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CNS-15-062

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10 CFR 50.55a

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

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station, Unit 1
Docket Number 50-413
Relief Request Serial Number 15-CN-001, Proposed Alternative Repair for Main Steam System Braided Flex-Hose - Submitted Pursuant to 10 CFR 50.55a(z)(2)

References:
1. Letters from Duke Energy to NRC, same subject, dated March 19, 2015 (ADAMS Accession Number ML15082A074) and May 4, 2015 (ADAMS Accession Number ML15127A170).
2. Electronic mail from NRC to Duke Energy, Acceptance Review for 15-CN-001, dated April 20, 2015.
3. Conference call between Duke Energy and NRC, dated June 8, 2015.

The March 19, 2015 Reference 1 letter submitted Relief Request 15-CN-001 requesting approval to use an alternative repair for a Main Steam System leaking braided flex hose. The Reference 2 electronic mail requested additional information to enable the NRC to complete its acceptance review of the March 19, 2015 Reference 1 letter. The May 4, 2015 Reference 1 letter responded to the Reference 2 request. During the Reference 3 conference call, the NRC requested Duke Energy to clarify its plans for performing the subject alternative repair. The purpose of this letter is to provide the requested clarification.

This clarification is provided as enclosed revised Relief Request 15-CN-001. The body of the relief request is being resubmitted in its entirety. The relief request attachments have not changed from the original March 19, 2015 Reference 1 submittal; therefore, they are not being resubmitted with this relief request revision.

There are no regulatory commitments contained in this letter or its enclosure.

If you have any questions or require additional information, please contact L.J. Rudy at (803) 701-3084.

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NRR

Very truly yours,

R. T. Simril for

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LJR/s

Enclosure

xc (with enclosure):

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Enclosure

Duke Energy Carolinas, LLC

Catawba Nuclear Station, Unit 1

Relief Request Serial No. 15-CN-001

Proposed Alternative Repair for Main Steam System Braided Flex-Hose, Submitted Pursuant to 10 CFR 50.55a(z)(2)

1.0 ASME Code Component Affected

Catawba Nuclear Station, Unit 1 ASME Class 2 Level Instrument Flex-Hose.

The following information is applicable to this component:

System:	Main Steam (SM)
Design Pressure:	1200 psia
Design Temperature:	600°F
Pipe Size and Material:	½"-Sch. 80 / SA376 TP304
Flex-Hose Size and Material:	½"-2500# / SA213 TP304
Root Valves:	½"-1500# / SA182, Gr. F316 / globe valve (Model 09J-574)

The flex-hose is part of level instrument 1SMLS-5710. This degraded flex-hose is located downstream of the two stainless steel root valves on each side of the flex-hose as detailed in Catawba Unit 1 Instrument Weld Isometric drawing CNI-SM-1571 (Attachment 2) and Weld Isometric drawing CN-1SM-0082 (Attachment 3).

Instrument 1SMLS-5710 is located within the containment isolation boundary of the Main Steam 1A Containment Penetration (M113) identified as Item No. 91 within UFSAR Table 6-77, Unit 1 Containment Isolation Valves Data.

2.0 Applicable Code Edition and Addenda

ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition with the 2000 Addenda

3.0 Applicable Code/Regulatory Requirements

3.1 IWC-3516 specifies that acceptance standards for Examination Category C-H, All Pressure Retaining Components are in the course of preparation, and that the standards of IWB-3522 may be applied. Duke Energy has chosen to apply the acceptance standards of IWB-3522 to address leakage from this Class 2 component.

3.2 IWA-4400 specifies requirements for welding, brazing, defect removal, and installation of pressure retaining items.

3.3 IWA-4133 provides alternative requirements for repairs using mechanical clamping devices using Mandatory Appendix IX.

3.4 The ASME Boiler and Pressure Vessel Code, Section XI, Appendix IX, Mechanical Clamping Devices for Class 2 and 3 Piping Pressure Boundary provides the requirements for using and designing mechanical clamping devices for piping pressure boundary.

4.0 Reason for Request

4.1 On August 12, 2014, a steam leak was discovered from a braided flex-hose part of level instrument 1SMLS-5710. This level switch is located in the Unit 1 Exterior Doghouse and controls valve 1SM-89 which dumps accumulated condensate to the Unit 1 main condenser as required. The flow diagram and weld isometric drawings showing this configuration are shown in Attachments 1, 2, and 3. The leaking ASME Class 2 flex-hose was isolated to comply with Catawba Selected Licensee Commitment (SLC) 16.5-5, "Structural Integrity – The structural integrity of the ASME Code Class 1, 2, and 3 components shall be maintained." Once isolated it was discovered that minor leakage still

occurred from the flex-hose due to valve seat leakage past the upstream root valves. To enable isolation of the leak to facilitate a code repair/replacement of the leaking flex-hose in accordance with IWA-4000, the alternative documented in this request is proposed. The proposed alternative consists of installing a mechanical clamping device using ASME Section XI, Appendix IX and injecting sealant into the pipe between the two root valves to fully isolate the leaking component. Following sealant injection, the leaking flex-hose shall be removed and replacement pressure boundary material (pipe caps or plugs) shall be attached to the downstream side of the root valves to which the flex-hose had been connected. Because Appendix IX prohibits the use of mechanical clamping devices on portions of piping systems that form the containment boundary, relief is required to permit use of this mechanical clamping device to facilitate the isolation of these valves to replace the leaking flex-hose.

4.2 As defined in UFSAR Table 6-77, "Unit 1 Containment Isolation Valve Data", the Main Steam containment isolation valves are not required to be leak rate tested. This is due to the Main Steam line being connected to the secondary side of the steam generator which is kept at a higher pressure than the primary side immediately after a LOCA occurs. Any leakage between the primary and secondary sides of the steam generator is directed inward to containment (i.e., main steam header pressure is maintained higher than peak containment pressure). This penetration is effectively sealed by steam header pressure against leakage from containment after a LOCA.

This Main Steam 1A Header leak has been evaluated against consequence during the postulated Steam Generator Tube Rupture (SGTR) Design Basis Accident. It was concluded that post SGTR radiation doses for this scenario would be bounded by those for the SGTR with a failed open Power Operated Relief Valve (PORV) on the ruptured Steam Generator. The presence of this steam leak does not invalidate any Safety Analysis calculations with regards to SGTR (i.e., an unanalyzed condition does not exist).

4.3 NRC Inspection Manual, Part 9900 Technical Guidance, Appendix C.12 Operational Leakage From ASME Code Class 1, 2, and 3 Components, states "The NRC staff does not consider through-wall conditions in components, unless intentionally designed to be there such as sparger flow holes, to be in accordance with the intent of the ASME Code or construction code and, therefore, would not meet code requirements, even though the system or component may demonstrate adequate structural integrity." The guidance provided in Part 9900 implies that the NRC does not accept that IWC-3000 of the ASME Code, Section XI allows through wall leakage in Class 2 components. Since a through-wall flaw in a flex-hose cannot be evaluated using an applicable and NRC endorsed code case, relief is required to comply with this guidance.

5.0 <u>Proposed Alternative and Basis for Use</u>

5.1 Proposed Alternative

In lieu of the requirement of Appendix IX, Article IX-1000, General,

(c) Clamping devices shall not be used on the following:

(2) portions of a piping system that forms the containment boundary

The following alternative requirements are proposed:

1. A non-code repair shall be performed to stop the leakage using a mechanical clamp, 1/8" NPT injection valve, and injection sealant. A mechanical clamp designed to the requirements of Appendix IX, Article XI-3000 shall be installed followed by installation of an ASME Class 1 injection valve. After installing the injection valve, a 3/16" diameter hole shall be drilled in the pipe to facilitate sealant injection. After sealant injection is completed, the mechanical clamp and closed injection valve shall serve as part of the Class 2 pressure boundary until a code repair/replacement activity complying with IWA-4000 can be performed. A drawing of the mechanical clamp is provided in Attachment 4. Additionally, after verification that the leak has been fully isolated, the damaged braided flex-hose shall be removed and code compliant caps (or plugs) shall be installed on the end of the outboard root valves. These caps (or plugs) will also serve as part of the Class 2 pressure boundary until the flex-hose, affected root valves, and piping can be replaced.

2. The mechanical clamp, injection valve, and sealant injection may be used between one or both sets of root valves located upstream and downstream of the degraded flex-hose to fully isolate the leakage.

5.2 Basis for Proposed Alternative

Hardship: A code-compliant repair cannot be performed without fully isolating and depressurizing the affected component. The root valves that are available to isolate this component are not leak-tight and are located upstream of the main steam isolation valves (MSIVs). Therefore, in order to isolate the affected component, a unit shutdown would be required to facilitate the repair. Therefore, the only way to perform a code-compliant repair in accordance with IWA-4000 would be to shut down the unit in order to depressurize the line and replace the affected components. Compliance with the specified requirement would result in hardship without a compensating increase in the level of quality and safety.

The proposed alternative to install an engineered clamp with an injection valve and inject sealant between the leaking root valves will enable the leaking component to be fully isolated and depressurized to permit removal of the leaking flex-hose and installation of pipe caps (or plugs) to restore the leak-tight integrity of the system. The piping between the leaking root valves has been evaluated and the structural and leak-tight integrity of this piping shall be maintained during the installation of the clamp, during leak injection, and during subsequent operation until a permanent repair/replacement activity can be performed.

6.0 Duration of Proposed Alternative

The proposed alternatives to the ASME Code are applicable for the third 10-year Inservice Inspection (ISI) Interval at Catawba Nuclear Station, Unit 1.

The Catawba Unit 1 third Inservice Inspection Interval began on June 29, 2005 and is currently scheduled to end on June 29, 2016.

Use of the proposed alternative is requested until Code repair/replacement activities can be performed on the level instrument piping and flex-hose during refueling

outage 1EOC22 (Fall 2015) or during a forced outage of sufficient duration before refueling outage 1EOC22.

7.0 References

7.1 1998 Edition through 2000 Addenda, ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

7.2 US NRC Regulatory Issue Summary 2005-20, Rev. 1, Revision to NRC Inspection Manual Part 9900 Technical Guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety".