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Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Notice of Enforcement Discretion (NOED) Request
Steam Generator Tubing Surveillances

Reference: Letter, WR McCollum (Duke) to USNRC, Steam Generator Tube End Anomalies - Interpretation of Technical Specification 5.5.10, § 8, dated September 7, 1999

Technical Specification (TS) 5.5.10 requires that steam generator tubes which exceed the repair limit shall be repaired by sleeving or rerolling, or shall be removed from service. Based on discussions with the NRC to resolve TS interpretation discrepancies described in the referenced letter, Duke Energy Corporation (Duke) has concluded Units 1 and 3 are operating contrary to the requirements of TS 5.5.10.

Thus, Oconee requests prompt support from the staff to avoid an unnecessary shutdown of Oconee Units 1 and 3. This request does not apply to Unit 2 since it is currently shutdown for a refueling outage. Oconee requests enforcement discretion apply to Units 1 and 3 until a license amendment can be submitted and approved by the staff which resolves the current situation. The details and justification of this request are provided in Attachment 1. Oconee is working diligently and expeditiously to prepare the LAR request to resolve this issue and will submit the proposed TSs to the staff on November 17, 1999. Attachment

A001

2 contains the preliminary proposed TS change marked up page.

Please address any questions to Larry Nicholson at (864-885-3292).

Very truly yours,

W. R. McCollum, Jr.
Site Vice President

Attachments (2)

xc: Mr. L. A. Reyes
Regional Administrator, Region II

Mr. M. C. Shannon
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Mr. D. E. LaBarge
ONRR, Project Manager

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Notice of Enforcement Discretion (NOED) Request

Oconee has used Administrative Letter 95-05 to develop this request for enforcement discretion. Relevant information supporting this request for enforcement discretion is provided below.

1. **Technical Specification (TS) that will be violated:**

TS 5.5.10, § e.6, requires in part:

> "Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unservicable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness."

Specifically, on November 15, 1999, at approximately 16:10, it was determined, based on discussions with the NRC, that a number of steam generator (SG) tube indications were not appropriately dispositioned. These indications, classified as tube end anomalies (TEA), are between the tube end and the primary face of the tubesheet cladding. These indications had not been previously repaired as a result of an interpretation that indications located between the end of the SG tube and the outer, primary side, of the SG tubesheet clad were outside of the pressure boundary. As a consequence, Duke did not either repair these indications during the prior refueling outages for Units 1, 2 or 3, or submit a License Amendment Request (LAR) to resolve the issue.

This request for enforcement discretion, if granted, would have the effect of permitting continued operation of Units 1 and 3 without having to repair SG tubes with TEAs. The enforcement discretion would continue until such time as the LAR to clarify TS 5.5.10 is submitted and approved by the NRC.

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2. **Circumstances surrounding the situation:**

In early May 1998, operating experience data based on events at Arkansas Nuclear One (ANO) were received by Duke Energy Corporation (Duke). This information indicated that previous Eddy Current (EC) indications classified as TEAs had exhibited primary-to-secondary leakage at ANO, thus indicating they were in the pressure boundary. As a consequence, in 1998, Duke redefined TEAs and developed an analysis methodology and guidelines capable of distinguishing between anomalies and indications in the tube sheet clad.

Discussions of the above issue with NRC personnel at NRR and Region II led Duke to incorrectly conclude that a consensus existed regarding axial TEA indications identified between the tube end and primary surface of the tube sheet clad. Specifically, Duke concluded that these indications were not part of the pressure boundary and could therefore be excluded from the TS inspection requirements. On September 7, 1999, Duke submitted the suggested interpretation letter which said, in part:

> "...the portion of a SG tube end that extends beyond the top of the cladding is not part of the pressure boundary since it is beyond the point of exit from the SG secondary side. The SG tube end beyond the top cladding is therefore excluded from the SG tube inspections."

On November 10, 1999, in a telephone conference with the NRC staff, the NRC suggested that the above position may be contrary to the requirements of TSs. Specifically, TS 5.5.10, SG Tube Surveillance Program, says in part:

> e.6. <u>Repair Limit</u> means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness.

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The NRC indicated that the above 40% repair criteria was applicable to the TEAs such that operation with the TEAs was contrary to TS. Oconee's TS require inspection of SG tubes from point of entry completely to the point of exit.

In a subsequent call with the NRC, on November 15, 1999, the NRC informed Duke that they had concluded that TS 5.5.10, § e.8 required the SG tubes to be inspected from point of entry completely to point of exit. Although Duke had inspected the SG tubes in Units 1, 2 and 3 completely from end to end, a methodology was established that allowed no repair of indications of potential SG tube defects between the tube end and the primary surface of the cladding.

3. **Safety Basis for Request**

Background

Inspections at plants with B&W designed steam generators have revealed indications near the ends of the expansion roll of the tubes' upper tube-to-tubesheet joints. These indications, originally defined as TEAs, were initially believed to be in the non-pressure boundary portion of the tube. During another B&W plant's outage of 1998, a reverse pressure bubble test identified leakage from two tubes, one in each generator. One of the leaking tubes at this plant had been identified as having a TEA during the previous inspection. Subsequent inspection of the leaking tubes using updated EC analysis guidelines indicated that some of these indications were in the portion of the tube adjacent to the tubesheet cladding and therefore are within the pressure boundary. The indications were identified as mixed mode cracking (axial and circumferential) in one steam generator and an axial crack in the other steam generator. Both of these indications were in the upper roll region just below the upper tubesheet seal weld. These crack-like indications located between the primary face of the tubesheet clad and the tubesheet clad to carbon steel interface are referred to as Tube End Cracks (TECs). These

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types of indications have been identified in NRC Information Notice 98-27, "Steam Generator Tube End Cracking" (dated July 24, 1998). It is noted that no laboratory examination data on tube-to-tubesheet rolled joints is available which would verify the indications are actually cracks. The indications are believed to be cracks based on the EC response and bubble test results.

During the spring 1998 outage at Oconee Unit 2, approximately 250 tubes with TEC indications were re-rolled. Based on re-analysis of EC data, some TEA indications were confirmed to extend below the primary surface of the tube sheet clad and into the clad and were in service on Unit 1 and Unit 3. These indications were re-classified as TEC indications. In 1998, Duke Energy Corporation requested and received both a Notice of Enforcement Discretion and a subsequent TS amendment from the NRC to allow the affected plants to operate the remainder of their current fuel cycles with TEC indications in service. The NRC granted these petitions with the commitment that the TEC indications would be repaired or plugged during the next outage of sufficient duration. All known TEC indications at Unit 1 and Unit 3 have since been repaired.

Characteristics of Tube End Cracks (TECs)

Based on a review of the EC data for tubes with TEC indications, the indications are typically characterized as crack-like and axially oriented. They are located between the primary surface of the tubesheet clad and the tubesheet clad to carbon steel interface. Circumferential indications have also been identified as well as a small number of volumetric indications. Multiple axial indications and combinations of axial and circumferential indications have also been identified.

Based on the EC data, the TECs are believed to initiate on the inside surface of the tube. They are typically short, axially oriented, and located in the rolled portion of the tube near the heat affected zone created by the tube-to-tubesheet weld. While no laboratory examination data on

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TECs is available, these indications have been verified as through wall cracks based on bubble tests performed at other B&W plants. The indications are believed to be cracks based on the EC response. The rolling and welding processes create residual stresses that may make the material more susceptible to Primary Water Stress Corrosion Cracking (PWSCC). For this reason, it is believed that the TECs are PWSCC initiated. The Oconee steam generators have been bubble tested during the last five refueling outages. Through wall TECs were detected in the Oconee Unit 1A OTSG. It should be noted that this OTSG has a flush welded upper tubesheet.

Characteristics of Tube End Anomalies (TEAs)

TEAs are axial ID indications located between the tube end and the primary face of the tubesheet clad. No laboratory examination data on TEAs is available and no TEAs have been identified with bubble testing. Therefore, these indications have not been confirmed to be actual degradation in the tube. The rolling and weld processes create residual stresses that may make the material more susceptible to Primary Water Stress Corrosion Cracking (PWSCC). For this reason, it is possible that the TEAs are PWSCC initiated. The Oconee steam generators have been bubble tested during the last five refueling outages for a total of ten tests. Leakage has not been identified from any TEA indication.

Safety impact of TEA indications

TEA indications are not predicted to contribute to primary-to-secondary leakage since the weld will remain functional and the TEA does not extend into the clad. A direct path from the primary side to the secondary side does not exist.

Primary-to-secondary leakage during normal operation is monitored according to the plants' TSs to ensure that any leakage remains less than the acceptable limit. Total operational leakage from TEAs left in service has remained well within the acceptable limits of the TSs and is zero. The presence of the tubesheet precludes the possibility of

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burst for tubes with TEAs. Therefore, the primary safety concern for primary-to-secondary leakage is postulated leakage during an accident condition. The axial loads and increased pressure differential during a postulated accident condition have the potential to increase the primary-to-secondary leakage, compared to the normal operating conditions. This can only occur if the TEAs that are left in service extend beyond the primary surface of the tubesheet clad. While growth into the clad and completely throughwall is not expected, the possibility can be assessed for potential impact.

If the TEAs are assumed to be due to PWSCC with the stress component being generated by the seal weld process, it is reasonable to assume that the growth rate of a TEA would be similar to that of a TEC. TEC growth rates were assessed in BAW-2346P Rev. 0 Alternate Repair Criteria for Tube End Cracking in the Tube-To-Tubesheet Roll Joint of Once Through Steam Generators. The growth rate indicated in the report is 0.0135 inches per 1.37 EFPY. Assuming a 0.0135 inch growth rate, 100 % throughwall, and no leakage reduction due to the tube to tubesheet roll, the leakage through these flaws can be conservatively calculated using the Tubeworks computer code. This code utilizes EPRI developed methodologies. This results in a Main Steam Line Break (MSLB) leakrate of 0.2 gpm for 10,000 indications.

Based on industry experience of Inconel 600 weld metal, it has been concluded that the tube-to-tubesheet weld (wire type 82T) is not likely to crack and that the weld would not be affected by the TEAs. If the tubing next to the seal weld is cracked due to PWSCC, the crack growth should slow as the remaining stresses (note the steam generators were full bundle stress relieved after the seal weld was installed) from the weld process are relieved. The crack is not expected to penetrate into the weld material. History of PWSCC failures of Inconel 600 material (control rod drive nozzles and pressurizer nozzles) in pressurized water reactors shows that the cracks typically initiate on the ID of the nozzle in the heat affected zone (created by the application of the weld) and propagates through the nozzle

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wall. The crack does not initiate at the weld. The crack initiates at the highest stress location and follows the stress profile. Based on industry experience, stress corrosion cracking (SCC) susceptibility of Inconel 600 materials can be ranked as follows:

1. Wrought Inconel 600 is most susceptible to SCC
2. Alloy 182 (wire type) weld metal is next
3. Alloy 82 (wire type) weld metal is least

The rankings provide additional evidence that the Inconel 600 tube is more likely to crack than the alloy 82 weld material.

The only known cracking of Inconel 600 weld material in pressurized water reactors has occurred in repair welds. The repair welds were installed as part of the replacement of an Inconel 600 nozzle that had failed due to PWSCC. For the repairs, portions of the original weld material were left in place. Although the cause of these weld failures is not known, it is believed the source may have been poor welds. There is also evidence the cracked welds were isolated to those repairs that used an alloy 690 nozzle.

This conclusion that the weld is not cracked is also supported by bubble tests of steam generator tubes performed at various plants. The tests clearly showed that the leakage was from a crack in the tube wall in or near the heat affected zone caused by the weld. There was no evidence of leakage from the weld and no evidence of cracks in the weld.

Summary

TEAs are indications in the tube end above the primary face of the clad. These indications are axial in nature. The true nature of these indications has not been determined but it is possible that the most likely cause is PWSCC.

TEAs that are left in service were evaluated for contribution to primary-to-secondary leakage during a

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postulated worse case accident condition. This contribution is expected to be zero. Burst and failure by bending are precluded by the presence of the tubesheet.

Accordingly, TEAs do not influence plant risk from a PRA perspective.

4. **The basis for the licensee's conclusion that the noncompliance will not be of potential detriment to the public health and safety and that neither an unreviewed safety question nor a significant hazard consideration is involved.**

Unreviewed Safety Question and No Significant Hazards Review:

1) **Increase the probability of an accident evaluated in the SAR?**

No. This evaluation addresses the potential effects of a missed repair opportunity for steam generator tubes. As described in the safety evaluation, operating with some steam generator tubes with TEAs in Units 1 and 3 does not increase the probability of an accident evaluated in the Safety Analysis Report (SAR) because this condition is not an accident initiator. There is no physical change to the plant Structures, Systems or Components (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. There is no increase in accident initiation likelihood, therefore analyzed accident scenarios are not impacted.

2) **Increase the probability of a malfunction of equipment important to safety evaluated in the SAR?**

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No. As described in the safety evaluation, operating with some steam generator tubes with TEAs in Units 1 and 3 does not increase the probability of a malfunction of equipment important to safety. As described above, there is no safety significance to the existence of these TEAs. This activity is not a test procedure and does not physically change or modify any plant system, structures, or components. The steam generators are QA condition 1 components that serve as part of the RCS pressure boundary and must meet the leakage requirements of the ONS Technical Specifications. Nothing is being done to inhibit their integrity or function. No valve manipulations, electrical alignments, or system configurations are required. No new hazardous materials or potential missiles are installed.

3) **Increase the consequences of an accident evaluated in the SAR?**

No. 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 zero. Therefore, 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.

4) **Increase the consequences of a malfunction of equipment important to safety evaluated in the SAR?**

No. No safety related or important to safety equipment necessary to place or maintain the plant in safe shutdown condition will be impacted by continued operation, absent the surveillance. As

described in the safety evaluation, operating with some steam generator tubes with TEAs in Units 1 and 3 does not increase the consequences of a malfunction of equipment important to safety. The total evaluated MSLB break leakage from the areas evaluated is zero. Therefore, 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.

5) **Create the possibility for an accident of a different type than any evaluated in the SAR?**

No. There is no increased risk of unit trip, or challenge to the Reactor Protection System or other safety systems. There is no physical effect on the plant, i.e., none on Reactor Coolant System 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 scenario has been thoroughly evaluated and the potential for damage to the steam generator tubes is not increased.

6) **Create the possibility for a malfunction of a different type than any evaluated in the SAR?**

No. There is no physical change to the plant SSCs or operating procedures. This change does not involve any plant changes, electrical lineups, or valve manipulations. No QA conditions or code requirements are degraded. No new equipment or components were installed. No credible new failures are postulated.

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7) **Reduce the margin of safety as defined in the bases to any Technical Specification?**

No. 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.

CONCLUSION

Based on the preceding evaluation, the noncompliance will not be of potential detriment to the public health and safety and neither an unreviewed safety question nor a significant hazard consideration is involved.

5. **The basis for the licensee's conclusion that the compliance will not involve adverse consequences to the environment.**

No environmental impact analysis is necessary since this request does not involve a significant hazards consideration, a significant change in the types/amounts of effluents that may be released offsite, or a significant increase in the individual/cumulative occupational radiation exposure.

6. **Any proposed compensatory measure(s).**

The site is not proposing any compensatory measures associated with this request for enforcement discretion. The non-compliance issue does not create any safety concerns that warrant compensatory measures as may be the case in other cases of enforcement discretion.

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7. **The justification for the duration of the noncompliance.**

Duke requests that enforcement discretion be granted for Oconee Units 1 and 3 until the license amendment is issued to the Technical Specifications. This license amendment will permit operation with unrepaired steam generator tubes with tube end anomalies that potentially meet the defect criterion as defined in Specification 5.5.10. Attachment 2 contains the proposed preliminary TS revised page. This revised page is in draft form and is subject to some minor revision. It is Duke's intent to submit this license amendment on November 17, 1999.

8. **A statement that the request has been approved by the Plant Operations Review Committee.**

On November 15, 1999, at approximately 16:10, the Plant Operations Review Committee completed its review and approval of this request for enforcement discretion.

9. **The request must specifically address how one of the NOED criteria for appropriate plant conditions is satisfied.**

Duke believes that this request satisfies the NOED criteria in that this request for enforcement discretion is necessary to avoid an undesirable shutdown. Compliance with TS 5.5.10 will require Oconee Units 1, and 3 to shut down and enter a steam generator outage to satisfy the surveillance requirements. As previously described, there is no safety significance associated with this compliance issue and requiring a shutdown of two units does not minimize potential safety consequences or operational risk.

On November 14, 1999, Duke concluded, based on conversations with the NRC regarding TS 5.5.10, that Units 1 and 3 were operating contrary to the SG tube repair criteria TS 5.5.10.

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Immediately upon reaching this conclusion, it was determined that Oconee Units 1 and 3 did not meet TS 5.5.10 in that these steam generator tubes had not been repaired or removed from service as required. Duke has acted timely and appropriately to resolve discrepancies between the language of TS 5.5.10 and various NRC letters to Duke. Duke's September 7, 1999 letter to the NRC attempted to resolve these discrepancies by summarizing Duke's conclusion from various NRC correspondence on the matter. This correspondence indicated that TEAs above the outer surface of the SG tube sheet outside the pressure boundary could be considered serviceable per TS 5.5.10

Therefore, no previous opportunity to identify the TS discrepancy existed. Thus, it is Duke's position that this issue fully conforms to the NOED criteria that have been issued by the staff.

10. **If a follow-up license amendment is required, the NOED request must include marked-up TS pages showing the proposed TS changes. The actual license amendment request must follow within 48 hours.**

A follow-up license amendment will be required to permit operation with unrepaired/unplugged steam generator tubes with tube end anomalies which potentially meet the defect criterion as defined in Specification 5.5.10. Attachment 2 contains the proposed preliminary TS revised page. This revised page is in draft form and is subject to some minor revision. It is Duke's intent to submit this license amendment on November 17, 1999.

ATTACHMENT 2

Preliminary Proposed Technical Specification Change
Marked-Up Pages

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube or a sleeve.

3. Degraded Tube means a tube or a sleeve containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the repair limit. A tube or sleeve containing a defect is defective.

6. Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness.

INSERT FROM FOLLOWING PAGE

The Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823P, Revision 1 will be used for sleeving repairs.

The rerolling repair process will only be used to repair tubes with defects in the upper tubesheet area. The rerolling repair process will be performed only once per steam generator tube using a 1 inch reroll length. The new roll area must be free of degradation in order for the repair to be considered acceptable. The rerolling process used by Oconee is described in the Topical Report, BAW-2303, Revision 3.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.10.d.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. The degraded tube above the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

Insert 1 to TS page 5.0-17

Axial tube imperfections observed beyond the outer, primary, surface of the tube sheet clad are excluded from this repair limit.