



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

June 20, 2019

Mr. Steve Snider
Vice President
Nuclear Engineering
Duke Energy
526 South Church Street, EC-07H
Charlotte, NC 28202

SUBJECT: CATAWBA NUCLEAR STATION, UNIT NOS. 1 AND 2; MCGUIRE NUCLEAR STATION, UNIT NOS. 1 AND 2; OCONEE NUCLEAR STATION, UNIT NOS. 1, 2, AND 3; SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1; AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ALTERNATIVE TO DEPTH SIZING QUALIFICATION EXAMINATION OF WELDS (EPID L-2018-LLR-0117)

Dear Mr. Snider:

By letter dated September 6, 2018, as supplemented by letters dated November 12, 2018, and February 11, 2019, Duke Energy Carolinas, LLC and Duke Energy Progress, LLC (the licensee) requested relief from the inspection requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, at Catawba Nuclear Station, Unit Nos. 1 and 2; McGuire Nuclear Station, Unit Nos. 1 and 2; Oconee Nuclear Station, Unit Nos. 1, 2, and 3; Shearon Harris Nuclear Power Plant, Unit 1; and H. B. Robinson Steam Electric Plant, Unit No. 2. The February 11, 2019, letter revised the request in the November 12, 2018, letter and supersedes the September 6, 2018, and November 12, 2018, letters in their entirety.

The licensee submitted for U. S. Nuclear Regulatory Commission (NRC) review and approval Relief Request 18-GO-001 to use an alternative depth sizing qualification in the ultrasonic examination of welds. The licensee requested relief from the depth-sizing uncertainty qualification requirement for ultrasonic testing examinations conducted from the inside diameter of pipes on the basis that the ASME Code requirement is impractical. Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g)(5)(iii), the licensee requested relief and to use alternative requirements for in-service inspection items on the basis that the code requirement is impractical.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, 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 Therefore, the NRC staff grants the relief for the facilities requested in the licensee's application, as superseded, for the duration of the applicable 10-year inservice inspection intervals.

If you have any questions, please contact the Duke Fleet Project Manager, Dennis Galvin at 301-415-6256 or via e-mail at Dennis Galvin@nrc.gov.

Sincerely,

/RA/

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369,
50-370, 50-269, 50-270,
50-287, 50-400, and 50-261

Enclosure:
Safety Evaluation

SUBJECT: CATAWBA NUCLEAR STATION, UNIT NOS. 1 AND 2; MCGUIRE NUCLEAR STATION, UNIT NOS. 1 AND 2; OCONEE NUCLEAR STATION, UNIT NOS. 1, 2, AND 3; SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1; AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ALTERNATIVE TO DEPTH SIZING QUALIFICATION EXAMINATION OF WELDS (EPID L-2018-LLR-0117) DATED JUNE 20, 2019

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELIEF REQUEST 18-GO-001
PROPOSED ALTERNATIVE FOR DEPTH SIZING QUALIFICATION
EXAMINATION OF WELDS
CATAWBA NUCLEAR STATION, UNITS 1 AND 2
MCGUIRE NUCLEAR STATION, UNITS 1 AND 2
OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2
DUKE ENERGY CAROLINAS, LLC AND DUKE ENERGY PROGRESS, LLC
DOCKET NOS. 50-413, 50-414, 50-369, 50-370,
50-269, 50-270, 50-287, 50-400, 50-261

1.0 INTRODUCTION

By letter dated September 6, 2018 (Reference 1), as supplemented by letters dated November 12, 2018, and February 11, 2019 (References 2 and 3, respectively), Duke Energy Carolinas, LLC and Duke Energy Progress, LLC (the licensee) requested relief from the inspection requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, at Catawba Nuclear Station, Units 1 and 2; McGuire Nuclear Station, Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3; Shearon Harris Nuclear Power Plant, Unit 1; and H.B. Robinson Steam Electric Plant, Unit 2. The licensee submitted for U. S. Nuclear Regulatory Commission (NRC) review and approval Relief Request 18-GO-001 to use an alternative depth sizing qualification in the ultrasonic examination of welds. The licensee requested relief from the depth-sizing uncertainty qualification requirement for ultrasonic testing (UT) examinations conducted from the inside diameter (ID) of pipes on the basis that the ASME Code requirement is impractical.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g)(5)(iii), the licensee requested relief and to use alternative requirements for in-service inspection items on the basis that the code requirement is impractical.

In the supplement dated February 11, 2019, the licensee revised Relief Request 18-GO-001 as a result of the NRC staff's request for additional information. The NRC staff's safety evaluation is based on the review of the revised relief request dated February 11, 2019.

2.0 REGULATORY EVALUATION

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 [RG] 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.

Section 50.55a(g)(6)(ii)(F) of 10 CFR requires the use of ASME Code Case N-770-2, "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," to examine dissimilar metal butt welds. The code case requires subsequent volumetric examination of all Inspection Item B welds at a frequency of every second inspection period not to exceed 7 years. The ASME Code, Section XI, 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.

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 18-GO-001

3.1.1 ASME Code Component(s) Affected

The affected components are ASME Code Class 1 Dissimilar Metal and Alloy 82/182 Welds listed in Tables 1A through 1G.

Table 1A Catawba Unit 1 Welds

Component ID	ASME Category or Code Case/ Inspection Item	Description	Nominal Nozzle Wall Thickness at Weld (Approximate)
1RPV-W52-01 1RPV-W52-02 1RPV-W52-03 1RPV-W52-04	N-770-2/B	Upper Head Injection Upper Tube to Lower Tube Welds (Auxiliary Head Adapter Welds)	0.65"
1RPV-W51-01-SE 1RPV-W51-02-SE 1RPV-W51-03-SE 1RPV-W51-04-SE	N-770-2/B	Reactor Vessel Closure Head to Upper Head Injection Lower Tube Welds (Auxiliary Head Adapter Welds)	0.65"

Table 1B Catawba Unit 2 Welds

Component ID	ASME Category or Code Case/ Inspection Item	Description	Nominal Nozzle Wall Thickness at Weld (Approximate)
2RPV-W79-101 2RPV-W80-101 2RPV-W81-101 2RPV-W82-101	N-770-2/B	Upper Head Injection Upper Tube to Lower Tube Welds (Auxiliary Head Adapter Welds)	0.65"
2RPV-W79-101SE 2RPV-W80-101SE 2RPV-W81-101SE 2RPV-W82-101SE	N-770-2/B	Reactor Vessel Closure Head to Upper Head Injection Lower Tube Welds (Auxiliary Head Adapter Welds)	0.65"
2RPV-201-121ASE 2RPV-201-121BSE 2RPV-201-121CSE 2RPV-201-121DSE	N-770-2/B	Reactor Vessel Cold Leg Nozzle to Safe End Welds	2.3"
2RPV-202-121ASE 2RPV-202-121BSE 2RPV-202-121CSE 2RPV-202-121DSE	N-770-2/A-2	Reactor Vessel Hot Leg Nozzle to Safe End Welds	2.4"

Table 1C McGuire Unit 1 Welds

Component ID	ASME Category or Code Case/ Inspection Item	Description	Nominal Nozzle Wall Thickness at Weld (Approximate)
1RPV3-445E-SE 1RPV3-445F-SE 1RPV3-445G-SE 1RPV3-445H-SE	N-770-2/A-2	Reactor Vessel Hot Leg Nozzle to Safe End Welds	2.5"
1RPV3-445A-SE 1RPV3-445B-SE 1RPV3-445C-SE 1RPV3-445D-SE	N-770-2/B	Reactor Vessel Cold Leg Nozzle to Safe End Welds	2.4"
1RPV1-462C-SE 1RPV1-462B-SE 1RPV1-462A-SE 1RPV1-462D-SE	N-770-2/B	Reactor Vessel Closure Head to Upper Head Injection Lower Tube Welds (Auxiliary Head Adapter Welds)	0.63"
1NI1FW-38-1 1NI1FW-38-2 1NI1FW-38-3 1NI1FW-38-4	N-770-2/B	Upper Head Injection Upper Tube to Lower Tube Welds (Auxiliary Head Adapter Welds)	0.63"

Table 1D McGuire Unit 2 Welds

Component ID	ASME Category or Code Case/ Inspection Item	Description	Nominal Nozzle Wall Thickness at Weld (Approximate)
2RPV-W51-01-SE 2RPV-W51-02-SE 2RPV-W51-03-SE 2RPV-W51-04-SE	N-770-2/B	Reactor Vessel Closure Head to Upper Head Injection Lower Tube Welds (Auxiliary Head Adapter Welds)	0.63"
2RPV-W52-01 2RPV-W52-02 2RPV-W52-03 2RPV-W52-04	N-770-2/B	Upper Head Injection Upper Tube to Lower Tube Welds (Auxiliary Head Adapter Welds)	0.63"

Table 1E Oconee Units 1, 2, and 3 Welds

Component ID	ASME Category or Code Case/ Inspection Item	Description	Nominal Nozzle Wall Thickness at Weld (Approximate)
1-RPV-WR-53 1-RPV-WR-53A	N-770-2/B and B-F ¹ /B5.10	Unit 1 Reactor Vessel Cold Leg Core Flood Nozzle-to-Safe End Welds	1.5"
2-RPV-WR-53 2-RPV-WR-53A	N-770-2/B and N-716-1, R-A/R1.11 and R1.15	Unit 2 Reactor Vessel Cold Leg Core Flood Nozzle-to-Safe End Welds	1.5"
3-RPV-WR-53 3-RPV-WR-53A	N-770-2/B and N-716-1, R-A/R1.11 and R1.15	Unit 3 Reactor Vessel Cold Leg Core Flood Nozzle-to-Safe End Welds	1.5"

Table 1F Robinson Unit 2 Welds

Component ID	ASME Category or Code Case/ Inspection Item	Description	Nominal Nozzle Wall Thickness at Weld (Approximate)
107/01DM 107A/01DM 107B/01DM	N-770-2/A-2 B-F ² /B5.10	Reactor Vessel Hot Leg Nozzle to Safe End Welds	2.4"
107/14DM 107A/14DM 107B/14DM	N-770-2/B B-F ² /B5.10	Reactor Vessel Cold Leg Nozzle to Safe End Welds	2.4"

Table 1G Harris Unit 1 Welds

Component ID	ASME Category or Code Case/ Inspection Item	Description	Nominal Nozzle Wall Thickness at Weld (Approximate)
II-RV-001 RVNOZAI-N-01SE II-RV-001 RVNOZBI-N-03SE II-RV-001 RVNOZCI-N-05SE	N-770-2/B	Reactor Vessel Cold Leg Nozzle to Safe End Welds	2.4"
II-RV-001 RVNOZAO-N-06SE II-RV-001 RVNOZBO-N-02SE II-RV-001 RVNOZCO-N-04SE	N-770-2/D	Reactor Vessel Hot Leg Nozzle to Safe End Welds	2.5"

Licensee's Footnotes:

¹ Oconee Unit 1 is in the process of implementing Code Case N-716-1, [Alternative Classification and Examination Requirements, Section XI, Division 1]. Category B-F, Item B5.10 will be replaced by the applicable Category R-A, Item Numbers for welds 1-RPV-WR-53 and 1-RPV-WR-53A when the inservice inspection plan and schedule are revised to implement this code case.

² Robinson Unit 2 is in the process of implementing Code Case N-716-1. Category B-F, Item B5.10 will be replaced by the applicable Category R-A, Item Numbers for the welds listed in Table 1F when the inservice inspection plan and schedule are revised to implement this code case.

3.1.2 Applicable Code Edition and Addenda

The applicable code of record is the ASME Code, Section XI, 2007 Edition with the 2008 Addenda. The inservice inspection (ISI) interval for the subject plants are identified in Table 2.

Table 2

Plant/Unit(s)	ISI Interval	Interval Start Date	Current Interval End Date
Catawba Nuclear Station Units 1 and 2	Fourth	08/19/2015 (Unit 1) 08/19/2015 (Unit 2)	12/06/2024 (Unit 1) 02/24/2026 (Unit 2)
McGuire Nuclear Station Units 1 and 2	Fourth	12/01/2011 (Unit 1) 07/15/2014 (Unit 2)	11/30/2021 (Unit 1) 12/14/2024 (Unit 2)
Oconee Nuclear Station Units 1, 2 and 3	Fifth	07/15/2014	07/15/2024
Robinson Nuclear Plant Unit 2	Fifth	07/21/2012	02/19/2023
Shearon Harris Nuclear Plant Unit 1	Fourth	09/09/2017	09/08/2027

3.1.3 Applicable Code Requirement

ASME Code Case N-770-2, as referenced in 10 CFR 50.55a(g)(6)(ii)(F), requires ultrasonic examination of Category A-2, B, and D welds fabricated from Alloy 82/182 material. Table 1, Note 4 of this code case requires that ultrasonic examinations meet the applicable requirements of the ASME Code, Section XI, Mandatory Appendix VIII.

For Category B-F welds, the ASME Code, Section XI, IWA-2232 requires that ultrasonic examinations be conducted in accordance with the ASME Code, Section XI, Mandatory Appendix I. Mandatory Appendix I, I-2220 requires that ultrasonic examinations be qualified by performance demonstration in accordance with the ASME Code, Section XI, Mandatory Appendix VIII.

For Category R-A welds (Oconee and Robinson), examinations are performed in accordance with ASME Code Case N-716-1. This case does not provide alternative requirements to those specified in IWA-2232, so the requirements of IWA-2232 apply.

ASME Code, Section XI, Mandatory Appendix VIII, Supplement 2, "Qualification Requirements for Wrought Austenitic Piping Welds," Paragraph 3.2(b) requires that examination procedures, equipment, and personnel are qualified for depth-sizing if the "...RMS error of the flaw depths estimated by ultrasonics, as compared with the true depths, do not exceed 0.125 in. [inches] (3 mm) [millimeters]..." RMS (root mean square) is defined in Mandatory Appendix VIII, VIII-3120.

ASME Code, Section XI, Mandatory Appendix VIII, Supplement 10, "Qualification Requirements for Dissimilar Metal Piping Welds," Paragraph 3.3(c) 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. (3 mm)..."

3.1.4 Impracticality of Compliance

The licensee stated that ASME Code Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds, Section XI, Division 1," is approved for use in RG 1.147, Revision 18, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1" (Reference 4). This code case provides alternatives to the requirements of Appendix VIII, Supplements 2 and 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 inches.

The licensee stated that the requirement for the 0.125 inches RMS error depth sizing accuracy criterion is impractical because, although examination vendors have qualified for detection and length sizing in accordance with the requirements for examinations from the ID surface, vendors have not met the established RMS error of 0.125 inches for indication depth sizing of welds 2.1 inches or greater in wall thickness. The licensee stated that several process enhancements including systems, new search units, and software modifications have been implemented, but these have not been successful in demonstrating the ability to meet the required measurement error accuracy. The licensee asserted that achieving the RMS error of 0.125 inches is impractical for use with the ID ultrasonic examination technology employed in the qualification efforts.

The licensee contended that compliance with the requirements of Appendix VIII, Supplement 2, paragraph 3.2(b) and Supplement 10, paragraph 3.3(c) is possible for examinations performed from the outside diameter surface. However, the licensee stated that examinations performed from the outside diameter surface result in significant and unnecessary personnel radiation exposure that can be avoided by performing these examinations remotely from the ID surface.

The licensee reported that its vendors have demonstrated RMS errors between 0.179 inches and 0.212 inches.

3.1.5 Proposed Alternative

In lieu of the ASME Code, Section XI, Appendix VIII requirements, the licensee proposed to use ASME Code Cases N-695-1, "Qualification Requirements for Dissimilar Metal Piping Welds, Section XI, Division 1," and N-696-1 "Qualification Requirements for Mandatory Appendix VIII Piping Examinations Conducted From the Inside Surface, Section XI, Division 1," to perform qualified ultrasonic examinations from the inside diameter surface of the subject welds.

The relief request will follow the specific provision in N-695-1 and N-696-1 as shown below:

Paragraph 3.3(d) of ASME Code Case N-695-1 states:

"...(d) For qualifications from the inside-surface, examination procedures, equipment, and personnel are qualified for depth sizing if the RMS error of the flaw depth measurements, as compared to the true flaw depths, does not exceed 0.125 in. (3 mm) for piping less than 2.1 in. (54 mm) in thickness, or 0.250 in. (6 mm) for piping 2.1 in. (54 mm) or greater in thickness..."

Paragraph 3.3(c) of ASME Code Case N-696-1 states:

"...(c) Supplement 2 examination procedures, equipment, and personnel are qualified for depth-sizing if the RMS error of the flaw depth measurements as compared to the true flaw depths, does not exceed 0.125 in. (3 mm) for piping less than 2.1 in. (54 mm) in thickness, or 0.250 in. (6 mm) for piping 2.1 in. (54 mm) or greater in thickness, when they are combined with a successful Supplement 10 qualification..."

In addition, the proposed alternative will satisfy the following requirements:

Personnel, procedures, and equipment shall satisfy all requirements of Code Cases N-695-1 and N-696-1.

Flaws detected and measured as less than 50 percent through-wall depth shall be sized using personnel, procedures, and equipment qualified to meet the requirements of ASME Code Cases N-695-1 and N-696-1.

For all welds listed in this request, if any inner diameter surface-breaking flaws are detected and measured (from the inner diameter surface) as 50 percent through-wall depth or greater, Duke Energy shall repair the indications or shall perform a volumetric examination from the outside diameter surface of the component to determine the flaw depth and shall perform flaw evaluations and shall submit the evaluations to the NRC for review and approval prior to reactor startup. The submitted flaw evaluation will include: (a) information concerning the mechanism that caused the flaw, (b) information

concerning the surface roughness and/or profile in the area of the examined pipe and/or weld, and (c) an estimate of the percentage of potential surface areas with UT probe "lift off" from the surface of the pipe and/or weld.

The licensee specifies that all other requirements of the ASME Code, Section XI and Code Case N-770-2 [as conditioned by 10 CFR 50.55a(g)(6)(ii)(F)] for which relief was not specifically requested apply, including the third party review by the Authorized Nuclear Inservice Inspector.

The licensee stated that the proposed alternative for welds less than 2.1 inches in thickness is essentially identical to that approved for use during the Catawba Unit 1 Third Inservice Inspection Interval, dated October 26, 2015 (Reference 5).

The licensee further stated that the proposed alternative may be used in lieu of the alternative approved in Relief Request RR-08, for the Robinson Nuclear Plant, Unit 2, Fifth Inservice Inspection Interval, dated July 16, 2013 (Reference 6).

In the February 11, 2019, letter, the licensee stated that it intends to use only vendors who have demonstrated acceptable RMS error in accordance with the limits specified in Code Cases N-695-1 and N-696-1.

3.1.6 Basis for Use

The technical basis of Code Cases N-695-1 and 696-1 is presented in Electric Power Research Institute (EPRI) Technical Report 3002000612, "Materials Reliability Program: Technical Basis for Change to American Society of Mechanical Engineers (ASME) Section XI Appendix VIII Root-Mean-Square Error (RMSE) Requirement for Qualification of Depth-Sizing for Ultrasonic Testing (UT) Performed from the Inner Diameter (ID) of Large-Diameter Thick-Wall Supplement 2, 10, and 14 Piping Welds (MRP-373), October 2013."

The key technical difference between Code Cases N-695 and N-695-1 is the allowance of a RMS error up to 0.250 inches for pipe components 2.1 inches or greater in wall thickness.

The two key technical differences between Code Cases N-696 and N-696-1 are: (1) the allowance of a RMS error up to 0.250 inches for components 2.1 inches or greater in wall thickness, and (2) the deletion of a requirement for the specimen set for Supplement 3 of the ASME Code, Section XI, Appendix VIII qualification to include at least three flaws in ferritic material. A statement was added that depth sizing qualification for ferritic piping shall be performed in accordance with Supplement 3.

The licensee explained that to date, for components 2.1 inches or greater in wall thickness, no UT method has met the ASME Code, Section XI, Appendix VIII qualification requirements of a RMS error smaller than or equal to 0.125 inches for the depth-sizing of flaws from the inner surface in reactor pressure vessel nozzles, according to Supplements 2, 10 or 14 of the ASME Code, Section XI, Appendix VIII. These efforts have shown the impracticality of obtaining the RMS error of 0.125 inches, given the challenges of weld geometry, rough surfaces, multiple materials, and microstructural anisotropies.

The licensee stated that EPRI implemented an alternate criterion that has been used by utilities in relief requests to the NRC on multiple occasions.

3.1.7 Duration of Proposed Alternative

The licensee requested this alternative for the inservice inspection intervals listed in Table 2 of the relief request and as shown in Table 2 of this safety evaluation.

3.2 NRC STAFF EVALUATION

Background

The NRC staff notes that ultrasonic examination of welds is to be conducted in accordance with the ASME Code, Section XI, Mandatory Appendix VIII, which contains various supplements for the examination qualification of various piping components. Supplement 2 provides qualification requirements for examining wrought austenitic piping welds. Supplement 3 provides qualification requirements for examining ferritic piping welds. Supplement 10 provides qualification requirements for examining dissimilar metal piping welds. Supplement 14 provides qualification requirements for coordinated implementation of supplements 2, 3, and 10 for examinations performed from the inside surface of piping.

The industry's ultrasonic examination technology cannot satisfy the RMS error requirement for the depth sizing as specified in Supplements 2, 3, and 10. On May 21, 2003, the ASME Committees published Code Case N-695, which provides alternatives to the requirements of Appendix VIII, Supplement 10. The NRC staff has approved the use of ASME Code Case N-695 in RG 1.147, Revision 18, as incorporated by reference in 10 CFR 50.55a(a). Code Case N-695 is specifically applicable to the ultrasonic examination of dissimilar metal butt welds.

On May 21, 2003, ASME Committees published Code Case N-696, which provides alternatives to the qualification requirements of Appendix VIII, Supplements, 2, 3, and 10. The NRC staff has approved the use of ASME Code Case N-696 in RG 1.147, Revision 18. Code Case N-696 is applicable to the ultrasonic examination of ferritic pipe welds.

On May 7, 2014, and December 31, 2014, ASME Committees published Code Cases N-696-1 and N-695-1, respectively. The NRC has not approved these two code cases for generic use.

The NRC staff notes that the ASME Code committees have approved Code Cases N-695-1 and N-696-1 to allow a maximum RMS error of 0.250 inches for components 2.1 inches or greater in wall thickness as a permanent solution to this issue. EPRI Technical Report 3002000612 contains the basis for this change.

Deterministic and probabilistic approaches were applied in EPRI Technical Report 3002000612 to show the acceptability of alternative depth-sizing RMS error requirements. The EPRI Technical Report stated that in a deterministic assessment, each input is set to a conservative value to account for uncertainty and variability. This methodology compounds various conservative margins that can lead to unrealistic results and mask the true extent of conservatism in the final calculation results. These deterministic evaluations demonstrate that a depth-sizing RMS error of 0.250 inches provides a structural margin for large-diameter PWR piping welds compared to that for large-diameter boiling water reactor (BWR) piping welds inspected with a depth-sizing RMS error of 0.125 inches. The RMS error of 0.125 inches currently required for the qualification of UT depth-sizing in accordance with Supplements 2, 10, and 14 of ASME Code, Section XI, Appendix VIII was originally a deterministic assessment based on the depth-sizing error that was achievable for UT of BWR piping welds in the 1980's.

The NRC staff finds that the use of probabilistic evaluations facilitates the incorporation of uncertainties, variability, and randomness important in the evaluation of leakage risk. Probabilistic assessment provides a direct uncertainty estimate for key outputs so the specific degree of conservatism in the result can be assessed. Probabilistic evaluations show that alternative depth-sizing RMS error requirement of 0.250 inches has little effect on probability of through-wall penetration of a primary water stress corrosion cracking (PWSCC) flaw. The probability of leakage due to through-wall PWSCC is a key indication of the effect of PWSCC on structural integrity as large flaws are necessary to produce both leakage and pressure-boundary rupture.

The NRC staff notes that EPRI Technical Report evaluated the effect of uncertainty in flaw sizing on structural integrity of piping systems through net section collapse calculation based on a circumferential flaw. The calculations of net section collapse are based on standard equations included in ASME Code, Section XI, for evaluating acceptability for continued service of piping systems with circumferential planar flaws connected to the inside surface.

Based on these facts, EPRI recommended and ASME approved changes incorporated in N-695-1 and N-696-1 that allow RMS error depth-sizing qualification to be changed from 0.125 inches to 0.250 inches for large-diameter PWR piping welds having a nominal wall thickness of at least 2.1 inches examined from the inside surface.

Root-Mean-Square Error

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 UT examination procedures performed from the inside surface of a pipe. The NRC staff understands that developing and qualifying the UT technology capable of meeting the 0.125 inches RMS error is impractical. The NRC staff concludes that the inability to qualify inside diameter ultrasonic examination techniques to meet the 0.125 inches RMS error acceptance criterion constitutes impracticality.

In 2012, to address the potential for undersizing of flaws by ID ultrasonic examination procedures that do not meet the 0.125 inches RMS error acceptance criterion, the NRC staff and U.S. nuclear utilities performance demonstration initiative (PDI) personnel examined the proprietary ultrasonic examination data set compiled from all attempts to qualify ID ultrasonic examination procedures to the RMS error acceptance criterion.

The NRC staff notes that the U.S. nuclear utilities have created the PDI to implement performance demonstration requirements contained in the ASME Code, Section XI, Appendix VIII. The industry's PDI has evolved into a PDI program for qualifying equipment, procedures, and personnel in accordance with the UT criteria of the ASME Code, Section XI, Appendix VIII. The NRC staff routinely assesses the PDI program for consistency with the current edition of the ASME Code Section XI, Appendix VIII. The NRC staff recognizes that the PDI program does not fully comport with the existing requirements of the ASME Code, Section XI, Appendix VIII. However, through periodic public meetings between the industry and NRC, the NRC staff has determined that the PDI program provides an acceptable level of quality and safety.

Based on its independent verification, the NRC staff concluded at the time 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. The correction factor is discussed further in this safety evaluation.

(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 that it is necessary to evaluate the flaws that have depth greater than 50 percent through-wall on a case-by-case basis.

The provisions of Code Cases N-695-1 and N-696-1 specify that for the inside surface examination, the RMS error of the flaw depth, as compared to the true flaw depths, does not exceed 0.125 inches for piping less than 2.1 inches in wall thickness, or 0.250 inches for piping 2.1 inches or greater in wall thickness.

The NRC staff finds that the licensee's inspection vendor was able to depth size with an RMS error between 0.179 inches and 0.212 inches, which is less than the RMS error of 0.250 inches for welds of 2.1 inches or greater. Therefore, the proposed alternative satisfies the required RMS error of 0.250 inches for welds of 2.1 inches or greater in wall thickness as specified in Code Cases N-695-1 and N-696-1.

The NRC staff recognizes that the vendor's RMS error does not satisfy the required RMS error of 0.125 inches for the welds that have wall thickness of less than 2.1 inches. Based on the review of EPRI data set, the NRC staff finds that the licensee vendor's RMS error is adequate to provide reasonable assurance of flaw depth sizing for the welds that has wall thickness less than 2.1 inches. The NRC staff has determined that the following three compensatory measures applied by the licensee to any UT examination of welds from the inside surface of a pipe provide reasonable assurance of structural integrity of examined welds:

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

The NRC staff finds that the proposed alternative satisfies the first compensatory measure because the licensee will examine the subject welds in accordance with Code Cases N-695-1 and N-696-1 which provides qualification requirements for ultrasonic examination.

(2) Repair the degraded weld in accordance with the ASME Code, Section XI, or, submit a flaw evaluation to NRC staff for review and approval prior to plant startup for flaw(s) with measured depth of greater than 50 percent of the wall thickness.

The NRC staff finds that in the revised relief request, the licensee states that for all welds listed in the relief request, if any ID surface-breaking flaws are detected and measured (from the ID surface) as 50 percent through-wall depth or greater, it will repair the indications or will perform a volumetric examination from the outside diameter surface of the component to determine the flaw depth and will perform flaw evaluations and will submit the evaluations to the NRC for review and approval prior to reactor startup. The NRC staff finds that the proposed alternative is consistent with the second compensatory measure and, therefore, is acceptable.

(3) In addition to information normally contained in flaw evaluations performed in accordance with the 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.

The NRC staff finds that in the revised relief request, the licensee stated that the submitted flaw evaluation will include: (a) information concerning the mechanism that caused the flaw, (b) information concerning the surface roughness and/or profile in the area of the examined pipe and/or weld, and (c) an estimate of the percentage of potential surface areas with UT probe “lift off” from the surface of the pipe and/or weld. The NRC staff finds that the proposed alternative is consistent with the third compensatory measure and therefore is acceptable.

Precedents

In the previously NRC-approved relief requests, the NRC staff required licensees to add a correction factor to the length of detected flaws. The correction factor is derived as the difference between the vendor procedure qualification RMS error and the required RMS error of 0.125 inches. However, after the publication of Code Cases N-695-1 and N-696-1, and after NRC’s review of the data set in the ultrasonic examination qualification procedures of the PDI program, the NRC staff finds that licensees do not need to add the correction factor to the flaw size. The NRC staff finds that both code cases provide sufficient requirements such that adequate flaw size could be appropriately measured by the ultrasonic examination performed in accordance with Code Cases N-695-1 and N-696-1 without adding the correction factor. Therefore, the NRC staff finds that the correction factor is no longer necessary for flaw sizing of the subject welds.

By letter dated October 26, 2015 (Reference 5), the NRC approved relief request 1-14-CN-003 that proposed to use Code Case N-695 at Catawba Nuclear Station Unit 1. At the time, the NRC required Catawba to perform eddy current examination(s), in addition to the ultrasonic examination, to confirm whether a flaw is connected to the inside surface of the pipe or weld. For the current Relief Request 18-GO-001, the NRC staff finds that the eddy current examination is no longer necessary to be performed with the ID ultrasonic examination of the subject welds because the provisions of Code Cases N-695-1 and N-696-1 provide sufficient accuracy in flaw sizing such that a flaw that is connected to the inside surface of the pipe or weld should be evident as a result of the ultrasonic examination. The NRC staff notes that in the past, licensees volunteered to perform the eddy current testing as a supplemental examination to verify whether the flaw is connected to the inside surface of the pipe when the embedded flaw is located near the inside surface of the pipe wall thickness.

Summary

In summary, the NRC staff concludes that the proposed alternative in Relief Request 18-GO-001 will provide reasonable assurance of the structural integrity and leak tightness of the subject welds because (1) based on the assessment of the PDI data, the NRC staff determines that the proposed RMS error is acceptable as it provides reasonable assurance that the ultrasonic examination has been qualified to measure the depth of flaws with a reasonable accuracy, (2) the licensee will use qualified ultrasonic examination technique in accordance with Code Cases N-695-1 and N- 696-1, and (3) the licensee will submit any flaw analysis for flaw(s) greater than 50 percent through-wall to the NRC staff for review and approval prior to startup.

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)(6)(i). Therefore, the NRC staff grants the use of Relief Request 18-GO-001, as supplemented February 11, 2019, at Catawba Nuclear Station, Units 1 and 2; McGuire Nuclear Station, Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3; Shearon Harris Nuclear Power Plant, Unit 1; and H.B. Robinson Steam Electric Plant, Unit 2 for the inservice inspection intervals listed in Table 2 of this safety evaluation.

The NRC staff notes that granting this relief request does not imply or infer the NRC's approval of Code Cases N-695-1 and N-696-1 for generic use.

All other requirements of the ASME Code, Section XI and Code Case N-770-2, as conditioned by 10 CFR 50.55a(g)(6)(ii)(F), 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.

5.0 REFERENCES

1. Donahue, J., Duke Energy, letter to U.S. Nuclear Regulatory Commission, "Relief Request in Accordance with 10 CFR 50.55a(G)(5)(lii) for an Alternative to the Depth Sizing Qualification Requirement of Appendix VIII, Supplements 2 and 10 (18-GO-001)," September 6, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18249A008).
2. Donahue, J., Duke Energy, letter to U.S. Nuclear Regulatory Commission, "Supplement to Relief Request for an Alternative to the Depth Sizing Qualification Requirement of Appendix VIII, Supplements 2 And 10 (18-GO-001)," November 12, 2018 (ADAMS Accession No. ML18316A035).
3. Snider, S., Duke Energy, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) Regarding Proposed Alternative to the Depth Sizing Qualification Requirement of Appendix VIII, Supplements 2 And 10 (18-GO-001)," February 11, 2019 (ADAMS Accession No. ML19042A314).
4. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.147, Revision 18, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," March 2017 (ADAMS Accession No. ML16321A336).
5. Pascarelli, R. J., U.S. Nuclear Regulatory Commission Duke Energy Carolinas, LLC, letter to Henderson, K., Duke Energy Carolinas, LLC, "Catawba Nuclear Station, Unit 1: Proposed Relief Request 14-CN-003, American Society of Mechanical Engineers (ASME) Boiler And Pressure Vessel Code (ASME Code), Code Case N-695 (CAC No. MF5447)," October 26, 2015 (ADAMS Accession No. ML15286A326).

6. Quichocho, J. F., U.S. Nuclear Regulatory Commission, letter to Gideon, W. R., Carolina Power & Light Company, "H. B. Robinson Steam Electric Plant, Unit No. 2 - Relief Request-08 From ASME Code Root Mean Square Error Value for the Fifth 10-Year Inservice Inspection Program Plan (TAC No. MF1015)," July 16, 2013 (ADAMS Accession No. ML13191A930).

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