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

August 27, 2015

Mr. Steven D. Capps
Vice President
McGuire Nuclear Station
Duke Energy Carolinas, LLC
12700 Hagers Ferry Road
Huntersville, NC 28078

SUBJECT: MCGUIRE NUCLEAR STATION, UNIT 1 - PROPOSED RELIEF REQUEST 15-MN-001 (TAC NO. MF5817)

Dear Mr. Capps:

By letter dated March 5, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. 15083A053), Duke Energy Carolinas, LLC (the licensee) submitted RR 15-MN-001, to certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) associated with inspection of reactor pressure vessel inlet cold leg nozzle to safe-end dissimilar metal butt welds at McGuire Nuclear Station, Unit No. 1 (MNS 1). The licensee requested authorization to use the proposed alternative pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1) on the basis that the proposed alternative would provide an acceptable level of quality and safety.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the proposed alternative provides an acceptable level of quality and safety. Therefore, the NRC staff authorizes the proposed alternative in accordance with 10 CFR 50.55a(z)(1) for the 4th 10-year inservice inspection interval, to extend inspections of the stated vessel cold leg dissimilar metal welds to no later than the 1EOC27 RFO scheduled for the fall of 2020.

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 In-service Inspector.

If you have any questions, please contact the Project Manager, G. Edward Miller at 301-415-2481 or via e-mail at ed.miller@nrc.gov.

Sincerely,

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-369

Enclosure:
Safety Evaluation

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. 15-MN-001

DUKE ENERGY CAROLINAS, LLC

MCGUIRE NUCLEAR STATION, UNIT 1

DOCKET NO. 50-369

1.0 INTRODUCTION

By letter dated March 5, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15083A053), Duke Energy Carolinas, LLC (the licensee) submitted an alternative, relief request 15-MN-001 (RR-15-MN-001), to certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) associated with inspection of reactor pressure vessel (RPV) inlet cold leg nozzle to safe-end dissimilar metal (DM) butt welds at McGuire Nuclear Station, Unit No. 1 (MNS 1). The licensee requested authorization to use the proposed alternative pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1) on the basis that the proposed alternative would provide an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

The inservice inspection (ISI) of ASME Code Class 1, 2 and 3 components is to be performed in accordance with Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code and applicable editions and addenda as required by, 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission.

10 CFR 50.55a(g)(6)(ii) states that the Commission may require the licensee to follow an augmented ISI program for systems and components for which the Commission deems that added assurance of structural reliability is necessary. 10 CFR 50.55a(g)(6)(ii)(F) requires, in part, augmented inservice volumetric inspection of class 1 piping and nozzle DM butt welds of pressurized water reactors in accordance with ASME Code Case N-770-1, subject to the conditions specified in paragraphs (2) through (10) of 10 CFR 50.55a(g)(6)(ii)(F).

Alternatives to requirements under 10 CFR 50.55a(g) may be authorized by the Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(z)(1) or 10 CFR 50.55a(z)(2). In proposing alternatives, the licensee must demonstrate that: (1) the proposed alternatives would provide an acceptable level of quality and safety; or (2) compliance with the specified

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requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on analysis of the regulatory requirements, the NRC staff finds that the regulatory authority exists to authorize the licensee's proposed alternative on the basis that the proposed alternative would provide an acceptable level of quality and safety. Accordingly, the NRC staff has reviewed and evaluated the licensee's request pursuant to 10 CFR 50.55a(z)(1).

3.0 TECHNICAL EVALUATION

3.1 Licensee Relief Request

3.1.1 Components for Which Relief Was Requested

Weld 1RPV3-445A-SE Primary Inlet Nozzle Weld - A-Loop
Weld 1RPV3-445B-SE Primary Inlet Nozzle Weld - B-Loop
Weld 1RPV3-445C-SE Primary Inlet Nozzle Weld - C-Loop
Weld 1RPV3-445D-SE Primary Inlet Nozzle Weld - D-Loop

3.1.2 Code Requirements for Which Relief is Requested

10 CFR 50.55a(g)(6)(ii)(F) requires, in part, a volumetric inspection of RPV inlet cold leg nozzle to safe-end DM welds of pressurized water reactors in accordance with ASME Code Case N-770-1 "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities," subject to the conditions specified in paragraphs (2) through (10) of 10 CFR 50.55a(g)(6)(ii)(F). ASME Code Case N-770-1, Table 1, Inspection Item B requires volumetric examination of essentially 100% of each weld once every second inspection period not to exceed 7 years.

3.1.3 Licensee's Proposed Alternative

The licensee proposes a onetime extension to the Code Case N-770-1, Table 1, Inspection Item B, volumetric examinations from a period of "every 2nd inspection period not to exceed 7 years" to a period not to exceed one ASME Section XI 10-year ISI interval for MNS 1. The licensee states that the inspection will be performed no later than the fall 2020 refueling outage (RFO), 1EOC27, approximately 10.5 years from the previous inspection during RFO 1EOC20 in May 2010.

3.1.4 Duration of Relief Request

This request is applicable to MNS 1 ISI Program for the 4th 10-year ISI interval. The proposed alternative is applicable until the fall 2020 refueling outage, 1EOC27. This is a one-time extension inspection frequency request.

3.1.5 Licensee's Basis for Relief

The licensee's overall basis used to demonstrate the acceptability of extending the inspection interval for Code Case N-770-1, Inspection Item B components is contained in Electric Power Research Institute MRP-349, "PWR Reactor Coolant System Cold-Loop Dissimilar Metal Butt Weld Reexamination Interval Extension," and the site-specific flaw evaluation performed for MNS 1. In summary, the licensee's basis for extending the inspection is: (1) there has been no service experience with cracking found in any RPV inlet cold leg DM welds, (2) crack growth rates in RPV inlet cold leg DM welds are slow, (3) the likelihood of initial cracking, crack growth and a subsequent through-wall leak is very small in these welds, and (4) the MNS 1 specific axial and circumferential flaw evaluation showing that any indication detected during the previous inspection which was performed in the spring 2010 RFO, as well as any flaw size which could have been reasonably missed during the spring 2010 RFO RPV inlet cold leg DM weld examination, would not grow to the allowable size flaw specified by ASME Section XI rules over the timeframe of the requested inspection interval. The licensee provided this technical basis to demonstrate that the re-examination interval can be extended while maintaining an acceptable level of quality and safety.

The licensee also provided a summary of the service experience of these welds at MNS 1 to support their proposal. The welds were inspected in May of 2010 using the qualification requirements of ASME Section XI, Appendix VIII including an examination volume of essentially 100% of the required inspection volume. The volumetric inspection found no unacceptable flaws in the welds. One fabrication indication at the clad to base metal interface adjacent to the DM weld was found; however, it was found to be acceptable per the acceptance criteria of IWB-3514 of ASME Section XI using the 1998 Edition through the 2000 Addenda. Additional supplemental eddy current examinations of the inside surface of the RPV inlet cold leg DM welds revealed no recordable indications on the surface.

The licensee further included details on the flaw analysis performed for the RPV inlet cold leg DM welds to support their proposal. The analysis was developed and based on the most recent guidance of MRP-287, "Primary Water Stress Corrosion Cracking (PWSCC) Flaw Evaluation Guidance." In development of the weld residual stresses, the licensee included the effects of a hypothetical 50%, in depth, inside surface weld repair. The licensee submitted the flaw evaluation and results to the NRC as part of the submittal under Westinghouse Letter LTR-PAFM-14-114-P. A non-proprietary version of the Westinghouse calculation was also submitted and is available under ADAMS Accession No. ML15083A046. In summary, the licensee states that the calculations show that in order for a flaw to have grown to an unacceptable ASME Code depth of 75% through-weld by the next inspection in the fall of 2020, an axial flaw would have been required to be 47% through-weld thickness or a circumferential flaw would have had to of been 38% through-weld thickness, during the previous inspection in May of 2010. The licensee states that based on the current inspection capabilities, the flaw sizes above are significantly larger than the flaw sizes that were identified during the previous inspection and flaw sizes that could have been reasonably missed during the previous inspection of these welds.

Finally, the licensee included a discussion on the probability of cracking or leakage from these welds. The licensee stated that there were analyses and sensitivity studies which showed the likelihood of cracking or through-wall leaks was very small and that more frequent inspections had only a small benefit in terms of risk. Further, the licensee noted that there was no

operational experience of cracking in RPV inlet cold leg DM welds, and the number of indications in RPV hot leg DM welds, a leading indicator of cracking due to temperature, was small considered against the number of those welds in service.

Therefore, the licensee concluded that extending the required MNS 1 RPV inlet cold leg DM weld volumetric examination until the fall of 2020 is justified. As such, the licensee found that the use of the proposed alternative would provide an acceptable level of quality and safety and requested that the NRC authorize the proposed alternative in accordance with 10 CFR 50.55a(z)(1).

3.2 NRC Staff Evaluation

The NRC staff notes that the generic rules for the frequency of volumetric examination of DM butt welds were established to provide reasonable assurance of the leak tightness and structural integrity of the reactor coolant pressure boundary. As such, the NRC staff finds that a plant-specific analysis could be used to provide a basis for inspection relief if the inspection frequency can be shown to maintain reasonable assurance of the leak tightness and structural integrity of the weld. As such, the NRC staff reviewed the licensee's proposed alternative under the requirements of 10 CFR 50.55a(z)(1), such that:

> "The proposed alternatives would provide an acceptable level of quality and safety."

The NRC staff reviewed each of the licensee's four bases; (1) there has been no service experience with cracking found in any RPV inlet cold leg DM welds, (2) crack growth rates in RPV inlet cold leg DM welds are slow, (3) the likelihood of initial cracking, crack growth and a subsequent through-wall leak is very small in these welds, and (4) the MNS 1 specific axial and circumferential flaw evaluation. While the first three items do provide a measure of assurance that previous cracking has not occurred in cold leg temperature welds of the RPV inlet nozzle, the NRC staff notes that cracking has been identified in cold leg temperature DM welds of smaller pipe size than the RPV inlet nozzle. Further, cracking has been found in other locations at cold leg temperatures in the reactor coolant pressure boundary. Additionally, reviews of the inspection data from those flaws have shown faster than average growth rates for the cold leg temperatures. The NRC staff notes that the MNS 1 RPV inlet cold leg DM welds are made with weld materials susceptible to primary water stress corrosion cracking. This type of cracking can initiate and grow at cold leg temperatures. Therefore, the NRC staff finds that for the licensee's first three items, alone, do not provide a sufficient basis to assume that cracking could not occur in these welds over time, and those flaws could not grow to a size that could challenge leak tightness or structural integrity. However, the licensee's fourth basis, the plant-specific axial and circumferential flaw evaluation, does assume a hypothetical flaw that could exist and provides an assessment of the potential growth of the flaw over time. The NRC staff finds this analysis can provide a basis to demonstrate leak tightness and structural integrity. Therefore, the NRC staff focused its review on this aspect of the licensee's basis for the proposed alternative.

The licensee's flaw analysis is composed of a stress analysis and then a flaw growth calculation. The NRC staff reviewed the licensee's stress analysis and found it followed the recommendations of MRP-287 and numerous NRC public meeting discussions with industry since November 19, 2009, on effective weld residual stress calculations to address PWSCC flaw analysis. Of note, for significant conservatism, a 50% inside surface weld repair 360°

around the circumference was simulated in the weld residual stress analysis. The fabrication sequence was simulated based on information provided in the plant-specific drawings. The NRC staff also found that the use of two stress paths, calculated for both hoop and axial stresses, was effective and consistent with NRC staff expectations. The NRC staff reviewed the final proprietary stress analysis through the thickness of the weld and found both the hoop and axial residual stress curve contours were consistent with analyses using similar geometries and fabrication methods. As such, the NRC staff's review found the licensee's plant-specific stress analysis for the subject welds to have conservative inputs and assumptions and, therefore, was adequate to be used in the flaw evaluation.

The NRC staff found that the licensee's crack growth calculation used reasonable inputs and industry methodologies to determine maximum end-of-evaluation period flaw sizes for both axial and circumferential flaws. The NRC staff found the licensee's use of the maximum allowable flaw size of 75% of the wall thickness in accordance with the requirements of ASME Code, Section XI, Paragraph IWB-3640, is an adequate approach. The NRC staff found the licensee's use of the MRP-115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds," adequate for the crack growth analysis.

The last component of the NRC staff's review concerned the licensee's flaw analysis results and licensee's conclusions to provide a technical basis to support the relief request. Figures 7-1 and 7-2 of the licensee's calculation, Westinghouse LTR-PAFM-14-114-P, provide the PWSCC crack growth curves through the thickness of the welds for both an axial and circumferential flaw, respectively. The licensee's flaw analysis shows that a flaw would have to be approximately 1-inch in depth or greater to grow to the allowable ASME Code flaw size limit in 10.5 effective full power years. The licensee used the results of the flaw analysis to support the conclusion that since no surface connected flaw was identified in the spring 2010 inspection of each subject weld, the next inspection can be delayed 10.5 calendar years to the fall 2020 RFO, while maintaining reasonable assurance of the structural integrity and leak tightness of each weld.

The NRC staff performed a separate crack growth analysis including a set of sensitivity calculations to address potential variables, such as loading and weld residual stress profiles. The NRC staff found that given the licensee's inspections in 2010, there is reasonable assurance that a surface connected flaw of 10% depth or greater should have been identified. Therefore, the NRC staff used an initial flaw size of approximately 10% depth in its crack growth analysis. The NRC staff's calculations showed less margin for a hypothetical axial or circumferential flaw to grow to an ASME Code allowable size of 75% through-wall depth than the licensee's calculations. However, the NRC staff's analysis, using conservative assumptions, still found that at least 11.7 effective full power years would be required to grow an initial flaw of 10% in depth to the ASME Code allowable depth of 75% through-wall. Given that a flaw at the ASME Code allowable depth would still insure leak tightness and structural integrity, the NRC staff's analysis showed sufficient margin to allow an extension of the inspection frequency for these welds for 10.5 calendar years.

Both the licensee's and NRC staff's flaw calculations provide results assuming that the plant is at operating temperature and pressure. Therefore, the associated times are in effective full power years. The licensee's proposed alternative is for approximately 10.5 calendar years. During this 10.5 calendar year period, the plant will have been shutdown for various time periods, most often for scheduled refueling outages. Therefore, depending on the amount of

time the plant is shutdown, 10.5 calendar years is always less than the same number of effective full power years.

In summary, the NRC staff reviewed the licensee's flaw evaluations and found both the inputs and analysis methodologies to be acceptable and within ASME Code requirements. The NRC staff's independent analysis has verified the licensee's conclusion that the largest hypothetical flaw that could have been missed during the 2010 inspections at MNS 1 would not grow to an ASME Code unacceptable size within the period of time requested in the licensee's proposed alternative. Therefore, the NRC staff finds that the licensee has provided an adequate technical basis to provide reasonable assurance of leak tightness and structural integrity for the extended inspection frequency, which increases the maximum inspection frequency for these welds from seven to 10.5 calendar years. Hence, the NRC staff finds that the licensee's proposed alternative would provide an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that the licensee provided sufficient technical basis to demonstrate that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1), the proposed alternative provides reasonable assurance of structural integrity and leak tightness, and is in compliance with the *Code of Federal Regulation's* requirements. Therefore, in accordance with 10 CFR 50.55a(z)(1) the NRC staff authorizes the licensee's proposed alternative, Relief Request 15-MN-001, at McGuire Nuclear Station, Unit 1, for the 4th 10-year inservice inspection interval, to extend inspections of the stated vessel cold leg dissimilar metal welds to no later than the 1EOC27 RFO scheduled for fall 2020.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributors: J. Collins, NRR

Date: August 27, 2015

If you have any questions, please contact the Project Manager, G. Edward Miller at 301-415-2481 or via e-mail at ed.miller@nrc.gov.

Sincerely,

/RA/

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-369

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Safety Evaluation

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