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

August 14, 2012

Mr. K. Henderson
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 - ALTERNATIVE REQUIREMENTS FOR PERFORMING SYSTEM LEAKAGE TESTS ON THE ASME CODE CLASS 1 PIPING AND COMPONENT SEGMENTS (TAC NOS. ME7182, ME7183, ME7184, ME7185, ME7186, AND ME7187)

Dear Mr. Henderson:

By letter dated September 13, 2011, as supplemented by letter dated April 11, 2012, Duke Energy Carolinas, LLC (the licensee) submitted relief request (RR) 11-CN-002, to the Nuclear Regulatory Commission (NRC) staff for the use of alternatives to certain requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code), Section XI, related to performing system leakage tests on the ASME Code Class 1 piping and component segments. RR 11-CN-002 was requested for the remainder of Catawba Nuclear Station, Units 1 and 2 (Catawba 1 and 2), third 10-year in-service inspection (ISI) interval. Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(a)(3)(ii), the licensee requested to use alternatives on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the 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 alternatives for system leakage tests would provide reasonable assurance of leak-tightness and structural integrity of the piping and component segments identified in RR 11-CN-002, and that complying with the specified ASME Code, Section XI, requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff authorizes the licensee's proposed alternatives for the third 10-year ISI interval at Catawba 1 and 2.

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, Jon H. Thompson at 301-415-1119 or via e-mail at Jon.Thompson@nrc.gov.

Sincerely,

John Boska

John Boska, Acting Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosure:
Safety Evaluation

cc w/encl: Distribution via ListServ

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. 11-CN-002 REGARDING ALTERNATIVE REQUIREMENTS FOR

PERFORMING SYSTEM LEAKAGE TESTS

ON THE ASME CODE CLASS 1 PIPING AND COMPONENT SEGMENTS

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated September 13, 2011, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11264A028) as supplemented by letter dated April 11, 2012, (ADAMS Accession No. ML12104A267) Duke Energy Carolinas, (the licensee) submitted relief request (RR) 11-CN-002 for the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's review and approval. The licensee requested the use of alternatives to certain requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code), Section XI, "Rules for In-service Inspection (ISI) of Nuclear Power Plant Components," related to performing system leakage tests on the ASME Code Class 1 piping and component segments. RR 11-CN-002 is requested for the remainder of Catawba Nuclear Station, Units 1 and 2 (Catawba 1 and 2), third 10-year ISI interval.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(a)(3)(ii), the licensee proposes alternative system pressure tests for the piping and component segments listed in RR 11-CN-002 in lieu of performing system leakage tests in accordance with the requirements of the ASME Code, Section XI, IWB-5221(a) or IWB-5222(b), as applicable. The licensee requested to use the proposed alternative on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

The regulation at 10 CFR 50.55a(g)(4) specifies that ASME Code Class 1, 2 and 3 components (including supports) "must meet the requirements, except design and access provisions and

preservice examination requirements, set forth in [ASME Code] Section XI ...to the extent practical within the limitations of design, geometry and materials of construction of the components."

The regulation at 10 CFR 50.55a(g)(4) further states in part that "... Components which are classified as Class MC pressure retaining components and their integral attachments, and components which are classified as Class CC pressure retaining components and their integral attachments must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of the ASME [*Boiler and Pressure Vessel Code*] BP&V Code and addenda that are incorporated by reference in paragraph (b) of this section, subject to the condition listed in paragraph (b)(2)(vi) of this section and the conditions listed in paragraphs (b)(2)(viii) and (b)(2)(ix) of this section, to the extent practical within the limitation of design, geometry and materials of construction of the components.

The regulation at 10 CFR 50.55a(g)(4)(i) states that "Inservice examinations of components and system pressure tests conducted during the initial 120-month [10-year] inspection interval must comply with the requirements in the latest edition and addenda of the [ASME] Code incorporated by reference in paragraph (b) of this section on the date 12 months before the date of issuance of the operating license under this part, or 12 months before the date scheduled for initial loading of fuel under a combined license under part 52 of this chapter (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, Revision 16, when using Section XI ... that are incorporated by reference in paragraph (b) of this section), subject to the conditions listed in paragraph (b) of this section."

Pursuant to 10 CFR 50.55a(g)(6)(ii), "The Commission may require the licensee to follow an augmented inservice inspection [ISI] program for systems and components for which the Commission deems that added assurance of structural reliability is necessary." The regulation at 10 CFR 50.55a(a)(3) allows for proposed alternatives to the requirements of paragraph (g) of 10 CFR 50.55a when authorized by the NRC and if the licensee demonstrates that "(i) The proposed alternatives would provide an acceptable level of quality and safety; or (ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

3.0 TECHNICAL EVALUATION

The licensee in RR 11-CN-002 requested approval of alternatives to the system leakage test for the ASME Code Class 1 piping and component segments connected to (or part of) the reactor coolant system (RCS) that are isolated from direct RCS pressure during normal operation. These segments are isolated from the RCS by their configuration because they are upstream of a check valve, between two or more check valves, or between two normally closed valves that remain closed when the unit is in normal operation. RR 11-CN-002 uses the following designation legend for the systems for which approval of an alternative is requested.

NC RCS
NV Chemical and Volume and Control System
ND Residual Heat Removal System (RHR)
NI Safety Injection System
WL Liquid Radwaste System

3.1.1 ASME Code Components Affected

Segment 1:

ASME Code Class: Class 1
Component: 2-inch NV Piping and Components Upstream of Auxiliary Spray Inboard Check Valve NV-38 up to and Including Outboard RCS Isolation Valves NV-37A (Globe Valve) and NV-861 (Check Valve)

System: Chemical and Volume and Control System

The specifications for Segment 1 piping and components at Catawba 1 and 2, for which approval of an alternative is requested, are listed in Table 1 of this SE.

Table 1. Segment 1, NV piping and component specifications

Nominal Piping Size (NPS)/ Schedule (Sch)	2-inch NPS/ Sch 160
Material Type / Grade	Stainless Steel SA-376 / Grade 304
Design Pressure	2500 pounds per square inch, absolute (psia)
Design Temperature	650°F
Approximate Length	70 feet (total for both units)

Segment 2:

ASME Code Class: Class 1
Component: 12-inch, 1-inch, and ¾-inch Piping and Components on the ND; Suction Line between (and Including) the RCS Double Isolation Gate Valves ND-1B and ND-2A ("A" Train ND Suction) and Flow Restrictors Upstream of Valves ND-4, ND-110, and ND-116

12-inch, 1-inch, and ¾-inch Piping and Components on the ND; Suction Line between (and Including) the RCS Double Isolation Gate Valves ND-36B and ND-37A ("B" Train ND Suction) and Flow Restrictors Upstream of Valves ND-39, ND-111, and ND-117.

System: Residual Heat Removal System

The specifications for Segment 2 piping and components at Catawba 1 and 2, for which approval of an alternative is requested, are listed in Tables 2a, 2b, and 2c of this SE.

Table 2a. Segment 2, ND piping and component specifications

Nominal Piping Size (NPS) / Schedule (Sch)	12-inch NPS / Sch 140
Material Type / Grade	Stainless Steel SA-376 / Grade 316
Design Pressure	2500 psia
Design Temperature	650°F
Approximate Length	305 feet (Total for both Units)

Table 2b. Segment 2, ND piping and component specifications

Nominal Piping Size (NPS)/ Schedule (Sch)	1-inch NPS / Sch 160 Pipe and 6000# Fittings
Material Type / Grade	Stainless Steel SA-376 / Grade 304 (Pipe); Stainless Steel SA-182 / Grade F304 (Fittings)
Design Pressure	2500 psia
Design Temperature	650°F
Approximate Length	6 inches (Total for both Units)

Table 2c. Segment 2, ND piping and component specifications

Nominal Piping Size (NPS)/ Schedule (Sch)	¾-inch NPS / Sch 160 Pipe and 6000# Fittings
Material Type / Grade	Stainless Steel SA-376 / Grade 304 (Pipe) Stainless Steel SA-182 / Grade F304 (Fittings)
Design Pressure	2500 psia
Design Temperature	650°F
Approximate Length	14 inches (Total for both Units)

Segment 3:

ASME Code Class: Class 1
Component: 1½-inch NI Piping and Components within the 4 RCS Loops, between (and Including) Double Isolation Check Valve Pairs Listed Below:
NI-15 and NI-351 for Loop A
NI-17 and NI-352 for Loop B
NI-19 and NI-353 for Loop C
NI-21 and NI-354 for Loop D
System: NI

The specifications for Segment 3 piping and components at Catawba 1 and 2, for which approval of an alternative is requested, are listed in Table 3 of this SE.

Table 3. Segment 3, NI piping and component specifications

Nominal Piping Size (NPS)/ Schedule (Sch)	1½-inch NPS / Sch 160
Material Type / Grade	Stainless Steel SA-376 / Grade 304
Design Pressure	2500 psia for Valves NI-15, NI-17, NI-19, and NI-21 2750 psia for Remaining Components in Segment 3
Design Temperature	650°F
Approximate Length	22 feet (Total for both Units)

Segment 4:

ASME Code Class: Class 1
Component: 2-inch and ½-inch NC Piping and Components between (and Including) Double Isolation Valves and Flow Restrictors Listed Below:
NC-4, NC-5 (Isolation Valves for Loop A) and Flow Restrictor Upstream of Valve NC-6
NC-94, NC-95 (Isolation Valves for Loop B) and Flow Restrictor Upstream of Valve NC-113
NC-13, NC-106, and NC-115 (Isolation Valves for Loop C)
NC-19, NC-20, and NC-111 (Isolation Valves for Loop D)
¾-inch and 3-inch NC Piping and Components between (and Including) the Following RCS Double Isolation Valve Pairs on the Reactor Vessel Head Vent Line Listed Below:
NC-298 and NC-299 (3 inch)
NC-311 and NC-312 (¾inch)
System: RCS

The specifications for Segment 4 piping and components at Catawba 1 and 2, for which approval of an alternative is requested, are listed in Tables 4a, 4b, 4c, and 4d of this SE.

Table 4a. Segment 4, NC piping and component specifications

Nominal Piping Size (NPS) / Schedule (Sch)	3-inch NPS / Sch 160
Material Type / Grade	Stainless Steel SA-376 / Grade 304
Design Pressure	2500 psia
Design Temperature	650°F
Approximate Length	5 feet

Table 4b. Segment 4, NC piping and component specifications

Nominal Piping Size (NPS)/ Schedule (Sch)	2-inch NPS / Sch 160
Material Type / Grade	Stainless Steel SA-376 / Grade 304
Design Pressure	2500 psia
Design Temperature	650°F
Approximate Length	34 feet

Table 4c. Segment 4, NC piping and component specifications

Nominal Piping Size (NPS)/ Schedule (Sch)	2-inch NPS / Sch 160 Pipe and 6000# Fittings
Material Type / Grade	Stainless Steel SA-376 / Grade 304 (Pipe) Stainless Steel SA-182 / Grade F304 (Fittings)
Design Pressure	2500 psia
Design Temperature	650°F
Approximate Length	2 feet

Table 4d. Segment 4, NC piping and component specifications

Nominal Piping Size (NPS)/ Schedule (Sch)	½-inch NPS / Sch 160
Material Type / Grade	Stainless Steel SA-376 / Grade 304
Design Pressure	2500 psia
Design Temperature	650°F
Approximate Length	4 feet

Segment 5:

ASME Code Class: Class 1
Component: 1-inch NC piping and components between (and including) the following RCS double isolation valves on the Reactor Vessel Head vent line listed below:
NC-250A, NC-251 B, NC-252B, and NC-253A
System: RCS

The specifications for Segment 5 piping and components at Catawba 1 and 2, for which approval of an alternative is requested, are listed in Tables 5 of this SE.

Table 5. Segment 5, NC piping and component specifications

Nominal Piping Size (NPS)/ Schedule (Sch)	1-inch NPS / Sch 160
Material Type / Grade	Stainless Steel SA-376 / Grade 304
Design Pressure	2500 psia
Design Temperature	650°F
Approximate Length	12 feet

3.1.2 Applicable Code Edition and Addenda

The code of record for the third 10-year ISI interval at Catawba 1 and 2 is the 1998 Edition through 2000 Addenda of the ASME Code, Section XI.

3.1.3 Applicable Code Requirement

The ASME Code, Section XI, IWB-2500, Table IWB-2500-1, Examination Category B-P, Item Nos. B15.50 and B15.70, requires a system leakage test in accordance with IWB-5220 and the visual (VT-2) examination of IWA-5240 be performed during each refueling outage.

IWB-5221(a) of the ASME Code, Section XI, requires that the system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power.

IWB-5222(b) of the ASME Code, Section XI, requires that the pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval shall extend to all ASME Code Class 1 pressure retaining components within the system boundary.

3.1.4 Licensee's Reason for Requesting An Alternative

The licensee stated that the ASME Code Class 1 piping segments identified in RR 11-CN-002 are equipped with valves that provide double isolation of the reactor coolant pressure boundary. Under normal operating conditions, these isolation valves are closed and these piping segments are subject to RCS pressure and temperature only if leakage through the inboard valves occurs. To perform the ASME Code, Section XI, required pressure tests, it would be necessary to place

the plant in an abnormal configuration by opening the inboard valves or installing temporary jumper hoses around check valves to pressurize the piping segments. Performing these tests under these conditions causes a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Furthermore, the licensee stated that the system leakage test conducted at the proposed reduced test pressures, and at or near the end of the inspection interval, for Segments 1, 2, 3, and 5 is acceptable because leakage (if it were to occur) would be detectable at a reduced rate. For all segments (Segments 1, 2, 3, 4, and 5), if leakage occurs past the first isolation valve, leakage in the piping segments would be evident during the system leakage tests and the VT-2 examinations performed each refueling outage. Also, boric acid inspections performed during refueling outages provide additional assurance that leakage from these components would be detected. For this reason, the proposed alternatives provide an acceptable level of assurance of the leak-tightness and structural integrity of the subject piping segments.

3.1.5 Licensee's Proposed Alternative

Segments 1, 2, 3, and 5

The licensee proposed that in lieu of test pressure requirement of IWB-5221(a) of the ASME Code, Section XI, the system leakage test will be performed at a test pressure not less than 300 pounds per square inch, guage (psig) for Segments 1, 2, and 5 and not less than 42 psig for Segment 3. This alternative is proposed only for the system leakage test conducted at or near the end of the inspection interval in accordance with IWB-5222(b) of the ASME Code, Section XI. The VT-2 examinations shall extend to and include the second closed valve at the boundary extremity.

Segment 4

The licensee proposed that in lieu of test boundary extension requirements of IWB-5222(b) of the ASME Code, Section XI, the system leakage test conducted at or near the end of the interval will be performed in accordance with the requirements of IWB-5222(a) of the ASME Code, Section XI, with the inboard and outboard isolation valves configured in their normal reactor startup position. The VT-2 examinations shall extend to and include the second closed valve at the boundary extremity.

3.1.6 Licensee's Basis for Hardship

Segment 1

The licensee stated that pressurizing Segment 1 piping and components to NC system operating pressure (2235 psig) during plant startup with the NC system at normal operating pressure and temperature would cause a hardship because of the high risk of an inadvertent pressurizer auxiliary spray initiation to the pressurizer. This design transient is undesirable because it would force static piping contents (cold water) into the pressurizer spray line, resulting in an additional thermal design cycle. The plant design only allows ten of these cycles over the plant design life. For this reason, performing the system leakage test at the pressure required by IWB-5221(a) is a hardship for Segment 1.

Segment 2

The licensee stated that Segment 2 piping and components cannot be pressurized to NC system normal operation pressure due to thermal relief check valves ND-116 and ND-117 that are routed to the NC system. Technical Specification (TS) Surveillance Requirement 3.4.14.2 requires verification of interlock to prevent opening the loop suction valves (ND-1B, ND-2A, ND-36B, and ND-37A) with NC pressure greater than or equal to 425 psig. Per the bases for Technical Specification TS 3.4.14, the purpose of the interlock is to prevent an intersystem loss-of-coolant accident (LOCA) due to inadvertent opening of a loop suction isolation valve. Opening a loop suction valve to pressurize Segment 2 piping from the NC System would defeat the required interlock. Additionally, opening valves ND-1B or ND-36B to pressurize this segment would violate the 10 CFR 50.55a(c)(2)(ii)-required double isolation valve barrier of the RCS boundary from the ND system. This would create an inability to mitigate a LOCA if a break were to occur in the 12-inch diameter piping, reducing the plant's margin of safety. Valve ND-1B or ND-36B could not be relied upon to close against the postulated flow from the RCS through a 12-inch line break. It would also subject ND system components to potential risk of damage with only a single valve isolating ND system from RCS pressure. For these reasons, performing the system leakage test at the pressure required by IWB-5221(a) of the ASME Code, Section XI, is a hardship for Segment 2.

Segment 3

The licensee stated that no intermediate test connection exists on the piping between the check valve pairs to measure the test pressure locally. Modifications would be necessary to add test gauges upstream and downstream of the check valves to verify NC pressure. Aligning an NV pump to the boron injection flow path in Mode 3 (during startup) and cracking open valve NI-3 would constitute a manual safety injection, diminishing the allowed number of cold leg thermal design transients. This action would risk degradation of piping and welds (due to thermal fatigue) for the sake of verifying the leak-tightness of the piping segment. For these reasons, performing the system leakage test at the pressure required by IWB-5221(a) of the ASME Code, Section XI, is a hardship for Segment 3.

Segment 5

The licensee stated that opening either of the isolation valves in each valve pair to permit pressure testing would violate the 10 CFR 50.55a(c)(2)(ii)-required double isolation valve barrier of the RCS boundary during plant operation. Opening the inner isolation valve would also expose personnel to unnecessary safety hazards because personnel would have to be stationed at or near the valves in lower containment with the RCS at normal operating pressure and temperature. Because there are no test connections to the piping in Segment 5, testing of the piping between the reactor head vent piping double isolation valves by hydro pump or temporary jumpers is not possible. For these reasons, performing the system leakage test at the pressure required by IWB-5221(a) of the ASME Code, Section XI, is a hardship for Segment 5.

Segment 4

The licensee stated that the isolation valves in Segment 4 are maintained in the closed position during normal operation. Opening either of the isolation valves in each valve pair to permit pressure testing would violate the 10 CFR 50.55a(c)(2)(ii)-required double isolation valve barrier of the RCS boundary during plant operation. Opening the inner isolation valve would also expose personnel to unnecessary safety hazards because personnel would have to be stationed at or near the valves in lower containment with the RCS at normal operating pressure and temperature. Because there are no test connections to the piping in Segment 4, testing of the piping between the double isolation valves by hydro pump or temporary jumpers is not possible. For these reasons, extending the test boundary in accordance with requirements of IWB-5222(b) of the ASME Code, Section XI, during the test conducted at or near the end of the interval is a hardship for components in Segment 4.

3.1.7 Duration of Relief

The licensee has requested RR 11-CN-002 for the third 10-year ISI interval of both Catawba 1 and 2. The third 10-year ISI interval for Catawba 1 commenced on June 29, 2005, and will end on July 14, 2014. The third 10-year ISI interval for Catawba 2 commenced on October 15, 2005, and will end on August 19, 2016.

3.2 NRC Staff Evaluation

The NRC staff has evaluated the information provided in RR 11-CN-002, as supplemented by letter dated April 11, 2012. The examination and inspection requirements of the ASME Code, Section XI, Article IWB-2500, Table IWB-2500-1, Examination Category B-P, Item Nos. B15.50 and B15.70, are that a system leakage test and VT-2 examination shall be performed in accordance with IWB-5220, "System Leakage Test," and IWA-5240, "Visual Examination," respectively, during each refueling outage. For system leakage test, IWB-5221(a) requires that the normal operating pressure associated with normal system operation shall be used. For the system leakage test conducted at or near the end of each inspection interval, IWB-5222(b) requires that the pressure retaining boundary shall extend to all ASME Code Class 1 pressure retaining components within the system boundary.

The NRC staff has evaluated the licensee's basis for hardship (summarized in Section 3.1.6 of this Safety Evaluation (SE)), in performing system leakage tests in accordance with IWB-5221(a)-required pressure criteria for piping Segments 1, 2, 3, and 5, and IWB-5222(b)-required extension of test boundary criteria for piping Segment 4. As documented in RR 11-CN-002, the components in each segment are connected to the RCS but are normally isolated from direct RCS pressure during normal operation. They are isolated from the reactor coolant loop by their location, either upstream of a check valve, between two check valves, or between two closed valves that must remain closed during the plant operation. There would be potential personnel safety hazards ranging from immediate physical exposure to temporary connections whose medium is pressurized to the RCS operating pressures (as high as 2235 psig) and temperatures (as high as 557°F). Operators will have to be stationed at opened manual vent/drain valves serving as RCS single isolation pressure and temperature barriers in order to maintain the RCS boundary redundant valve protection requirement of 10 CFR 50.55a(c)(ii) during the test. In addition, the station operators setting up system leakage testing activities

and positioned near subject valves during the testing would be subject to radiation exposure. Therefore, the NRC staff has determined that the potential challenges (i.e., safe plant operation and personnel safety hazards and radiation exposure) that would be placed on the facility and plant personnel, for compliance with the requirements (i.e., IWB-5221(a) for piping Segments 1, 2, 3, and 5, and IWB-5222(b) for piping Segment 4) create hardship or usual difficulty to the licensee without a compensating increase in the level of quality or safety.

The NRC staff has evaluated the licensee's analysis for adequate quality and safety. By letter dated April 11, 2012, in its response to RAI question 1.a., the licensee provided a table with the pressures that piping segments 1, 2, 3, 4, and 5 could experience during normal operating, stagnant, and potential accident and fault conditions, as well as justifications for reasonable assurance of leak-tightness and structural integrity with the use of the proposed alternatives. The licensee stated that for a reasonable assurance of leak-tightness and structural integrity of the piping systems, the nondestructive examinations (NDE), as required by the ASME Code, Section XI, have been performed on selected welds in piping Segments 1, 2, 3, 4, and 5 and no recordable indications were identified. As an additional assurance, the Catawba 1 and 2 boric acid corrosion control program would detect evidence of any leakage during the operating cycle by identifying boron deposits on the outside surface of the components within the subject piping segments. For Segments 1, 2, 3, and 5, the leakage, if it were to occur, would be detected at the proposed reduced test pressures at reduced leakage rates.

The licensee has noted that this position is consistent with the NRC staff SE dated June 23, 2005, for RR 04-CN-004 (ADAMS Accession No. ML051780164), which requested the use of alternatives to the system leakage tests for the subject piping systems during the second (previous) 10-year ISI interval for Catawba 1 and 2. For Segment 4, if any leakage occurs past the first isolation valve, leakage in the piping segment would be evident during the system leakage tests and the VT-2 examinations performed during each refueling outage.

The NRC staff has considered this precedent and determined that the licensee's NDE examinations, corrosion control program, and the VT-2 examinations provide reasonable assurance for leak-tightness and structural integrity of the system.

By letter dated April 11, 2012, in its response to RAI question 3 the licensee provided a table with the number and type of welded connections in piping Segments 1, 2, 3, 4, and 5, the NDE methods used for the inspections of the welds, and any industry or plant-specific operating experience regarding potential degradation mechanisms. The licensee stated that there is potential for stress corrosion cracking (SCC) to occur given the operating conditions of systems containing borated water and stainless steel materials.

Regarding operating experience, the licensee stated that SCC was identified in one weld (not part of piping segments listed in RR 11-CN-002) in systems of a stagnant portion of the NI system during ISI examinations during the Catawba 1 and 2 refueling outage (1EOC18) in 2009. A recordable indication was detected by ultrasonic testing (UT) examinations. The flaw was inner-diameter surface-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. During the same refueling outage, the UT examinations performed on 36 additional welds located in stagnant portions of the NI system, found no additional recordable indications. Furthermore, a review of the

construction history of Catawba 1 and 2 welds located in the stagnant portions of NI piping inside the containment was performed to determine which welds had been repaired. Subsequently, the UT examinations were performed on 37 butt welds at Catawba 1 and on 44 butt welds at Catawba 2 that had records of repair. No recordable indications were identified as a result of these subsequent examinations. In addition, the licensee stated that the Catawba 1 and 2 site-specific and industry operating experience related to thermal fatigue failures and overloading revealed no leakages attributable to thermal or vibration fatigue for the subject piping segments.

The NRC staff has determined that the licensee has demonstrated by previous NDE of the welds in the subject piping that the structural integrity of the subject piping has been maintained and that the potential susceptibility of the subject piping and component segments to SCC as a result of the operating conditions and stainless steel materials will be monitored periodically by future NDE. The boric acid corrosion program at Catawba 1 and 2 monitors environment and substances that cause SCC in stainless steel materials. This program also monitors the subject piping and component segments for potential SCC.

In summary, the NRC staff has determined that compliance with the ASME Code, Section XI, requirements regarding test pressure during system leakage tests for the subject piping and component segments would not result in an increase in assurance of structural integrity commensurate with the potential challenges that would be placed on the facility and plant personnel. In the unlikely event of not detecting a leak in the subject piping segments during a refueling outage, the instrumentation available to the operators for detection and monitoring of RCS leakage would provide prompt qualitative information to permit them to take immediate corrective action during operation. Therefore, the NRC staff has determined that compliance with the ASME Code, Section XI, requirements regarding test pressure during system leakage tests for the piping and component segments identified in RR 11-CN-002 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff has determined that the proposed alternatives for system leakage tests will provide reasonable assurance of leak-tightness and structural integrity of piping and component segments identified in RR 11-CN-002. The NRC staff further determines that complying with the specified ASME Code, Section XI, requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii), and is in compliance with the ASME Code's requirements. Therefore, the NRC staff authorizes the licensee's proposed alternatives for the third 10-year ISI interval at Catawba 1 and 2.

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 Inservice Inspector.

Principal Contributor: A. Rezai, NRR

Date: August 14, 2012

If you have any questions, please contact the Project Manager, Jon H. Thompson at 301-415-1119 or via e-mail at Jon.Thompson@nrc.gov.

Sincerely,

/RA/

John Boska, Acting Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

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DATE	08/13/12	08/13/12	05/15/12	08/14/12

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