March 9, 2001

Mr. G. R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION, UNIT 2 RE: SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN REQUEST FOR RELIEF NO. 00-001 FOR CATAWBA NUCLEAR STATION, UNIT 2 (TAC NO. MA9125)

Dear Mr. Peterson:

The Nuclear Regulatory Commission (NRC) staff with technical assistance from its contractor, the Idaho National Engineering and Environmental Laboratory (INEEL), has reviewed and evaluated the information provided by Duke Energy Corporation by letter dated May 23, 2000, proposing its second 10-Year Interval Inservice Inspection Program Plan Request for Relief No. 00-001 for Catawba Nuclear Station, Unit 2.

The staff concludes that certain inservice examinations cannot be performed to the extent required by the American Society of Mechanical Engineers Code, Section XI (the Code) at Catawba Nuclear Station, Unit 2. For the items discussed in Relief Request No. 00-001 (Parts A and B), the staff has determined that the Code requirements are impractical to meet, and that reasonable assurance of the structural integrity of the subject components has been provided by the examinations that have been completed. Therefore, relief is granted pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g)(6)(i). The staff has also determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property, or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

The NRC staff's evaluation and conclusions are contained in Enclosure 1. Enclosure 2 summarizes both portions of the relief request. Enclosure 3 is the INEEL Technical Letter Report.

Sincerely,

/RA/

Maitri Banerjee, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures: As stated

cc w/encls: See next page

March 9, 2001

Mr. G. R. Peterson	Distribution:	RidsNrrLACHawes (paper copy)
Site Vice President	PUBLIC	RidsNrrPMCPatel (paper copy)
Catawba Nuclear Station	PDII-1 Rdg.	RidsOgcRp
Duke Energy Corporation	RidsNrrDlpmLpdii	GHill (4)
4800 Concord Road	RidsNrrDlpmLpdii1 (paper copy)	
York, South Carolina 29745-9635	MBanerjee	RidsAcrsAcnwMailCenter
	SRosenberg	RidsRgn2MailCenter

SUBJECT: CATAWBA NUCLEAR STATION, UNIT 2 RE: SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN REQUEST FOR RELIEF NO. 00-001 FOR CATAWBA NUCLEAR STATION, UNIT 2 (TAC NO. MA9125)

Dear Mr. Peterson:

The Nuclear Regulatory Commission (NRC) staff with technical assistance from its contractor, the Idaho National Engineering and Environmental Laboratory (INEEL), has reviewed and evaluated the information provided by Duke Energy Corporation by letter dated May 23, 2000, proposing its second 10-Year Interval Inservice Inspection Program Plan Request for Relief No. 00-001 for Catawba Nuclear Station, Unit 2.

The staff concludes that certain inservice examinations cannot be performed to the extent required by the American Society of Mechanical Engineers Code, Section XI (the Code) at Catawba Nuclear Station, Unit 2. For the items discussed in Relief Request No. 00-001 (Parts A and B), the staff has determined that the Code requirements are impractical to meet, and that reasonable assurance of the structural integrity of the subject components has been provided by the examinations that have been completed. Therefore, relief is granted pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g)(6)(i). The staff has also determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property, or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

The NRC staff's evaluation and conclusions are contained in Enclosure 1. Enclosure 2 summarizes both portions of the relief request. Enclosure 3 is the INEEL Technical Letter Report.

Sincerely,

/RA/

Maitri Banerjee, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures: As stated

cc w/encls: See next page

DOCUMENT NAME: C:\RRA9125.wpd

OFFICE	PDII-1:PM	PDII-1:LA	SC(A):PDII-1	OGC
NAME	CPatel:cn	EDunnington for CHawes	MBanerjee	RHoefling
DATE	2/13/2001	2/13/2001	2/22/2001	3/2/2001

OFFICIAL RECORD COPY

Catawba Nuclear Station

cc:

Mr. Gary Gilbert
Regulatory Compliance Manager
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Ms. Lisa F. Vaughn
Legal Department (PB05E)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

Anne Cottingham, Esquire
Winston and Strawn
1400 L Street, NW
Washington, DC 20005

North Carolina Municipal Power
Agency Number 1
1427 Meadowwood Boulevard
P. O. Box 29513
Raleigh, North Carolina 27626

County Manager of York County
York County Courthouse
York, South Carolina 29745

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

North Carolina Electric Membership
Corporation
P. O. Box 27306
Raleigh, North Carolina 27611

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
4830 Concord Road
York, South Carolina 29745

Virgil R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental
Control
2600 Bull Street
Columbia, South Carolina 29201-1708

Mr. C. Jeffrey Thomas
Manager - Nuclear Regulatory
Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Saluda River Electric
P. O. Box 929
Laurens, South Carolina 29360

Mr. Peter R. Harden, IV
VP-Customer Relations and Sales
Westinghouse Electric Company
5929 Carnegie Blvd.
Suite 500
Charlotte, North Carolina 28209

cc:

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of
Environment, Health, and
Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR

SECOND 10-YEAR INTERVAL INSERVICE INSPECTION

REQUEST FOR RELIEF NO. 00-001

FOR

CATAWBA NUCLEAR STATION, UNIT 2

DUKE ENERGY CORPORATION

DOCKET NO. 50-414

1.0 INTRODUCTION

Inservice inspection (ISI) of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

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 Code of record for the Catawba Nuclear Station, Unit 2, second 10-year ISI interval is the 1989 Edition of the ASME B&PV Code.

2.0 EVALUATION

The Nuclear Regulatory Commission (NRC) staff, with technical assistance from the Idaho National Engineering and Environmental Laboratory (INEEL), has reviewed the information concerning second 10-Year Inservice Inspection (ISI) Program Plan Request for Relief 00-001[1] (Parts A and B) for Catawba Nuclear Station, Unit 2, provided in a Duke Energy Corporation (the licensee) letter dated May 23, 2000.

The staff adopts the evaluations and recommendations for granting relief from the Code requirements contained in the Technical Letter Report (TLR), included as Enclosure 3, prepared by INEEL.

Enclosure 2 summarizes both portions of the relief request.

For the Catawba Nuclear Station, Unit 2, the staff determined that because of access limitations, the Code requirement of 100-percent volumetric examination is impractical to perform for the subject welds discussed in Request for Relief No. 00-001 (Parts A and B). To gain access for examination of the subject welds, design modifications would be required. Imposition of this requirement would create an undue burden on the licensee.

For Request for Relief No. 00-001 (Part A), the licensee has volumetrically examined a significant portion of these welds, obtaining 75-percent coverage. In addition, the licensee completed 100 percent of the Code-required surface examination. Based on the coverage obtained, the staff determined that any existing patterns of degradation would have been detected by the examinations completed, and the examinations performed provide reasonable assurance of structural integrity of the subject welds.

For Request for Relief No. 00-001 (Part B), the licensee has volumetrically examined a significant portion of the weld, obtaining 88.34-percent coverage for the subject weld. Based on the coverage obtained, the staff determined that any existing patterns of degradation would have been detected by the examination completed, and the examination performed provides reasonable assurance of structural integrity of the subject weld.

3.0 CONCLUSION

The Catawba Nuclear Station, Unit 2, Request for Relief No. 00-001 (Parts A and B) from certain Code requirements has been reviewed by the NRC staff with the assistance of its contractor, INEEL. The TLR provides INEEL's evaluation of these requests for relief. The staff has reviewed the TLR and adopts the evaluations and recommendations for granting relief.

The staff concludes that certain inservice examinations cannot be performed to the extent required by the Code at Catawba Nuclear Station, Unit 2. For the items discussed in Relief Request No. 00-001 (Parts A and B), the staff has determined that the Code requirements are impractical to meet, and that reasonable assurance of the structural integrity of the subject

1. For ease of evaluation the staff has divided Request for Relief No. 00-001 into parts A and B.

components has been provided by the examinations that have been completed. Therefore, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i). The staff has also determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property, or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

Principal Contributor: T. McLellan

Date: March 9, 2001

Catawba Nuclear Station, Unit No. 2
Second 10-Year ISI Interval

TABLE 1
SUMMARY OF RELIEF REQUESTS

Relief Request Number	INEEL TLR Sec.	System or Component	Exam Category	Item No.	Volume or Area to be Examined	Required Method	Licensee Proposed Alternative	Relief Request Disposition
00-001 (Part A)	2.1	Class 1 Dissimilar Metal Welds	B-F	B5.70 B5.130	SG Nozzle-to-Safe End Welds Safe End to Pipe Welds	Volumetric/Surface	Partial volumetric examination be found acceptable	Granted 10 CFR 50.55a(g)(6)(i)
00-001 (Part B)	2.2	Class 2 Pressure Vessels	C-A	C1.20	Volume Control Tank Lower Head-to-Shell Weld	Volumetric	Partial volumetric examination be found acceptable	Granted 10 CFR 50.55a(g)(6)(i)

TECHNICAL LETTER REPORT
SECOND 10-YEAR INTERVAL INSERVICE INSPECTION
REQUEST FOR RELIEF NUMBER 00-001
FOR
DUKE ENERGY CORPORATION
CATAWBA NUCLEAR STATION, UNIT 2
DOCKET NUMBER: 50-414

1. INTRODUCTION

By letter dated May 23, 2000, the licensee, Duke Energy Corporation, submitted Request for Relief Number 00-001 from the requirements of the ASME Code, Section XI, for the Catawba Nuclear Station, Unit 2 second 10-year inservice inspection (ISI) interval. The Idaho National Engineering and Environmental Laboratory (INEEL) staff's evaluation of the subject request for relief is in the following section.

2. 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. The Code of record for the Catawba Nuclear Station, Unit 2, second 10-year ISI interval, which began August 19, 1996, is the 1989 Edition of Section XI of the ASME Boiler and Pressure Vessel Code.

2.1 Request for Relief No. 00-001 (Part A)[1], Examination Category B-F, Pressure Retaining Dissimilar Metal Welds

Code Requirement: Examination Category B-F, Items B5.70, and B5.130 require 100% volumetric and surface examinations of steam generator nozzle-to-safe end welds and dissimilar metal butt welds as defined by Figure IWB-2500-8.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100% volumetric examination of the welds listed below.

Unit	Component ID	Description	Category Item No.	Exam Coverage	Limitation
2	2SGB-INLET-SE B05.070.003	Safe End to Nozzle	B-F B5.70	75%	Volumetric Examination limited due to nozzle geometry and Material Characteristics
2	2SGB-OUTLET-SE B05.070.004	Safe End to Nozzle	B-F B5.70	75%	Volumetric Examination limited due to nozzle geometry and Material Characteristics
2	2NC11-02 B05.130.006	Safe End to Pipe	B-F B5.130	75%	Volumetric Examination limited due to nozzle geometry and Material Characteristics
2	2NC11-03 B05.130.007	Safe End to Pipe	B-F B5.130	75%	Volumetric Examination limited due to nozzle geometry and Material Characteristics

[1] For ease of evaluation the INEEL Staff has divided Request for Relief No. 00-001 into parts A and B

<u>Licensee's Basis for Requesting Relief</u> (as stated):

"During the ultrasonic examination of the 2SGB Inlet and Outlet Nozzle-to-Safe-End and Safe-End-to-Pipe Welds, 2SGB-INLET-SE, 2SGB-OUTLET-SE, 2NC11-02 and 2NC11-03 (Item Numbers B05.070.003, B05.070.004, B05.130.006 and B05.130.007 respectively) shown in Attachments[2] 2, 3, 4 and 5, greater than 90% coverage of the required examination volume could not be obtained. Material characteristics and single sided access caused by component geometry prevents two-beam path direction coverage of the examination volume and limits the examination coverage to 75%. The most effective ultrasonic technique for the examination of dissimilar metal welds and cast stainless steel welds uses refracted longitudinal waves. The longitudinal wave is preferred as the austenitic weld metal and buttering create highly attenuative barriers to shear wave ultrasound. The longitudinal wave is less affected by these difficulties. However, the longitudinal wave is affected by mode conversion when it strikes the inside surface of the safe end or pipe at any angle other than a right angle to the surface.

"...the mode conversion process creates two sound beams of differing intensities reflecting off the inside surface. At incident angles greater than 30 degrees, the shear wave will predominate. However, the shear wave is attenuated and scattered by the austenitic weld metal and the layer of buttering. The examination sensitivity is degraded to such an extent that any examination using the second sound path leg is meaningless. Therefore, the two-beam path direction coverage requirement is impractical.

"In order to obtain the required two-beam path direction coverage, welds would have to be re-designed to allow scanning from both sides.

"Although the examination volume requirements as defined in ASME Section XI 1989 Edition with no addenda, Figure IWB-2500-8, Examination Volume C-D-E-F for ID Numbers 2SGB-INLET-SE, 2SGB-OUTLET-SE, 2NC11-02 and 2NC11-03 (Item Numbers B05.070.003, B05070.004, B05.130.006 and B05.130.007 respectively) could not be met, the amount of coverage obtained for these examinations provides an acceptable level of quality and integrity. For results of the examinations, reference Attachments 2, 3, 4, and 5.

"The nozzle to safe-end and safe-end to pipe welds on the Steam Generator Inlet and Outlet Nozzles are located inside containment and are part of the reactor coolant system pressure boundary. General Design Criterion 30, 'Quality of Reactor Coolant Pressure Boundary,' of Appendix A to 10 CFR Part 50, 'General Design Criteria for Nuclear Power Plants,' mandates that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. If a leak were to develop at these weld

---

2. Attachments, sketches, and examination reports submitted by the licensee are not included in this report.

locations discussed in this relief request, the instrumentation available to the operators for detection and monitoring of leakage would provide a prompt and qualitative information necessary to permit them to take immediate corrective action. If a leak should develop in these aforementioned locations, the only corrective action would be shutdown and depressurize the reactor coolant system, since the welds are non-isolable.

“Plant Technical Specifications dictate that a reactor coolant system water inventory balance be performed on a regular basis. A normal operating practice is to perform this computer based mass balance on a daily frequency and/or whenever the operators suspect any abnormal changes to other leakage detection systems. Plant Technical Specifications require that if the leak rate cannot be reduced below 1 gpm unidentified that the plant be put in hot standby within 6 hours and in cold shutdown within the following 30 hours. Leakage as a result of a failed weld discussed in this section would show up as unidentified leakage and subject to the 1 gpm limit.

“Other leakage detection systems available to the operator and dictated per plant technical specifications are:

- Containment Atmosphere Gaseous and Particulate Radioactivity Monitoring System (EMF monitors 38 & 39) which would detect airborne radiological activity;
- Containment Floor and Equipment Sump Level and Flow Monitoring Subsystem where unidentified accumulated water on the containment floor would be monitored and evaluated as sump level changes;
- Containment Ventilation Unit Condensate Drain Tank Level Monitoring Subsystem which collects and measures as unidentified leakage the moisture removed from the containment atmosphere.

“Additionally, other indicators are also available to the operator that a leak exists or may be developing:

- Containment Atmosphere Iodine Monitor (EMF 40);
- Charging/Letdown system mismatches;
- Containment humidity indications;
- Pre-Cycle walkdowns performed each outage while system is at operating temperature and pressure prior to criticality;
- Post-Cycle walkdowns performed at operating temperature and pressure performed during unit shutdown.”

Licensee’s Proposed Alternative Examination (as stated):

“No additional examinations are planned during the current interval for ID Numbers 2SGB-INLET-SE, 2SGB-OUTLET-SE, 2NC11-02, 2NC11-03... Duke Energy Corporation will continue to use the most current ultrasonic techniques available to obtain maximum coverage for future examinations of these ID Numbers.”

Evaluation: The Code requires 100% volumetric and surface examination of the subject Steam Generator nozzle-to-safe end and safe-end to pipe butt welds. Review of the sketches and examination reports submitted by the licensee demonstrates that complete volumetric examination coverage is impractical due to restricted access caused by nozzle configuration and material characteristics of the austenitic weld metal and buttering, which create highly attenuative barriers to shear wave ultrasound. To meet the Code requirements, the subject nozzles and/or materials would require design modification to allow access to the subject welds. Therefore, the volumetric examination of the subject welds is impractical to perform to the extent required by the Code. Imposition of this requirement would create a considerable burden on the licensee.

The licensee has completed a significant portion (75%) of the Code required volumetric examinations on the subject welds. In addition, the licensee has performed 100% of the required surface examination of the subject welds. Based upon the limited volumetric coverages achieved, and the 100% surface examinations performed, it is reasonable to conclude that existing patterns of degradation, if present, would have been detected, providing reasonable assurance of the structural integrity of the welds. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.2 Request for Relief No. 00-001 (Part B), Examination Category C-A, Item C1.20, Pressure Retaining Welds in Pressure Vessels

Code Requirement: Examination Category C-A, Item C1.20, requires 100% volumetric examination of Head Circumferential Welds of Pressure Vessels as defined by Figure IWC-2500-1.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100% volumetric examination of the weld shown in the table below.

Unit	Component ID	Description	Category Item No.	Exam Coverage	Limitation
2	2VCT-LH-SH C01.020.010	Volume Control Tank Lower Head-to-Shell Weld	C-A C1.20	88.34%	Volumetric Examination limited due to four support legs.

Licensee's Basis for Requesting Relief (as stated):

"Although the examination volume requirements as defined in ASME Section XI, Appendix III, Paragraph III-4420, 1989 Edition with no addenda, for ID Number 2VCT-LH-SH (Item Number C01.020.010) could not be met, the amount of coverage obtained provides an acceptable level of quality and integrity. For results of the examination, reference Attachment 6.

"The Volume Control Tank (VCT) is used in power operations. The VCT is located in the Auxiliary Building adjacent to the unit mechanical penetration room on floor elevation 560 feet. During power operations and unit refueling outages, the VCT is accessible for visual inspections.

"If a leak were to occur at the weld in question (lower head to shell weld), there are several periodic tests and evaluations that are performed by established procedures that should identify the leakage for prompt OPS/ENG evaluation:

- During power operation, any leakage from the VCT would be identified as a mass loss in reactor coolant system water inventory balance. As described above, a normal operating practice is to perform this computer based mass balance on a daily frequency and/or whenever the operators suspect any abnormal changes to other leakage detection systems. Plant Technical Specification requires that if the leak rate cannot be reduced below 1 gpm unidentified that the plant be put in hot standby within 6 hours and in cold shutdown within the following 30 hours. Leakage as a result of a failed weld discussed in this section would show up as unidentified leakage and subject to the 1 gpm limit.

- If a leak were to occur at the subject weld, the water would spill on floor in VCT room and flow to floor drain and then to Floor Drain Tank. Our Chemistry department periodically monitors the tank level and evaluates unidentified leakage for correction.

- Weekly visual inspections are made by Operations into the VCT Room per PT/1(2)/A/4150/02 (Visual Inspection of Radioactive Components Outside Containment). Any leaks are required to be reported and evaluated per this Periodic Test.

- Quarterly walkdowns by the System Engineer include a check of the VCT and related components.

- Periodically, visual material condition inspections in accordance with NSD 104 are made in the VCT room by the site owner of the Aux. Bldg. Elev. 560 area. Identified leakage would be reported for evaluation.

- At a frequency of each refueling outage, visual leakage inspections of the VCT and charging system are made per PT procedure PT/1(2)/A/4202/06, 'Leak Rate Determination for NV System.' Any NV components identified with external leakage are documented for evaluation, including the VCT."

<u>Licensee's Proposed Alternative Examination</u> (as stated):

"No additional examinations are planned during the current interval for ID Numbers ...and 2VCT-LH-SH. Duke Energy Corporation will continue to use the most current ultrasonic techniques available to obtain maximum coverage for future examinations of these ID Numbers."

<u>Evaluation</u>: The Code requires 100% volumetric examination of the subject lower head-to-shell weld. Complete examination coverage is not possible due to the four component supports attached near the subject weld. Therefore, the volumetric examination is impractical to perform to the extent required by the Code. To meet the Code requirements, the volume control tank supports would have to be redesigned,

refabricated, and reinstalled. Imposition of the examination requirement to obtain an additional two percent[3] of coverage would create a considerable burden on the licensee.

The licensee has completed a significant portion of the Code required volumetric examination (88.34%). Based upon the significant volumetric examination coverage obtained, it is determined that existing patterns of degradation would have been detected and reasonable assurance of the structural integrity of the subject weld was provided. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

3. CONCLUSION

The INEEL staff evaluated the licensee's submittal and concluded that certain inservice examinations cannot be performed to the extent required by the Code at the Catawba Nuclear Station, Unit 2. For Request for Relief No. 00-001, Parts A and B, it is concluded that the Code requirements are impractical for the subject welds. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

3. The licensee has adopted Code Case N-460 which defines "essentially 100%" as greater than 90% coverage.