Mr. Craig G. Anderson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

May 2, 2000

SUBJECT: DETERMINISTIC OPERATIONAL ASSESSMENT OF STEAM GENERATOR TUBING FOR THE REMAINDER OF CYCLE 14, ARKANSAS NUCLEAR ONE, UNIT 2 (TAC NO. MA1951)

Dear Mr. Anderson:

By letter dated February 11, 2000, you submitted, for Arkansas Nuclear One, Unit 2 (ANO-2), an "Operational Assessment of ANO-2 Steam Generator Tubing for the Remainder of Cycle 14," and the "Assessment of Burst Pressure for ANO-2 SG [Steam Generator] B, [Tube] R72C72." In addition, you discussed the submittal and provided additional material during a public meeting on February 17, 2000, at U.S. Nuclear Regulatory Commission (NRC) Headquarters. In the submittal and during the meeting, you discussed the in-situ testing of six indications. Five indications exceeded the acceptance criterion of 4369 psi (3ΔP, 3 times the differential pressure between the primary and secondary systems). One flaw was only taken to 4147 psi due to leakage in excess of the pump capacity. However, your subsequent evaluation of the flaw in tube R72C72 determined that the tube would have met 3ΔP if the test pump had a higher capacity. You concluded that your deterministic operational assessment demonstrates that ANO-2 can safely operate until the September 2000 refueling and SG replacement outage with adequate margin.

After reviewing the submittal and meeting material discussed above, the NRC concludes that the results of the tests and analyses performed by Entergy Operations, Inc. (Entergy) presently do not demonstrate compliance with the expectation that SG tubes will maintain a strength margin of a factor of 3 against burst at the normal operating pressure differential for the portion of the operating cycle remaining after about June 25, 2000. This conclusion rests on the in-situ pressure test results for tube R72C72 in SG B. The NRC does agree with Entergy that the results of the eddy current tests following the in-situ pressure test indicate that tube R72C72 was not completely burst by the pressure test. However, the NRC believes that the incomplete burst is due to the limitations of the in-situ pressure test rather than the inherent strength of the flawed tube. The information provided suggests to the NRC that the tube was at the point of incipient burst when the test was terminated due to leakage in excess of the test system capacity. The NRC further concludes that the test most likely indicates a burst pressure of about 3900 to 4025 psi. The target pressure to simulate the normal operating "3ΔP" criterion was 4369 psi.

The basis for NRC's position is as follows:

1. At no time, when testing at pressures above 3900 psi, was Entergy able to hold pressure while maintaining a constant leak rate. When ramping beyond a reported test pressure of 4025 psi, the leak rate experienced a rapid increase from 1 gpm to 3.7 gpm

at a reported test pressure of 4147 psi, whereupon the pressure dropped suddenly to 600 psi. The NRC believes this behavior to be consistent with a condition of incipient burst at the time the test was terminated. The rapid drop off in pressure precluded driving the crack to a fish-mouth rupture.

2. Entergy did not describe its procedure for accounting for the head loss between the water supply location and the crack, nor did you describe the leak rate that was considered when making this adjustment. However, it is the NRC's understanding that these adjustments assume equilibrium conditions exist rather than a rapidly changing crack geometry and leak rate. Thus, the NRC is concerned that at reported test pressures beyond 3900 psi, the actual pressure at the flaw may be somewhat smaller than the reported test pressure, particularly for reported test pressures above 4025 psi.

3. The burst analyses of plus-point crack profiles B5534 and S5971 indicate a nominal burst pressure for the subject tube in the range of 3752 to 4311 psi, based on the Westinghouse burst model in WCAP-15128, Revision 1, "Depth-Based SG Tube Repair Criteria for Axial PWSCC [Primary Water Stress Corrosion Cracking] at Dented TSP [Tube Support Plate] Intersections," dated August 1999. This is consistent with the test results.

4. The effective crack lengths (as defined in WCAP-15128) for profiles B5534 and S5971 were 0.905 inches and 0.72 inches, respectively. These lengths exceeded the critical crack lengths of 0.74 inches and 0.63 inches for differential pressure loadings of 3752 and 4311 psi, respectively. This means that failure of the remaining crack ligaments leads directly to burst without any further increase in pressure. It is possible for a partial ligament failure to occur prior to burst as was indicated by the fact that initial leakage was observed during the test at a pressure of 3890 psi. However, the test scenario was consistent with a state of incipient total ligament failure and incipient burst when the test was terminated.

5. Entergy has estimated ligament tearing pressures of 3125 and 3752 psi for profiles B5534 and S5971, respectively, using the Argonne National Laboratory (ANL) model in NUREG/CR-6575. These estimates are based on ligament effective lengths of 1.065 and 0.86 inches, respectively. These lengths exceed the critical crack lengths (0.9 and 0.74 inches, respectively) associated with differential pressures of 3125 and 3752 psi, respectively. Therefore, at these lengths, failure of the ligaments should lead directly to burst without any additional increase in pressure. The NRC has not yet determined why the ANL model leads to a more conservative estimate of burst pressure than the burst model in WCAP-15128. However, results with the ANL model are consistent with those from the WCAP model, which indicate there is no pressure increase between the point of failure of the remaining ligament and tube burst.

6. The NRC applied the same burst prediction model to the post-test plus-point profile. The calculated burst pressure is 1933 psi with the WCAP-15128 model. This is consistent with the hypothesis that the crack was at the point of incipient total ligament failure and incipient burst at the time the test was terminated, and that the crack had undergone significant ligament damage and degradation of its subsequent burst pressure capability.

7. The NRC observes that the post-test crack profile appears fairly consistent over its entire 1.49-inch length, ranging from 90% through-wall (TW) to 100% TW. This contrasts with the two pre-test profiles which exhibit more irregularity over this length. Pre-test profile B5534 exhibits no detectable depth over the lowermost 0.2 inches of this length, steps up to about 70% depth over the next 0.25 inches, and then steps up to about 85% depth for the remainder of this length. Pre-test profile S5971 exhibits a depth of about 65% for the initial 0.5 inches of this length, steps up to about 85% for the next 0.75 inches, and then steps down to about 70% for the remainder of this length. Based on the post-test profile, it would appear that the actual pre-test crack profile was likely longer than indicated by the measured pre-test profile B5534 and of more consistent depth than indicated by either pre-test profile. This further calls into question estimated differences between ligament failure pressure and burst pressure based on the measured pre-test profiles.

8. The electric discharge machine (EDM) notched specimens contained 80% TW EDM notches with lengths of 0.5 inches and 0.7 inches. These specimens appear non-representative of the likely pre-test crack profile, as discussed in item 7 above, and do not provide a credible basis for evaluating the pressure differences between ligament failure and burst.

Given the information provided, the NRC concludes that the results of tests on SG B tube R72C72 indicate that ANO-2 did not achieve a strength margin of a factor of 3 against burst during the first half of fuel cycle 14. This margin against burst is included in NEI 97-06, Revision 1b, “Steam Generator Program Guidelines,” dated January 2000, as an expected structural margin that plants shall maintain (the licensees for all pressurized water reactors, including ANO-2, have committed to follow NEI 97-06). Because ANO-2 has not achieved a strength margin of a factor of 3 during several previous tube inspections, this conclusion is consistent with previous performance, as well as with the current data. According to your initial deterministic operational assessment dated December 21, 1999, which utilized an assumption of burst for tube R72C72, an operating run-time of 7.0 effective full power months is the point at which the worse case flaw would exceed the strength margin of a factor of 3 criterion. According to your March 9, 2000, “Proposed License Change For Cycle 14 Risk-Informed Operation,” the 7.0 effective full power months run-time will expire around June 25, 2000.

With respect to our progress in reviewing your March 9, 2000, risk-informed license amendment request, we understand from our previous discussions that you would need to begin preparations for a SG tube inspection outage on May 1, 2000, should you decide that another mid-cycle SG inspection is warranted. We are unable to commit to that schedule for completion of our review of your risk-informed application. As previously discussed, that review is a first-of-a-kind process due to the more challenging nature of the ANO-2 reactor with respect to the integrity of the SG tubes during severe accident sequences. That is why we urged you since last November to submit your proposed risk-informed application soon, or at least to pre-submit background information so that the review could be started. Although you recently submitted a very comprehensive and well organized risk-informed application, it is a time-consuming process to review and verify the completeness and accuracy of the application. We urge you to make every effort to expedite your responses to our requests for supporting information and to take advantage of our recommendation to discuss your development of those responses to ensure that they are sufficient when submitted.

We will consider additional information in response to this letter, if such information adequately addresses the NRC position set out above. If you have any questions on this matter, please contact Robert A. Gramm at 301-415-1010.

Sincerely,

/RA/

Stuart A. Richards, Director
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-368

cc: See next page

We will consider additional information in response to this letter, if such information adequately addresses the NRC position set out above. If you have any questions on this matter, please contact Robert A. Gramm at 301-415-1010.

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