OPPD

Omaha Public Power District
1623 Harney Omaha, Nebraska 68102
402/536-4000

February 2, 1985
LIC-85-040

Mr. Robert D. Martin
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

References: (1) Docket 50-285
(2) Letter from OPPD (W. C. Jones) to NRC (J. T. Collins) dated June 19, 1984 (LIC-84-196)
(3) Letter from NRC (J. T. Collins) to OPPD (W. C. Jones) dated June 22, 1984
(4) Letter from OPPD (R. L. Andrews) to NRC (J. T. Collins) dated July 17, 1984 (LIC-84-228)

Dear Mr. Martin:

Mid-Cycle Inspection of the
Fort Calhoun Station Steam Generators

The Omaha Public Power District notified the NRC by letter dated May 22, 1984 that a steam generator tube failure had occurred at the Fort Calhoun Station on May 16, 1984. Subsequently, the District performed Eddy Current (EC) Examinations on the accessible tubes in both steam generators and removed the failed section of tubing for metallurgical analysis. The results of the EC examination and metallurgical analysis were provided to the NRC in References (2) and (4).

The District received Reference (3) which provided the Commission's safety evaluation of the incident and permission to restart Fort Calhoun Station. Within Reference (3) is a requirement to conduct EC and profilometry examinations of the Fort Calhoun Station steam generators nine (9) months following initial power operation unless justification is provided that such inspections are not warranted. The purpose of this letter is to provide the results of the ongoing investigations as required by Reference (3) and to provide information on the programs implemented and planned which will significantly reduce the probability of recurrence of intergranular stress corrosion cracking (IGSCC) and IGSCC induced steam generator tube failure. This information is presented in Attachment A as justification that a mid-cycle inspection of the Fort Calhoun Station steam generators is not warranted.

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Attachment A provides data supporting the postulated causative mechanism of the Fort Calhoun steam generator tube failure as the concentration of caustic species from the secondary coolant in crevices around the steam generator heat transfer tubes. Combustion Engineering, Inc. has induced denting and IGSCC of Alloy 600 tubes in a laboratory setting by concentration of caustic species in tubesheet crevices. In addition, Attachment A provides District commitments to remove contaminants from the secondary coolant and the steam generators, to maintain more conservative secondary coolant chemistry limits, and to take prompt and prudent action in the event chemistry limits are exceeded or primary-to-secondary leakage is detected.

The contents of Attachment A provide conclusive evidence that the steam generators are being operated in a very conservative manner that has substantially reduced the probability of additional tube failures resulting from IGSCC. The District believes that the work completed to date and current operating practices as detailed in Attachment A, will allow continued safe operation until the next refueling outage. The stability and integrity of the steam generators was demonstrated by the shutdown and startup transients during the month of November, 1984 also described in Attachment A.

Therefore, the Omaha Public Power District, holder of Operating License DPR-40, respectfully requests that the requirement for mid-cycle inspection of the Fort Calhoun Unit No. 1 steam generators, as presented in the Commission's letter of June 22, 1984, be waived on the basis of the information presented in Attachment A. The District is available to assist in the expedient review and disposition of this matter.

Sincerely,

R L Andrews

R. L. Andrews
Division Manager
Nuclear Production

RLA/CWN/dao

Attachment

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, DC 20036

Mr. E. G. Tourigny, NRC Project Manager
Mr. L. A. Yandell, NRC Senior Resident Inspector

Attachment A

Request for Waiver of NRC Requirement
for Mid-Cycle Steam Generator Inspections
at the Fort Calhoun Station

Request for Waiver of NRC Requirement
for Mid-Cycle Steam Generator Inspections
at the Fort Calhoun Station

Executive Summary

On June 19, 1984, the Omaha Public Power District submitted a report of the steam generator tube rupture at the Fort Calhoun Station. Since that time, the District, in conjunction with consultants, has continued to investigate the cause of and corrective actions for the IGSCC and has initiated operational changes to minimize the probability of recurrence.

The NRC safety evaluation related to restart of the unit following the tube rupture was forwarded in a letter from Mr. J. T. Collins to Mr. W. C. Jones dated June 22, 1984. This letter contains a requirement for mid-cycle inspection of steam generator tubes at the Fort Calhoun Station unless the District can provide additional justification to demonstrate that such inspections are not warranted. The attached report provides the requested justification.

Additional laboratory work has strengthened the position that the IGSCC was the result of concentration of sodium-based caustic compounds. The ingress of these compounds occurred prior to the tube failure as a result of low-level leakage of condenser cooling water from the Missouri River. The District has made a strong commitment to steam generator and condenser integrity. Steps have been taken to reduce the contaminant inventory in the steam generators and to ensure that the condenser is leak tight. Additional monitoring equipment is being installed to enhance the ability to rapidly detect condenser leakage. More restrictive guidelines have been maintained on the chemistry parameters for steam generator operation. Further investigative and operational actions have also been taken.

The improved steam generator operating conditions, the operational actions which have been taken, the leak before break consideration relative to a steam generator tube rupture and the strong management commitment to ensuring steam generator integrity have led the District to conclude that mid-cycle steam generator inspection is not warranted. Considerable detail is provided in the attached report.

Based on the information provided in the submittal, the District requests that the requirement for mid-cycle inspection of the steam generators at the Fort Calhoun Station be waived.

Request for Waiver of NRC Requirement
for Mid-Cycle Steam Generator Inspections
at the Fort Calhoun Station

I. Introduction

The following report provides the basis to conclude that mid-cycle eddy current and profilometry examinations of steam generator tubes at the Fort Calhoun Station are not warranted. This report is submitted in response to a letter from Mr. J. T. Collins to Mr. W. C. Jones dated June 22, 1984.

The material presented in this report includes a synopsis of the information submitted previously, a presentation of technical information which has been developed, a discussion of the operational actions which have been taken to limit the probability of recurrence, commitments by District management, the technical and licensing bases for waiver of mid-cycle inspections, and the District's conclusions, which culminate in a request for waiver of this inspection requirement. In addition, information is presented regarding the scope of the planned steam generator inspections for the 1985 refueling outage. Other steam generator action items in progress or under consideration by the District are also presented.

II. Synopsis of Information Previously Submitted

On May 16, 1984, the Fort Calhoun Station was in the process of conducting a hydrostatic test of the reactor coolant system during heatup following the 1984 refueling outage. With the reactor coolant system at approximately 1800 psia and 400°F, a tube failure occurred in the "B" steam generator. Operations personnel performed the actions dictated by emergency procedures and safely placed the plant in a refueling shutdown condition (Mode 5). The performance of the Operations staff and the content of the procedures used in mitigating this event were judged to be adequate.

The failed tube was in the second peripheral row from the outside of the tube bundle. The actual failure location was within a 4" wide vertical support strap at the top of the U-bend on the hot leg side of the generator. The failure resulted in a fish-mouthed tube opening approximately 1-1/4" long at the six o'clock position in the tube. Since the failed tube was in a relatively accessible location, the failed section was removed from the secondary side and transported to Combustion Engineering's laboratory for destructive metallurgical analysis.

The metallurgical analysis showed that the failure was the result of intergranular stress corrosion cracking (IGSCC) of the mill annealed Inconel 600 tubing. In addition to the main failure, there was an adjacent smaller defect oriented at 45° to the failure. Based upon Scanning Electron Microscopy (SEM) analysis, the IGSCC had propagated approximately 95% of the way through the tube wall. The remaining 5%

of the failure was by ductile tearing. The defect was 100% IGSCC. Microscopic and x-ray diffraction measurements were conducted in an effort to determine the chemical species which caused the failure. The exact causative agent could not be pinpointed at that time, however, all indications pointed to concentration of caustic species as being the most likely cause. Subsequent work has strengthened the belief that the failure was caused by caustic-induced IGSCC.

The failed tube had been eddy current inspected in December of 1982 with no apparent defect indications. The tube was also included in the inspection pattern for the March, 1984, examination. Review of the ECT data tapes from the March inspection clearly showed the presence of a 99% through-wall defect within the hot leg vertical support which had not been reported by the data analyst. The second defect in the tube also was apparent on review of the ECT tapes. As a result of this incident, all of the ECT data from the March, 1984, inspection was independently reanalyzed in order to determine if any other defect signals had not been reported. No additional conclusive defect indications were discovered as a result of this reanalysis. One tube with an anomalous indication within the vertical support was later judged to be defective based on the results of additional specialized testing.

In order to determine the extent of defect indications within the vertical support straps in the Fort Calhoun steam generators, the District embarked upon a test program which ultimately examined all of the accessible tubes in both steam generators. Of the 5,005 tubes in each steam generator, 4,957 were tested in S/G A and 4,968 in S/G B. (All tubes with support geometry similar to that of the failed tube were inspected.) Including the failed tube, a total of four indications were found in each steam generator within the hot leg vertical support. In addition to the failed tube, only one of these indications exceeded the plugging criterion of 40% through-wall penetration. This was the tube with the ambiguous indication which was discussed in the previous paragraph.

Profilometric examinations were conducted on several hundred tubes in order to determine the extent of ovalization of tubes within the vertical support straps. A majority of the examined tubes in the outer rows of the tube bundle showed measurable ovality in one or more of the vertical supports.

Prior to restart of the Fort Calhoun Station, the eddy current test data was analyzed and independently verified and the necessary corrective actions were taken. These actions included plugging of all of the tubes with any indication within the hot leg vertical support (four tubes per generator, including the failed tube). The profilometry data required computerized analysis which was not completed prior to restart of the unit. (Analysis has subsequently been completed, and the results are discussed in Section III.) Approval to restart was received on June 22, 1984, and criticality was achieved on July 11, 1984. During the course of this startup, a hydrostatic test of the reactor coolant system to 2150 psia (50 psi greater than normal operating pressure) was conducted. No additional steam generator problems were found.

II. Synopsis of Information Previously Submitted (Continued)

The District submitted a detailed report of this event and the preliminary metallurgical analyses on June 19, 1984. Minor corrections to this report were submitted on September 20, 1984. The final metallurgical report on the failed tube was submitted on July 17, 1984.

III. Recent Technical Information

The District, in conjunction with Combustion Engineering, has continued its root cause failure analysis and has developed operational actions which will limit the possibility of recurrence of this type of failure. Significant technical information has been developed, and an operational program which strongly emphasizes the importance of secondary chemistry control has been implemented. The recent technical information will be summarized in the following section of this report.

A. Corrosion Scenario/Causative Mechanism

Additional technical work by Combustion Engineering has strengthened the conclusion that the IGSCC was the result of the concentration of caustic species in low flow, high steam quality regions of the steam generator. A corrosion scenario has been developed. The following paragraphs will detail this scenario and provide the technical bases in support of the conclusion.

Visual inspection of the upper portions of the Fort Calhoun steam generators have been conducted routinely since 1975. These inspections revealed that the crevices between the heat transfer tubes and the batwing supports, vertical strap supports and scalloped bar supports have become increasingly fouled by feedtrain corrosion products. Fouling of crevices between heat transfer tubes and the support system increased the tendency for concentration of bulk water impurities in the crevices. Inleakage of condenser cooling water from the Missouri River provided a source of impurities. Concentration of impurities in the crevices produced a caustic environment, which resulted in corrosion of the support system and led to tube denting. The interaction of the tube deformation and caustic environment, particularly at the scalloped bar/vertical strap supports, resulted in IGSCC.

In the case of the tube intersections with the vertical strip/scalloped bar support system, resultant tube deformation occurred at the 3 and 9 o'clock position on the tubing, i.e., point of contact with the vertical strips, and produced a hoop tensile stress at the 6 o'clock position. The interaction of the denting induced tensile stress, the caustic environment in the crevice and the susceptible mill annealed Alloy 600 tubing produced IGSCC.

The corrosion scenario is supported by the following data:

1. Steam generator secondary chemistry data were reviewed for fuel cycles 7 and 8 and for the current cycle 9 through November 1984. Monitored parameters were generally within

III. Recent Technical Information (Continued)

A. Corrosion Scenario/Causative Mechanism (Continued)

1. (Continued)

specifications. Sodium concentrations, however, increased substantially during periods of condenser inleakage. The significant increase in the ratio of sodium concentration to chloride concentration during recovery conditions suggests that sodium is present as a caustic. Computer simulation of the effects of concentration of condenser cooling water (Missouri River) indicates that the steam blanketed regions contain alkaline solutions compared to bulk water concentrations. It should be noted that there has been no detectable leakage of Missouri River water into the condenser to date during Cycle 9.

These simulations predict a high temperature pH as high as 8.4 compared to neutral pH of 5.7 at 572°F (300°C), i.e., over two orders of magnitude more alkaline than neutral chemistry. The calculations also predict precipitation of $Mg(OH)_2$, $Ca(OH)_2$, and $Ca(SO_4)$.

2. ASTM A-508 low alloy steel induced denting has been observed in a Combustion Engineering laboratory test faulted with Missouri River water. The test configuration consisted of a seven tube apparatus with full depth crevices prototypic of some steam generators. Operation of the apparatus faulted with additions of Missouri River water produced tube denting (1 mil) approximately one inch below the secondary face of the tubesheet. The denting was the result of corrosion of the A-508 low alloy steel tubesheet, which has been shown in previous tests to be more resistant to denting than the carbon steel used in steam generator support systems. A second test operated with sodium hydroxide additions resulted in a similar observation. Hence, it is believed that caustic faulted conditions will promote denting of the Alloy 600 tubes within the carbon steel support system in the Fort Calhoun steam generators.

3. Stress corrosion cracking of Alloy 600 and 800 tubes was also observed in the tubesheet crevices following the test described above. Alloy 600 failures were axially aligned intergranular stress corrosion cracks at several locations in the tubesheet crevices. Alloy 800 failures were transgranular stress corrosion cracks along the length of the tubes and at the roll transition zone at the base of the crevice. The combination of intergranular stress corrosion cracking of Alloy 600 and transgranular stress corrosion cracking of Alloy 800 indicates that caustic was the causative species for the failures.

III. Recent Technical Information (Continued)

A. Corrosion Scenario/Causative Mechanism (Continued)

4. Hideout recovery during operational transients at Fort Calhoun indicates that the recovered ratio of sodium to chloride strongly favors caustic presence in the hideout regions of the steam generators.

5. Profiles of the failed tube, in the field and laboratory, indicated that the tube was dented prior to failure. Laboratory measurements demonstrated that the tube was compressed at the 3 and 9 o'clock positions. The tube failure was an axially aligned intergranular stress corrosion crack located at the 6 o'clock position.

B. Profilometry

Several profilometry inspection programs were conducted in the Fort Calhoun steam generators. This included a specialized test to examine the horizontal run vertical strap intersections which was performed following the tube rupture. This test revealed that a majority of the vertical strap intersections were ovalized, with the largest size and the greatest occurrence of denting at the hot leg vertical support strap. The profilometry was performed to complement the bobbin coil eddy current examinations which had been performed. This was done to provide improved detection and characterization of the dent signals. The readings from the eight eddy current coils of the profilometry probe were processed with a computer algorithm to produce a curve fit that represents the actual shape of the tube.

At each support elevation, including eggcrates, partial drilled hole support plates and vertical straps, denting is present to some degree. The batwing area (#9 support) which bisects the tube bends, could not be analyzed due to interference caused by the bend geometry. In the area of the horizontal run vertical strap intersections (#10-12 supports), denting appeared to be more severe than in the vertical straight run sections (those that intersect #1-7 supports). Of the examinations performed in the horizontal run, #10 support, the hot leg vertical support strap was determined to have the highest magnitude of denting. This type of behavior can be found in a caustic-induced denting environment since the highest concentration of caustic agents can be expected in this region. Of the examinations performed in the vertical section, #7 support appeared to be the most severely dented. In comparing the magnitude of denting in the #7 and #10 support regions, average values indicate that denting in the #10 support region is more severe, although the standard deviations associated with the readings overshadow any conclusions that may be drawn concerning the comparison. The attached Figure 1 shows the support elevations in the Fort Calhoun steam generators.

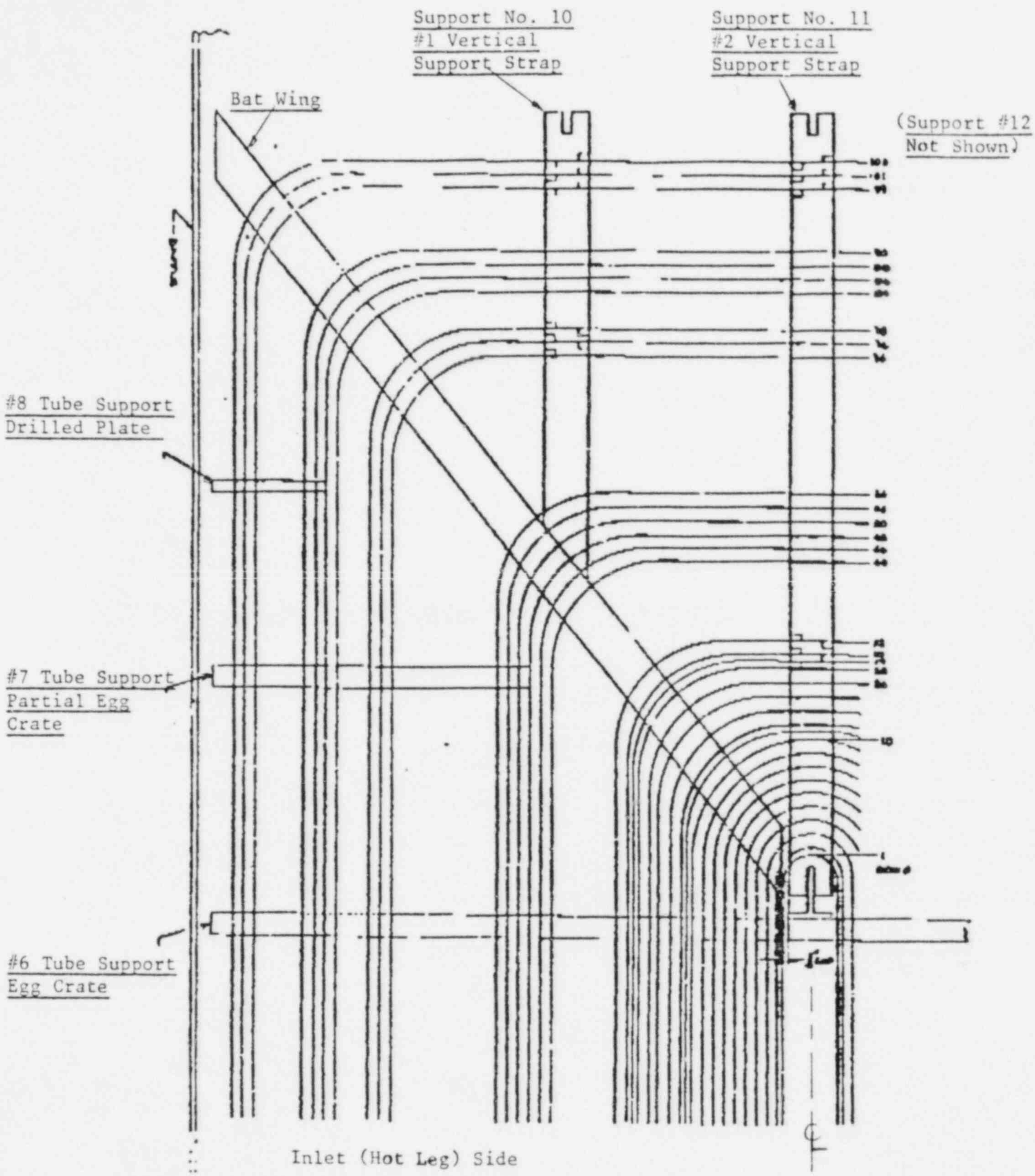

Figure No. 1 - Steam Generator Support Structure in 'U' Bend & Vertical Support Region

III. Recent Technical Information (Continued)

C. Thermal Hydraulic Analysis

A thermal hydraulics analysis of the Fort Calhoun steam generators was conducted to identify the thermal and hydraulic conditions in the proximity of the failed tube and to investigate the relationship, if any, between the thermal hydraulic conditions and the tube failure. This analysis was conducted using the ATHOS computer code, a three-dimensional, two-phase heat transfer flow distribution code. This analysis did confirm the existence of a relatively high steam quality, 48.77%, in the area of the failed tube. This result supports the postulation that the presence of the vertical support structure may have resulted in local dry-out and concentration of chemical species on the tube and adjacent support surfaces.

IV. Operational Actions to Limit Probability of Recurrence

During fuel cycle 9, the District has initiated a number of operational actions to improve the condition of the steam generators and to significantly reduce the likelihood of IGSCC-initiated tube failure.

During the heatup from the 1984 refueling outage, primary and secondary temperatures were maintained at approximately 400°F for a period of approximately 12 hours. During this temperature hold, two major operational evolutions were conducted. First, the reactor coolant system pressure was increased to 2150 psia (50 psi above normal operating pressure) for purposes of conducting a hydrostatic test. Installed instrumentation was observed, and a detailed leak inspection was made. No indications of primary-to-secondary leakage were noted during this test. The second operational evolution which occurred during this hold period was the blowdown of the steam generators at as high a rate as possible in order to remove chemical contaminants from the steam generators in the temperature range in which they are most soluble.

During November, 1984, the Fort Calhoun Station was removed from operation and taken to cold shutdown for repair of leaking pressurizer spray control valves. During the heatup following the first disassembly and repair of these valves, samples of the feed train and the blowdown were collected at periodic temperature intervals for analysis of the hideout recovery of contaminants from the steam generators. Because of further spray valve leakage, the plant was returned to cold shutdown midway through this heatup process. The valves were again repaired and another heatup was initiated. Hideout recovery samples were also taken during this heatup, and a soak/blowdown operation was conducted at approximately 400°F. During this period, a hydrostatic test to 2,150 psia was performed to ensure the integrity of the entire reactor coolant system, including the spray valves and steam generator tubes. Analysis of the samples taken during the heatup shows that significant recovery of contaminant compounds does occur at mid-range temperatures during heatup. The soak and recovery operations have been effective in removing significant quantities of potentially damaging contaminants from the steam generators.

IV. Operational Actions to Limit Probability of Recurrence (Continued)

District management is strongly committed to operating the Fort Calhoun Station with prudent chemistry control. More restrictive secondary chemistry guidelines and operating limits, which are consistent with the current recommendations of both Combustion Engineering and Steam Generator Owners Group II, have been formally adopted. Hold points for chemistry during startup have been mandated to ensure optimum chemistry conditions in the generators . These guidelines and limits include corrective action levels, shutdown levels, and the actions necessary to return chemistry parameters within specifications.

The Fort Calhoun Station has operated within the revised secondary chemistry guidelines since the beginning of cycle 9 (present cycle). Corrective actions in accordance with the guidelines have been implemented when necessary.

In addition to taking steps to reduce the contaminant concentrations in the Fort Calhoun steam generators, the District has also taken steps to reduce their ingress from the condenser and from the condensate makeup water system. During the 1984 refueling outage, a hydrostatic test of the condenser was conducted with a fluorescent dye indicator. Corrective actions were taken for those tubes and tube/tubesheet joints which were shown to be leaking as a result of this hydrostatic test. Also, the mixed bed resins in the water plant at the Fort Calhoun Station have been replaced with new, more efficient resin to improve the quality of the condensate makeup water.

As part of the overall chemistry improvement program, the District is in the process of upgrading its monitoring capabilities. Additional laboratory equipment, including an ion chromatograph, has been placed into service. On-line sodium analyzers and cation conductivity analyzers have been ordered for installation on steam generator blowdown sample lines. It is expected that the sodium analyzers will be operational by February 15, 1985. The cation conductivity analyzers have arrived on site. It is expected that they will be operational by April 1, 1985. This monitoring equipment will enable the District to promptly detect condenser inleakage or other upset in secondary chemistry so that corrective action can be initiated in a timely manner.

V. Commitments by District Management

The steam generator tube rupture which occurred on May 16, 1984, has had a significant operational and financial impact on the District. The continued integrity of the steam generators at the Fort Calhoun Station is a matter of prime concern to OPPD. The District has concluded that the caustic-induced IGSCC in the steam generators was the result of low levels of condenser inleakage, during previous operating cycles, and this condition will not again be allowed to persist.

The District is committed to prompt and prudent corrective action in the event that primary to secondary leakage is detected and confirmed or that secondary chemistry operating limits, including those relating to condenser inleakage, are exceeded. As an example of this commitment, prompt investigative action was taken during December when an

V. Commitments by District Management (Continued)

increase in the dissolved oxygen content in the condensate was noted. This problem was traced to air inleakage at a condensate pump. The pump was removed from service and corrective maintenance was performed. The dissolved oxygen content was brought back within specifications within the action level time frame specified in the chemistry procedures.

VI. Technical Bases for Waiver of Mid-Cycle Inspection Requirement

The District provides the following bases as justification that mid-cycle inspection of steam generator tubes at the Fort Calhoun Station is not warranted:

A. Management commitment to prompt action.

B. Likelihood of detectable leak before break.

C. Reduced inventory of contaminants.

D. Cycle 9 operational improvement.

Each of these four aspects of the request is detailed below:

A. Management Commitment

The commitment of the District's management to an ongoing program to ensure steam generator and condenser integrity includes adoption of more restrictive chemistry limits, improvements in monitoring capabilities, soak/recovery operations to reduce contaminant concentrations in the steam generators and a policy of prompt corrective action. This commitment is re-emphasized at this point to stress that any additional indications of primary-to-secondary or condenser leakage will be treated as a serious matter and that prompt and prudent actions will be taken.

B. Likelihood of Leak Before Break

For approximately two weeks prior to shutdown for the 1984 refueling outage, the Fort Calhoun Station detected a small primary-to-secondary leak in the "B" steam generator. Since all accessible tubes in this generator were eddy current inspected at least once during the 1984 refueling outage, it has been concluded from analysis of the ECT data that the tube which leaked prior to shutdown is the same one which subsequently failed. This failure of a defective tube during a pressure transient shows that there is a strong likelihood that a tube will exhibit a stable and detectable leak prior to proceeding to ultimate failure. This is particularly true if the tube in question is not subjected to any large temperature and/or pressure transients. This conclusion is supported qualitatively by test data from Combustion Engineering. The tests involved subjecting tube samples with pre-existing through-wall cracks to large pressure transients. In the large majority of cases, stable leakage was measured at differential pressures comparable to those which are seen during plant oper-

VI. Technical Bases for Waiver of Mid-Cycle Inspection Requirement (Continued)

B. (Continued)

ation prior to a further increase in pressure which ultimately resulted in tube rupture. The District believes that in the event of an additional IGSCC induced tube failure, a small detectable leak will occur and that the unit can be placed in a safe shutdown condition prior to rupture of the leaking tube.

C. Reduced Inventory of Contaminants

The District has taken several actions during the present operating cycle which will lessen the likelihood of IGSCC in the steam generators. One of these actions has been the soaks at mid range temperatures which have been conducted during all of the startups during the present operating cycle. While the amount of contaminants that has been removed from the steam generators cannot be quantified, these efforts to reduce contaminant concentrations will have a definite impact on retarding further incidence and propagation of IGSCC. The information gained from the hideout recovery program conducted in November, 1984, will be used to enhance the removal of additional contaminants during subsequent shutdowns and startups. Along with reducing the concentrations of contaminants which were in the steam generators, the District has placed a strong emphasis on condenser integrity which will prevent ingress of further contaminant species. The improvements in chemistry monitoring instrumentation systems which have been implemented or are in progress will greatly improve the District's ability to detect and respond in the event that condenser inleakage or other secondary chemistry problems occur.

D. Operational Improvements

Operational actions have also been taken during cycle 9 to minimize the probability of further tube failures. These actions include the implementation of revised secondary chemistry operating limits, including action levels. The goal of this program is to maintain the concentrations of potentially adverse chemical species as low as practicable. As discussed previously, all secondary chemistry parameters have been maintained within the new, more restrictive guidelines throughout cycle 9. Also, there have been no indications of primary-to-secondary leakage to date during cycle 9. One additional operational action which has been taken is a slight (8°F) reduction in T_C. There is limited laboratory data which indicates that the propagation rate of caustic-induced IGSCC can be reduced by reducing temperature. This reduction in T_C results in a slight performance penalty for the unit, which the District has decided to accept until such time as further contaminants are soaked from steam generator crevices.

VI. Technical Bases for Waiver of Mid-Cycle Inspection Requirement (Continued)

E. Summary

The District believes that the actions that have been taken to date have substantially reduced the probability of a steam generator tube rupture resulting from IGSCC. In order to offer the highest degree of protection for the health and safety of the public, however, additional operator training with respect to awareness of and response to steam generator tube ruptures has been conducted. The plant chemistry staff is highly aware of the need for prompt detection, confirmation and reporting of a primary-to-secondary leak, should it occur. The District does not believe that the health and safety of the public will be jeopardized by postponing steam generator inspections until the scheduled 1985 refueling outage.

For the reasons expressed above, the District requests that the present NRC requirement for mid-cycle inspection of steam generator tubes at the Fort Calhoun Station be waived.

VII. Licensing Bases for Waiver of Mid-Cycle Inspection Requirement

Initial information regarding the tube failure at the Fort Calhoun Station was presented to the NRC staff in a meeting on May 31, 1984. At that meeting, it was stated that the District does not believe that a safety problem exists. The District's tube failure report, dated June 19, 1984, included an analysis which concluded that continued operation of the Fort Calhoun Station did not constitute an unreviewed safety question. None of the investigative and operational actions taken since the June 19, 1984, report have provided any information which would alter this position. The District, therefore, reiterates these conclusions.

The plant safety analysis for the steam generator tube rupture event is based on conservative assumptions of coolant system activities and a double ended tube rupture. Results of the analysis show that the calculated off-site doses are well below the guidelines of 10 CFR Part 100.

The tube failure of 5/16/84 was handled by the plant operating staff with no threat to the health and safety of the plant staff or public, thereby demonstrating a capability for handling any future similar event, however unlikely, in a safe manner.

Laboratory data exists which demonstrates that IGSCC can propagate rapidly under the proper set of material, environmental and stress conditions. Thus, a mid-cycle inspection of steam generator tubes at the Fort Calhoun Station would provide only slight assurance that a tube failure would not occur later in the operating cycle. The District asserts that this slight level of assurance is not offset by the large negative economic impact of the outage, or the increased man-rem exposure which personnel will accumulate during the outage. It is estimated that the cost to the District for inspection services, additional

VII. Licensing Bases for Waiver of Mid-Cycle Inspection Requirement (Continued)

personnel and replacement power is a minimum of $2,000,000. It is also estimated that 20 man-rem of personnel radiation exposure will be received by people directly involved in the inspection.

In addition to these factors, the originally planned 18 month length of Cycle 9 has been reduced due to the startup delay caused by the tube rupture and subsequent inspections and repairs. Present plans are to shut down for refueling in October 1985 when the Cycle 9 burnup meets the Cycle 10 design basis. This results in a Cycle 9 length of 15 months. Since there has already been one cold shutdown during Cycle 9, as discussed in Section IV, plant operation will not be continuous during this cycle. Also, the District is requesting an extension of approximately six months prior to the scheduled 1985 refueling outage and the planned steam generator inspections.

Because of the acceptable results of the conservative safety analysis for a double ended steam generator tube rupture, the demonstrated capability of operators to handle the event properly, and the additional factors stated above, we do not believe that waiver of the mid-cycle testing requirements will significantly affect the health and safety of the public. The District, therefore, requests that the mid-cycle tube inspection requirement be waived.

VIII. Inspection Plans

The District, in conjunction with Combustion Engineering, has developed an inspection plan for the next steam generator tube eddy current examination, regardless of whether this examination is conducted during a mid-cycle outage or during the 1985 refueling outage, scheduled to begin in October, 1985. The major elements of this examination plan are as follows:

1. Multifrequency bobbin coil eddy current examination, including a 100 KHz absolute test, of approximately 500 tubes in each steam generator. Since a complete multifrequency examination was done during 1984, the inspection pattern for this examination will be concentrated in the outer areas of the tube bundle to search primarily for incipient indications in the vertical support strap regions. The pattern will also include examination of those tubes with nonplugable indications and a selection of tubes which will reveal if any of the types of problems which have been noted in other Combustion Engineering steam generators are present at Fort Calhoun.

2. Profilometry of approximately 200 tubes per steam generator. The tubes to be profiled will be selected from those which were profiled during 1984. The purpose of these examinations is to determine the amount of denting or ovalization which has occurred since the last inspection. Since standard bobbin coil ECT measures the average radius change within the tube, it has a tendency to underestimate the degree of ovalization which may be occurring within the eggcrates or vertical tube supports. Profilome-

VIII. Inspection Plans (Continued)

2. (Continued)

try provides the best measure available of the extent of distortion which exists.

The District believes that this examination program will provide an accurate assessment of the condition of the Fort Calhoun steam generators. The results of the eddy current phases of this program will be analyzed and independently verified, as was done with the 1984 post-failure inspections. In the event that the results of this inspection program show that the scope of this inspection should be expanded, additional inspections will be conducted.

The overall program of steam generator inspections to be conducted during the 1985 refueling outage will also include a visual inspection of the secondary side of each of the generators. The amount of deposits present on the tubes, on the tube supports, and at tube/support intersections will be of particular interest. The position of the No. 8 partial drilled hole support plate following operation subsequent to the rim cut modification will be measured for growth. This is an indirect assessment of dent progression.

IX. Other Related Actions

As part of its commitment to the continued integrity of the steam generators at the Fort Calhoun Station, the District intends to become a participant in Steam Generator Owners Group II. The primary purposes for this membership are to enhance the knowledge of the District personnel who are responsible for the steam generator integrity program and to ensure that the inspection and maintenance work performed in the Fort Calhoun steam generators is State of the Art. The District fully intends to be an active and supportive participant in the program.

The District is now a member of the Electric Power Research Institute. This membership will enhance the District's access to current research throughout the industry and will enhance awareness of potential problems and prospective solutions in the industry for appropriate consideration in the operation of the Fort Calhoun Station.

The District has investigated the advisability of applying boric acid neutralization to the steam generators. Limited laboratory data by both Combustion Engineering and Westinghouse indicates that boric acid may effectively neutralize caustic-induced denting and/or IGA/IGSCC. Equipment modifications and procedures for application of boric acid passivation at the Fort Calhoun Station are presently being developed in the event that it is determined to be desirable.

The present low pressure feedwater heaters at the Fort Calhoun Station are equipped with copper alloy tubes. The District has purchased replacement stainless steel tube bundles, which will be installed during the 1985 refueling outage. This will reduce the further deposition of copper and copper oxides in the steam generators and will allow operating chemistry parameters to be adjusted.

X. Comparative Health of Fort Calhoun Steam Generators

The steam generators at the Fort Calhoun Station are in generally good condition when compared with other steam generators with similar operating histories. Combustion Engineering has recently evaluated the structural integrity of the steam generators and has conclude hat the generators are acceptable for continued operation. In addition to the IGSCC-induced tube failure, the steam generators, however, do have other problems, as listed below:

1. Corrosion of the carbon steel support system has occurred. This is evidenced by an increase in the average dent size in the No. 8 partial drilled hole support plate. Tube deformation at the hot leg batwing and vertical supports has also occurred, and this may have been aggravated by differential thermal expansion of tubes locked within the No. 8 drilled support plates in the vertical tube section and reacting against the upper supports in the horizontal tube section.

2. As in other generators, ovalization of eggcrate supports has occurred at Fort Calhoun. This is present primarily on the hot leg side of each steam generator.

3. Some minor tube wall indications have been noted at mid span and sludge pile locations.

When compared with the condition of other recirculating steam generators throughout the industry, the problems at the Fort Calhoun Station are relatively minor. The tube microstructure of the Fort Calhoun steam generators is not highly susceptible to intergranular attack (IGA), there is not extensive pitting of tubes within the sludge pile region, the eggcrate supports have not been severely degraded, and there are no indications of the tight radius U-bend problems which have been noted in other generators. Less than 0.3% of the tubes in the Fort Calhoun steam generators have been plugged for dents or defects since initial operation in 1973. Even so, the District does view the problems at the Fort Calhoun Station as being serious and worthy of prompt corrective action and continued attention. This concern has formed the basis for the actions detailed in this report.

XI. Summary

The District has presented in this report a synopsis of the actions that have been, are being, and will continue to be taken in order to ensure the continued integrity of the steam generators at the Fort Calhoun Station. We understand that the NRC basis for requiring mid-cycle testing of steam generator tubes is to locate and plug any tube degradation that might otherwise lead to a tube rupture before completion of this operating cycle. We believe, however, that the inherent nature of tube failures and the corrective measures already taken, combine to make the benefits of this approach extremely small. The leak before break consideration, the improved steam generator chemistry conditions, the commitment of management to ensuring steam generator integrity, and the operational actions discussed above have led us to conclude that mid-cycle inspection of steam generator tubes at the Fort Calhoun Station is not warranted. Therefore, the District respectfully requests that the requirement for mid-cycle inspection of these steam generators, as expressed in the letter from Mr. J. T. Collins to Mr. W. C. Jones dated June 22, 1984, be waived on the basis of the information provided in this submittal.

RL Andrews