May 1, 2000

MEMORANDUM TO: File

FROM: Thomas W. Alexion, Project Manager, Section 1 /RA/
Project Directorate IV & Decommissioning
Division of Licensing Project Management

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 2 RE: PROPOSED LICENSE AMENDMENT ON SETPOINT CHANGES AND RELATED ITEMS FOR THE REPLACEMENT STEAM GENERATORS (TAC NO. MA7299)

The U. S. Nuclear Regulatory Commission (NRC) staff has had discussions with Entergy Operations, Inc., the licensee, on its November 29, 1999, application on “Proposed Technical Specification Changes And Resolution of Unreviewed Safety Question Associated With Applicable Limits And Setpoints Supporting Steam Generator Replacement.” The steam generators are scheduled to be replaced during the 2R14 refueling outage, currently scheduled for September 2000.

In order to facilitate these discussions, the NRC provided the licensee with a preliminary list of questions. These draft questions do not represent final NRC positions and may get revised/eliminated 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

May 1, 2000

MEMORANDUM TO: File

FROM: Thomas W. Alexion, Project Manager, Section 1 ***/RA/***
Project Directorate IV & Decommissioning
Division of Licensing Project Management

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 2 RE: PROPOSED LICENSE AMENDMENT ON SETPOINT CHANGES AND RELATED ITEMS FOR THE REPLACEMENT STEAM GENERATORS (TAC NO. MA7299)

The U. S. Nuclear Regulatory Commission (NRC) staff has had discussions with Entergy Operations, Inc., the licensee, on its November 29, 1999, application on “Proposed Technical Specification Changes And Resolution of Unreviewed Safety Question Associated With Applicable Limits And Setpoints Supporting Steam Generator Replacement.” The steam generators are scheduled to be replaced during the 2R14 refueling outage, currently scheduled for September 2000.

In order to facilitate these discussions, the NRC provided the licensee with a preliminary list of questions. These draft questions do not represent final NRC positions and may get revised/eliminated 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

DISTRIBUTION:
PUBLIC
PDIV-1 RF

Accession No.: ML003709458

OFFICE	PDIV-1/PM	PDIV-1/LA
NAME	TAlexion	DJohnson
DATE	05/01/00	5/1/00

DOCUMENT NAME: C:\MEMA7299.wpd
OFFICIAL RECORD COPY

<u>DRAFT QUESTIONS</u>

<u>SETPOINT CHANGES AND RELATED ITEMS FOR THE</u>

<u>REPLACEMENT STEAM GENERATORS</u>

<u>ARKANSAS NUCLEAR ONE, UNIT 2</u>

<u>Safety Analysis Questions</u>

1. Please provide references to confirm that no plans currently exist for Arkansas Nuclear One, Unit 2 (ANO-2) to allow plant startup and power operation with less than four reactor coolant pumps (RCPs) operating.

2. Please provide the minimum thermal margin reserved by the updated Core Operating Limits Supervisory System and the Departure from Nucleate Boiling Limit Plot for offsetting the increased required thermal margin during Loss of Primary Flow Events.

3. Please explain why the Main Steam Line Break and Feedwater Line Break analyses were found to be bounding for offsite dose considerations among all events analyzed. Please discuss radiological consequences of a steam generator tube rupture accident and a seized rotor event relative to the above conclusion.

4. Please discuss the bases for determining the amount of increase in the required thermal margin for the decrease in reactor coolant system (RCS) flow during pump coastdown, increase in maximum initial RCS flow, and increase in core protection calculator response time during a loss of primary flow event.

5. Please provide justification on why a loss of offsite power is not assumed in the analysis of a seized RCP shaft accident.

6. Please discuss the radiological consequences of a steam generator tube rupture event when a stuck open atmospheric steam dump valve associated with the failed steam generator is assumed as a most limiting single failure. Provide justification for not assuming this limiting single failure at ANO-2.

7. In the reanalyses of various events, different initial pressures of RCS and steam generators were assumed. Please provide method of determining these initial conditions in light of conservativeness in analyses.

8. Please provide the following information regarding your proposed feedwater line break analysis:

    a. initial conditions and justifications including instrument uncertainty discussions,
    b. sequence of events,
    c. trip setpoints including instrument uncertainties,
    d. safety system actuation setpoints including instrument uncertainties,
    e. single failure and loss of offsite power discussions,
    f. codes/method used for analyses, approval reference for these codes,

g. major transient curves, and
h. explain in detail how the break size was determined for the current and revised analyses and justify the differences.

9. Show that the large and small break loss-of-coolant accident (LOCA) analyses methodologies referenced in the November 29, 1999, submittal apply to ANO-2 by confirming that the ANO-2/ANO-2 LOCA analysis vendors have ongoing processes to assure that the values of peak cladding temperature-sensitive parameters input to the LOCA analyses bound the as-operated plant values for those parameters.

10. Why are the large break LOCA analyses performed assuming no steam generator tube plugging, whereas the small break analyses are done assuming 10% steam generator tube plugging?

11. Identify whether the LOCA methodologies used to perform the ANO-2 LOCA analyses described in the November 29, 1999, submittal differ from the approved and referenced versions.

Dose Questions

1. Your submittal does not address the impact of the proposed changes on the ability of the ANO-2 control room to meet the habitability requirements of General Design Criterion (GDC)-19 (or the ANO-2 commitments to NUREG-0737 Item III.D.3.4). In a telephone call with the staff, you stated that the maximum hypothetical accident (i.e., LOCA) was the limiting accident for control room habitability at ANO-2. Please provide a statement describing your evaluation of the impacts on the habitability of the ANO-2 control room that justifies your conclusion that the LOCA is still the limiting accident. Your statement should address differences in control room isolation actuation for the different accidents and differences in X/Q values for the different release points involved.

2. The ANO-2 Safety Analysis Report (SAR) indicates that the control room unfiltered in-leakage is limited to 10 cfm. The staff considers in-leakage to be subject to the design control measures of 10 CFR Part 50, Appendix B, Criterion 3. Several power reactors (representing about 20% of the U.S. plants) have performed testing of their control room in-leakage. In all but one case, the test results showed in-leakage in excess of the facility’s design basis. Please provide an explanation supporting your conclusion that the unfiltered in-leakage at ANO-2 is limited to only 10 cfm.

3. In Section 15.1.0.5.5.H of the marked-up SAR, there is a statement that ICRP-30 values for dose conversion factors were used to determine whole body doses for events with fuel failure. This appears to be an error. ICRP-30 addresses internal exposures due to uptake of radioactive material.

4. Section 15.1.0.5.5.K of the marked-up SAR indicates that a pre-existing and an event generated iodine spike was assumed for non-fuel failure postulated accidents. Table 15.1.14-40 presents the dose results for the cycle 15 feedwater line break event but does not provide the doses for the spiking cases. The analysis description in the

SAR mark-up and that on page 40 of 172 of your submittal does not indicate that fuel damage is postulated. Please provide the doses for the spiking cases.

5. Page 25 of 172 of your submittal states "For this evaluation, the reported fuel failure for the radiological doses are back-calculated from the 10 CFR dose criteria." Please explain how the fuel damage estimate arrived in this manner reflects the fuel damage projected by the thermo-dynamic analysis of the transient.

6. Your re-analyses incorporate iodine spiking. In early 1999, Beaver Valley submitted an LER regarding non-conservatisms in the calculation of the accident-generated iodine spike appearance rate. In summary, Beaver Valley determined that its contractor had used minimum values for purification flow rate and demineralizer efficiency which resulted in a iodine appearance rate which was not bounding for all plant operating conditions. The staff notes that the cycle 12 main steam line break analysis submitted to the staff on December 12, 1997, assumes the purification flow to be 40 gpm. However, your system design allows for flow up to 128 gpm. Please confirm that your recent analyses used the appropriate flow rate.

7. The staff has performed independent calculations of the seized rotor event, feedwater line break, and main steam line break using the parameters and values documented in your submittal. Our results are reasonably consistent with the results presented in the submittal, with the exception of the seized rotor event. Please provide the following additional information:

   a. Please briefly explain how the Ci/pin data in Table 15.1.0-3C (per 15.1.0.5.5.D) are converted to RCS activity or RCS concentration in your analysis.

   b. Section 15.1.0.5.5.J states that an iodine partitioning of 100 is assumed. Previous ANO-2 analyses reviewed by the staff have included an iodine flash fraction. Is such a fraction included in your seized rotor event analysis?

   c. What is the RCS mass and steam generator liquid mass assumed in your seized rotor event?

   d. Are the steam mass releases of 15.1.0.5.5.M valid for the seized rotor event?

   e. What density was assumed in converting the volume units of the 150 gpd technical specification to mass units? Is this value consistent with the density used in surveillance procedures to demonstrate compliance with the technical specifications?

   f. Please include any other description that you believe would help the staff resolve the differences between our analyses.