February 17, 2005

Mr. D. M. Jamil
Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: REQUEST FOR RELIEF 03-001, SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN (TAC NOS. MB9141 AND MB9142)

Dear Mr. Jamil:

By letters dated May 22, 2003, and September 8, 2004, Duke Energy Corporation, the licensee for Catawba Nuclear Station (Catawba), Units 1 and 2, submitted a request for relief, Relief Request No. 03-001, from the requirements of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, 1989 Edition with no addenda. Specifically, the licensee requested relief associated with the Code-required 100 percent volumetric examination of the regenerative heat exchanger head-to-shell and tubesheet-to-shell welds. The licensee’s proposed alternative is to perform visual VT-2 examinations during Code-required system leakage tests in lieu of the volumetric examinations for the second 10-year inservice inspection (ISI) intervals at Catawba, Units 1 and 2.

The Nuclear Regulatory Commission (NRC) staff, with technical assistance from its contractor, the Pacific Northwest National Laboratory, has reviewed the information provided for this relief request. The enclosed Safety Evaluation contains the NRC staff's evaluation and conclusions. Based on the information provided in the relief request, the NRC staff concludes that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the proposed alternative is authorized pursuant to Title 10 of the *Code of Federal Regulations*, Section 50.55a(a)(3)(ii) for the second ISI intervals at Catawba, Units 1 and 2.

Sincerely,

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosure: As stated

cc w/encl: See next page

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Docket Nos. 50-413 and 50-414

Enclosure: As stated

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Date: February 17, 2005

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF NUMBER 03-001

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DUKE ENERGY CORPORATION

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) staff, with technical assistance from its contractor, the Pacific Northwest National Laboratory (PNNL), has reviewed and evaluated the information provided by Duke Energy Corporation (Duke, the licensee) in its letter dated May 22, 2003, that proposed its Second 10-Year Interval Inservice (ISI) Inspection Program Plan Request for Relief No. 03-001 for Catawba Nuclear Station (Catawba), Units 1 and 2. The licensee provided additional information by letter dated September 8, 2004.

2.0 REGULATORY EVALUATION

ISI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable Code of record for the second

10-year ISI for Catawba, Units 1 and 2 is the 1989 Edition of the ASME B&PV Code, Section XI. The Catawba, Unit 1 second 10-year ISI interval began on June 29, 1995, and Catawba, Unit 2 second 10-year ISI interval began on August 19, 1996.

3.0 TECHNICAL EVALUATION

The ASME Code, Examination Category C-A, Items C1.20 and C1.30, requires essentially 100 percent volumetric examination, as defined by Figures IWC-2500-1 and -2, of the length of Class 2 head circumferential and tubesheet-to-shell welds for the regenerative heat exchanger.

The licensee proposed to eliminate the Code-required 100 percent volumetric examination of the Catawba, Units 1 and 2 regenerative heat exchanger head-to-shell and tubesheet-to-shell welds because of the high radiation exposure that the examiners receive. An estimated 9.975 man-rem of exposure would be received for each unit. Furthermore, Duke proposed that the pressure testing currently being performed under Examination category C-H, “All Pressure Retaining Components” Visual Examination, VT-2 and monitoring of these vessels for leakage during normal plant operations be considered as a basis for approval of its request for relief.

The regenerative heat exchanger and associated system piping, designed and constructed to meet the Class 2 requirements of the 1974 edition of ASME Section III, have a low probability of failure throughout their design life. The regenerative heat exchanger was fabricated from austenitic stainless steel (Type 304/316) and is resistant to base and weld metal degradation in the primary reactor coolant environment. The licensee strictly limits oxygen levels and contaminants in the primary system, thereby greatly reducing the susceptibility to stress corrosion cracking. Industry operating experience does not indicate that these stainless materials are susceptible to significant corrosion in the primary water environment.

The Catawba Technical Specifications place limits on the amount of reactor coolant leakage allowed during system operation, and Catawba has a system in place to detect any variation in the reactor coolant inventory, including the water present in both the tube and shell side of the regenerative heat exchanger, as well as its associated piping. Therefore, any weld failure would be detected by the reactor coolant leak detection system, and procedures and automatic system actions are in place to ensure that the heat exchanger would be isolated. The regenerative heat exchanger is isolable from the reactor coolant system by valves operated from the control room and/or automatic closure signals. The licensee performs the Code-required system leakage test and the VT-2 visual examination each outage. During the latest refueling outages for Unit 1 (EOC12) and Unit 2 (EOC11), the VT-2 visual examinations did not reveal any evidence of leakage.

The NRC staff determined that, based on its review of the licensee’s submittal, to require the licensee to perform the ASME Code-required examinations on the subject components of the regenerative heat exchanger would be a hardship without a compensating increase in quality and safety. Furthermore, the NRC staff determined that the licensee’s proposed alternative provides reasonable assurance of the continued structural integrity of the regenerative heat exchangers for both Catawba, Units 1 and 2.

4.0 CONCLUSION

The Catawba, Units 1 and 2 Request for Relief No. 03-001 from the Code requirements has been reviewed by the NRC staff, with the assistance of its contractor, PNNL.

The attached Technical Letter Report provides PNNL's evaluation of these requests for relief. The NRC staff has reviewed the TLR and adopts the evaluations and recommendations for authorizing the licensee's request for relief.

For Request for Relief 03-001, the NRC staff has concluded that compliance with the Code requirements would result in a hardship or unusual difficulty without a compensating increase in quality and safety. The alternative proposed by the licensee provides reasonable assurance of the continued leak tightness or structural integrity of the subject component. Therefore, Request for Relief 03-001, is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second 10-year ISI interval at Catawba, Units 1 and 2. All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Attachment: As stated

Principal Contributor: T. McLellan

Date: February 17, 2005

TECHNICAL LETTER REPORT

ON THE SECOND 10-YEAR INTERVAL INSERVICE INSPECTION

REQUEST FOR RELIEF 03-001

DUKE ENERGY CORPORATION

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NUMBERS: 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated May 22, 2003, the licensee, Duke Energy Corporation, submitted Request for Relief 03-001 from requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*. In response to an NRC Request for Additional Information, the licensee provided clarification and component drawings in a letter dated September 8, 2004. This request was submitted as part of the inservice inspection (ISI) program for the second 10-year inservice inspection (ISI) intervals at Catawba Nuclear Station, Units 1 and 2 (Catawba 1-2). The Pacific Northwest National Laboratory (PNNL) has evaluated the subject request for relief in the following section.

2.0 REGULATORY REQUIREMENTS

Inservice inspection of the ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME *Boiler and Pressure Vessel Code* (B&PV Code), and applicable addenda, as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). The regulation at 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), 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 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The code of record for Catawba 1-2 second 10-year interval inservice inspection programs, which began on

June 29, 1995 (Unit 1) and August 19, 1996 (Unit 2), is the 1989 Edition of Section XI of the ASME Boiler and Pressure Vessel Code.

3.0 EVALUATION

The information provided by Duke Energy Corporation in support of the request for relief from Code requirements has been evaluated and the basis for disposition is documented below.

3.1 Request for Relief 03-001, Examination Category C-A, Items C1.20 and C1.30, Pressure Retaining Welds in Pressure Vessels, Regenerative Heat Exchanger

Code Requirement: Examination Category C-A, Items C1.20 and C1.30, require essentially 100% volumetric examination, as defined by Figures IWC-2500-1 and -2, of the length of Class 2 head circumferential and tubesheet-to-shell welds. “Essentially 100%,” as clarified by ASME Code Case N-460, is greater than 90% coverage of the examination volume, or surface area, as applicable.

Licensee’s Proposed Alternative to Code: Based on the hardship of examining these welds, the licensee has proposed an alternative, in accordance with 10CFR50.55a(a)(3)(ii), to the Code-required 100% volumetric examination of Catawba 1-2 regenerative heat exchanger head-to-shell and tubesheet-to-shell welds. The licensee’s alternative is to perform visual VT-2 examinations during Code-required system leakage tests in lieu of the volumetric examinations.

Licensee’s Basis for Alternative (as stated):

> Due to high radiation dose rates in the area of the regenerative heat exchanger, it is station managements' request that these welds not be examined. To complete the examinations on the regenerative heat exchanger, an estimated 9.975 man-rem of exposure would be received for each unit.
>
> Listed below is a break-down of the examination tasks and their respective estimates [of exposure] as developed by the Catawba ALARA Staff and Inservice Inspection Coordinator. The estimates assume dose rates at the time of examination will be comparable to dose rates measured during previous outages.
>
> The average radiation level in the vicinity of the regenerative heat exchanger is 700 millirem per hour. To achieve this dose rate, the letdown line must be isolated prior to peroxide injection (induced crud burst). Also, a successful flush of the letdown line and regenerative heat exchanger using water from the reactor make-up water storage tank would be required. Both of these initiatives are routinely performed each outage.

Activity	Man-Hours	Average Dose Rate	Activity Exposure Estimate (mrem)
Erect/Remove scaffold	3	700	2100
Remove/Restore Insulation	2.5	700	1750
Weld Prep (assumes no grinding)	2.5	700	1750
NDE	6	700	4200
RP Support	0.25	700	175
Estimated Total Exposure			9975

The use of temporary shielding in the area of the heat exchanger has been considered. However, preliminary evaluations using typical methods and materials suggest that the amount of exposure incurred during installation and removal would be equal to or greater than the amount of exposure saved.

In addition, structural steel supporting the heat exchangers would have to be removed to facilitate the examination process on 6 of the 12 welds, or perform a limited coverage examination. The estimate shown above does not include removal and replacement of any structural steel.

Given there is no alternative volumetric or surface exam available due to similar radiation concerns, in lieu of implementing the requirements of Examination Category C-A, it is proposed that the pressure testing currently being performed under Examination category C-H, "All Pressure Retaining Components" (Visual Examination, VT-2) be considered as a basis for approval of this request.

Approval of the alternative testing provided by this relief request would significantly reduce unnecessary radiological exposure to plant personnel when complying with the volumetric examination requirements, without decreasing the confidence level in the operability of the Regenerative Heat Exchanger.

The alternative testing would not result in a reduction of the level of quality and safety for the following reasons:

1. The Regenerative Heat Exchanger and associated system piping, having been designed and constructed to meet the Class 2 requirements of the 1974 edition of ASME Section III, have a low probability of failure throughout their design life. It was fabricated from austenitic stainless steel (Type 304/316). This material is resistant to base and weld metal degradation of the heat exchanger in the primary reactor coolant environment. The 12 welds for each unit of Catawba are not dissimilar metal welds and thus are not subjected to primary water stress corrosion cracking associated with other materials. Oxygen levels in the primary system are strictly limited, thereby greatly reducing the susceptibility to intergranular stress corrosion cracking. Furthermore, there has been no industry operating experience

that has identified these stainless materials as susceptible to significant corrosion in the primary water environment.

2. Thermal fatigue has been considered in the design of the heat exchanger. No thermal cycling, stratification, or striping conditions have ever been identified to invalidate the qualification of the heat exchanger. While flow induced vibration of the connected letdown piping has been observed in the past, the structural integrity of the twelve shell to head and tubesheet welds is not affected. Vibrational forces originating at the orifices are attenuated at the HX by the configuration and distance between the orifices and HX. Furthermore, past modifications have minimized the vibration levels in the letdown piping. Based on industry operating and plant specific experience, there are no known degradation mechanisms identified for these welds.

3. Catawba Technical Specifications place conservative limits on the amount of reactor coolant leakage allowed during system operation. The reactor coolant leak detection system is in place to detect any variation in the reactor coolant inventory, including the water present in both the tube and shell side of the Regenerative Heat Exchanger, as well as its associated piping. Any weld failure would be detected by the reactor coolant leak detection system, and procedures and automatic system actions are in place to ensure that the heat exchanger would be isolated.

4. Regenerative Heat Exchanger is isolable from the reactor coolant system by valves either operated from the control room and/or automatic closure signals. The shell side of the heat exchanger is isolable from the reactor coolant system by two fail closed, air operated gate valves in series. These valves are provided a safety signal to automatically close on a Pressurizer Low-Level setpoint, which would be present with a significant leak from a Regenerative HX Shell-to-Head or Shell-to-Tubesheet weld failure. The tube side is isolable from the high pressure charging system by two motor operated gate valves in series, which are controlled from the Control Room and/or automatically close on a Safety Injection Signal (SS), which would be present with a significant HX weld leak. Regenerative Heat Exchanger is located inside the Containment Building, which is designed to contain any leakage.

5. Visual examinations associated with Pressure Testing of the Regenerative Heat Exchangers during the latest refueling outages for Unit 1 (EOC12) and Unit 2 (EOC11) did not identify any evidence of weld leakage.

<u>Response to Request for Additional Information (as stated):</u>

ASME Section XI, Paragraph IWC-1222(b), applies to component connections nominal pipe size four inches and smaller (including nozzles, socket fittings, and other connections) in vessels, piping, pumps, valves, and other components. Chemical and Volume Control System piping welded to the Regenerative Heat Exchanger Nozzles is three inches in diameter. Therefore, these nozzle-to-shell welds are exempt from volumetric and/or surface examination requirements per IWC-1222.

All Category C-A welds identified in Request for Relief 03-001 were volumetrically inspected by radiography during vessel fabrication. A weld repair was performed by the vessel manufacturer on the Catawba Unit 1 vessel, shell number 2, girth weld number 2. The repair was limited to one area contained within one four-inch RT film interval (4-5). In addition, a weld repair was performed on the Catawba Unit 2 vessel, shell number 3, girth weld number 1. The repair was limited to one area contained within two four-inch film intervals (4-5 and 5-6). The weld repair areas were re-examined by radiography and found to be acceptable. The remaining welds on the Catawba Unit 1 and 2 vessels were found to be radiographically acceptable during the fabrication process without performing weld repair activities.

All other Category C-A welds on Class 2 vessels have been reviewed and confirmed to be examined in accordance with Code requirements.

The regenerative heat exchanger is isolable from the Reactor Coolant System by valves either operated from the control room or by valves that receive automatic closure signals. The shell side of the heat exchanger is isolable from the Reactor Coolant System by two fail-closed, air operated gate valves in series. These valves are provided a safety signal to automatically close on a pressurizer low level, which would be present with a significant leak from a regenerative heat exchanger shell-to-head or shell-to-tubesheet weld failure. The tube side is isolable from the high pressure charging system by two motor operated gate valves in series, which are controlled from the control room and/or which automatically close on a safety injection signal. A safety injection signal would occur with a significant heat exchanger weld leak.

<u>Evaluation</u>: The Code requires that essentially 100% of the length of all Class 2 vessel shell-to-head and tubesheet-to-shell welds be volumetrically examined once during each ISI interval. This includes examination of 24 welds on the regenerative heat exchangers (RHXs) at Catawba 1-2 (12 welds per heat exchanger at each unit). However, because of the manner in which these heat exchangers operate, particulates from the reactor coolant system accumulate in low-flow regions of the vessels during normal service conditions. This causes the vessels and surrounding area to become highly radioactive. To require the licensee to examine the subject heat exchanger welds would present a significant hardship due to excessive personnel radiation exposure.

The RHXs at Catawba 1-2 are Class 2 and consist of a shell and tube design with three separate vessels stacked vertically and piped in series. The licensee considers all three vessels to be one heat exchanger. Each component possesses a head-to-shell and shell-to-tubesheet weld on either end, for a total of 12 welds per heat exchanger. The RHX is part of the plant chemical and volume control system, and is designed to recover heat from the letdown stream by reheating the charging stream during normal operation. The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flows through the tubes. The unit is made of austenitic stainless steel, and is of all-welded construction. Other than the subject shell-to-head and tubesheet-to-shell welds, no other welds are required by Code to be examined by volumetric or surface methods. This is because Class 2 inlet and outlet nozzle welds on the RHXs are less than NPS 4-inches in diameter, which are exempt from all examinations except visual VT-2 during pressure tests. The licensee estimates that approximately 10 man-Rem of radiation exposure will be incurred during examination of these welds on each heat exchanger at Catawba 1-2. This is due to activities associated with the examination such as erection and removal of scaffolding, insulation removal and replacement, surface preparation of the welds, and the actual examination process.

Several potential forms of degradation have been considered for these welds, however, no aggressive mechanisms can be identified that may challenge the structural integrity of the RHXs, based on materials of construction and operating environments. It is concluded that, once the subject shell and head welds have been thoroughly examined during preservice or prior inservice inspections, failure probabilities are very low, and that exposure of plant personnel to the high levels of radiation to support continued volumetric examinations is unwarranted.

In addition, the RHXs can be quickly isolated from the primary coolant system by valves if leakage is detected. Furthermore, in a brief review of international databases[1] and readily available literature to-date, no service-induced pressure boundary failures have been experienced for shell and/or head welds on this type of RHX. Therefore, Duke's proposal to continue to perform visual VT-2 examinations during system leakage tests, and to monitor these vessels for leakage during normal plant operations, provides reasonable assurance that the RHXs will continue to function as designed at Catawba 1-2.

To require the Code volumetric examinations of the subject RHX shell welds would subject the licensee to a significant hardship, with no compensating increase in quality or safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), it is recommended that the licensee's proposed alternative found in Request for Relief 03-001 be authorized for the second interval at Catawba 1-2.

4.0 CONCLUSIONS

For Request for Relief 03-001, it has been shown that compliance with the Code requirements would result in a hardship or unusual difficulty with no compensating increase in quality or safety. The alternative proposed by the licensee provides reasonable assurance of the continued leakage or structural integrity of the subject component. Therefore, for Request for Relief 03-001, it is recommended that the licensee's alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second interval at Catawba 1-2.

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1. Databases searched include:

NUREG/CR-5779 (1992), a survey of operating experience with non-power cycle heat exchangers performed by Oak Ridge National Laboratory based on three data sources: 1) LERs through 1991, 2) the Nuclear Plant Reliability Data System (NPRDS) which is now the Equipment Performance and Information Exchange (EPIX), and 3) Nuclear Power Experience.

PIPExp™, a commercially available database which lead in 2002 to the establishment of the OECD/Nuclear Energy Agency Pipe Failure Data Exchange (OPDE) project. As an international cooperative effort, OPDE is supported by 12 countries and 19 organizations as a forum for component failure data exchange and analysis.

EGG-SSRE-9639 (1991). This report by the Idaho National Engineering Laboratory (INEL) provides leak and rupture frequency estimates for heat exchanger shells. The estimates are based on reviews of information extracted from nuclear power experience. The raw data provided in an appendix to the report shows zero heat exchanger pressure boundary failures.