Entergy

Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802
Tel 479-858-4619

Dale E. James
Acting, Director,
Nuclear Safety Assurance

0CAN100403

October 28, 2004

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Response to Generic Letter 2004-01, Requirements for Steam Generator Tube Inspections
Plant Name Arkansas Nuclear One, Units 1 and 2
Docket Nos. 50-313 and 50-368
License No. DPR-51 and NPF-6

REFERENCES:

1. NRC letter from NRC dated August 30, 2004, *Generic Letter 2004-01, Requirements for Steam Generator Tube Inspections* (0CNA080412)

2. Entergy letter to NRC dated August 3, 2004, Once Through Steam Generator Inservice Inspection Report (1CAN080401)

Dear Sir or Madam:

Per Reference 1, the NRC issued Generic Letter 2004-01 regarding steam generator tube inspections. The NRC requested that all pressurized water reactors (PWRs) who have not ceased operation provide information within 60 days of the date of the generic letter regarding past and proposed practices on inspection of steam generator tubes using the most appropriate 10CFR50, Appendix B inspection methods. The response to the requested information for Arkansas Nuclear One (ANO), Units 1 and 2 is provided in the attachment to this letter.

Entergy is not making any commitments as a result of our response to this letter. If you have any questions or require additional information, please contact Steve Bennett at 479-858-4626.

A115

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 28, 2004.

Sincerely,

DEJ/sab

Attachment:: Response to Generic Letter 2004-01 for ANO-1 and ANO-2

cc: Dr. Bruce S. Mallett
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Arkansas Nuclear One
P. O. Box 310
London, AR 72847

U. S. Nuclear Regulatory Commission
Attn: Mr. Thomas W. Alexion
MS O-7 D1
Washington, DC 20555-0001

U. S. Nuclear Regulatory Commission
Attn: Mr. Drew Holland
MS O-7 D1
Washington, DC 20555-0001

Mr. Bernard R. Bevill
Director Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street
Little Rock, AR 72205

Attachment

to

Response to Generic Letter 2004-01 for ANO-1 and ANO-2

Response to Generic Letter 2004-01 for ANO-1 and ANO-2

NRC Generic Letter (GL) 2004-01 *Requirements for Steam Generator Tube Inspections*, dated August 30, 2004 was sent to all holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

The generic letter requested the following information within 60 days:

NRC Requested Information 1

Addressees should provide a description of the SG tube inspections performed at their plant during the last inspection. In addition, if they are not using SG tube inspection methods whose capabilities are consistent with the NRC's position, addressees should provide an assessment of how the tube inspections performed at their plant meet the inspection requirements of the TS in conjunction with Criteria IX and XI of 10 CFR Part 50, Appendix B, and corrective action taken in accordance with Appendix B, Criterion XVI. This assessment should also address whether the tube inspection practices are capable of detecting flaws of any type that may potentially be present along the length of the tube required to be inspected and that may exceed the applicable tube repair criteria.

ANO-1 Response:

Background

ANO-1 has two Babcock & Wilcox designed 177FA once through steam generators (OTSGs). Each OTSG contains 15,531 sensitized Inconel-600 tubes that have an outer diameter of 0.625 inch with a nominal wall thickness of 0.037 inch. Each tube is supported by 15 TSPs that are 1.5-inches thick carbon steel and have trifoil broached holes, except for the 15th TSP, which has drilled holes for the tubes at the outer periphery of the tube bundles. The upper and lower tube ends are roll-expanded to a minimum depth of 1.0 inch from the primary face of the tubesheet and a fillet weld exists between the primary face of the tubesheet and the tube end. A repair roll has been qualified for installation in the upper tubesheet or the lower tubesheet (LTS) to repair indications of primary water stress corrosion cracking (PWSCC) and/or intergranular attack (IGA). After installation, the repair roll becomes the new pressure boundary. The unit operates on an 18 month fuel cycle and the last inspection was during the unit's 18th refueling outage (1R18) in the spring of 2004.

Previous Inspection Information

The ANO-1 spring 2004 refueling outage steam generator tube inspection report was provided to the NRC in Reference 2 and is summarized in the following table. In addition to the technical specification (TS) inspection requirements, a degradation assessment was performed (using the EPRI *PWR Steam Generator Examination Guidelines* in effect at the time of the inspection, and available industry data for steam generators of similar design) to identify possible damage mechanisms that may exist in the OTSGs. Once the possible damage mechanisms were identified, qualified inspection techniques were used to inspect for flaws in the respective areas.

Item	Steam Generator Region	Inspection Probe	Inspection Scope and Extent
1	Full Length of Tube (Note 1)	Bobbin	100% - full length
2	Dents ≥ 2 Volts Superheat Region (Note 2)	Plus Point	100% down to coldest elevation with reported degradation plus 100% sample at next five elevations
3	Dents ≥ 2 Volts Lower Tubesheet Region (Note 2)	Plus Point	21% of the kidney region
4	Sludge Pile / Lower Tube Sheet Crevice / Kidney Region *[LTS face +4" to -4"]*	Plus Point	21% of Sludge Pile
5	Upper Tube Ends, Upper Original Roll Transition, or Lowest Repair Roll Transition (Note 3)	Plus Point	100% of the upper tubesheet tube ends (ARC), original rolls and re-rolls
6	Lower Tube Ends, Lower Original Roll Transition, or Highest Repair Roll Transition (Note 3)	Plus Point	50% Radius > 41 inches 20% Radius < 41 inches
7	Upper Tubesheet	Plus Point	100% recorded indications and new indications from Bobbin
8	Bobbin Indications (Note 4)	Plus Point	100% Non-Quantifiable Indications (I-Codes), wear, impingement, and potential loose parts from Bobbin
9	Alloy 600 Sleeves – Upper Roll Expansion	Plus Point	100% In-service sleeves
10	Alloy 600 Sleeves – Lower Roll Expansion	Plus Point	100% In-service sleeves

Notes for table:

1. Full-length of the tube is defined as: completely from point of entry to point of exit. The previously existing tube and tube roll, outboard of a new roll area in the tube sheet, is excluded from future periodic inspection requirements because it is no longer part of the pressure boundary after a repair roll is installed.
2. ANO-1 does not use the "ding" nomenclature; all indications of mechanical tube deformation are called "dents."
3. Inspection of tube ends and original roll transition is not required in a tube that has a repair roll installed. Plus Point inspection to 1inch beyond inboard roll transition is required.
4. Plus Point probe inspection was performed on bobbin coil indications having possible degradation, all recorded permeability variation indications, all recorded pilgering process indications, and all recorded dent indications. Wear indications on bobbin coil inspection were confirmed with Plus Point.

Findings

During the 1R18 refueling outage in the spring of 2004, Entergy performed a supplemental inspection of the lower tubesheet for tube end cracking (TEC) as a result of industry information at two other OTSG plants. Cracking had been identified in the lower (cold leg) side of the tubesheet with the predominance of these being in the outer periphery. Entergy did not believe that ANO-1 had a similar concern regarding extensive cracking of the lower tube ends. A sampling of the lower tube ends was performed using the Plus Point probe in both OTSGs as discussed in the response to Question 3. A 50% sample (100% of half of the SG) was conducted on the outer periphery of each OTSG from 41 inches out. A smaller 20% sampling of the tubes on the inner 41 inch radius was conducted since industry data did not identify significant central tube cracking. The ANO-1 inspection in 1R18 identified only four total TECs in the outer periphery of the "B" SG and none in the "A" SG. This confirmed that ANO-1 did not have a similar concern with tube end cracking. None of the TECs extended into the cladding (pressure boundary) however, they were determined to have circumferential extent. These tubes were plugged [Entergy has an Alternate Repair Criterion (ARC) for TECs in the lower tubesheet but is only applicable for axial flaws]. A further sampling (expansion) was not conducted since this inspection was not performed for compliance to the TSs and the indications found did not pose a substantial leakage or a safety concern. A deviation from the EPRI steam generator guidelines was documented in the ANO-1 SG 1R18 inspection findings as allowed by NEI 97-06, *Steam Generator Program Guidelines*. The Cycle 19 Operational Assessment included leakage from comparable findings that may have existed in the other 50% sample of the "B" OTSG.

Conclusion:

The inspection method in the lower tubesheet for detection of tube end cracking is conservatively concluded to be inconsistent with the NRC's position even though the ANO-1 condition is not discussed in the considerations for GL 2004-01. Entergy did use the appropriate 10CFR50, Appendix B inspection methods (Plus Point) and did not limit our inspection to particular tubesheet depth. However, TEC indications were identified in the initial 50% sample of the "B" SB and similar results could be concluded to exist in the remainder of the "B" SG. Therefore, Entergy is reporting this information as if it does not meet the NRC position for tube inspections within the generic letter.

ANO-2 Response:

Steam generator tube inspections performed at ANO-2 are consistent with the NRC's position regarding tube inspections.

The ANO-2 steam generators are U-Bend Westinghouse Delta 109 design. Each steam generator contains 10,637 tubes made from thermally treated Alloy 690 material with an outside diameter of 11/16 inches and a wall thickness of 0.040 inches. The first 17 rows had the U-Bend area stress relieved after bending. The tubes are hydraulically expanded the full depth of the tube sheet. There are eight full flat-contact trifoil broached tube support plates (TSPs) constructed of Type 405 ferritic stainless steel. Five sets of Type 405 stainless steel anti-vibration bars are assembled in the U-bend region of the tubes to provide support in the regions susceptible to degradation due to U-bend vibration and wear.

The ANO-2 steam generators were replaced in 2000. A baseline inspection of the SG tubes was conducted after the first operating cycle (2R15). Entergy was granted an amendment to the TSs from performing a SG tube inspection in the refueling outage (2R16) following the second operating cycle. ANO-2 uses operating plant experience and engineering analysis to determine where specialty probes, such as the Plus Point probes are appropriate. The specialty probes are used in those areas that have a potential for degradation or to resolve bobbin coil data. In the previous refueling outage in the spring of 2002 (2R15), ANO-2 performed the following tube inspection scope in both steam generators with no flaws detected:

- 100% full length bobbin inspection (except R1 and R2 U-bends)
- 100% bobbin in straight sections of R1 and R2
- 100% small radius (Row 1 and Row 2) U-bend with the Plus Point probe
- Plus Point examination of all "I-code" indications that were new or changed from baseline

As a result of the 2R15 tube inspection, the ANO-2 thermally treated Alloy 690 tubing has not experienced any PWSCC or other SG tube related degradation.

<u>Conclusion</u>:

Entergy is consistent with the NRC position provided in GL 2004-01.

Requested Information 2

If addressees conclude that full compliance with the TS in conjunction with Criteria IX, XI and XVI of 10 CFR Part 50, Appendix B, requires corrective actions, they should discuss their proposed corrective actions (e.g., changing inspection practices consistent with the NRC's position or submitting a TS amendment request with the associated safety basis for limiting the inspections) to achieve full compliance. If addressees choose to change their TS, the staff has included in the Attachment suggested changes to the TS definitions for a tube inspection and for plugging limits to show what may be acceptable to the staff in cases where the tubes are expanded for the full depth of the tubesheet and where the extent of the inspection in the tubesheet region is limited.

ANO-1 Response:

As noted in the response to NRC Request 1 above, Entergy has concluded that the previous inspection conducted was not consistent with NRC's position defined in the generic letter. ANO-1 has documented the condition in our corrective action program. However, Entergy will be replacing the existing OTSGs with thermally treated Alloy 690 tubing in the next refueling outage in the fall of 2005. Entergy does not need to take any exception to the TSs or 10CFR50, Appendix B for the replacement steam generators. The results of the safety assessment have demonstrated SG operability is maintained for the remaining life of the SGs per response to question 3 below.

ANO-2 Response:

Steam generator tube inspections performed at ANO-2 are consistent with the NRC's position regarding tube inspections; therefore, this question does not apply.

Requested Information 3

For plants where SG tube inspections have not been or are not being performed consistent with the NRC's position on the requirements in the TS in conjunction with Criteria IX, XI, and XVI of 10 CFR Part 50, Appendix B, the licensee should submit a safety assessment (i.e., a justification for continued operation based on maintaining tube structural and leakage integrity) that addresses any differences between the licensee's inspection practices and those called for by the NRC's position. Safety assessments should be submitted for all areas of the tube required to be inspected by the TS, where flaws have the potential to exist and inspection techniques capable of detecting these flaws are not being used, and should include the basis for not employing such inspection techniques. The assessment should include an evaluation of (1) whether the inspection practices rely on an acceptance standard (e.g., cracks located at least a minimum distance of x below the top of the tube sheet, even if these cracks cause complete severance of the tube) which is different from the TS acceptance standards (i.e., the tube plugging limits or repair criteria), and (2) whether the safety assessment constitutes a change to the "method of evaluation" (as defined in 10 CFR 50.59) for establishing the structural and leakage integrity of the joint. If the safety assessment constitutes a change to the method of evaluation under 10 CFR 50.59, the licensee should determine whether a license amendment is necessary pursuant to that regulation.

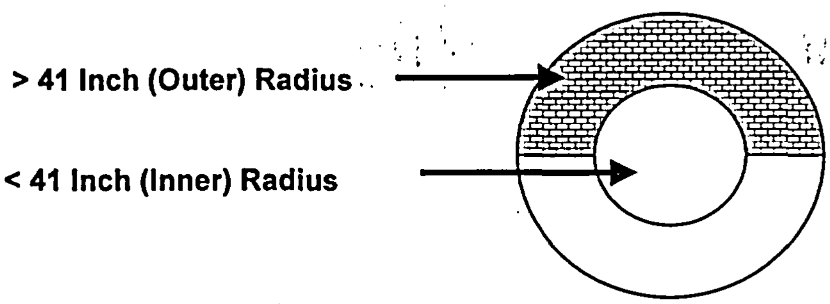
ANO-1 Response:

The following information provides the ANO-1 safety assessment for the TEC inspection location in the lower tubesheet or cold leg. The safety assessment provides adequate basis for SG operability and that ANO-1 is in compliance with the program elements of NEI 97-06.

Historically, ANO-1 has been tracking tube end cracking in the hot leg for several cycles. During the most recent inspection, a sample of the tube ends in the cold leg was tested with the Plus Point probe. Based on industry data, the following initial inspection was performed:

- 50% (100% of one half of the SG) of the tubes in the outer periphery (radius > 41 inches)
- 20% of the tubes in the inner section (radius < 41 inches)

The inspection is depicted in the following diagram:



No cracking was identified in the inner section (radius < 41 inches) but four circumferential indications were identified in the 50% sample in the "B" steam generator. The following table lists the indications:

Row	Tube	Arc	Depth (in.)	Radius	Assigned Leakage (gpm)
115	4	0.19	50	55.36	0.000823
125	2	0.24	72	56.39	0.00137
136	78	0.20	97	55.82	0.000823
137	75	0.21	97	55.69	0.000823

<u>Safety Assessment:</u>

Since there is no concern associated with burst, leakage was addressed in the operational assessment by extrapolating the un-inspected population in the remaining 50%. First, the number of indications was doubled to represent the un-inspected 50% of the tube ends. Additionally, a probability of detection (POD) adjustment of 18% associated with the ARC used in the upper tubesheet was applied. The worst case leakage (tube 125-2) was conservatively used for all indications.

(4 indications in initial 50%) X (2 – remaining 50% un-inspected) X 1.18 = <u>10 total indications</u>

There was a 30% increase in the number of new tube end cracks in the upper tubesheet so the value was increased by 30% for added conservatism:

10 indications X 1.3 = 13 indications X 0.00137gpm/indication = <u>0.018 gpm</u>

This estimated leakage of 0.018 gpm at steam line break conditions was added to the total leakage for all mechanisms which is equal to 0.776 gpm. The site limit is 1.0 gpm. Therefore, the SGs are operable with adequate safety margin still available. The safety assessment does not constitute a change to the "method of evaluation" (as defined in 10CFR50.59) for establishing the structural and leakage integrity of the joint.

ANO-2 Response:

Steam Generator tube inspections performed at ANO-2 are consistent with the NRC's position regarding tube inspections; therefore, this question does not apply to ANO-2.