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1CAN030402

March 1, 2004

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

Subject: Response to NRC Request for Additional Information Regarding ANO-1 Steam Generator Tube Inservice Inspection Report from 1R17
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

REFERENCES:

1. Entergy letter dated January 17, 2003, *Once Through Steam Generator Inservice Inspection Report* (1CAN010301)
2. NRC letter dated March 28, 2001, *Arkansas Nuclear One, Unit No. 1 - Issuance of Amendment Re: The Use of the Reroll Repair Process for Steam Generator Tubes* (TAC NO. MB0097) (1CNA030105)

Dear Sir or Madam:

On January 17, 2003, Entergy transmitted the 1R17 Once Through Steam Generator Inservice Inspection Report (Ref. 1) which included the reporting of the calculated maximum total best-estimate large break loss of coolant accident (LBLOCA) leakage of 1.87 gpm. The leakage values and acceptance criteria assigned to flaws were established based on the Entergy commitment referenced in the ANO-1 Safety Evaluation Report dated March 28, 2001 (Ref. 2) which states:

> *Entergy will verbally notify the NRC of... determination of the best-estimate total leakage that would result from an analysis of the limiting LBLOCA based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet reroll repairs, and heat affected zones of seal welds as found during each inspection.*

On December 5, 2003, and January 29, 2004, Entergy received requests for additional information (RAIs) from the NRC Staff regarding the content and methods used in determining the best estimate leakage values and acceptance criteria. Draft responses were discussed with members of the NRC Staff during several conference calls. The responses to these RAIs are contained in the attachment to this letter.

This letter contains no commitments. If you have any questions or require additional information, please contact Steve Bennett at (479) 858-4626.

A047

Sincerely,

DEJ/sab

Attachment: Response to NRC Request for Additional Information (RAI) on ANO-1 Steam Generator Inservice Inspection Report from 1R17

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Attachment

1CAN030402

Response to NRC Request for Additional Information (RAI) on ANO-1 Steam Generator Inservice Inspection Report from 1R17

Response to NRC Request for Additional Information (RAI) on ANO-1 Steam Generator Inservice Inspection Report from 1R17

Response to NRC RAIs of December 5, 2003

Reference: Letter (1CAN010301) dated January 17, 2003 from Sherrie R. Cotton, Entergy Operations, Inc. to NRC transmitting the 1R17 Once Through Steam Generator Inservice Inspection 90 Day Report.

1. In Table 2.1 of the referenced report, volumetric indications are reported at the lower re-roll transitions. What is your assessment concerning the defect mechanism and cause of these indications? Were these indications present during previous inspections or are they new indications? If these volumetric indications are potentially intergranular attack (IGA) related, why are these indications considered a separate population from those indications labeled in Table 2.1 as "volumetric IGA indications in the UTS" which you have shown are not exhibiting growth at the present time?

   Response: The cold working of the steam generator (SG) tubing appears to have caused several of the "rerolled" tubes to crack after a relatively short service time. The indications appear after heatup and cooldown cycling of the once-through steam generators (OTSG) tubing. The primary area where flaws are noticed are in the heel of the re-roll which is outside the pressure boundary. The indication of cracking is not present immediately following the installation of the re-roll since all re-rolls have a baseline bobbin and motorized rotating pancake coil (MRPC) examination performed prior to being placed into service. Entergy also looks at the data in the tubing to ensure no IGA or indications are present in the area of the re-roll prior to installation of the re-roll. The volumetric indications are not included in the alternate repair criteria (ARC) since that population (reroll tubing) was not included in the approved IGA ARC. Entergy believes this population of tubes should be kept separate.

2. In Table 2.1, please provide a breakdown of "upper roll/transition cracking" in terms of number of axial and circumferential indications. Similarly, please provide a breakdown of "re-roll cracking - Upper Transition (OPB)" and "re-roll cracking - other re-roll indications within the pressure boundary" in terms of the number of axial, circumferential, and volumetric indications."

Response:

SG Tube Location	A SG	B SG
Upper roll/transition cracking	40 total (38 axials, 2 SVI)	41 total (1 circ, 35 axial, and 3 SVI) *[39 actual total (One TEC and one reroll heel transition crack were counted twice)]*
Upper Transition (OPB)	1457 total (200 circs, 1252 axials, and 5 vols)	1094 total (106 circs and 988 axials)
Lower Transition Toe	11 vols	10 axials – 1 vol
Other reroll indications (IPB)	14 vols	3 vols

Legend: SVI - single volumetric indication; TEC - tube end crack; Axial – axially oriented flaws; IPB – inside pressure boundary; Circ - circumferentially oriented flaws; Vol – volumetric indications; OPB – outside pressure boundary

3. Table 3.1 refers to "TSP cracking circumferential" for which 0.025 gallon per minute (gpm) leakage is projected for the end of the current operating cycle. Table 2.1 makes no mention of this circumferential cracking mechanism at the tube support plates, nor is there any discussion of this mechanism in the report. Were any circumferential indications identified during 1R17, apart from those at the tube ends, tube hard rolls, or tube re-rolls? If so, provide the number, size, and location of these circumferential indications.

   Response: The tube support plate (TSP) cracking circumferential was included as a conservative assessment for the end of cycle leakage. No circumferential flaws were found during 1R17 apart from that mentioned above.

4. Tables 2.3 and 2.4 report the condition monitoring leakage estimates for the upper tubesheet TEC. Table 2.9 reports the condition monitoring leakage estimates for upper tubesheet IGA. Were there other mechanisms that also contributed to total condition monitoring estimate of accident induced leakage? If so, what were the contributions from these other mechanisms? What was the condition monitoring estimate of total accident induced leak rate from all mechanisms?

Response: PWSCC flaws from the upper roll transition were also included. The total 1R17 SG condition monitoring leakage is as follows

SG Tube Location	A SG	B SG
UTS IGA ARC	0.11 gpm	0.10 gpm
Upper Roll PWSCC	0.12 gpm	0.11 gpm
HAZ TEC ARC	0.55 gpm	0.50 gpm
TOTAL	**0.78 gpm**	**0.71 gpm**

Legend:
UTS – upper tubesheet
IGA - intergranular attack
TEC – tube end cracking
HAZ – heat affected zone
PWSCC – primary water stress corrosion cracking
ARC – alternate repair cracking

5. The January 17, 2003, letter reports that the calculated maximum total best estimate LBLOCA leakage is 1.87 gpm. Describe the basis by which this leakage was determined to be acceptable; i.e., that this best estimate leakage would not result in a significant increase of radionuclide release (e.g., in excess of 10 CFR 100 limits). In addition, please provide a summary of the assessment performed for the circumferential cracks found during 1R17 in the original tube-to-tubesheet rolls, tube-to-tubesheet re-roll repairs, and heat affected zones of seal welds to establish their contribution to the calculated 1.87 gpm leakage.

Response: The leakage for the first two minutes of the event was determined to be 1.87 gpm and 1.46 gpm for the remaining 30 days.

The LBLOCA "best estimate" leakage calculation is accomplished using a detailed spreadsheet during the outage. The location of the flaw and details of the flaw are input into the spreadsheet once the circumferential flaws are "sized." These details, along with the structural integrity of the flaw, help determine the actual leak rate. Most of the circumferential flaws found during 1R17 were located above a reroll in the upper tubesheet. Only one circumferential flaw was found in any other location which was located at the upper roll transition in the "B" OTSG. That flaw contributed 1.47 gpm to the "best estimate" 2-minute leakage. All other flaws contributed the remainder of the 0.40 gpm.

See additional discussion in responses below regarding leakage acceptability.

Response to NRC RAIs of January 29, 2004

1. What is the acceptance criterion that is being used?

Response: The acceptance criteria established for the "best estimate" large break loss of coolant accident (LBLOCA) from the ANO-1 once through steam generators is 9.0 gpm for the initial 2 minutes and 3.0 gpm for the remaining 30 days. This is the bounding assumption used in the offsite and control room (CR) dose calculations.

The leakage values and acceptance criteria assigned to flaws are in accordance with the ANO-1 Safety Evaluation Report (SER) dated March 28, 2001 which states:

> *Determination of the best-estimate total leakage that would result from an analysis of the limiting LBLOCA based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet reroll repairs, and heat affected zones of seal welds as found during each inspection.*
>
> *For the purpose of this evaluation, "acceptable leakage" means the best estimate leakage (in the event of a LBLOCA) would not result in a significant increase in radionuclide release (e.g., in excess of 10 CFR Part 100 limits). In addition, the leakage would not be expected to jeopardize long-term operation of the ECCS or accident management (e.g., control room habitability issues such as those in addressed in GDC 19).*

The commitment to perform a best estimate LBLOCA leakage assessment was based on the interim need to assure that flaws removed from the pressure boundary by reroll repairs would not have created unacceptable LBLOCA doses. The NRC SER also states:

> *These regulatory commitments will ensure that the licensee will perform an adequate evaluation to demonstrate that gross structural failure and leakage of the reroll repair joints will not occur in the event of a LBLOCA pending the resolution of this issue during the review of BAW-2374. This evaluation will demonstrate that adequate safety margins and defense-in-depth are maintained in the design and installation of the reroll repairs at the ANO-1. Entergy recognizes that further NRC review of BAW-2374 may require it to modify the regulatory commitment or otherwise involve additional actions to comply with the final NRC conclusion on the topical report. The staff has concluded that adequate controls for these actions are provided by the licensee's commitment management program and that additional regulatory requirements, unless defined as a result of the continuing review of BAW-2374, are not warranted.*

Therefore, the reporting of a "best estimate" LBLOCA leakage is not required to comply with the ANO-1 licensing basis, but is to provide interim assurance for maintaining doses within Part 100 limits if circumferential flaws are identified in the steam generator tube pressure boundary.

2. What is the basis / derivation of the acceptance criterion?

Response: The most significant parameter for tube leakage under LBLOCA conditions is the tube tensile load created by the tube to tubesheet temperature differences when the tubesheet dilation occurs. The large axial tube loads pose a concern to tubes found to have circumferential cracks. Therefore, the predominant contributor to LBLOCA leakage was from circumferential cracks. A small amount of leakage was also assigned

to repair products (i.e., sleeves & plugs) from 1R17. From the initial 1R16 LBLOCA best estimate leakage, Entergy chose a bounding "best estimate leakage acceptance criteria, which was conservatively assumed to be 9 gpm for two minutes (duration of maximum tubesheet bore dilation) and 3 gpm for the remainder of the 30-day event.

The actual ANO-1 "best estimate" LBLOCA leakage values determined for 1R17 were:

	2 min.	30 day
1R17	1.87 gpm [1]	1.46 gpm

[1] - Only the 2 minute leakage value was reported in the January 17, 2003 OTSG ISI report for 1R17.

3. If the acceptance criterion is expressed in terms of post-accident dose or if the acceptance criteria was back-calculated from dose, please explain the significant assumptions and inputs that are used to calculate the dose from the measured leakage and vice versa.

   Response: See response to RAI 4 below

4. Are the methodology, assumptions and inputs of these analyses consistent with regulatory guidance for design basis accidents?

   Response: The most probable response to a LBLOCA would be that the main steam isolation valves (MSIVs) would close and isolate the secondary side. Since the main steam safety valves would remain closed as the primary and secondary depressurize, only a very small amount of the primary leakage would escape into the environment. Therefore, it is expected that there would not be significant additional dose above that calculated for the licensing basis.

   As a defense in depth measure, however, a bounding best-estimate assessment was performed of the primary to secondary leakage which might occur due to SG tube loads induced by the LBLOCA transient without crediting MSIV closure. As stated in the SER dated March 28, 2001, the leakage to be evaluated is the "total leakage that would result from an analysis of the limiting LBLOCA". The limiting LBLOCA for calculating tube loads is a break in the candy cane portion of the system. This break location is well above the top of the core which limits fuel damage. This break location is also bounded by the location assumed in the 10CFR50.46 LBLOCA licensing basis analysis which occurs at the RCP discharge where only part of the cladding is breached. Therefore, the dose assumptions used for this analysis are the same as those established in the ANO-1 Safety Analysis Report (SAR) 14.2.2.5.6 for estimating LBLOCA doses with the total release of the fuel gap inventory. The SAR dose results bound the best estimate LBLOCA doses. Please note that this is not TID-14844 source term which was assumed for the MHA (Maximum Hypothetical Accident). The MHA is a non-mechanistic worst case accident. Therefore, using the gap inventory as a source term is conservative.

The offsite doses calculation conservatively assumes that 10% of the inventory that leaks into the secondary from the reactor coolant system (RCS) is released to the environment. This fraction is consistent with assumptions described in the bases of the Unit 1 Technical Specifications and is considered conservative here given the depressurization of the RCS and secondary side. Consistent with the evaluation of the dose from emergency core coolant system (ECCS) leakage, it is assumed that only iodine is in solution in the sump inventory. The noble gases in the reactor building (RB) have already been accounted for in the LBLOCA dose. Therefore, only the thyroid dose was assumed for the purposes of this analysis. The same diffusion/dispersion rates (X/Qs), breathing rates, and dose conversion factors were used for the offsite dose consequences calculation. Control room doses were similarly developed.

The control room doses for a LBLOCA were not reported in the SAR. Using the same inputs for the control room doses as for the MHA and replacing the MHA source term with the LOCA source term reported in SAR 14.2.2.5.6 resulted in a CR thyroid dose of 1.21 Rem. This case was then re-evaluated for the primary to secondary leakage. The analysis for the 9 gpm leakage (2 minute) and 3 gpm leakage (30-day) acceptance criteria result in the following doses:

	EAB	LPZ	CR Thyroid
SAR Reactor Building Leakage Dose (rem)	7.01	2.66	1.21[1]
Primary to Secondary Leakage (rem)	22.22	19.34	13.61
Total (rem)	**29.23**	**22.00**	**14.82**

[1] – In Entergy calcs, not reported in ANO-1 SAR.

The resulting exclusion area boundary (EAB) and low population zone (LPZ) doses from this leakage, when added to the doses reported in Table 14-49 of the ANO-1 SAR, are far less than the 10CFR100 thyroid limit of 300 rem, and were bounded by the EAB and LPZ doses reported for the maximum hypothetical accident (MHA) in Table 14-52 (148.68 and 52.38 rem, respectively) of the Unit 1 SAR.

Using the same control room model as used for the MHA, the thyroid dose to the control room operator from the above scenario is well below the GDC-19 guideline of 30 rem and less than the MHA control room thyroid dose reported in section 14.2.2.6.7 of the Unit 1 SAR (18.93 rem).

5. Provide summary description of how the 1.87 gpm leak rate estimate was determined for LBLOCA. This should include the following information:

   a. Does this estimate consider leakage from flaws other than circumferential cracks?

   Response: A small amount of leakage was assigned to repair products for 1R17. This is the only other leakage assigned besides circumferential flaws.

b. Brief description of method used to calculate leakage for circumferential cracks in the original roll or heat affected zone adjacent to the tube-to-tubesheet seal weld if no reroll is present.

Response: The reported leakage associated with the rolled joint or the heat affected zone is the calculated leakage through the rolled joint as determined by loading and predicted dilations for LBLOCA for that specific radial location (ratio added for LBLOCA loads). The leakage for a specific flaw is determined by calculating the stress on the flaw, determining the percent degraded area of the flaw and assigning leakage. Circumferential flaws are evaluated to determine whether they could sever. If they are determined to sever, that leakage is used unless the flaws are located behind a roll (original or reroll). If the circumferential flaw is behind the roll, the lesser leakage by the roll is used.

c. Brief description of method used to calculate leakage for circumferential cracks at or below lower transition of original roll if no reroll is present.

Response: The leakage is determined by sizing the flaw and determining which section of the flaw could leak without taking any credit for the rolled area. The leakage is determined by taking the lesser of the leakage associated with the actual flaw or the leakage through the annulus.

d. Brief description of method used to calculate leakage for circumferential cracks at reroll repair.

Response: A reroll length as short as 0.86" has been leak tested by Framatome under operational and accident conditions. This leakage is applied if the flaw has moved into the heel transition (outboard or non-pressure boundary). Again, depending upon the radial distance of the tube with the flaw, a leak rate is assigned through the reroll joint or the leak associated with the actual flaw, whichever is less. No circumferential cracks were found in rerolled tubes during 1R17.

e. Brief description of method used to calculate leakage for circumferential cracks inboard of reroll repair.

Response: The leakage is determined by sizing the flaw and determining which section of the flaw could leak without taking any credit for the rolled area. Again, annulus leakage was assigned as discussed in item "c" above.

f. Brief description of method used to calculate leakage for circumferential cracks outboard of reroll repair. Are the portions of tubing outboard of reroll repairs subject to inspection?

Response: Yes, the outboard portion of the tubing above the reroll is inspected for circumferential cracking. As discussed above, the leakage assigned is the leakage through the reroll joint or the leakage through the flaw, whichever is less.