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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, Units 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Alternative Request Number 11-CN-002 for the Third Ten-Year Inservice Inspection Interval
Response to NRC Request for Additional Information
(TAC Nos. ME7182 through ME7187)

Reference: Letter from Duke Energy to NRC dated September 13, 2011

The reference letter requested NRC approval of proposed alternative testing for the remainder of the third ten-year inservice inspection interval at the Catawba Nuclear Station. On March 1, 2012, Requests for Additional Information (RAIs) were electronically received from the NRC. The purpose of this letter is to formally respond to these RAIs. The attachment to this letter contains Duke Energy's response. The format of the response is to restate each RAI question, followed by the response.

This submittal document contains no regulatory commitments.

If there are any questions or if additional information is needed, please contact L.J. Rudy at (803) 701-3084.

Very truly yours,

James R. Morris

James R. Morris

Attachment

A047
NRR

xc (with attachment):

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ATTACHMENT

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

REQUEST FOR RELIEF 11-CN-002

OFFICE OF NUCLEAR REACTOR REGULATION
REQUEST FOR ADDITIONAL INFORMATION
RELIEF REQUEST 11-CN-002
PROPOSED ALTERNATIVE REQUEST NUMBER 11-CN-002 FOR THE
THIRD TEN-YEAR INSERVICE INSPECTION INTERVAL
DUKE ENERGY CAROLINAS, LLC
CATAWBA NUCLEAR STATION, UNITS 1 AND 2
DOCKET NOS. 50-413 AND 50-414

By letter dated September 13, 2011, Duke Energy Carolina, LLC (the licensee) submitted Relief Request (RR) 11-CN-002, "Proposed Alternative Request Number 11-CN-002 for the Third Ten-Year Inservice Inspection Interval" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11264A028) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. In the subject RR, the licensee proposed alternative pressure testing for the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) Class 1 piping and component segments connected to (or part of) the reactor coolant system (RCS) in lieu of requirements of the ASME Code, Section XI, pressure testing. The proposed alternative is requested for the remainder of the third 10-year in service inspection (ISI) interval of Catawba 1 (which commenced on June 29, 2005, and will end on July 14, 2014) and Catawba 2 (which commenced on October 15, 2005, and will end on August 19, 2016).

The NRC staff has reviewed the information provided by the licensee in RR 11-CN-002 and finds the following additional information is needed to complete its review.

1. RR 11-CN-002 documented that the design pressure for piping and components in Segments 1, 2, 3, 4, and 5 is 2500 psig, while the section titled "Proposed Alternative" states that a system leakage test will be performed at a pressure not less than 300 psig for Segments 1, 2, and 5 and not less than 42 psig for Segment 3. The NRC staff notes that the section titled "Bases for the Proposed Alternative" states, "The proposed system leakage test conducted at a pressure of at least 300 psig (Segments 1, 2 and 5) and at least 42 psig (Segment 3) is acceptable because leakage (if it were to occur) would still be detectable at this reduced pressure, although at a reduced rate."

   a. Provide the maximum pressure that the subject piping and piping components for Segments 1, 2, 3, 4, and 5 would experience during normal operating, stagnant, accident, and fault conditions.

Duke Energy Response:

Area	**Normal Operating(1) (psig)**	**Stagnant(1) (psig)**	**Accident(2) (psig)**	**Fault Conditions(2) (psig)**
Segment 1	**2235**	**2235**	**2235**	**2235**
Segment 2	**2235**	**2235**	**385**	**385**
Segment 3	**2235**	**2235**	**1845**	**1845**
Segment 4	**2235**	**2235**	**(see footnote 3)**	**(see footnote 3)**
Segment 5	**2235**	**2235**	**2485**	**2485**

(1) Maximum pressure assuming leakage from first isolation valve off the RCS.

(2) Maximum pressure assuming segment is placed in service under accident or faulted condition.

(3) **The segment 4 valves remain closed during accident operation; therefore, accident and fault pressure are not applicable to these segments.**

b. In light of the documented design pressure of 2500 psig for Segments 1, 2, 3, 4, and 5, and maximum pressures (identified in the response to RAI question 1a), provide justification for performing a system leakage test at such a reduced pressure to ensure the structural integrity of the system.

Duke Energy Response:

Nondestructive Examination (NDE) has been performed on selected welds in piping segments 1, 2, 3, 4, and 5 as required by the ASME Code.

In addition, the Boric Acid Corrosion Control program would detect evidence of any leakage during the previous operating cycle by identifying boron deposits on the outside surface of the components within these piping segments.

Performing the system leakage tests at the proposed reduced test pressures is acceptable because leakage, if it were to occur, would be detected at the reduced pressures, although at a reduced leakage rate. This position is consistent with the basis documented in the Safety Evaluation Report for Relief Request 04-CN-004 (ADAMS Accession No. ML051780164).

For the above reasons, Duke Energy maintains that the proposed alternative to conduct the system leakage testing at the reduced pressures provides an acceptable level of assurance of the leak-tight and structural integrity of piping segments 1, 2, 3, 4, and 5.

2. On pages 8 and 9 of the subject RR, several related industry RRs are cited. Discuss whether during the second (previous) 10-year ISI interval of Catawba 1 and 2, a RR for pressure testing requirements was submitted to the NRC staff for the same piping and piping components of Segments 1, 2, 3, 4, and 5.

Duke Energy Response:

Duke Energy submitted and received approval to use Relief Request 04-CN-004 for seven piping segments during the 2nd 10-year interval as documented in a Safety Evaluation dated June 23, 2005 (ADAMS Accession No. ML051780164). Five of the seven piping segments listed in Relief Request 04-CN-004 are similar to those listed in Relief Request 11-CN-002. Duke Energy has determined that relief is not needed during the 3rd 10-year interval for the other two segments identified in Relief Request 04-CN-004.

3. Are there any welded connections in piping and components for Segments 1, 2, 3, 4, and 5? If the answer is yes, provide number and type (e.g., full penetration butt weld and fillet weld) of welds. Discuss any nondestructive examinations (NDEs) that were performed on the welded connections. Discuss any industry or plant-specific operating experience regarding potential degradation (e.g, fatigue, stress corrosion cracking, overloading, and corrosion) of welds in piping and components for the subject segments.

Duke Energy Response:

The answer to question 3 is "Yes". The table below provides the number of welds, type

of welds, and type of NDE listed in the ISI Plan for the welds.

Area	Dwg	Unit	No. of Welds in Segment	No. of Welds Examined	Type of Weld	NDE Performed
Seg# 1	CN-1554-1.0	1	22	7	Socket	Surface
	CN-2554-1.0	2	24	7	Socket	Surface
Seg# 2	CN-1561-1.0	1	12	3	Butt	Surface & Volumetric
			3	0	Socket	Exempt
	CN-2561-1.0	2	16	3	Butt	Surface & Volumetric
	CN-1561-1.1	1	15	3	Butt	Surface & Volumetric
	CN-2561-1.1	2	15	4	Butt	Surface & Volumetric
Seg# 3	CN-1562-1.0	1	16	4	Socket	Surface
	CN-2562-1.0	2	18	5	Socket	Surface
Seg# 4	CN-1553-1.0	1	36	7	Socket	Surface
			2	0	Butt	None Selected
	CN-2553-1.0	2	28	8	Socket	Surface
			2	2	Butt	Surface
Seg# 5	CN-1553-1.1	1	10	0	Socket	Exempt
	CN-2553-1.1	2	10	0	Socket	Exempt

The recordable indication identified in one weld of a stagnant portion of the safety injection system (see OE on corrosion) was not in any of the piping segments listed in Relief Request 11-CN-002.

Duke Energy Response Regarding OE on Corrosion:

Duke Energy recognizes there is potential for stress corrosion cracking to occur given the operating conditions of systems containing borated water and stainless steel materials. Stress corrosion cracking was identified during inservice inspection during the Catawba 1EOC18 refueling outage in 2009. A recordable indication was identified in one weld of a stagnant portion of the safety injection system by ultrasonic examination. The flaw was ID connected and located in the heat affected zone of the butt weld that had been repaired during construction. The flaw was determined to be the result of intergranular stress corrosion cracking of the stainless steel material.

Ultrasonic examinations were performed on 36 additional welds located in stagnant portions of the safety injection system during the 1EOC18 outage. No additional recordable indications were identified. Additionally, a review of the construction history for welds on Catawba Units 1 and 2 located in the stagnant portions of safety injection piping located inside containment was performed to determine which welds had been repaired. Ultrasonic examinations were performed during the next refueling outage on 37 butt welds for Unit 1 and 44 butt welds for Unit 2 which had a history of repair. No recordable indications were identified as a result of these subsequent examinations.

Other than the single indication identified in the safety injection system, Catawba Nuclear Station has not detected evidence of stress corrosion cracking or other corrosion degradation in borated water systems containing stainless steel welds or stainless steel piping.

Duke Energy Response Regarding OE on Fatigue and Overloading:

A search of the database of Duke Energy's Corrective Action Program[2] and Industry Operating Experience[3] related to thermal fatigue failures and overloading[1] revealed no site-specific leakages attributable to thermal or vibration fatigue for the subject piping segments. However, Duke Energy acknowledges there have been 1) industry fatigue failures in non-isolable portions of branch lines off the RCS due to thermal stratification cycling and 2) thermal fatigue failures in RHR system mixing tees; Duke Energy is managing these per EPRI MRP-146/MRP-192 guidelines, respectively. However, the piping segments in Relief Request 11-CN-002 are not within the scope of piping affected by these guidelines.

Footnotes:

1. **It is assumed that the term "overloading" in the RAI refers to any known past identified loadings that fell outside the bounds of the original analysis and design, such as cyclic loadings due to thermal stratification.**
2. **Duke Energy's Corrective Action Program (Problem Investigation Process or PIP) as administered through Nuclear Policy Manual Directive NSD 208.**
3. **Industry Operating Experience as related to thermal fatigue failures is taken from EPRI document MRP-85 "Operating Experience Regarding Thermal Fatigue of Piping Connected to PWR Reactor Coolant Systems".**

4. NRC Information Notice (IN) 2011-04, "Contaminants and Stagnant Conditions Affecting Stress Corrosion Cracking [SCC] in Stainless Steel Piping in Pressurized water Reactors," (ADAMS Accession No. ML103410363), discusses potential SCC in stainless steel piping. Discuss the potential for SCC in piping and piping components for Segments 1, 2, 3, 4, and 5.

Duke Energy Response:

Duke Energy recognizes there is potential for stress corrosion cracking to occur in the piping and piping components for Segments 1, 2, 3, 4, and 5, due to the operating conditions of each system and the stainless steel materials present. However, controls exist to prevent stress corrosion cracking of these segments of piping by controlling the environment and substances which cause stress corrosion cracking of stainless steel materials. Additionally, system leakage tests and walkdowns looking for evidence of leakage inside containment are performed every refueling outage. If evidence of leakage is detected, actions are taken to identify the source of leakage and resolve the cause.

NRC Information Notice (IN) 2011-04 has been distributed to both the Catawba and Duke Energy Nuclear fleet Boric Acid Corrosion Program engineers for awareness. Catawba has also implemented the NEI 03-08 good practice recommendation documented in PWROG Letter OG-10-436 dated December 20, 2010 and PA-MSC-0563. This good practice ensured a consistent and minimum level of awareness was communicated to the plant staff regarding the stainless steel outside diameter stress corrosion cracking

events that have occurred in the industry.

5. ASME Code Case N-731, "Alternative Class 1 System Leakage Test Pressure Requirements," approved for use in Regulatory Guide (RG) 1.147, Rev. 16 (ADAMS Accession No. ML101800536), provides an acceptable alternative to existing provisions of the ASME Code, Section XI. Discuss whether piping and piping components for Segments 1, 2, 3, 4, and 5 for which relief is requested, meet the requirements of ASME Code Case N-731.

Duke Energy Response:

Code Case N-731 allows Class 1 system leakage test pressure requirements to be lowered for portions of Class 1 safety injection systems where the portions are continuously pressurized during an operating cycle to a lower pressure than that pressure currently required by IWB-5221(a). Code Case N-731 does not apply to the segments listed in Relief Request 11-CN-002 because those segments are either not part of the safety injection system or are not continuously pressurized during plant operation.