry Instrumentation

The inspectors toured the secondary chemistry laboratory and reviewed the capabilities for on-line monitoring of secondary water chemistry, the procedures for directing action following detection of a chemistry anomaly, the capability to track out-of-specification chemistry conditions, and the general knowledge of personnel regarding detection and repair of condenser tube leaks.

7.4.1 On-Line Monitoring Systems and Tracking

The licensee utilized a MicroMax system for continuous monitoring of chemistry parameters in the condenser hotwells, heater drains, the condensate pump discharge, the condensate polisher effluent, low pressure feedwater heater influent, main feedwater pump discharge, and steam generators. Parameters monitored included pH, sodium, cation conductivity, total conductivity, and oxygen. The equipment appeared to be in good physical condition. The detection thresholds and sensitivity of the monitoring equipment appeared to the inspectors to be sufficiently low for early identification of changes in chemistry parameters.

Licensee personnel also manually entered chemistry results into a separate data base that contained information dating back to 1985. The availability of this data base provided a convenient means to the licensee for performing reviews of operational chemistry history. Out-of-specification values were highlighted for easy retrieval and recognition. Chemistry values that exceed Technical Specification or NPDES limits were required to be entered into the licensee's corrective action system by Procedure CDP-ZZ-00200, "Chemistry Schedule and Water Specs."

7.4.2 Condenser Tube Leak Detection and Repair

The licensee's MicroMax monitoring system provides early indication of a condenser tube leak, with the first observable increase in sodium observed in either the condenser hotwell (depending on the MicroMax sample sequence) or condensate pump discharge (continuously monitored). The alarm setpoints for sodium concentration were 0.3 ppb for the condensate pump discharge and 1.0 ppb for the condenser hotwells. The stated lower range of the sodium monitor was 0.01 ppb. Procedure CDP-ZZ-00200, "Chemistry Schedule and Water Specs," provided the limits and specifications for operation and action levels. Procedure APA-ZZ-01021, "Secondary Chemistry Program," provided guidance for actions based on exceeding chemistry action level limits.

The inspectors reviewed the referenced documents and then held discussions with a secondary chemistry supervisor to ascertain the individual's familiarity with condenser tube leak indications, procedures to be utilized, and the actions to be taken in response including general methodology for the tube repair. The inspectors concluded that the supervisor was appropriately conversant with identification and response actions.

7.5 Off-Normal Secondary Chemistry History

The inspectors reviewed the off-normal chemistry history for commercial operation that was prepared from Steam Generator A blowdown chemistry results during performance of Surveillance SP96-015. The results indicated overall good chemistry performance subsequent to Cycle 2, with no significant out-of-specification conditions noted that would be expected to contribute to tube degradation.

The total hours during Cycles 1 and 2 when water chemistry parameters exceeded current EPRI secondary water chemistry guideline Action Level 1 values are listed in Table 7. The results showed that the number of hours was significant, particularly in Cycle 1. The inspectors ascertained from review of Surveillance Report SP96-015 that the tabulated Action Level 1 hours also included time periods when blowdown chemistry values exceeded current Action Level 2 and 3 values. For sodium, the hours included, respectively, 70 hours and 14 hours above the Action Level 2 value during Cycles 1 and 2. Eight of the 14 hours in Cycle 2 that were above Action Level 2 pertained to time above the Action Level 3 limit. The auditors ascertained during their review of history that the plant did not follow EPRI secondary water guideline recommendations when significant off-normal sodium conditions occurred (i.e., power was typically held for Action Level 2 values versus the recommended power reduction, and power was reduced to 30 percent for the Action Level 3 occurrence versus the recommended shutdown). With respect to cation conductivity, the Action Level 2 value was exceeded for a total of 154 hours in Cycle 1 and 56 hours in Cycle 2. The Cycle 2 hours included 4 hours when the Action Level 3 value was exceeded. The plant actions were the same as noted for sodium, which was assumed by the inspectors to be because the elevation of sodium and cation conductivity occurred concurrently. This assumption was not specifically validated by the inspectors.

Level 2 chloride limits were exceeded for 3 hours during Cycle 2, with the highest level reached being 272 ppb. In this case, the licensee did reduce power. The sulfate hours recorded in Table 7 included 186 hours in Cycle 1 and 59 hours in Cycle 2 above the current Action Level 2 value. An average sulfate concentration of 326 ppb occurred during the Cycle 2 time above Action Level 2, with a peak concentration recorded of 898 ppb. As noted in Table 7, Action Level values were not established by the EPRI for sulfate in the time frame of Cycles 1 and 2. The absence of specific program requirements, thus, provides an explanation of why the accrued hours reached the levels noted in Table 7. The inspectors considered the sulfate information from Cycles 1 and 2 to be germane because of the known causal effects of sulfates with respect to stress corrosion cracking and intergranular attack.

Table 7

STEAM GENERATOR A BLOWDOWN CHEMISTRY ACTION LEVEL 1 HOURS				
Cycle	Sodium	Sulfate[1]	Chloride	Cat. Cond.[2]
1	1188	3938	268	1809
2	33	931	26	496

1 - Current action level requirements were used to compile the sulfate Action Level 1 hours. EPRI Action Level values were not established in the time frame of Cycles 1 and 2.

2 - Cation conductivity.

Overall, the inspectors considered the early off-normal water chemistry as a possible contributor to tube degradation. Licensee actions have significantly improved chemistry controls from Cycle 3 onwards.

ATTACHMENT 3

December 17, 1996, Facsimile From Licensee

December 17, 1996

Ian Barnes
USNRC Region IV
611 Ryan Plaza Drive
Suite 400
Arlington, Texas 76011

Dear Ian:

During the telephone exit on December 9, you reiterated concerns with two items:

1. Our eddy current analysis guidelines, Step 5.1.3.6, instruct "For RPC, adjust the probe rotation so that probe motion is horizontal." This is inconsistent with Appendix H guidelines and Westinghouse's Appendix H technique qualification for use with the +Point probe.

2. It is not clear how Westinghouse met their WCAP requirement to monitor and record the essential variables identified in Code Case N-395, particularly in the case of laser power (wattage).

Attached is SOS 96-1936, which was written to address the first item. We have solicited information on the second item from Westinghouse. However, due to the travel schedule of certain key personnel, we have not yet obtained it. Westinghouse has committed to responding to us on this item by Wednesday, December 18. We then hope to fax you a response on that item by the next day.

If you have any further questions, please contact Tim Pettus at (573) 676-8158 or me at (573) 676-8241.

Sincerely,

T. E. Herrmann

T.E. Herrmann

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######   #####   ######
### ##  ##   ##  ### ##
###     ##   ##  ###
   ###  ##   ##     ###
## ###  ##   ##  ## ###
######   #####   ######
```

( ) SUGGESTION
(X) OCCURRENCE Z170.0007

PART 1: IDENTIFICATION OF CONCERN

A) DISCOVERY DATE: 19961119 OCCURRED DATE/TIME: 19961020 / 1200

B) FULL DESCRIPTION/IMMEDIATE ACTION TAKEN:

S/G ANALYSIS GUIDELINES (ETP-BB-01309), IN STEP 5.1.3.6, DIRECT "FOR__ RPC, ADJUST THE PROBE ROTATION SO THAT PROBE MOTION IS HORIZONTAL."___ THIS IS INCONSISTENT WITH EPRI APPENDIX H GUIDELINES AND WESTINGHOUSE_ APPENDIX H TECHNIQUE QUALIFICATION FOR THE +POINT RPC PROBE. THE USE_ OF THIS TECHNIQUE FOR THE +POINT COULD AFFECT THE ANALYSIS OF THE DATA.

D) IDENTIFICATION

SYS	COMPONENT	COMPONENT NAME	FUNC FAIL
BB	EBB01A	RCS STEAM GENERATOR A	
BB	EBB01B	RCS STEAM GENERATOR B	
BB	EBB01C	RCS STEAM GENERATOR C	
BB	EBB01D	RCS STEAM GENERATOR D	

E) REFERENCES

SOS 96-0280 PROC ETP-BB-01309
QAA SP96-015 SOS 96-0278

F) ORIGINATOR DATE DEPT

TIMOTHY W. PETTUS 19961209 NEDN

G) PART 1 REVIEWER DATE DEPT

UNION ELECTRIC COMPANY

TOTAL SOS RECORDS PRINTED => 1
TOTAL NUMBER OF PAGES PRINTED => 2

~~12/27/96~~
06:24

UNION ELECTRIC COMPANY
SUGGESTION OCCURRENCE SOLUTION SYSTEM
P A R T 2: REPORTABILITY

SLSP3123
PAGE: 1
SOS: 96-1936

SOS TYPE : O   SUBJECT : AC   RPT. LVL.: 5   PRIORITY: 17
SOS STATUS: 50   TARGET DATE: 19980201   LFSL ENGR: FINK M.N.

SUMMARY S/G ET ANALYSIS GUIDELINES DID NOT PROVIDE ADEQUATE +POINT SETUP INSTR
DESC.: S/G ANALYSIS GUIDELINES (ETP-BB-01309), IN STEP 5.1.3.6, DIRECT "FOR RPC, ADJUST THE PROBE ROTATION SO THAT PROBE MOTION IS HORIZONTAL." THIS IS INCONSISTENT WITH EPRI APPENDIX H GUIDELINES AND WESTINGHOUSE APPENDIX H TECHNIQUE QUALIFICATION FOR THE +POINT RPC PROBE. THE USE OF THIS TECHNIQUE FOR THE +POINT COULD AFFECT THE ANALYSIS OF THE DATA

A) PLANT MODE: 1 RX POWER: 100 PLANT EFFECT G   ESF ACTUATION:

TRANSIENT PER EDP-ZZ-01007: N LCO:

ACTION INFORMATION: COMBINED RESPONSE: N

DEPARTMENT	STATUS	DUE DATE	ASSIGNED ID	MGR.APV.REQ	RESP. APVD.
NEDC	54	19980201	24356		19961210
ORC	54	19961219			

POTENTIAL FUNC FAILURE? N   FUNC FAIL REVIEW DEPT:

B) LESS THAN 48 HOURS PER REPORTABILITY DETERMINATION PER APA-ZZ-00520

REPORT TYPE: REPORT NO.:   REPORT DATE :

C) NOTIFICATIONS (SS)

CALL : N   ULNRC :   S.S. NOTIFIED?: N

D) ADDITIONAL REPORTABILITY PER APA-ZZ-00520

LFSL REVIEW: HUGH D BONO   DATE : 19961216   TIME: 1453

E) PLANT PRIORITY CALCULATION

REG	NUCLEAR SAFETY	IND/RAD SAFETY	EFFICIENCY/ RELIABILITY	MANAGEMENT DISCRETION	EMERG	TOTAL
X 3	X 5	X 5	X 3	X 2	X 99	
3	1		1			17

K E Y W O R D S:
STEAM GENERATOR HTEXCH   TEST

TOTAL SOS RECORDS PRINTED => 1
TOTAL NUMBER OF PAGES PRINTED => 1

SUBJECT: AC REPORT LEVEL: 5 PRIORITY: 17

****

PART 3: ACTION ASSIGNMENT

****

ACTION ORGANIZATION(S)

COMBINED RESPONSE? N

DEPT: NEDC DUE DATE: 19980201
ORC 19961219

****

PART 4: LFSL/QA COMMENTS AND CAUSES

****

A) LFSL/QA COMMENTS

B) DESCRIPTION OF CAUSES

NEDC

The cause of the occurrence was an oversight by the guideline writer and reviewers. The guidelines (ETP-BB-01309) were prepared for Refuel 8 as follows:

Gary Henry of EPRI was asked to revise and rewrite the guidelines to incorporate the latest technology and techniques, including the use of Appendix H methodology. Gary is the EPRI Project Manager for Steam Generator NDE. He is a former Level III eddy current analyst and has been an integral part of the EPRI Appendix G, Appendix H, and QDA program development.

After receipt of his draft, the guidelines were reviewed as part of the UEQA surveillance (SP96-015) of the steam generator program. This review yielded SOS's 96-0278 and 96-0280. SOS 96-0280, which dealt with perceived technical deficiencies of the guidelines, was sent to Gary Henry for his comments and suggestions of any further revisions that may have been necessary for the guidelines. He responded with some more changes/clarifications.

The procedure was then sent to Westinghouse (our primary contractor) and to Framatome Technologies (our contractor for a secondary review of the data). At Westinghouse, the guidelines were reviewed by Greg Turley (at that time the Manager, NDE Field Services). Greg is also a certified Level IIA analyst. In addition, they were reviewed by Craig Bowser (the Level III lead analyst for the Callaway outage). At Framatome, the guidelines were reviewed by Mike Chambers (the lead Level III analyst assigned to Callaway). Craig and Mike are both known as experienced and competent analysts. In fact, Mr. Chambers is widely considered to be one of the top eddy current analysts anywhere. Their comments were resolved and incorporated into the document.

Even with all the above efforts, specific direction for the setup of

the+Point probe failed to be added to the guidelines.

ORC

****

PART 5: REMEDIAL CORRECTIVE ACTIONS AND C A T P R

****

A) REMEDIAL CORRECTIVE ACTION COMPLETED (BEYOND FULL DESCRIPTION/ACTION TAKEN)

NEDC

The deficiency of the guidelines to provide proper and explicit direction for the setup of the +Point coil was not identified until after the outage. At that point, discussions were held with the lead analysts for Westinghouse (Craig Bowser) and Framatome (Mike Chambers). Westinghouse indicated that no specific direction was provided to their analysts for the setup of the +Point, but that review of a significant amount of the data indicated they had been properly set up in accordance with industry practice. Framatome, it turns out, had actually provided their analysts with a two page supplemental guideline that provided proper set up instructions for the +Point coil.

Therefore, we have reasonable assurance that the inspection was performed correctly in spite of the omission.

ORC

B) SOLUTION/CORRECTIVE ACTION TO PREVENT RECURRENCE (CATPR)

NEDC

ETP-BB-01309 will be revised to provide proper guidance for the setup of the +Point coil. In addition, experts solicited by Union Electric will be specifically directed to review the guidelines with respect to Appendix H qualifications and current industry practices, to ensure the guidelines are current and adequate. To further ensure those analysts examining data at Callaway are familiar with the procedure, they will be required to pass a written test administered during the analyst qualification demonstration prior to the inspection. This test will result in assurance that the written guidelines have been thoroughly examined by additional qualified analysts, who are likely to identify any deficiencies at that time, and will provide Union Electric with assurance that the guidelines are accurate and are actually being used by the analysts.

RE-ASSIGNED BY T.W.PETTUS ON 12/10/96 TO TIMOTHY E HERRMANN

A CATPR date of 19971231 is requested to support revisions for Refuel 9

APPROVER: K W KUECHENMEISTER SELECTED BY HERRMANN, TIM E. ON 12/10/96

APPROVED BY: K.W.KUECHENMEISTER ON 19961210 -- CATPR DATE = 19980201
RE-ASSIGNED BY HERRMANN, TIM E. ON 12/10/96 TO TIMOTHY W PETTUS

ORC

END OF SOS: 96-1936
TOTAL NUMBER OF PAGES PRINTED => 3

ATTACHMENT 4

November 26 and December 19, 1996, Facsimiles From Licensee

Westinghouse
Electric Corporation

Energy Systems

Nuclear Services Division

Box 158
Madison Pennsylvania 15663-0158

NSD-ST-96-416

November 26, 1996

TO: Tim Pettus
Union Electric - Callaway Plant

SUBJECT: Callaway Laser Welded Sleeving: Monitoring of Essential Variables During Field Welding

Westinghouse WCAP-14596* stipulates that the essential laser welding variables per the ASME BPVC Code Case N-395 "are monitored and documented for field weld acceptance". This letter provides clarification of the intent and the manner in which that requirement was implemented during the recent Callaway Laser Welded Sleeving campaign and in prior sleeving campaigns.

The specific essential variable in question is the laser beam power at the work piece. The power specified in the Welding Procedure Specification (WPS) is 255W to 265W for the first pass, and 324W to 361W for the second pass. These power ranges were qualified based on extensive process testing, to yield a minimum weld width at the sleeve/tube interface of 0.015 in. (The UT acceptance criteria that has been qualified consistently rejects weld widths that are less than 0.015 in. All welds are required to pass the UT examination).

Monitoring and documentation of the laser beam power is performed prior to start of welding (with every weld head) using a calibrated Ophir laser power meter to measure the power delivered by the weld head. The most common cause of loss of laser power is due to deterioration of the reflective copper mirror in the weld head. Based on extensive tests, a conservative mirror replacement interval of 30 welds had been established prior to the implementation of infrared weld quality monitoring. The power at the weld head was measured and documented each time the mirror was replaced to verify compliance with the WPS.

SUBJECT: Callaway Laser Welded Sleeving: Monitoring of Essential Variables During Field Welding

During the Callaway LWS campaign real time weld quality monitoring was performed by analyzing the infrared feedback from the weld pool. This method had been used in a prior laser welded sleeving campaign at Byron-1 in 1995. The infrared weld quality monitoring technique was shown to be a more conservative weld acceptance technique than the UT examination. The data from the infrared monitoring system was therefore used as an indicator of when a weld head mirror had to be replaced. At Callaway, this permitted more than 30 welds to be made with the same mirror. In all cases, the laser beam power at the weld head was measured and documented prior to start of welding with a weld head and after every mirror replacement.

It is therefore the Westinghouse position that the manner of monitoring and documentation of the laser beam power described above fulfills the intent of Reference 1.

- WCAP-1 4596, Laser Welded Elevated Tubesheet Sleeves for Westinghouse Model F Steam Generators, March 1 996

Bala R. Nair, Manager
Service Technology

December 19, 1996

Ian Barnes
USNRC Region IV
611 Ryan Plaza Drive
Suite 400
Arlington, Texas 76011

Dear Ian:

This is a follow-up to my letter of December 17, which addressed the concern you raised on the analysis guidelines. Your second concern was:

> **It is not clear how Westinghouse met their WCAP requirement to monitor and document the essential variables identified in Code Case N-395, particularly in the case of laser power (wattage).**

You sought answers to four specific questions. Attached is a response from Westinghouse on those questions. Please advise me as to what additional information, if any, you would like to see concerning this item.

During our conversation of December 9, you stated that a post-weld power burn would have alleviated your concerns on this matter. As we have evaluated that possibility, it is not clear why this would be so. For example, a good post-weld power burn would not justify accepting a weld with a bad UT examination, nor would a bad post-weld power burn justify rejecting a weld that UT indicated was acceptable. Therefore, it seems reasonable that the initial power burn and the UT verification of weld width meet the intent of the WCAP statement that "essential variables per Code Case N-395 are monitored and documented for field weld acceptance."

If you have any further questions, please contact Tim Pettus at (573) 676-8158 or me at (573) 676-8241.

Sincerely,

T. E. Herrmann

T.E. Herrmann

Responses to NRC Questions on Callaway LWS Infrared Feedback process Qualification

1. What was done by Westinghouse to qualify the Infrared Feedback process? Wants documentation or phone call to Ian Barnes from the NRC to explain what was done.

Response: The Infrared Feedback process qualification was conducted in conjunction with the acceptance testing of 5 field laser weld heads that were built for use at Callaway. Typically, about 30 welds were made with each weld head using the qualified WPS process parameters. Laser power at the weld head was measured at the start of welding with each and after any mirror replacement. The infrared pulse shapes were monitored for each weld and recorded. The weld quality predictions from infrared feedback were compared to the UT examination results. The qualification program showed that the infrared feedback permitted the identification of acceptable welds, welds with insufficient width (at the sleeve/tube interface), and welds with perforations. These are the characteristics that are identified by the UT examination. Neither the infrared feedback nor the UT examination can identify hot cracking in the welds. This is left to the EC inspection.

Documentation of the infrared qualification is provided separately.

2. How were the infrared monitoring requirements proceduralized?

Response: A special field procedure was prepared for the installation, checkout, and operation of the infrared feedback equipment. The field procedure for laser welding provided the Technical Advisor at site with the option to replace the weld head mirror (or other corrective action) based on the infrared feedback data.

3. What training/certification is required of the infrared operators.

Response: All infrared system operators were provided training at Waltz Mill site in the operation of the system and interpretation of the feedback data. A formal certification/recertification program has not been established, however, it is planned to provide the training/retraining every 6 months along with laser operator training.

4. Was there a QA programmatic requirement to maintain the infrared data? If so, Ian wants to see the data/charts?

Response: The infrared feedback was used as a means of monitoring weld quality, with the final acceptance of the weld joint being solely based on UT and EC inspection results. There was therefore no stipulated QA programmatic requirement to maintain this infrared data. However, the infrared data has been maintained on computer discs for all welds from which charts and other information can be readily obtained.

B. R. Nair