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Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
License Renewal Application RAI Clarifications (TAC No. MA8054)

Gentlemen:

The following discussions provide either clarifications or corrections to previous requests for additional information (RAIs) from the NRC Staff. This supplemental information responds to concerns or questions raised by the Staff during the review of the RAI responses.

By letter dated September 6, 2000 (1CAN090002), Entergy Operations provided responses to a RAI concerning the reactor coolant system sections of the Arkansas Nuclear One, Unit 1 (ANO-1) License Renewal Application (LRA). Subsequent to this submittal, Entergy Operations determined that one of the RAIs contained incorrect information. Please find attached a revised response to RAI 3.3.2.2.2.2-1(d).

By letter dated August 24, 2000 (1CAN080003), Entergy Operations provided responses to RAIs concerning the reactor vessel. To clarify the response to RAI 4.2.3-3, the determination of the fluence at the inner diameter of the reactor vessel was not determined by using the technique reviewed and approved by the staff in BAW 2241A. The generic reactor vessel aging management report, BAW-2251, was completed prior to finalization of the BAW-2241A approach. Fluence evaluations were subsequently completed for ANO-1 using the method described in BAW-2241A. The 48 EFPY fluence estimates reported in BAW-2251 for the most limiting locations within the beltline region are conservative with respect to the fluence estimates obtained using the BAW-2241A method. Reference: Babcock and Wilcox Owners Group letter, OG-1677, dated October 21, 1997.

Also, in the August 24, 2000, correspondence, RAI 2.3.1-4 requested information concerning the reactor vessel level monitoring system probe. Subsequent to this submittal, the NRC Staff

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requested additional clarifying information concerning this RAI; therefore, please find attached a revised response to RAI 2.3.1-4.

By letter dated August 30, 2000 (1CAN080007), Entergy Operations provided responses to RAIs concerning the structural scoping section of the ANO-1 LRA. To clarify the response to RAI 2.4-5, Section 2.4 of the ANO-1 LRA should have included the turbine building within the scope of license renewal because the turbine building contains 10CFR50.48-required components and commodities.

By letter dated September 12, 2000 (1CAN090004), Entergy Operations provided responses to RAIs pertaining to Section 3 of the ANO-1 LRA. To clarify the response to RAI 3.3.3.1-7, the sodium hydroxide utilized at ANO-1 contains approximately 3.1 ppm chloride and 2 ppm sulfur as impurities. There is no mechanism for concentrating these impurities in the sodium hydroxide system. Although stress corrosion cracking is possible with carbon steel in the presence of aqueous chlorides, one of the most reliable methods of preventing this effect is to select material with a yield strength of less than 100 ksi. The yield strength of carbon steels typically used in this application is on the order of 30 – 45 ksi. Industry data does not indicate a problem of stress corrosion cracking in low strength carbon steels.

In the September 12, 2000, correspondence, RAI 3.3.3.1-2(b) credited the Preventative Maintenance (PM) Program for inspections of the borated water storage tank (BWST). The Staff requested that more detail of the specific visual inspection be provided. At times in the past, the visual inspection of the BWST has been performed with cameras; however, the PM Program requires only a visual inspection to the maximum extent practical from the manway on top of the tank. Under the PM task, the internal surface of the tank is inspected for evidence of deterioration, such as cracking, splitting or severe corrosion.

In the September 12, 2000, correspondence, RAI 3.3.3.1-6(a) addressed materials and environments for components associated with the reactor building cooling and purge system. To provide further clarification of the response to this item, boric acid crystals have been observed primarily in portions of the system (chilled water coils) that are not within the scope of license renewal. Although most of the boric acid plates out on the chilled water coils, boric acid will be considered a potential environment for components in this system that are subject to aging management review. In the portions of the reactor building cooling and purge system within the scope of license renewal, inspections have not revealed significant corrosion due to boric acid.

Also, in the September 12, 2000, correspondence, in the response to RAI 3.3.1.4.4-2 Entergy Operations stated that the Corrective Action Program, which includes the confirmation process, and the administrative (document) control program, that governs site procedures is controlled under 10CFR Part 50, Appendix B and applies to the aging management programs credited for license renewal. This information will be included in a revision to the safety analysis report (SAR) supplement which is Appendix A in the January 31, 2000 (1CAN010003), ANO-1 LRA submittal. Entergy Operations plans to submit the SAR supplement revision with the annual update to the ANO-1 LRA.

Should you have any further questions, please contact me.

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

Executed on October 3, 2000.

Very truly yours,

Jimmy D. Vandergriff
Director, Nuclear Safety Assurance

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Attachment

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Revised RAI Responses

3.3.2.2.2.2-1 Table 2.3-2 of the LRA contains the ANO-1 response to the renewal applicant action items identified in the staff's safety evaluation concerning BWOG report BAW-2243A. BAW-2243A addresses the reactor coolant system piping. In its safety evaluation, the staff indicated that a license renewal applicant would have to provide additional details regarding its augmented inspection program for small-bore piping. In addition, Table 2.3-4 contains the ANO-1 response to the renewal applicant action items addressed in the staff's safety evaluation concerning BAW-2244A. BAW-2244A addresses the pressurizer. An augmented inspection program for the pressurizer small-bore piping nozzles and safe-ends was also identified as a renewal applicant action item. The augmented inspection program addressing the action items from both topical reports is described in Appendix B, Section 4.3.8 of the LRA. For the reactor coolant system piping and the pressurizer small-bore piping nozzles and safe-ends, provide the following:

(d) Discuss the results of previous inspections performed on the small-bore or Alloy 600 piping, nozzles and safe-ends.

Selected stainless steel welds in the 2.5-inch nominal pipe size (NPS) pressurizer spray line, 2.5-inch NPS makeup and purification lines, and 2.5-inch NPS letdown line receive volumetric inspection at ANO-1 each interval in accordance with the ANO-1 risk-informed Inservice Inspection (ISI) plan for American Society of Mechanical Engineers (ASME) Class 1 piping. No reportable indications have been identified from these inspections.

In 1990, on the ANO-1 pressurizer, an instrumentation nozzle (pressurizer level tap), MK-30, was modified due to reactor coolant system (RCS) leakage. The nozzle is periodically examined utilizing a demonstrated ultrasonic technique in accordance with the ANO-1 Augmented ISI Program. The inspections have found no indication of corrosion or erosion of the ferritic material. In addition, the Alloy 600 nozzles attached to the pressurizer have received visual inspection each refueling outage since 1990 and no indications of leakage have been observed.

In March 2000, cracks were discovered at a number of ANO-1 Alloy 600 RCS hot leg level instrumentation nozzles during a visual inspection. These nozzles were field-installed in 1986 and were a different design than other RCS nozzles. The nozzle assemblies were two piece designs with a corrosion barrier sleeve welded on the inside

diameter and outside diameter of the hot leg pipe and a nozzle inserted in the sleeve welded to the outside diameter of the hot leg pipe. Unlike other RCS piping nozzles, these field-installed nozzles were not stress relieved. The cracking occurred in the weld material and not the nozzle or sleeve. A root cause evaluation determined the cracking was caused by primary water stress-corrosion cracking (PWSCC). The design of the nozzles resulted in high thermal and residual stresses at the root of the Alloy 182 weld (a weld material similar to Alloy 600) that connects the Alloy 600 nozzle to the ferritic piping. The high stresses led to cracking of the welds. These specific welds were not directly evaluated in the Alloy 600 susceptibility model as operating experience indicated that nozzles and sleeves are more susceptible to PWSCC than the Alloy 82/182 welds that attach the Alloy 600 nozzles and sleeves to ferritic steel. In addition, the residual stresses due to field fabrication were not accounted for when conducting the susceptibility ranking due to administrative oversight. Proper consideration of residual stresses due to field installation at the hot leg level tap locations would have resulted in the highest susceptibility rating for these items. These subject hot leg level taps are the only Alloy 600 RCS piping penetrations that have not been stress relieved.

Thus, the Alloy 600 susceptibility model did not indicate a high susceptibility to cracking at these welds since they were not specifically modeled. Apparently, the thermal stresses imposed by the sleeves caused bending stresses in the welds that promoted PWSCC. A new design was developed to eliminate the high thermal stresses for the replaced nozzles. Nozzle replacements have been performed on all but one of these nozzles using Alloy 152 (a weld material similar to Alloy 690) and Alloy 690 nozzle material, which has been demonstrated to be very resistant to PWSCC. A NRC-approved temporary weld repair was performed on the one remaining nozzle that will be replaced during 1R16.

Entergy Operations maintains that the three locations most susceptible to PWSCC have been identified and will be monitored during the period of extended operation. Monitoring the top three locations will bound the locations that are not monitored. The repaired hot leg level taps have a low susceptibility to cracking since the repairs were made using materials that are resistant to PWSCC.

2.3.1-4 **Please clarify whether the reactor vessel level monitoring system probe itself is subject to an AMR. If not, provide a justification for excluding the level probe from an AMR.**

The reactor vessel level monitoring system is part of the inadequate core cooling (ICC) instrumentation required by NUREG-0737, item II.F.2. The Bases for ANO-1 Technical Specification 3.5 state, "The function of the ICC instrumentation is to increase the ability of the plant operators to diagnose the approach to and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37 and are not required by the accident analysis, nor to bring the plant to cold shutdown conditions."

Because the instrumentation is not required for response to accident conditions, the technical specifications allow plant operation to continue after a failure of one or both of the reactor vessel level monitoring channels. Specifically, the action required by the ANO-1 Technical Specifications reads, "If repair is not feasible without shutting down, operation may continue and a special report pursuant to specification 6.12.5 shall be submitted to the NRC within 30 days following the failure; describing the action taken, the cause of the inoperability, and the plans and schedules for restoring the channels to operable status during the next scheduled refueling outage."

Operators are trained to detect voids by monitoring the reactor coolant system (RCS) level response to pressurizer spray and heater operation, even without the use of the reactor vessel level monitoring instrumentation. Manual actions, if required for accident mitigation, are based primarily on RCS pressure, temperature, and subcooling margin.

The reactor vessel level monitoring probes are classified safety related, but only for the passive function of maintaining the RCS pressure boundary. For the pressure boundary function, the results of the aging management review for the reactor vessel level monitoring system adapter flange and closure assembly are provided under control rod drive mechanisms in Table 3.2-1 of the ANO-1 LRA.