UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 31, 2014

Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO EXTEND INTEGRATED LEAK RATE TESTING INTERVAL (TAC NO. MF3279)

Dear Sir or Madam:

By letter dated December 20, 2013 (Agencywide Documents Access and Management System Accession No. ML13358A195), Entergy Operations, Inc., submitted a proposed license amendment request (LAR) for Arkansas Nuclear One, Unit 1 (ANO-1). The LAR proposes to extend the interval for containment integrated leak rate testing from 10 to 15 years on a permanent basis.

The Nuclear Regulatory Commission staff has been reviewing the submittal and has determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). The questions were sent via electronic transmission on June 19, 2014, and July 10, 2014, to Mr. Robert Clark, of your staff. The draft questions were sent to ensure that they were understandable, the regulatory basis was clear, and to determine if the information was previously docketed. The draft questions were discussed with your staff in teleconferences on June 25, 2013, and July 23, 2014, and it was agreed that a response would be submitted within 30 days of the date of this letter.

If you have any questions, please contact me at (301) 415-2833 or by e-mail at Peter.Bamford@nrc.gov.

Sincerely,

Peter J. Bamford, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST PROPOSING TO EXTEND THE

CONTAINMENT INTEGRATED LEAK RATE TESTING FREQUENCY

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

By letter dated December 20, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13358A195), supplemented by letter dated March 11, 2014 (ADAMS Accession No. ML14070A399), Entergy Operations, Inc. (Entergy, the licensee), submitted a license amendment request (LAR) proposing a change to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs). The proposed change would allow for the 10-year frequency of the ANO-1 Type A or Integrated Leak Rate Test (ILRT) that is required by TS 5.5.16, "Reactor Building Leakage Rate Testing Program," to be extended to 15 years on a permanent basis. In order for the U.S. Nuclear Regulatory Commission (NRC) staff to complete its review of the LAR, a response to the following request for additional information is requested.

1. According to Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007 (ADAMS Accession No. ML070650428), the NRC staff expects that licensees fully address all scope elements with Revision 2 of Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009 (ADAMS Accession No. ML090140014), by the end of its implementation period (i.e., 1 year after the issuance of Revision 2 of RG 1.200). Revision 2 of RG 1.200 endorses, with exceptions and clarifications, the combined American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) probabilistic risk assessment (PRA) standard (ASME/ANS RA-Sa-2009).

   (a) In Section 4.5.2 of Attachment 1 to the LAR, the licensee stated that an industry peer review of the updated PRA model has been performed. Please clarify the scope of the peer review and the version of the ASME/ANS standard and RG 1.200 used for the peer review.

   (b) In Section 4.5.2 of Attachment 1 to the LAR, the licensee stated the ANO-1 internal events model has been updated to meet standards of RG 1.200, Revision 1, dated January 2007 (ADAMS Accession No. ML070240001). Given that the implementation date of RG 1.200, Revision 2, was April 2010 and the LAR was submitted in December 2013, if the peer review was not completed against RG 1.200, Revision 2, please describe any gaps between the peer review of the PRA model used in this application and RG 1.200, Revision 2, that are relevant to this submittal and also describe the impact of any gaps on this application.

2. Revision 2 of RG 1.200 endorses, with exceptions and clarifications, ASME/ANS RA-Sa-2009. In Regulatory Position 4.2 of RG 1.200, Revision 2, the NRC staff stated that it expects licensees to submit a discussion of the resolution of the peer review findings that are applicable to the parts of the PRA required for the application. The licensee stated in the LAR that an industry peer review of the updated PRA model has been performed.

    (a) For the PRA model used to support this application, please provide a list of Findings and Observations (F&Os) from the peer review relevant to this submittal.

    (b) Please explain how these F&Os were addressed for this application and the impact of remaining open items on this application.

3. The application refers to Electric Power Research Institute (EPRI) TR-1009325, Revision 2-A, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated October 2008 (ADAMS Accession No. ML14024A045). EPRI TR-1009325, Revision 2-A, states, in part, that

    > The most relevant plant-specific information should be used to develop population dose information. The order of preference shall be plant-specific best estimate, Severe Accident Mitigation Alternative (SAMA) for license renewal, and scaling of a reference plant population dose.

    (a) Given that plant-specific population dose estimates were available as part of the ANO-1 SAMA analysis, please discuss the reasons for the decision to estimate the population dose based on scaling of Surry population doses.

    (b) Please discuss whether using information from SAMA analysis would significantly change the estimated increase in population dose resulted from extending the Type A frequency.

4. EPRI TR-1009325, Revision 2-A states, in part, that

    > Where possible, the analysis should include a quantitative assessment of the contribution of external events (for example, fire and seismic) in the risk impact assessment for extended ILRT intervals. For example, where a licensee possesses a quantitative fire analysis and that analysis is of sufficient quality and detail to assess the impact, the methods used to obtain the impact from internal events should be applied for the external event...This assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an

order of magnitude estimate for contribution of the external event to the impact of the changed interval.

(a) Given that a Fire PRA (FPRA) model of ANO-1 has been used in the LAR to adopt National Fire Protection Association (NFPA) 805 performance-based standard for fire protection submitted to NRC on January 29, 2014 (ADAMS Accession No. ML14029A438), please discuss the reasons that the FPRA model was not used to estimate the contribution of fire to large early release frequency (LERF) for this application.

(b) Please discuss whether using the FPRA model would significantly change the total estimated LERF.

5. Section 5.1.5.1 of EPRI TR-1009325, Revision 2-A uses the Calvert Cliffs methodology in evaluating the impact of liner corrosion on the extension of ILRT testing intervals. This assessment was based on two observed corrosion events at North Anna Power Station, Unit 2, and Brunswick Steam Electric Plant, Unit 2. As there have been additional instances of liner corrosion that could be relevant to this assessment, please provide a more complete accounting of all observed corrosion events relevant to ANO-1 containment, and an evaluation of the impact on risk results when all relevant corrosion events are included in the risk assessment.

6. The LAR states that there is one primary containment surface associated with the area around the equipment hatch that requires augmented examinations in accordance with the ASME *Boiler and Pressure Vessel Code* (ASME Code) Section XI, IWE-1240. Please provide information regarding the findings that led to the augmented examination. Also, please provide information that would demonstrate proper and effective monitoring and managing of this condition.

7. Attachment 4 of the LAR, Tables 4-2 and 4-3, include a brief description of the results of reactor building interior and exterior structural inspections and ASME Code, Section XI, Subsection IWE inspections. Both tables indicate that numerous deficiencies were noted; however, they do not include details regarding these deficiencies. Please discuss highlights of the significant findings from the ASME Code, Section XI, Subsection IWE and IWL examinations performed since the last Type A test on the containment pressure-retaining structures and components, in accordance with the ANO-1 containment in-service inspection (CISI) program, and actions taken to disposition them. In the response, provide information that would demonstrate proper and effective implementation of the ANO-1 CISI program in monitoring and managing degradation to ensure that containment structural and leak-tight integrity has been, and will continue to be, maintained through the service life of the plant. The response should include relevant highlights of examinations performed on the containment penetrations (with seals, gaskets, and bolted connections), the containment steel liner, moisture barrier, and the reinforced concrete containment structure. Also, please discuss highlights of findings from recent

inspections from the ANO-1 containment coating inspection program and actions taken to disposition them.

8. Please provide the schedule of inspections, including the corresponding refueling outage, that were, or will be, performed on the containment structure in accordance with ASME Section XI, Subsection IWE and IWL, and explain how it meets the provisions in Section 9.2.3.2 of Nuclear Energy Institute (NEI) 94-01, Revision 2-A, "Industry Guideline for Implementing the Performance-Based Option of 10 CFR Part 50, Appendix J," and Condition 2 in Section 4.1 of the NRC safety evaluation dated June 25, 2008 (ADAMS Accession No. ML081140105) for topical report NEI 94-01, Revision 2-A.

9. Please provide information of instances during implementation of the ANO-1 CISI program in accordance with ASME Code, Section XI, Subsections IWE/IWL, where existence of or potential for degraded conditions in inaccessible areas of the concrete containment structure and steel liner were identified and evaluated based on conditions found in accessible areas, as required by Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, paragraph 55a(b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A). If there were any instances of such conditions, discuss the findings and corrective actions taken to disposition the findings.

10. As stated in ANO-1 Safety Analysis Report, Section 5.2.2.1.4, "Special Penetrations," expansion joint bellows at the fuel transfer tube provide for the relative movement between the reactor building, internals, and the auxiliary building. For any bellows used on penetrations through containment pressure-retaining boundaries at ANO-1, please provide information on their location, inspection, testing and operating experience with regard to detection of bellows leakage.

11. Attachment 5 to the LAR contains a summary table of components that did not meet the administrative limit for Type B and Type C testing. Please describe the causes and corrective actions taken to address the components that did not demonstrate acceptable performance in accordance with the ANO-1 Reactor Building Leakage Rate Testing Program.

12. Please provide the following information for penetrations/components that are subject to Type B and Type C testing:

    (a) Total number of penetrations/components subject to Type B test.

    (b) Total number of penetrations/components subject to Type C test.

    (c) Total number of penetrations/components that are on an extended performance-based test interval compiled by their test interval (120-month, 60-month, 30-month, etc.)

If you have any questions, please contact me at (301) 415-2833 or by e-mail at Peter.Bamford@nrc.gov.

Sincerely,

/RA/

Peter J. Bamford, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosure:
Request for Additional Information

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ADAMS Accession No. ML14209A085 ***via email** ****via memo**

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OFFICE	NRR/DRA/APLA/BC**	NRR/DORL/LPL4-1/BC(A)	NRR/DORL/LPL4-1/PM
NAME	HHamzehee	EOesterle	PBamford
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