June 21, 2001

MEMORANDUM TO: Robert A. Gramm, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Thomas W. Alexion, Project Manager, Section 1
Project Directorate IV ***/RA/***
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 2 RE: PROPOSED LICENSE AMENDMENT ON EXTENDED POWER UPRATE (TAC NO. MB0789)

The U. S. Nuclear Regulatory Commission (NRC) staff has had discussions with Entergy Operations, Inc., the licensee, on its December 19, 2000, "Application for License Amendment to Increase Authorized Power Level." The requested power level increase is 7.5%.

In order to facilitate these discussions, the NRC provided the licensee with a preliminary request for additional information (RAI) on mechanical and civil engineering issues. This RAI does not represent final NRC positions and it may get revised as a result of discussions with the licensee. The purpose of this memorandum is to place the attachment in the Public Document Room.

Docket No. 50-368

Attachment: As stated

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Request for Additional Information on
Mechanical and Civil Engineering Issues
Extended Power Uprate License Amendment Application
Arkansas Nuclear One, Unit 2 (ANO-2)

1. In reference to Section 5.3.3.2 of the application, provide the calculated maximum stresses and fatigue usage factors at the critical locations of the control element drive mechanisms for all operating conditions shown in Table 5-3 as a result of the power uprate. Also, provide the allowable Code limits for the critical components evaluated, and the Code and Code edition used for the evaluation. If different from the Code of record, justify and reconcile the differences.

2. Section 5.4 describes the mechanical and thermal analyses performed to determine the response of the reactor cooling system (RCS) main coolant loop and components, including the reactor vessel (RV), reactor coolant pumps (RCPs), replacement steam generators (RSGs), hot and cold leg piping, and component (RV, RCP, pressurizer and RSG) supports. The piping is discussed separately in Section 5.8. Provide the methodology, assumptions and loading combinations used for evaluating the RV, the pressurizer, the RCPs, the RSGs and their supports. Also provide the calculated maximum stresses and cumulative usage factors at critical locations of each component for the power uprate condition, including the allowable Code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

3. As a result of the RSGs and the power uprate, the feedwater flow and pressure in the feedwater system have to increase from those required for the RSGs at the current and uprate power levels. Discuss the potential for flow-induced vibration of the RSG tubes due to various mechanisms, including, in particular, the fluid-elastic instability in the RSGs at the current power level. Provide an evaluation of the flow-induced vibration of the tubes in the RSGs at the power uprate condition regarding the analysis methodology, damping value of the tubes and the computer code used in the analysis, results of the predicted vibration levels during the normal operating condition and the worst case transient condition, and the calculated fluid-elastic instability ratios. Explain whether or not the current analysis considers the potential for a possible degraded RSG condition.

4. In regard to Section 5.2.2, you stated that for the holddown ring evaluation, rocking and sliding margins were calculated using the revised hydraulic input loads and moments, in combination with holddown ring loads derived from recent field ring deflection measurement data. Confirm whether and how the holddown ring is acceptable to provide adequate reactor vessel internal (RVI) hold down force and provide technical basis that the margin factors of 2 and 1.5 are considered acceptable as stated in Section 5.2.2. Also, in regard to Section 5.2.2, provide an assessment of flow-induced vibration of the RVI components due to the power uprate.

5. In reference to Section 5.7.1, you stated that following the application of leak-before-break (LBB), the remaining pipe breaks in the mechanical design basis of the RCS are all primary and secondary side branch line pipe breaks (BLPBs) interfacing with the RCS. Of these, the limiting breaks with respect to RCS structural considerations are

breaks in the largest tributary pipes such as main steam line, feedwater line, surge line, safety injection line and shutdown cooling line. Clarify whether the thermal transient effects due to large-bore RCS pipe-break loss-of-coolant accidents (LOCAs) were considered in current licensing basis for the design of the ANO-2 RSGs. If not, explain why they were not considered (note that the approved LBB condition applies only to dynamic effects). Also, provide the stress analysis results for the primary side components of the RSGs including the RSG tubes to demonstrate the adequacy of the ANO-2 RSGs for the effects of thermal transients arising from postulated large-bore RCS pipe-break LOCAs during the power uprate.

6. In reference to Section 5.7.2, you stated that for the RCS with the RSGs, non-linear response time history analyses were performed to calculate the RCS response to the limiting BLPBs following the application of LBB technology. You also stated that a more detailed model of the RVI was included in the primary side pipe break model, because these pipe breaks cause RV blowdown loads. This RVI model included hydro-mass and coupling terms, as well as additional nodes for RV blowdown input loadings. Confirm whether the analyses of the RV blowdown forcing functions and the non-linear structural responses due to the RSGs and the power uprate were performed by computer codes that were approved by the NRC or used in the analysis of record at ANO-2. Identify the computer codes that were used for the analyses of pipe breaks, seismic and transients events, that are different from those used in the original design basis analysis, and provide a justification that the new code was bench-marked for this application.

7. In reference to Section 5.7.2, you indicated that for the pipe break analysis of the RCS with RSGs, two three-dimensional ANSYS models of the entire RCS were developed from the RCS seismic model, one for secondary side breaks and one for primary side breaks. For the secondary side pipe break model, the representation of the RVI remained essentially the same as that for the seismic model, because secondary side breaks do not cause RV blowdown. A more detailed model of the RVI was included in the primary side pipe break model, because these pipe breaks cause RV blowdown loads. This RVI model included hydro-mass and coupling terms, as well as additional nodes for RV blowdown input loadings. The response of the entire RCS to pipe breaks was calculated using non-linear response time history analysis. The ANSYS computer code was used to perform the time history analyses due to BLPBs, using the modal superposition method and constant 3% modal damping. Clarify whether the ANSYS computer code was used to perform the non-linear time history analysis, using the modal superposition method. Describe the nonlinear parameters used in analysis. Also, provide a summary of analysis with a detailed model of the reactor internals to account for the depressurization blowdown loading in the BLPB analysis.

8. In reference to Section 5.8, provide, for the most critical RCS piping systems evaluated, the calculated maximum stresses and fatigue usage factor, and code allowable limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide the necessary justification. Were the analytical computer codes used in the stress analysis different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

9. In reference to Section 2, you stated that the balance-of-plant (BOP) structures, systems and components have been evaluated for the impact of the 107.5 percent power uprate and in general found acceptable. Those requiring modifications due to power uprate consideration are provided in Table 2-2. Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

10. Provide the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the Code of record and Code edition used for the power uprate conditions. If different from the Code of record, justify and reconcile the differences.

11. In reference to Section 2.4.5.3, you stated that the feedwater heaters have been evaluated for the power uprate condition for extractions, design pressures, pressure drops, and drain, tube and nozzle velocities. You also stated that feedwater heater vibration characteristics and shell-side relief valve capacities have been evaluated. The main steam and feedwater flow rate increase about 10 percent for the power uprate as shown in Table 3-1. Discuss the potential for flow-induced vibration in the main steam and feedwater pipe and the BOP heaters and heat exchangers following the power uprate.

12. Discuss the functionality of safety-related mechanical components (i.e., all safety related valves and pumps, including power-operated relief valves) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) in your Generic Letter (GL) 89-10 MOV program at ANO-2 will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which functionality at the uprated power level was not evaluated. Also, discuss effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for GL 95-07 and on the evaluation of overpressurization of isolated piping segments for GL 96-06.

13. Confirm whether the steam generator replacement and the proposed power uprate will increase the accident temperature, pressure and sub-compartment pressurization that affect the design basis analyses for steel and concrete in the containment, steam tunnel and the spent fuel pool. If the structural steel and concrete will be affected, provide the design basis margin and margins after considering increased accident loading due to the steam generator replacement/power uprate.