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

August 29, 2016

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

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 – RELIEF REQUEST NO. ANO1-ISI-025, RELIEF FROM AMERICAN SOCIETY OF MECHANICAL ENGINEERS SECTION XI TABLE IWB-2500-1 REQUIREMENTS (CAC NO. MF7625)

Dear Sir or Madam:

By letter dated July 13, 2016 (Agencywide Documents Access and Management System Accession No. ML16195A270), Entergy Operations, Inc. (Entergy, the licensee), submitted to the U.S. Nuclear Regulatory Commission (NRC) a request for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), for Section XI Table IWB-2500-1 requirements at Arkansas Nuclear One, Unit 1 (ANO-1).

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g)(5)(iii), the licensee requested the use of an inspection procedure at ANO-1 with a depth sizing error that is greater than the requirements of the ASME Code Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds," for the fourth 10-year inservice inspection (ISI) interval. The licensee requested relief from the requirements for ISI items on the basis that the ASME Code requirement is impractical.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that Entergy has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(g)(5)(iii). Therefore, the staff grants Relief Request No. ANO1-ISI-025 for the ANO-1, 1R26 refueling outage scheduled for the fall of 2016.

If you have any questions, please contact Stephen Koenick at (301) 415-6631 or by e-mail at Stephen.Koenick@nrc.gov.

Sincerely,

Shaun M. Anderson, Acting Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. ANO1-ISI-025

FOR THE FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL

ENTERGY OPERATIONS, INC

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated July 13, 2016 (Agencywide Documents Access and Management System Accession No. ML16195A270), Entergy Operations, Inc. (Entergy, the licensee) submitted Relief Request No. ANO1-ISI-025 to the U.S. Nuclear Regulatory Commission (NRC) requesting the use of an inspection procedure at Arkansas Nuclear One, Unit 1 (ANO-1) with a depth sizing error that is greater than the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds," for the fourth 10-year inservice inspection (ISI) interval. This letter supersedes in its entirety the original request dated April 22, 2016 (ADAMS Accession No. ML16116A175).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g)(5)(iii), the licensee requested relief from the depth-sizing uncertainty qualification requirement for ultrasonic testing (UT) examinations conducted from the inside diameter (ID) of pipes (i.e., root mean square (RMS) error not greater than 0.125 inches), contained in ASME Code Case N-695. The licensee requested relief from the requirements for ISI items on the basis that the ASME Code requirement is impractical.

2.0 REGULATORY EVALUATION

In its letter dated July 13, 2016, the licensee requested relief from the 0.125 inch RMS error depth-sizing acceptance criteria contained in ASME Code Case N-695 pursuant to 10 CFR 50.55a(g)(5)(iii).

ASME Code Case N-695 is accepted for use in NRC Regulatory Guide (RG) 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1" (ADAMS Accession No. ML13339A689) and incorporated by reference in 10 CFR 50.55a(a).

Enclosure

Section 50.55a(g)(4)(ii) of 10 CFR states, in part, that "Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (a) of this section 12 months before the start of the 120-month inspection interval (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147...."

Section 50.55a(g)(5)(iii) of 10 CFR states, in part, that licensees may determine that conformance with certain Code requirements is impractical and that the licensee shall notify the Commission and submit information in support of the determination.

Section 50.55a(g)(6)(i) of 10 CFR states, in part, that the Commission will evaluate determinations under paragraph (g)(5) of this section that Code requirements are impractical and that the Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property.

ASME Code Case N-770-1, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR [pressurized-water reactor] Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities," requires subsequent volumetric examination of all Inspection Item B welds, as defined in Table 1 of the code case, at a frequency of every second inspection period not to exceed 7 years. Table IWB-2500-1, Examination Category B-F, "Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles," Item B5.10 requires a volumetric and surface examination of the weld volume as identified in Figure IWB-2500-8.

The licensee is using a risk-informed ISI program as per ASME Code Case N-716-1 "Alternative Piping Classification and Examination Requirements." ASME Code Case N-716-1 classifies these welds as R-A Item R1.11/15 which requires the same volumetric examination as Category B-F, Item B5.10 but no surface examination.

The volumetric examination is to be conducted in accordance with ASME Section XI, Mandatory Appendix VIII; Supplement 10, as modified by ASME Code Case N-695. ASME Code Case N-695 is listed as acceptable for use in RG 1.147, Revision 17. This code case provides alternatives to the requirements of Appendix VIII, Supplement 10, but Paragraph 3.3(c) of this case requires that "Examination procedures, equipment, and personnel are qualified for depth-sizing when the RMS error of the flaw depth measurements, as compared to the true flaw depths, do not exceed 0.125 in. (3mm)."

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to grant, the relief requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 The Licensee's Relief Request

The welds covered by ANO1-ISI-025 are the two core flood piping safe-end to nozzle dissimilar metal welds.

The current Code of record governing inservice inspection for ANO-1 is the 2001 Edition with the 2003 Addenda.

During the upcoming 2016 fall refueling outage (1R26), the licensee will perform UT of the two core flood piping safe-end to nozzle dissimilar metal welds. These examinations will be performed from the ID of the weld. Code Case N-695 will be used as the basis for performing these examinations. To date, although examination vendors have qualified for detection and length sizing on these welds, the examination vendors have not met the RMS error requirement for depth sizing. The licensee proposes to use Code Case N-695 with a RMS error of 0.189 inches instead of the 0.125 inches specified for depth sizing in the code case.

Further, as part of the proposed alternative, if a reportable flaw is detected and determined to be ID surface connected during examination of the welds in accordance with this relief request, Entergy will provide a flaw evaluation including the measured flaw size as determined by ultrasonic examination for NRC review. Eddy current testing will be used to determine if flaws are surface connected.

The licensee is requesting authorization to perform the proposed alternative to the Code requirement during the ANO-1, 1R26 refueling outage scheduled for the fall of 2016.

3.2 NRC Staff Evaluation

The licensee will use NRC-approved Code Case N-695 to satisfy the requirements of ASME Code, Section XI, Appendix VIII, Supplement 10. Code Case N-695 requires that procedures used to inspect welds from the ID of the pipe be qualified by performance demonstration. The acceptance criterion in Code Case N-695 specifies that the RMS error of the examination procedures shall not be greater than 0.125 inches. The licensee's inspection vendor was able to depth size with an RMS error of 0.189 inches. The licensee is requesting relief from the 0.125 inch depth sizing requirement in ASME Code Case N-695 in accordance with 10 CFR 50.55a(g)(5)(iii).

The NRC staff has confirmed that since 2002, the industry has not been able to satisfy the RMS error acceptance criterion of less than 0.125 inches when qualifying the volumetric examination inspection procedures performed from the ID of a pipe. Developing new technology capable of meeting the 0.125 inch RMS error and qualifying the new technology to meet the requirements of ASME Code Case N-695 is impractical. The NRC staff concludes that this repeated inability to qualify ID UT inspection techniques in accordance with ASME Code Case N-695 constitutes an impracticality as described in 10 CFR 50.55a(g)(5)(iii).

To address the issue of increased potential for undersizing of flaws by ID UT inspection procedures that do not meet ASME Code Case N-695 acceptance criterion, in 2012, the NRC staff, in conjunction with personnel from the Performance Demonstration Initiative, examined the proprietary UT examination data set compiled from all attempts to date to qualify ID UT inspection procedures to the acceptance criterion contained in ASME Code Case N-695. Based on this examination, the NRC staff concluded that:

(a) For flaw depths less than or equal to 50 percent pipe wall thickness, a flaw could be appropriately depth sized if a correction factor is added to the measured flaw depth such that the adjusted flaw depth is equal to the measured flaw depth plus the difference between the vendor procedure qualification RMS error and 0.125 inches.

(b) For flaw depths greater than 50 percent wall thickness, the variability of sizing errors is sufficiently large so that no single mathematic flaw size adjustment formula is sufficient to provide reasonable assurance of appropriate flaw depth-sizing. As a result, the NRC staff finds it necessary to evaluate the flaws that have depth greater than 50 percent through-wall on a case-by-case basis.

To provide reasonable assurance of the structural integrity of examined welds, the NRC staff determined that the following compensatory measures shall be applied to any inspection not meeting the 0.125 inch RMS error for depth sizing to address the measurement uncertainty in flaw depth-sizing when examining welds from the inside surface:

(1) Examine the welds under consideration using a UT technique that is qualified for flaw detection and length sizing.

(2) For flaw(s) with a measured depth of less than 50 percent of the wall thickness, the depth shall be adjusted by adding the measured flaw depth to the difference between the procedure qualification RMS error and 0.125 inches.

(3) For flaw(s) with measured depth of greater than 50 percent of the wall thickness, either the degraded weld needs to be repaired in accordance with the ASME Code, or, a flaw evaluation needs to be submitted to the NRC staff for review and approval prior to reactor startup.

(4) In addition to information normally contained in flaw evaluations performed in accordance with ASME Code, Section XI, IWB-3600, the submitted flaw evaluation shall include (a) information concerning the degradation mechanism that caused the crack, (b) information concerning the surface roughness and/or profile in the area of the examined pipe and/or weld, and (c) information concerning areas in which the UT probe may "lift off" from the surface of the pipe and/or weld.

(5) Perform eddy current examination(s) to confirm whether a flaw is connected to the ID of the pipe and/or weld.

The licensee included the following items as requirements of the request and as a regulatory commitment:

> If a reportable flaw is detected and determined to be inside diameter (ID) surface connected during examination of the welds in accordance with the Relief Request ANO-1-ISI-025, Entergy will provide a flaw evaluation including the measured flaw size as determined by ultrasonic examination for NRC review. Eddy current testing will be used to determine if flaws are surface connected. Additional data including details of the surrounding ID surface contour in the region of the flaw and percentage of the examination area where ultrasonic testing (UT) probe lift-off is evident, if any, will be included.
>
> In the event that any flaw(s) requiring depth sizing are detected during examination of welds in accordance with the Relief Request ANO1-ISI-025, the following criteria shall be implemented:
>
> - Flaws detected and measured as less than 50 percent through-wall in depth shall be adjusted by adding a correction factor to the flaw depth such that the adjusted flaw depth is equal to the *measured flaw depth* + (applicable vendor [RMS error] – 0.125 in.), prior to comparison to the applicable acceptance criteria;
>
> - For flaws detected and measured as 50 percent through-wall depth or greater and to remain in service without mitigation or repair, Entergy shall submit flaw evaluation(s) for review and approval prior to reactor startup. The flaw evaluation shall include:
>
>   - Information concerning the mechanism that caused the flaw
>
>   - Information concerning the inside surface roughness and/or profile of the region surrounding the flaw in the examined piping weld
>
>   - Information concerning areas where UT probe lift-off is observed, if any.

These regulatory commitments are considered conditions of the relief as per 10 CFR 50.55a(g)(6)(i).

The NRC staff concludes that the licensee's alternative is consistent with the compensatory measures discussed above, because (1) the licensee will add the correction factor to the crack tip(s); (2) the licensee will use eddy current testing to verify whether an embedded flaw is connected to the inside surface; and (3) the licensee will submit any flaw analysis for flaws greater than 50 percent through-wall to the NRC staff for review and approval prior to startup.

Therefore, the NRC staff determines relief from the depth-sizing RMS error acceptance criterion of ASME Code Case N-695 and using a vendor with a 0.189 inch RMS error for depth sizing provides reasonable assurance of the structural integrity and leak tightness in the subject welds.

4.0 CONCLUSION

As set forth above, the NRC staff determines that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(g)(5)(iii). Therefore, the NRC staff grants the licensee's Relief Request No. ANO1-ISI-025 at Arkansas Nuclear One, Unit 1 for the ANO-1, 1R26 refueling outage scheduled for the fall of 2016.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Stephen Cumblidge

Date: August 29, 2016

If you have any questions, please contact Stephen Koenick at (301) 415-6631 or by e-mail at Stephen.Koenick@nrc.gov.

Sincerely,

/RA/

Shaun M. Anderson, Acting Chief
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosure:
Safety Evaluation

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DATE	8/25/16	8/25/16	8/16/16	8/29/16

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