ATTACHMENT 1

TECHNICAL SPECIFICATIONS

Remove Page	Insert Page
4.17-1	4.17-1

4.17 STEAM GENERATOR TUBING SURVEILLANCE

Applicability

Applies to the surveillance of tubing of each steam generator.

Objective

To ensure integrity of the steam generator tubing through a defined inservice surveillance program, and to minimize exposure of personnel to radiation during performance of the surveillance program.

Specification

4.17.1 Examination Methods

Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

4.17.2 Acceptance Criteria

The steam generator shall be considered operable after completion of the specified actions. All tubes examined exceeding the repair limit shall be repaired by sleeving or rerolling or removed from service (e.g., plugged, stabilized).

For Units 1 and 3, there are a number of steam generator tubes which exceed the tube repair limit as a result of tube end anomalies. These tubes are temporarily exempted from the requirement for sleeving, rerolling or removal from service, until repaired during or before the next Unit 1 and Unit 3 refueling outages (Unit 1 EOC 18, and Unit 3 EOC 17 refueling outages, respectively). An analysis has been performed which confirms that operability of Units 1 and 3 will not be impacted with these tubes in service until the next refueling outage on each of these units.

4.17.3 Selection and Testing

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.17.1. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.17.4 and the inspected tubes shall be verified acceptable per Specification 4.17.5. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators, with one or both steam generators being inspected. The tubes selected for these inspections shall be selected on a random basis except:

a. The first sample inspection during each inservice inspection of each steam generator shall include:

1. All tubes that previously had detectable wall penetrations (>20%) and have not been plugged or sleeve repaired in the affected area.

2. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

3. A tube adjacent to any selected tube which does not permit passage of the eddy current probe for tube inspection.

ATTACHMENT 2

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ATTACHMENT 3

TECHNICAL JUSTIFICATION

Background

During the recent Unit 2 refueling outage, operating experience data based on events at Arkansas Nuclear One (ANO) were received by the Duke Steam Generator Engineering staff. This information indicated that previous eddy current indications classified as tube end anomalies (TEA's) had exhibited primary-to-secondary leakage at ANO, thus indicating they were in the pressure boundary. Subsequent evaluation of the ANO data by Duke analysts indicated a potential for indications to extend into the pressure boundary based on analyst guidelines which were in effect at the time of the original analysis. The guidelines were not specific in identifying the landmarks that should be used to determine that indications were outside the pressure boundary. A review of the eddy current data during the later stages of the Unit 2 outage identified some indications that were reclassified from TEA's to repairable indications. These tubes were included in the reroll repairs performed during the Unit 2 outage.

Based on this new information, Duke initiated a Problem Investigation Process (PIP) report on May 6, 1998. Engineering immediately began to assess the operability implications of this information with respect to Units 1 and 3. An evaluation of the results of the previous steam generator inspection results from the refueling outages on Units 1 and 3 indicated a number of tubes with TEA indications that were not repaired during those respective outages. The operability evaluation for Units 1 and 3 conservatively assumed that all the identified TEA's would result in leakage at rates determined by previous measurements on mockups performed by Framatome Technologies (FTI). The operability evaluation, completed on May 9, 1998, concluded that the predicted leakage was well below the leakage assumed in design basis steam line break accident analysis.

An action plan was identified to initiate a reanalysis of the Units 1 and 3 data obtained during the previous outages to establish the extent of the TEA indications. The action plan used the operating experience based on the ANO indications and results of steam generator inspections during the recent Unit 2 refueling outage. The following steps were employed to analyze the issue of TEA indications for Units 1 and 3.

1) FTI Constructed Mock Up for Eddy Current Evaluation
2) Developed Analysis Guidelines
3) Selected Analyst
4) Trained Analyst
5) Conducted Site Specific Testing
6) Performed Review of eddy-current test (ECT) Data for Units 1 and 3
7) Resolved any indications as result of review
8) Corrected database
9) Completed leakage evaluation for any remaining indications

While several activities were performed in parallel, the controlling activity for completing the data review was completion of the mock up. This detailed mockup of the upper tube sheet and clad was constructed with tubes installed with a geometry identical to the ONS steam generators. Machined defects were included in identified areas. This mockup was used to verify eddy current analyst guidelines were appropriate and comprehensive. The results of this reanalysis indicate that 372 indications out of 2,951 TEA's not previously repaired for Unit 1 and 61 out of 66 TEA's not previously repaired on Unit 3 extended beyond the upper surface of the tube sheet clad. These indications would have met Duke's criteria for repair during the outage by reroll. Ultimately, confirmation of ECT indications in the rolled area that met the repair criteria prompted this Notice of Enforcement Discretion (NOED) request. Based on preliminary results from the review, the Operations Shift Manager was briefed on June 2, 1998, regarding the revised inspection results for Units 1 and 3. The OSM logged the missed surveillance at 1715 hours on June 2, 1998 and Engineering completed its evaluation later that evening.

Evaluation

The steam generators are QA condition 1 components. The steam generators serve as part of the RCS pressure boundary and must meet the leakage requirements of the ONS Technical

Specifications. The pressure boundary function of the steam generator tubes, particularly the upper tubesheet rolled joints, have been evaluated for continued operation of Units 1 and 3 within their present conditions. These TEAs are eddy current testing calls for indications at the hot leg tube end, typically between the primary face cladding and the protruding tube end. TEA indications can only be identified when the upper tubesheet roll joints are inspected with a rotating pancake coil (RPC) eddy current probe or a plus-point eddy current probe. To determine if indications may exist in the Unit's 1 and 3 steam generators, a review of the ONS-1 EOC-17 and ONS-3 EOC-16 steam generator eddy current testing inspections was conducted and an operability evaluation was performed. TEA's are characterized as axial eddy current testing indications at the end of the tube outside of the pressure boundary. Some TEA's in the Unit 1 and 3 steam generators may be in the pressure boundary, which means they must be reclassified as repairable indications. The leak rates from repairable indications are considered in the following analysis.

<u>Reanalysis Results</u>

The review of the TEA's for Oconee Units 1 and 3 is complete using clarified analysis guidelines. The results are as follows with all flaws in the clad except where noted:

<u>ONS3 SG "A"</u>

3 TEA's were reported during the EOC-16 RFO
2 are single axially oriented indications (SAI)

<u>ONS3 SG "B"</u>

63 TEA's were reported during the EOC-16 RFO
55 are single axially oriented indications (SAI)
4 are multiple axially oriented indication (MAI),

<u>ONS1 SG "A"</u>

1,020 TEA's were reported during the EOC 17 RFO
221 are multiple axial indication (MAI)
    1 MAI in the carbon steel with four indications

96 are single axial indication (SAI)
1 is a single circumferentially oriented indication (SCI),
1 is mixed mode indications(MMI)
10 stayed as TEA's

<u>ONS1 SG "B"</u>

1,931 TEA's were reported during the EOC-17 RFO
1,817 remained as TEA's
35 are single axial indications (SAI)
  1 SAI is in the carbon steel at the roll transition
3 are single circumferential indications (SCI)
5 are mixed mode indications (MMI)
10 are multiple axial indication (MAI)

FTI has performed testing for Arkansas Nuclear One to measure the leakage from axial electrical discharge machining (EDM) notches (0.25 inches x 0.005 inches) in typical rolled joint mockups at 2500 psi at room temperature. The resulting leakage was 0.0066 cubic inches per hour. Tubes were also tested in typical roll mockups with a 360^{0} sever in a ¾ inch deep roll with a resulting leakage of 0.02 $inches^{3}$/hour at room temperature at 2500 psi. Note that room temperature leak tests are conservative since higher temperatures increase the joint tightness due to thermal expansion differences between the I600 tube and the carbon steel tubesheet. Additionally, when the primary system saturation pressure is larger than the actual secondary pressure, the leak flow will flash and the leak rate will be limited to a critical value by a choking process. The leakage from these size notches are conservative due to the width of the notches with respect to cracks.

For this review the indications are conservatively combined for a total leakage value through one steam generator. Current TEA's will not be considered for leakage because they are outside the pressure boundary.

The first consideration is tube burst. The indications are contained within the tubesheet. The tubesheet provides reinforcement to any indication. Therefore, the tubes are not expected to burst under normal and accident conditions. The tube integrity requirements are satisfied.

The second consideration is tube leakage at main steam line break loading conditions. By inspection, Oconee Unit 1 is the most limiting case with respect to leakage due to the number of indications present.

Leakage Potential

Circumferential Indications
Four circumferential indications were identified in th review. These four indications are located interior to the bundle. Therefore, tubesheet hole dilation will not be considered. They have been reported as having circumferential extents of less than 41^{0} and located in the clad of the tubesheet. For leakage, they will be considered 360^{0} and the FTI test results will be utilized.

Mixed Mode Indications
There have also been 6 mixed mode indications identified. The data has again been reviewed to determine if they are clear axial or have potential circumferential involvement. For this analysis they will be considered as circumferential indications.

(4 SCI + 6 MMI) X 0.02 inches^3/hour X (2575/2500)^2/231 inches^3/gallon / 60 minutes /hour = 0.000015 gpm

Indications in carbon steel
Two axial indications have been identified in the carbon steel region of the tubesheet. The axial extent of these indications are 0.2 inch and 0.06 inches long. One of these indications is located in the roll transition. From the EPRI Primary Water Sress Corrosion Cracking report the tube is not going to burst due to the reinforcing effect of the tube sheet and is contained within the tubesheet. These indications are not expected to grow to a point of concern with an industry growth rate of 0.03 inches/EFPY. Leakage will be considered for accident conditions for these indications at the rate identified in recent in situ pressure test results for an indication in the tubesheet. This leaking tube had a leak rate of 0.006 gpm to 0.01 gpm at main steam line break (MSLB) temperature and pressures.

The axial flaw length was 0.33 inches. The leak rate will be assumed to be 0.01 gpm.

2 indications X 0.01 gpm = 0.02 gpm

Multiple Axial Indications
Multiple axial indication are considered by assuming that the number of indications around the circumference is four, which by inspection is conservative.

231 MAI's X 4 indication /tube = 924 axial indications

Single Axial indications
There are 129 indications to be considered.

(924+ 129) X 0.0066 inches^3/hour X $(2575/2500)^2$/ 231 inches^3/gallon / 60 minutes /hour = 0.00053 gpm

The total leakage for Unit 1 is 0.000015 gpm for circumferential indications + 0.00053 gpm from axial indications + 0.02 gpm from the roll transition flaw = 0.02 gpm

Additional main steam line break leakage from rerolls.

Unit 1:

4166 rerolls **X** 0.0055 inches^3/ hour at room temperature at 2330psi **X** $(2575/2320)^2$ **/** 231inches^3/gallon **/** 60 minutes /hour = 0.002 gpm

Unit 3:

0 rerolls =0 gpm

The total predicted main steam line break tube leakage is 0.023 gpm for Unit 1. The resulting leakage is significantly less than that assumed in the off site dose analysis of 0.7 gpm at room temperature for each unit. Since Unit 1 is bounding, both units meet the MSLB leakage requirements for steam generator integrity. Therefore, the analysis concludes that the steam generators are capable of performing their intended safety function during normal operation and postulated accident conditions.

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Pursuant to 10 CFR 50.91, Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated:

This evaluation addresses the potential effects of a missed surveillance and repair opportunity for steam generator tubes. As described in the technical justification, operating with some steam generator tubes with TEAs and repairable indications in Units 1 and 3 does not increase the probability of an accident evaluated in the SAR because this condition is not an accident initiator. There is no physical change to the plant SSCs or operating procedures. Neither electrical power systems, nor important to safety mechanical SSCs will be adversely affected. The steam generators have been evaluated as operable for normal and accident conditions. There are no shutdown margin, reactivity management, or fuel integrity concerns.

This activity will not adversely affect the ability to mitigate any SAR described accidents. The total evaluated main steam line break leakage from the areas evaluated is 0.023 gpm for Unit 1 which is the limiting unit. The resulting leakage was considerably less than that assumed in the off site dose analysis of 0.7 gpm for each unit. Therefore both Units 1 and 3 met the MSLB leakage requirements for steam generator integrity with no compensatory actions required. There is no adverse impact on containment integrity, radiological release pathways, fuel design, filtration systems, main steam relief valve setpoints, or radwaste systems.

There is no increase in accident initiation likelihood or consequences, therefore analyzed accident scenarios are not impacted.

(2) <u>Create the possibility of a new or different kind of accident from any kind of accident previously evaluated:</u>

There is no increased risk of unit trip, or challenge to the RPS or other safety systems. There is no physical effect on the plant, i.e., none on RCS temperature, boron concentration, control rod manipulations, core configuration changes, and no impact on nuclear instrumentation. There is no increased risk of a reactivity excursion. No new failure modes or credible accident scenarios are postulated from this activity. The MSLB screnario has been evaluated and the potential for damage to the steam generator tubes is not increased.

(3) <u>Involve a significant reduction in a margin of safety.</u>

No function of any important to safety SSC will be adversely affected or degraded as a result of continued operation. No safety parameters, setpoints, or design limits are changed. There is no adverse impact to the nuclear fuel, cladding, RCS, or required containment systems. Therefore, the margins of safety as defined in the bases to any Technical Specifications are not reduced as a result of this change.

Duke has concluded, based on the above, that there are no significant hazards considerations involved in this amendment request.

ATTACHMENT 5

ENVIRONMENTAL IMPACT ANALYSIS

Pursuant to 10 CFR 51.22 (b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22 (c) 9 of the regulations. The proposed amendment does not involve:

1) A significant hazards consideration.

   This conclusion is supported by the No Significant Hazards Consideration evaluation that is contained in Attachment 4.

2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

   The proposed amendment will not significantly change the types or amounts of any effluents that may be released offsite.

3) A significant increase in the individual or cumulative occupational radiation exposure.

   The proposed will not significantly increase the individual or cumulative occupational radiation exposure.

In summary, the proposed amendment request meets the criteria set forth in 10 CFR 51.22 (c) 9 of the regulations for categorical exclusion from an environmental impact statement.