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Serial No: MNS-15-014

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10 CFR 50.55a

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Duke Energy Carolinas, LLC (Duke Energy)
McGuire Nuclear Station, Unit 1
Docket No. 50-369
Relief Request 15-MN-001
Reactor Vessel Cold Leg Nozzle Dissimilar Metal (DM) Weld Inspection

Pursuant to 10 CFR 50.55a(z), Duke Energy hereby requests U.S. Nuclear Regulatory Commission (NRC) approval of relief from the requirements of 10 CFR 50.55a(g)(6)(ii)(F). 10 CFR 50.55a(g)(6)(ii)(F) requires that licensees of existing, operating pressurized water reactors implement the requirements of American Society of Mechanical Engineers (ASME) Code Case N-770-1. Inspection Item B of this code case requires unmitigated butt welds at cold leg operating temperature to be volumetrically examined every second inspection period not to exceed 7 years. The next required examination for the reactor vessel cold leg nozzle DM welds will be during the Spring 2016 Refueling Outage (RFO). Duke Energy proposes a one-time extension for the next examination of these welds to be performed no later than the Fall 2020 RFO.

The proposed alternative would provide an acceptable level of quality and safety. The requested relief is to allow this examination to coincide with ASME Code Section XI Inservice Inspection of the reactor vessel and the Materials Reliability Program (MRP)-227-A reactor internal inspection associated with license renewal commitments. Duke Energy requests NRC approval of this relief request by September 30, 2015, to facilitate scheduling of this examination. This submittal contains no regulatory commitments.

Attachment 1 contains the relief request.

Attachment 2 contains one copy of LTR-PAFM-14-114-P, Revision 0, "Technical Justification to Support Extended Volumetric Examination Interval for McGuire Unit 1 Reactor Vessel Inlet Nozzle to Safe End Dissimilar Metal Welds." (Proprietary)

Attachment 3 contains one copy of LTR-PAFM-14-114-NP, Revision 0, "Technical Justification to Support Extended Volumetric Examination Interval for McGuire Unit 1 Reactor Vessel Inlet Nozzle to Safe End Dissimilar Metal Welds." (Non-Proprietary)

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NRR

Attachment 2 to this letter contains proprietary information.
Withhold from public disclosure under 10 CFR 2.390.
Upon removal of Attachment 2, this letter is uncontrolled.

Attachment 4 contains the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-15-4089, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Attachment 2 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-15-4089 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

If you have any questions or require additional information, please contact P.T. Vu of Regulatory Affairs at (980) 875-4302.

Sincerely,

Steven D. Capps

Attachments

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Attachment 1

Duke Energy Carolinas, LLC

McGuire Nuclear Station, Unit 1

Relief Request Serial # 15-MN-001

Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1) for Reactor Vessel Cold Leg Dissimilar Metal Weld Inspections

1.0 ASME Code Component(s) Affected

The affected components are McGuire Nuclear Station (MNS), Unit 1 Reactor Vessel Cold Leg nozzle to safe-end welds. These welds are Alloy 82/182 type welds covered by Code Case N-770-1, Inspection Item B.

N-770-1 INSPECTION ITEM	COMPONENT ID	NOZZLE SIZE	DESCRIPTION
B	1RPV3-445A-SE	27.5" ID (Nom.)	Primary Inlet Nozzle Weld – A-Loop
B	1RPV3-445B-SE	27.5" ID (Nom.)	Primary Inlet Nozzle Weld – B-Loop
B	1RPV3-445C-SE	27.5" ID (Nom.)	Primary Inlet Nozzle Weld – C-Loop
B	1RPV3-445D-SE	27.5" ID (Nom.)	Primary Inlet Nozzle Weld – D-Loop

Component materials and nozzle weld configuration is shown in Figure 1 below. The inlet nozzle operating temperature is 557.5°F with a thickness of 2.526 inches.

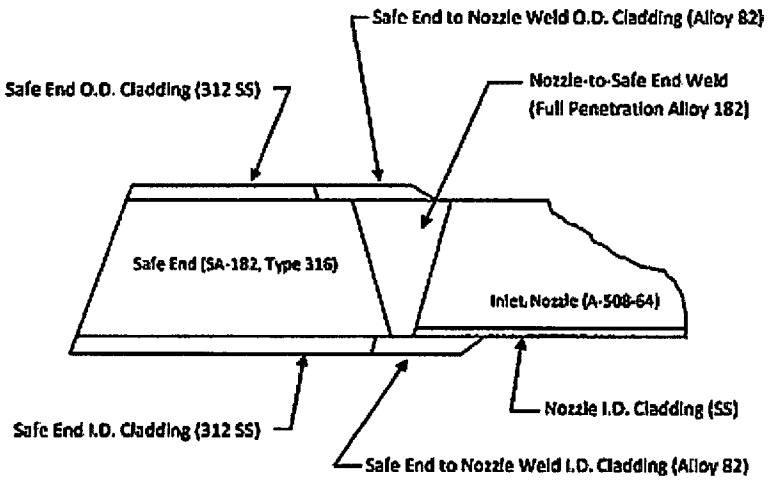

Figure 1. Inlet Nozzle to Safe-End Welds (Not to Scale)

2.0 Applicable Code Edition and Addenda

The applicable Code edition in the 4th ISI interval for MNS, Unit 1 is the 2007 Edition of ASME Section XI with 2008 Addenda.

3.0 Applicable Code Requirement

10 CFR 50.55a(g)(6)(ii)(F) requires that licensees of existing, operating pressurized water reactors implement the requirements of ASME Code Case N-770-1. Inspection Item B of Code Case N-770-1 requires unmitigated butt welds at Cold Leg operating temperatures

(≥ 525°F and < 580°F) to be volumetrically examined every second inspection period not to exceed 7 years.

4.0 Reason for Request

The next required examination for the Unit 1 Reactor Vessel Cold Leg nozzle to safe-end welds will be during the end of cycle 24 refueling outage (1EOC24 RFO) in Spring 2016. Relief is being requested at this time to extend the Cold Leg inspection up to 1EOC27 RFO scheduled for Fall 2020 for MNS, Unit 1. This extension request is to allow this inspection to coincide with ASME Code Section XI Inservice Inspection (ISI) of the reactor vessel and the Materials Reliability Program (MRP)-227-A reactor internals inspection associated with license renewal commitments.

The Reactor Vessel Cold Leg welds (N-770-1, Inspection Item B) for MNS, Unit 1 are inspected from the Inner Diameter (ID) of the pipe. Therefore, to access the Cold Leg welds for examination, a critical core-barrel lift must be performed. By receiving relief to move the Cold Leg inspection, the site can leverage performing the inspection with the aforementioned required ASME Section XI ISI of the reactor vessel and MRP-227-A inspection which also require this critical core barrel lift. Additionally, personnel dose savings for the site can be realized with this extension, promoting As Low As Reasonably Achievable (ALARA) practices associated with this significant evolution.

EPRI document MRP-349 (Reference 2) and a site-specific crack growth evaluation for MNS, Unit 1 (Reference 3, Attachment 2) provides the overall basis for extension of the current volumetric inspection interval for the Reactor Vessel Cold Leg Dissimilar Metal (DM) welds. This technical basis demonstrates that the re-examination interval can be extended to the requested interval length while maintaining an acceptable level of quality and safety.

5.0 Proposed Alternative and Basis for Use

10 CFR 50.55a(z) states, in part:

> "Alternatives to the requirements of paragraphs (b) through (h) of this section or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation, or Director, Office of New Reactors, as appropriate. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that:
> (1) Acceptable level of quality and safety. The proposed alternative would provide an acceptable level of quality and safety;"

Duke Energy believes that the proposed alternative of this request provides an acceptable level of quality and safety.

MNS, Unit 1 proposes a one-time extension to the Code Case N-770-1, Table 1, Inspection Item B, volumetric examination from every 2nd inspection period not to exceed 7 years. The extension requested is for a period not to exceed one ASME XI ISI interval. The inspection will be performed no later than the Fall 2020 refueling outage, approximately 10.5 years from the previous inspection (1EOC20 in May 2010).

Technical Basis

The 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 MRP-349 and the site-specific flaw evaluation performed for MNS Unit 1. In summary, the basis for extending the inspection is: (1) there has been no service experience with cracking found in any Reactor Vessel Cold Leg DM welds, (2) crack growth rates in Reactor Vessel Cold Leg DM welds are slow, (3) the likelihood of initial cracking, crack growth and a subsequent through-wall leak is very small in Reactor Vessel Cold Leg DM welds, and (4) the MNS specific axial and circumferential flaw evaluation showing any indication detected during the Spring 2010 RFO inspection, as well as any flaw size which could have been reasonably missed during the Spring 2010 RFO inlet nozzle weld examination would not grow to the allowable size flaw specified by ASME Section XI rules over the timeframe of the requested inspection interval. This technical basis demonstrates that the re-examination interval can be extended while maintaining an acceptable level of quality and safety.

Service Experience

Inspection of the MNS Unit 1 reactor vessel Cold Leg nozzle welds was performed in 1EOC20 (May, 2010) in accordance with ASME Section XI and MRP-139 guidance. Cold Leg welds 1RPV3-445A-SE, 1RPV3-445B-SE, 1RPV3-445C-SE, and 1RPV3-445D-SE were examined and met the ASME Section XI, Appendix VIII requirements including examination volume of essentially 100%.

Automated ultrasonic examination of the Cold Leg nozzle welds revealed one (1) recordable indication. No recordable indications were identified in DM welds 1RPV3-445A-SE, 1RPV3-445B-SE and 1RPV3-445C-SE. Examination of weld 1RPV3-445D-SE revealed 1 fabrication indication at the clad to base metal interface adjacent to the DM weld. This recordable subsurface indication was acceptable per IWB-3514 of ASME Section XI 1998 edition through the 2000 addenda and was characterized to be 0.16 inch deep and 0.48 in long.

Supplemental eddy current examinations of the Cold Leg DM welds revealed no recordable indications.

Crack Growth Rates (Flaw Tolerance)

Site-specific analysis has been performed for the reactor vessel nozzles using through wall residual stress distributions that were developed and based on the most recent guidance of MRP-287, entitled "Primary Water Stress Corrosion Cracking (PWSCC) Flaw Evaluation Guidance". For development of the residual stress profile, the sequence of 1) initial welding of the nozzle to safe-end joint, 2) an inside surface weld repair with a repair depth of 50% through the dissimilar metal weld thickness and 3) the adjacent stainless steel weld between the safe-end and piping was included. The subsequent crack growth analyses have shown that the flaw tolerance of the location is high and a postulated axial or circumferential flaw will not reach the maximum ASME allowable depth within the requested extension time period.

The crack growth curves and maximum allowable flaw sizes developed for MNS, Unit 1 are shown in Figures 7-1 and 7-2 of Reference 3. The maximum allowable end-of-evaluation

period flaw sizes are also shown on these figures for the axial and circumferential flaw configurations analyzed. Based on these crack growth results, the maximum allowable initial flaw sizes for the axial and circumferential flaws are tabulated in the table below.

Maximum Allowable Initial Flaw Sizes

	Axial Flaw (Aspect Ratio = 2)	Circumferential Flaw (Aspect Ratio = 10)
Maximum Allowable Initial Flaw Size (a/t)	0.47	0.38
Flaw Depth (inches)	1.19	0.96
Flaw Length (inches)	2.38	9.60

The maximum allowable flaw sizes shown in the table above are the largest axial and circumferential flaw sizes that could be left behind in service and remain acceptable for a service life of 10.5 years from 1EOC20 to 1EOC27. Based on current inspection capabilities, the maximum flaw sizes listed above are significantly larger than the flaw size that was identified during 1EOC20 and flaw sizes that could have been reasonably missed during the 1EOC20 RFO inspection of the Reactor Vessel nozzle DM welds.

During 1EOC20 inspection, D-Loop weld 1RPV3-445D-SE revealed 1 fabrication indication at the clad to base metal interface adjacent to the DM weld. This indication was characterized to be 0.16 inch deep and 0.48 in long. The subsurface indication was acceptable per the IWB-3514-1 Table of ASME Section XI 1998 edition through the 2000 addenda and the flaw remains acceptable by the ASME Section XI 2007 edition with the 2008 addenda. The initial flaw sizes determined in the site-specific analysis are far larger than the IWB-3500 allowable flaw sizes. Therefore, the fabrication defect of weld 1RPV3-445D-SE is bounded by the allowable initial flaw sizes shown in the table above and would remain acceptable through the 1EOC27 RFO.

The maximum allowable initial flaw sizes that have been analyzed and summarized in the table above are larger than the detected indication size and the flaw sizes that might have been reasonably missed during the 1EOC20 inspections. Thus deferral of volumetric examination for the Reactor Vessel Cold Leg nozzle DM welds from the Code Case N-770-1 required 7 years to the requested 10.5 years is technically justified.

Probability of Cracking or Through Wall Leaks

Analyses have been performed to calculate the probability of failure for Alloy 82/182 welds using both probabilistic fracture mechanics and statistical methods. Both approaches have shown that the likelihood of cracking or through-wall leaks, in large-diameter cold leg welds, is very small. Furthermore, sensitivity studies performed using probabilistic fracture mechanics have shown that even for the more limiting high temperature locations, more frequent inspections than required by Section XI, such as that in MRP-139 or Code Case N-770, have only a small benefit in terms of risk.

Though past service experience may not be an absolute indicator of the likelihood of future cracking, the experience does give an indication of the relative likelihood of cracking in cold leg temperature locations versus hot leg temperature locations. While there is some amount of Primary Water Stress Corrosion Cracking (PWSCC) service experience in hot leg locations, the number of indications in large-bore butt welds is still small relative to the

number of potential locations. Therefore, if hot leg PWSCC is a leading indicator for cold leg PWSCC, and the higher frequency of inspections will be maintained for the hot leg locations, it is reasonable to conclude that a moderately less rigorous inspection schedule would be capable of detecting any cold leg indications before they became large enough to be a concern.

Conclusions

Extending the required MNS, Unit 1 Cold Leg DM weld volumetric examination from 1EOC24 to 1EOC27 is justified given (1) there has been no service experience with cracking found in any Reactor Vessel Cold Leg DM welds, (2) crack growth rates in Reactor Vessel Cold Leg DM welds are slow, (3) the likelihood of initial cracking, crack growth and a subsequent through-wall leak is very small in Reactor Vessel Cold Leg DM welds, and (4) the MNS specific axial and circumferential flaw evaluation showing a detected fabrication indication at the cladding to nozzle base metal interface of the D-Loop Reactor Vessel Cold Leg nozzle and any undetected flaw size that could have been reasonably missed during the 1EOC20 RFO examination would not grow to the allowable flaw size specified by ASME XI rules over the timeframe of the requested inspection interval. The use of this proposed alternative will provide an acceptable level of quality and safety. For these reasons, it is requested that the NRC authorize this proposed alternative in accordance with 10 CFR 50.55a(z)(1).

6.0 Duration of Proposed Alternative

This request is applicable to MNS, Unit 1 Inservice Inspection Program for the 4th interval. The proposed alternative is applicable until the Fall 2020 RFO.

7.0 Related Industry Relief Requests

Similar proposed alternatives were previously approved by the NRC for the following licensees:

1. Joseph M. Farley Nuclear Plant, Units 1 and 2 – (FNP-ISI-ALT-15, Version 1) Alternative to Inservice Inspection Regarding Reactor Pressure Vessel Cold-Leg Nozzle Dissimilar Metal Welds (TAC Nos. MF3687 and MF3688), dated December 5, 2014 (ML14262A317).

2. Arkansas Nuclear One, Unit No. 1- Request for Alternative ANO1-ISI-023 to ASME Code Case N-770-1 Volumetric Examination Frequency Requirements for the Fourth 10-Year Inservice Inspection Interval (TAC No. MF3176), dated October 29, 2014 (ML14282A479).

3. Indian Point Nuclear Generating Unit No. 3 – Relief Request IP3-ISI-RR-07 for Reactor Vessel Cold Leg Nozzle to Safe-End Weld Examination (TAC No. MF3346), dated August 4, 2014 (ML14199A444).

4. Indian Point Nuclear Generating Unit No. 2 – Request for Relief Request No. IP2-ISI-RR-14, Code Case N-770-1, Reactor Coolant System Cold Leg Nozzle Weld Inspection Frequency Extension (TAC No. ME6801), dated February 2, 2012 (ML120260090).

8.0 References

1. 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 Section XI, Division 1.

2. Materials Reliability Program (MRP): PWR Reactor Coolant System Coolant System Cold-Loop Dissimilar Metal Butt Weld Reexamination Interval Extension (MRP-349), August 2012.

3. Westinghouse LTR-PAFM-14-114-P, "Technical Justification to Support Extended Volumetric Examination Interval for McGuire Unit 1 Reactor Vessel Inlet Nozzle to Safe End Dissimilar Metal Welds." (Proprietary)

4. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), December 2011.

5. Materials Reliability Program: Primary Water Stress Corrosion Cracking (PWSCC) Flaw Evaluation Guidance (MRP-287), December 2010.