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1CAN101407

October 28, 2014

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Clarifications to Response to Request for Additional Information Associated with Technical Specification Change to Extend the Type A Test Frequency to 15 Years
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

REFERENCES;
1. Entergy letter to NRC, "License Amendment Request Technical Specification Change to Extend the Type A Frequency to 15 Years," dated December 20, 2013 (1CAN121302) (ML13358A195)

2. NRC letter to Entergy, "Arkansas Nuclear One, Unit No. 1 – Request for Additional Information Regarding License Amendment Request to Extend Integrated Leak Rate Testing Interval" (TAC No. MF3279) (1CNA081402) (ML14209A085).

3. Entergy letter to NRC, "Response to Request for Additional Information Associated with Technical Specification Change to Extend the Type A Test Frequency to 15 Years," dated September 2, 2014 (1CAN091401)

4. NRC letter to Nuclear Energy Institute, "Request Revision to Topical Report NEI 94-01, Revision 3-A, 'Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,'" dated August 20, 2013 (ML13192A394)

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) requested an amendment to Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TS). Specifically, the proposed change would allow the ten-year frequency of the ANO-1 Type A, or Integrated Leak Rate Test, that is required by TS 5.5.16 to be extended to 15 years on a permanent basis (Reference 1).

During review of the request, the NRC determined that additional information is needed to complete the review. Reference 2 provides the NRC's request for additional information (RAI). The response to the request was provided in Reference 3. During the review of the information that was provided, there were several conference calls between representatives of the NRC and ANO. During one of the teleconferences, the NRC asked that the responses to RAIs 7 and 8 be clarified. The purpose of this submittal is to provide the requested clarifications.

In an email dated October 9, 2014, the NRC informed Entergy of an open request to NEI to revise NEI 94-01 (Reference 4). It is the NRC's opinion that NEI 94-01, Revision 3-A, is incomplete in that it does not incorporate the limitations and conditions in the Safety Evaluation for Revision 2. As such, the NRC requested that the proposed TS change, presented in Reference 1, be revised to address this issue. The revised marked up and clean TS pages are included.

In accordance with 10 CFR 50.91(b)(1), a copy of this application is being provided to the designated Arkansas state official.

The responses do not contain any new regulatory commitments. With respect to the original Entergy request (Reference 1), changes included in this letter have been evaluated and Entergy has determined that the changes do not invalidate the assessment of the no significant hazards consideration included in the reference letter.

If you have any questions or require additional information, please contact Stephenie Pyle at 479-858-4704.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on October 28, 2014.

Sincerely,

ORIGINAL SIGNED BY TERRY A. EVANS FOR JEREMY G. BROWNING

JGB/rwc

Attachments:

1. Response to Request for Additional Information License Amendment Request Proposing to Extend the Containment Integrated Leak Rate Testing Frequency

2. Replacement Markup of Technical Specification Page

3. Replacement Clean (Revised) Technical Specification Page

cc: Mr. Marc L. Dapas
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Attachment 1 to

1CAN101407

Response to Request for Additional Information
License Amendment Request Proposing to Extend the
Containment Integrated Leak Rate Testing Frequency

Clarifications to Responses to Request for Additional Information Items 7 and 8 License Amendment Request Proposing to Extend the Containment Integrated Leak Rate Testing Frequency

By letter dated December 20, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13358A 195), 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.

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.**

   Clarification: The response mainly focused on tendon surveillance. It does not appear that the deficiencies in Table 4-2 of the LAR "Reactor Building Interior and Exterior Structural Inspections" have been covered.

   The following response supplements the response provided in the RAI response. This clarification is associated with the most recent Type A test on the ANO-1 reactor building pressure system that was conducted in 2005 during refueling outage 1R19.

   Table 4-2, December 14, 2005 – All deficiencies noted were judged to be non-significant.

Details of the deficiencies:

This inspection was performed just prior to the last ILRT that was performed. A Professional Engineer reviewed these deficiencies and found all to be acceptable.

- Many areas were identified where top coat of liner had peeled off or had begun to bubble. Minor stains throughout building.
- Coating / paint chipped in the reactor building dome was determined to be surface level and not considered a major deficiency.
- Concrete chipped and crumbled in numerous areas identified above was determined to be non-significant.
- An ~0.025 inch wide crack on the west side of the dome about 5 feet long was determined to be a non-significant surface crack.
- Minor paint flakes in the tendon gallery were determined to be non-significant.
- Very minor crack in the concrete in the area of the safety relief valves.
- Grease stains on the wall.
- Abandoned supports in the area.
- Rust stains and calcium deposits on the wall in several locations.

None of the items identified above were determined to be significant or needing to be addressed prior to the ILRT.

Table 4-2, November 3, 2008 – Several deficiencies were noted during the inspection. All were evaluated and it was determined that none of the deficiencies would affect reactor building integrity.

Details of the deficiencies:

- Noted hairline cracks in concrete of several handrail attachment points were determined to be non-significant.
- Three mechanically induced holes / voids approximately 3/8 inch to 3/4 inch in diameter on the interior wall between vertical tendons V45 and V44. Cause of these holes is unknown, but thought to be the result of past attempts to mount equipment to wall. Determined to be non-significant.

- Coating topcoat flanking / peeling on floors and walls. Identified and noted during past inspection in December 2005.

- Water intrusion issues with some of the lower elevations. Previously identified.

- Identified a crack in the concrete around penetration WR-6-7 from approximately the 9 o'clock position to the 12 o'clock position. The crack has apparently existed for many years as evidenced by the fact that it has been painted over and does not appear to be active.

- The liner plate exhibits blistering of the topcoat of the coating in numerous areas. The prime coat of the coating system is still intact.

- A conduit was identified in the basement exiting the moisture barrier seal between the floor and the liner plate wall.

- Grease indications on the roof and buttresses.

- Bolt holes on the exterior surface left unfilled / patched.

- Cracked / flaking paint on exterior of building.

None of the items presented structural integrity concerns.

Table 4-2, March 2013 – Several deficiencies were noted during the inspection. All were evaluated and it was determined that none of the deficiencies would affect reactor building integrity.

Details of the deficiencies:

This inspection was the 40 Year IWL Tendon Examination and Concrete Inspection.

While no indications were found that challenged the current structural integrity or leak tightness of the reactor building, four indications were found that required evaluation under the American Society of Mechanical Engineers (ASME) Code, Section XI, Article IWL-3300.

<u>Tendon 3D104 did not meet elongation requirements during retensioning</u>: During retensioning of the tendon, the measured elongation was not within 10% of the recorded value of the last measurement as required by ASME Section XI, Subsection IWL-3221.1(d). An evaluation of the variation was completed considering the testing of a tendon wire, the tendon liftoff values, and other factors, and the condition was found to be acceptable.

Tendon 31H05 was found with one missing buttonhead: During the inspection of the tendon, it was discovered that the field end was missing one buttonhead. The requirement of IWL-3221.3(c) states that the condition is acceptable if the detached buttonhead was documented and accepted during a preservice examination, or during a previous inservice examination. Because this condition had not previously been identified, an evaluation was completed, and the condition was determined to be acceptable.

Three protruding wires on Tendon 31H19: After wire removal and retensioning of the tendon, three buttonheads did not fully seat against the anchorhead. The requirement of IWL-3221.3(c) states that the condition is acceptable if the unseated wires were documented and accepted during a preservice examination, or during a previous inservice examination. Because this condition had not previously been identified, an evaluation was completed, and the condition was determined to be acceptable.

Two cracks wider than 0.010” were detected at the shop end of Tendon 31H08: During the inspection of Tendon 31H08, it was discovered that the field end had two cracks emanating from the bearing plate corners. The upper crack was 0.060”, and the bottom crack was 0.020” at the widest point of the cracks. The requirement of IWL-3221.3(d) states that the condition is acceptable if the cracks do not exceed 0.01” in width. This condition was evaluated and determined to be acceptable. The result of the evaluation was the sealing of the cracks to eliminate the degradation mechanism, and the establishment of an increased monitoring plan to insure there are no changes in the cracks.

1R21 Coatings Assessment Inspection

Degraded Coatings - 5 areas - 23 ft^2

Unqualified Coatings – 5 areas – Electrical Panels - 23 ft^2
Unqualified Coatings – 6 areas – Valve Operators - 48 ft^2
Unqualified Coatings – 1 area – Hanger HS101 – 3.5 ft^2
Unqualified Coatings – 1 area – Misc. Wall Stenciling –1000 ft^2

Based on observations, no coatings requiring “Immediate Repair” were identified, thus none were categorized as such.

1R22 Coatings Assessment Inspection

Degraded Coatings - 7 areas on liner plate – Blistering – 235 ft^2
Degraded Coatings – 1 area top of D-ring wall (S Cavity) – cracking – 10 ft^2

Unqualified Coatings – 6 areas – Elevator doors and frame – 297.5 ft^2
Unqualified Coatings – 7 areas – Graffiti / Spray paint on liner plate – 63.5 ft^2
Unqualified Coatings – 7 areas – Graffiti / Spray paint on steel structure - 10 ft2

New ZOI Primer-only Coatings – 2 items – Handrails (S Cavity) - 156 ft^2

Repaired Degraded Coatings – 4 items - 18 ft^2

Repaired Unqualified Coatings – 2 items - 6 ft^2

For areas of Degraded coatings on the liner plate, it is the topcoat that is degraded. The primer coat is still intact and provides full corrosion protection of the liner plate. For areas that are considered to have an Unqualified coating, the coating is still intact and providing protection of the surfaces that they are applied to.

The estimated total volume of Degraded Qualified, Unqualified, and Primer-Only coatings that were newly identified during 1R22 and not previously accounted for was determined to be within the available remaining margin for coatings.

1R23 Coatings Assessment Inspection

Degraded Coatings - 5 areas on liner plate – Flaking – 14 ft^2
Degraded Coatings – 2 areas on hangers – Flaking – 4 ft^2
Degraded Coatings – 3 areas on instrument racks – Flaking – 9 ft^2
Degraded Coatings - 1 area on personnel hatch – Flaking – 2 ft^2

Unqualified Coatings - 4 areas Graffiti/Spray paint – 47 ft^2

Repaired Unqualified Coatings – 12 items - 305.75 ft^2

There were no gross coating failures identified and only minor isolated deficiencies were noted (degraded coatings). Thus, these areas are categorized as “Monitor and Trend”.

1R24 Coatings Assessment Inspection

Based on the available coatings margin of 0.623 ft^3, inspection frequency should be every other refueling outage. Inspections were last performed during 1R23; therefore, no coatings assessment inspections were performed during 1R24. However, inspections are required during 1R25.

No repairs to degraded/unqualified coatings were performed.

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. ML0811401 05) for topical report NEI 94-01, Revision 2-A.**

   Clarification: It is not clear that information provided in Table 8 satisfy the visual inspection requirements (one inspection prior to Type A test plus at least 3 other inspections before the next Type A test) of the NEI 94-01 and Condition 2 of the NRC safety evaluation.

   The following response supersedes the response provided in the RAI response.

   Section 9.2.3.2 of NEI 94-01, Revision 3-A, and Condition 2 in Section 4.1 of the NRC Safety Evaluation for topical report NEI 94-01, Revision 2-A, require supplemental general visual inspections of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity. These inspections must be conducted prior to each Type A test and during at least three other outages before the next Type A test, if the interval for the Type A test has been extended to 15 years.

   The examinations performed in accordance with the ANO-1 ASME Code, Section XI, Subsection IWE/IWL program satisfy the general visual examinations requirements specified in 10 CFR 50, Appendix J, Option B. ASME Code, Section XI, Subsection IWE, assures that at least three general visual examinations of metallic components will be conducted before the next Type A test, if the Type A test interval is extended to 15 years. This meets the requirements of Section 9.2.3.2 of NEI 94-01, Revision 3-A, and Condition 2 in Section 4.1 of the NRC Safety Evaluation for NEI 94-01, Revision 2.

   Visual examinations of accessible concrete containment components in accordance with ASME Code, Section XI, Subsection IWL are performed every five years, resulting in at least three IWL examinations being performed during a 15-year Type A test interval.

   In addition to the IWL examinations, ANO-1 performs a visual inspection of the accessible interior and exterior of the ANO-1 Reactor Building structure prior to any Type A test. This examination is performed in sufficient detail to identify any evidence of deterioration which may affect the reactor building's structural integrity or leak tightness. The areas that are inspected include the external surface of the building, the tendon access area, the basement of the building, the wall inside the main steam safety enclosure, and the interior liner plate surface of the reactor building. The examinations of the inside of the building are performed during Mode 5 or 6. The exterior portion of the inspection may be performed during any mode but must be performed following any repair or modification to the reactor building and prior to pressurization of the building. The examination is conducted in accordance with approved plant procedures to satisfy the requirements of the 10 CFR 50, Appendix J, Testing Program. The activity is coordinated with the IWL examinations to the extent possible.

Together, these examinations assure that at least three general visual examinations of the accessible containment surfaces (exterior and interior) and one visual examination immediately prior to a Type A test will be conducted before the next Type A test, if the Type A test interval is extended to 15 years, thereby meeting the requirements of Section 9.2.3.2 of NEI 94-01, Revision 3-A, and Condition 2 in Section 4.1 of the NRC Safety Evaluation for NEI 94-01, Revision 2.

Table 8 provides an approximate schedule for the containment surface examinations, assuming the Type A test frequency is extended to 15 years.

Table 8

Calendar Year (Outage)	Type A Test (ILRT)	General Visual Examination of Accessible Exterior Surface	General Visual Examination of Accessible Interior (Liner) Surface
2005 (1R19)	X	X	X
2006			
2007 (1R20)			X
2008 (1R21)		X	
2009			X
2010 (1R22)			
2011 (1R23)			X
2012			
2013(1R24)		X	X
2014			
2015(1R25)			X
2016(1R26)			X
2017			
2018 (1R27)		X	X
2019 (1R28)	X	X	X

Attachment 2 to

1CAN101407

Replacement Markup of Technical Specification Page

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.16 Reactor Building Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the reactor building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995~~NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 2012 and Section 4.1 "Limitations and Conditions for NEI TR 94-01, Revision 2" of the NRC Safety Evaluation Report in NEI 94-01, Revision 2A, dated October 2008.~~, except that t~~ The next Type A test performed after the December 16, 2005~~April 16, 1992~~ Type A test shall be performed no later than December 16, 2020 ~~April 15, 2007~~.

In addition, the reactor building purge supply and exhaust isolation valves shall be leakage rate tested once prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days.

The peak calculated reactor building internal pressure for the design basis loss of coolant accident, P_a, is 54 psig.

The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Reactor Building leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $< 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$;

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at $\geq$ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Reactor Building Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Reactor Building Leakage Rate Testing Program.

Attachment 3 to

1CAN101407

Replacement Clean (Revised) Technical Specification Page

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.16 Reactor Building Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the reactor building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 2012 and Section 4.1 "Limitations and Conditions for NEI TR 94-01, Revision 2" of the NRC Safety Evaluation Report in NEI 94-01, Revision 2A, dated October 2008. The next Type A test performed after the December 16, 2005 Type A test shall be performed no later than December 16, 2020.

In addition, the reactor building purge supply and exhaust isolation valves shall be leakage rate tested once prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days.

The peak calculated reactor building internal pressure for the design basis loss of coolant accident, P_a, is 54 psig.

The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Reactor Building leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $< 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$;

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at ≥ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Reactor Building Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Reactor Building Leakage Rate Testing Program.