July 12, 2000

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

SUBJECT: SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN REQUEST FOR RELIEF NO. 99-02 (PARTS A, B, C, D, AND E) FOR CATAWBA NUCLEAR STATION, UNIT 1 (TAC NO. MA6316)

Dear Mr. Peterson:

The Nuclear Regulatory Commission 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 August 18, 1999, proposing its Second 10-Year Interval Inservice Inspection Program Plan Request for Relief No. 99-02 (Parts A, B, C, D, and E) for Catawba Nuclear Station, Unit 1.

The staff concludes that Request for Relief No. 99-02 (Parts A, B, C, D, and E) as evaluated by by the staff will provide reasonable assurance of structural integrity of the subject components in your requests for relief. The staff has determined that granting relief pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 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.

The staff's evaluation and conclusions are contained in Enclosure 1. Enclosure 2 lists each request for relief and the status of approval. Enclosure 3 is the INEEL Technical Letter Report.

Sincerely,

/RA/

L. Raghavan, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-413

Enclosures: As stated

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR

SECOND 10-YEAR INTERVAL INSERVICE INSPECTION

REQUEST FOR RELIEF NO. 99-02 (PARTS A, B, C, D, AND E)

FOR

CATAWBA NUCLEAR STATION, UNIT 1

DUKE ENERGY CORPORATION

DOCKET NO. 50-413

1.0 INTRODUCTION

Inservice inspection 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 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 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 ten-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) twelve 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 1 second 10-year ISI interval is the 1989 Edition of the ASME Boiler and Pressure Vessel Code.

Enclosure 1

2.0 EVALUATION

The Nuclear Regulatory Commission Staff, with technical assistance from Idaho National Engineering and Environmental Laboratory (INEEL), has reviewed the information concerning inservice inspection (ISI) program Request for Relief No. 99-02 (Parts A, B, C, D, and E) submitted for the second ten-year interval for Catawba Nuclear Station, Unit 1 in a Duke Energy Corporation (the licensee) letter dated August 18, 1999.

The NRC staff adopts the evaluations and recommendations for granting relief contained in the Technical Letter Report (TLR), Enclosure 3, prepared by INEEL. Enclosure 2 lists each relief request and the status of approval. Results of the review are provided in Section 2.0 of the TLR.

In making this determination, the staff has considered the impracticality of performing the required examination, the burden on the licensee if the requirements were imposed. For the Catawba Nuclear Station, Unit 1 relief is granted from the inspection requirements which have been determined to be impractical to perform. Request for Relief No. 99-02 (Parts A, B, C, D, and E) is granted for the second 10-year interval inservice inspection program plan.

3.0 CONCLUSION

The Catawba Nuclear Station, Unit 1 ISI program Request for Relief No. 99-02 (Parts A, B, C, D, and E) from the Code requirements have been reviewed by the staff with the assistance of its contractor, INEEL. The TLR provides INEEL's evaluation of Request for Relief (Parts A, B, C, D, and E). The staff has reviewed the TLR and concurs with the evaluations and recommendations for granting relief. A summary of Request for Relief No. 99-02 (Parts A, B, C, D, and E) determinations is presented in Enclosure 2.

The staff concludes that Request for Relief No. 99-02 (Parts A, B, C, D, and E) as evaluated by this Safety Evaluation will provide reasonable assurance of structural integrity of the subject components in the licensee’s requests for relief. The staff has 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 are imposed on the facility.

Principal Contributor: T. McLellan

Date: July 12, 2000

CATAWBA NUCLEAR STATION, UNIT 1
Third 10-Year ISI Interval

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
99-02, Part A	2.1	Steam Generator	B-D	B3.140	Nozzle Inner Radius Section	Volumetric	None	Granted 10 CFR50.55a(g)(6)(i)
99-02, Part B	2.2	Steam Generator	B-F	B5.70	Nozzle-to-Safe End Welds	Volumetric/Surface	None	Granted 10 CFR50.55a(g)(6)(i)
99-02, Part C	2.3	Class 1 Piping	B-J	B9.11 B9.31	Circumferential Piping Welds Branch Connection Welds	Volumetric/Surface	None	Granted 10 CFR50.55a(g)(6)(i)
99-02, Part D	2.4	Volume Control Tank	C-A	C1.20	Head-to-Shell Weld	Volumetric	None	Granted 10 CFR50.55a(g)(6)(i)
99-02, Part E	2.5	Class 2 Piping	C-F-1	C5.11	Circumferential Welds in SS Piping	Volumetric/Surface	None	Granted 10 CFR50.55a(g)(6)(i)

TECHNICAL LETTER REPORT
ON THE SECOND 10-YEAR INTERVAL INSERVICE INSPECTION
REQUEST FOR RELIEF SERIAL NO. 99-02
FOR
DUKE ENERGY CORPORATION
CATAWBA NUCLEAR STATION, UNIT 1
DOCKET NUMBER: 50-413

1. INTRODUCTION

By letter dated August 18, 1999, the licensee, Duke Energy Corporation, submitted Request for Relief Serial No. 99-02 seeking relief from the requirements of the ASME Code, Section XI, for the Catawba Nuclear Station, Unit 1. This relief request is for the 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 1, second 10-year ISI interval, which began June 29, 1995, is the 1989 Edition of Section XI of the ASME Boiler and Pressure Vessel Code.

2.1 Request for Relief Serial No. 99-02, Part A, Examination Category B-D, Item B3.140, Steam Generator Nozzle Inside Radius Sections

Code Requirement: Examination Category C-D, Item B3.140 requires 100% volumetric examination, as defined by Figure IWB-2500-7, for steam generator nozzle inner radius (IR) sections.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code requirements for the welds listed below.

Comp. ID	Component Description	ASME Category	Item Number	Aggregate Coverage	Limitation
B03.140.003	Steam Generator 1B Inlet (IR)	B-D	B3.140	83.28%	Volumetric examination limited due to the nozzle OD to the vessel thickness
B03.140.004	Steam Generator 1B Outlet(IR)	B-D	B3.140	83.28%	Volumetric examination limited due to the nozzle OD to the vessel thickness
B03.140.007	Steam Generator 1D Inlet (IR)	B-D	B3.140	83.28%	Volumetric examination limited due to the nozzle OD to the vessel thickness
B03.140.008	Steam Generator 1D Outlet (IR)	B-D	B3.140	83.28%	Volumetric examination limited due to the nozzle OD to the vessel thickness

Licensee's Basis for Requesting Relief (as stated):
"During the ultrasonic examination of the Steam Generator 1B and 1D Inlet and Outlet Nozzle Inside Radius Sections, 1SGB-Inlet, 1SGB-Outlet, 1SGD-Inlet and 1SGD-Outlet (Item Nos. B03.140.003, B03.140.004, B03.140.007 and B03.140.008 respectively) shown in Attachments 2, 3, 4 and 5, greater than 90% coverage of the required examination volume could not be obtained. Limitations were caused by the ratio of the nozzle OD to the vessel thickness, which limits the ultrasonic coverage to the 83.28% of the required examination volume. When the nozzle OD is small in relation to the vessel thickness, more coverage can be obtained when scanning from the vessel side.

"Nozzle inner radius sections were examined with the ultrasonic method to the maximum extent practical from the vessel wall. Calibration blocks and procedures were in accordance with ASME Section V, Article 4.

"Although the examination volume requirements as defined in ASME Section XI 1989 Edition with no addenda, Figure IWB-2500-7(b), Examination Volume M-N-O-P for ID Numbers 1SGB-Inlet, 1SGB-Outlet, 1SGD- Inlet and 1SGD-Outlet (Item Nos. B03.140.003, B03.140.004, B03.140.007 and B03.140.008 respectively) could not be met, the amount of coverage obtained for these examinations provides an acceptable level of quality and integrity.

"The nozzle inner radius....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 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 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.

"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 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):
None

Evaluation: The Code requires 100% volumetric examination of the subject nozzle inner radius sections. However, complete examination of these areas is limited by nozzle configuration (i.e. nozzle bore size and vessel wall thickness). As supported by figures attached to the licensee's submittal, these restrictions limit access and make the Code examination requirements impractical for the subject inner radius sections. To meet the Code requirements, design modifications would be necessary to provide access for examination. Imposition of the Code requirements would result in an undue hardship on the licensee.

The licensee has examined a significant portion (83.28%) of each of the subject nozzle examination areas. As a result, any existing patterns of degradation would have been detected and reasonable assurance of the continued structural integrity has been provided. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.2 Request for Relief Serial No. 99-02, Part B, Examination Category B-F, Items B5.70, Dissimilar Metal Welds

Code Requirement: Examination Category B-F, Item B5.70 requires 100% volumetric and surface examination, as defined by Figure IWB-2500-8, for steam generator nozzle-to-safe end dissimilar metal welds 4-inch NPS or larger.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the

licensee requested relief from the Code requirements for the welds listed below

Comp. ID	Component Description	ASME Category	Item Number	Aggregate Coverage	Limitation
B05.070.003	Steam Generator 1B Inlet (IR)	B-F	B5.70	75%	Volumetric examination limited due material characteristics and single side access
B05.070.004	Steam Generator 1B Outlet(IR)	B-F	B5.70	75%	Volumetric examination limited due material characteristics and single side access
B05.070.007	Steam Generator 1D Inlet (IR)	B-F	B5.70	75%	Volumetric examination limited due material characteristics and single side access
B05.070.008	Steam Generator 1D Outlet (IR)	B-F	B5.70	75%	Volumetric examination limited due material characteristics and single side access

Licensee's Basis for Requesting Relief (as stated):
"During the ultrasonic examination of the 1SGB and 1SGD Inlet and Outlet Nozzle-to-Safe-End Welds, 1SGB-INLET-W5SE, 1SGB-OUT-W6SE, 1SGD-INLET-W5SE and 1SGD-OUT-W6SE (Item Nos. B05.070.003, B05.070.004, B05.070.007, B05.070.008 respectively) shown in Attachment[1] 6, 7, 8, and 9 respectively, 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 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.

"As shown in the graph on page 6 of 16, the mode conversion process creates two sound beams of differing intensities reflecting off the inside surface. At incident angles greater than 30° 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 redesigned to allow scanning from both sides."

Licensee's Proposed Alternative Examination (as stated):

1 Figures and attachments not included with this report.

None

Evaluation: The Code requires 100% surface and volumetric examination for dissimilar metal safe end welds. However, complete examination of the subject welds was limited by component geometry (one-sided access) and material properties. As supported by figures attached to the licensee's submittal, these restrictions limit access and make the Code volumetric coverage requirements impractical for the subject dissimilar metal welds. To meet the Code coverage requirements, design modifications would be necessary to provide access for examination. Imposition of the Code requirements would result in an undue hardship on the licensee.

The licensee has examined a significant portion (75%) of each of the subject dissimilar metal welds, in addition to the complete surface examination. As a result, any existing patterns of degradation would have been detected and reasonable assurance of the continued structural integrity has been provided. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.3 Request for Relief Serial No. 99-02, Part C, Examination Category B-J, Items B9.11, and B9.31, Pressure Retaining Welds In Piping

Code Requirement: Examination Category B-J, Items B9.11 and B9.31 require 100% surface and volumetric examination of circumferential and branch connection welds in pressure-retaining piping NPS 4 or larger, as defined by Figure IWB-2500-8, -9, -10, and -11 each inspection interval.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee has requested relief from performing the volumetric examination to the extent required by the Code for the welds identified in the following table.

Comp. ID	Component Description	ASME Cat.	Item #	Aggregate Coverage	Limitation
1NC22-02	Steam Generator 1B Inlet Nozzle Elbow-to-Nozzle Weld	B-J	B9.11	65.08%	Volumetric examination limited due to cast austenitic material characteristics and single sided access
1NC22-03	Steam Generator 1B Outlet Nozzle Elbow-to-Nozzle Weld	B-J	B9.11	53.60%	Volumetric examination limited due to cast austenitic material characteristics and single sided access
1NC23-02	Steam Generator 1D Pipe-to-Safe-End Weld	B-J	B9.11	28.19%	Volumetric examination limited due to cast austenitic material characteristics and single sided access
1NC23-03	Steam Generator 1D Pipe-to-Safe-End Weld	B-J	B9.11	24.55%	Volumetric examination limited due to cast austenitic material characteristics and single sided access
1NC22-WN7	NC System Branch-to-Pipe Weld	B-J	B9.31	22.87%	Volumetric examination limited due to cast austenitic material characteristics and single sided access

Comp. ID	Component Description	ASME Cat.	Item #	Aggregate Coverage	Limitation
1NC24-WN9	NC System Branch-to-Pipe Weld	B-J	B9.31	29.64%	Volumetric examination limited due to cast austenitic material characteristics and single sided access

Licensee's Basis for Requesting Relief (as stated):
"During the ultrasonic examination of the Class 1 Piping: Elbow-to-Nozzle Weld 1NC22-02, Elbow-to-Nozzle Weld 1NC22-03, Pipe-to-Safe-End Weld 1NC23-02, Pipe-to-Safe-End Weld 1NC23-03, Branch Connection Weld 1NC-WN7 and Branch Connection Weld 1NC-WN9 (Item Nos. B09.011.049, B09.011.050, B09.011.051, B09.011.052, B09.031.001 and B09.031.003 respectively) shown in Attachments[2] 10, 11, 12, 13, 14 and 15 respectively, greater than 90% coverage of the required examination volume could not be obtained. Cast austenitic material characteristics and single-sided access caused by the component geometry prevents two-beam path direction coverage of the examination volume and limits the examination coverage to the percentages shown on Attachment 1. In order to obtain the required two-beam path direction coverage, welds would have to be re-designed to allow scanning from both sides.

"Ultrasonic techniques used for cast stainless steel are the same as those for dissimilar metal welds.

"Although the examination volume requirements as defined in ASME Section XI, Appendix III, Paragraph III-4420, 1989 Edition with no addenda, for ID Numbers 1NC22-02, 1NC22-03, 1NC23-02, 1NC23-03, 1NC22-WN7 and 1NC24-WN9 (Item Nos. B09.011.049, B09.011.050, B09.011.051, B09.011.052, B09.031.001 and B09.031.003 respectively) shown in Attachments 10, 11, 12, 13, 14 and 15 respectively could not be met, the amount of coverage obtained provides an acceptable level of quality and integrity.

"The class 1 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 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.

2 Figures and attachments not included with this report.

"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 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.

"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 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."

<u>Licensee's Proposed Alternative Examination</u> (as stated):
None

<u>Evaluation</u>: Examination Category B-J, Items B9.11 and B9.31 require 100% surface and volumetric examination of circumferential and branch pipe connection welds NPS 4 or larger in Class 1 pressure-retaining piping. However, complete examination of the subject welds was limited by component geometry (one-sided access) and material properties (attenuative austenitic grain structure). As supported by figures attached to the licensee's submittal, these restrictions limit access and make the Code examination requirements impractical for the subject Item B9.11 and B9.31 welds. To complete the

examinations to the extent required by the Code, the licensee would have to redesign and modify the subject piping. Imposition of the Code requirements would result in a considerable burden on the licensee.

For the subject Item B9.11 and B9.31 welds, the licensee has performed 22.87%-65.08% of the Code-required volumetric examinations and 100% of the surface examinations. In addition, other B-J Category welds have been examined to Code requirements. Based upon the limited portion of the welds examined volumetrically, the 100% surface examination coverage, and the completion of other welds in this Code Category, it is concluded that significant patterns of degradation would have been detected, and reasonable assurance of the structural integrity of these welds is provided.

Based on the impracticality of meeting the Code requirements for the subject examination areas, and the reasonable assurance provided by the examinations that were completed, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i) for the subject welds.

2.4 Request for Relief Serial No. 99-02, Part D, Examination Category C-A, Item C1.20, Pressure Retaining Weld in Volume Control Tank

Code Requirement: Examination Category C-A, Item Number C1.20 requires a volumetric examination of the Volume Control Tank head-to-shell weld. The examination coverage shall include essentially 100% of the weld length.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(g)(6)(i), the licensee requested relief from the volumetric examination, to the extent required by the Code, of Volume Control Tank bottom head-to-shell Weld 1VCT-LH-SH.

Licensee's Basis for Requesting Relief (as stated):
"During the ultrasonic examination of the Volume Control Tank Lower Head-to-Shell Weld 1VCT-LH-SH (Item Number C01.020.016) shown in Attachment[3] 16, greater than 90% coverage of the required examination volume could not be obtained. Coverage was limited to 88.34% of the required examination volume because of the proximity of four support legs. In order to achieve greater than 90% coverage, more access would have to be provided by moving the support legs.

"Although the examination volume requirements as defined in ASME Section XI, Appendix III, Paragraph III-4420, 1989 Edition with no addenda, for ID number 1VCT-LH-SH (Item Number C01.020.016) could not be met, the amount of coverage obtained provides an acceptable level of quality and integrity.

"The Volume Control Tank (VCT) is used in power operations. The VCT is located in the Auxiliary Building adjacent 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 welds in question (lower head to shell weld), there

3 Figures and attachments not included with this report.

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 welds, 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):
None

<u>Evaluation</u>: The Code requires 100% volumetric examination of the subject bottom head-to-shell weld. Complete examination coverage was not possible due to four component supports attached near the 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 this requirement would create a considerable burden on the licensee.

The licensee has completed a significant portion of the Code-required volumetric examination (88.34%). Consequently, the INEEL staff concludes 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).

2.5 Request for Relief Serial No. 99-02, Part E, Examination Category C-F-1, Item C5.11, Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping

Code Requirement: Examination Category C-F-1, Item C5.11 requires 100% surface and volumetric examination, as defined by Figure IWC-2500-7, for circumferential welds in piping ≥3/8 in. nominal wall thickness and >NPS 4.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required volumetric examination of feedwater nozzle transition ring-to-elbow Weld 1CA66-35.

Licensee's Basis for Requesting Relief (as stated):
"During the ultrasonic examination of the Class 2 Steam Generator 1A Auxiliary Feedwater Elbow-to-Nozzle Weld 1CA66-35 (Item Number C05.011.012) shown in Attachment 17, 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 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 calculations[4] below show that a 45° refracted longitudinal wave striking the inside surface of a pipe will produce a 22.9° refracted shear wave in addition to the normally expected 45° reflected longitudinal wave.

"As shown in the graph on page 6 of 16, the mode conversion process creates two sound beams of differing intensities reflecting off the inside surface. At incident angles greater than 30° 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 direction coverage requirement is impractical.

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

"The use of radiography as an alternate volumetric examination method for is not practical due to component thickness and geometric configurations. Other

4 Figures, calculations, and attachments not included with this report.

restrictions making radiography impractical are the necessity to use double wall techniques due to inaccessibility of the ID surface and physical barriers prohibiting access for placement of source, film, number bands, etc.

“If a leak were to occur at the weld in question (Steam Generator 1D Auxiliary Feedwater (CA) Nozzle to Transition Piece, Weld 1SGD-W261), there are methods by which the leak could be identified for prompt Engineering evaluation.

“The CA nozzles are at final feedwater pressure and temperature. In Modes 1 (Power</= approximately 17%), 2, and 3, all feedwater is provided to the steam generators via the CA nozzle (CF-to-CA Bypass Flow). In Mode 1 (Power >/= approximately 17% power), a small amount of feedwater (Tempering Flow) is directed to the CA nozzle of the steam generators to keep the nozzles at final feedwater temperature to reduce the thermal shock to the nozzles associated with CA system operation and transfer of feedwater flow to the nozzles during unit shutdowns. A leak at the CA nozzle would result in the following:

- Increased S/G enclosure temperature. This parameter is monitored periodically by the Containment Ventilation System (VV) Engineer per the associated ‘Engineering Support Program’.

- Increased input into the Ventilation Unit Condensate Drain Tank (VUCDT). This parameter is monitored continuously by Operations via an OAC alarm and also periodically by the Liquid Radwaste System (WL) Engineer and Reactor Coolant System (NC) Engineer per the associated ‘Engineering Support Program’.

- Note: The above parameters would be used to identify a leak in the steam generator enclosure, but could not specifically identify the CA nozzle as the source of leakage. A containment entry would be required to identify the exact source of the leakage.

- Also, a containment walkdown is performed when the unit reaches Mode 3 (full temperature / pressure) during the unit shutdown for each refueling outage. This walkdown should identify any leak at the weld in question.

“Concerning the consequences of a leak at the CA nozzle (affects on CA system operation): Any leakage would result in a portion of the CA flow bypassing the

steam generator, and therefore being unavailable to maintain steam generator levels. Very small leaks (<1 gpm) would have no discernible effect on CA system operation. Leaks that approach 5 gpm would need to be evaluated for system operability effects.”

<u>Licensee’s Proposed Alternative Examination</u> (as stated):
None

<u>Evaluation</u>: Examination Category C-F-1, Item C5.11 requires 100% surface and volumetric examination of the subject circumferential weld. However, complete

examination of this weld was limited by component geometry (one-sided access) and material properties (attenuative austenitic grain structure). As supported by figures attached to the licensee's submittal, these restrictions limit access and make the Code examination requirements impractical for the subject weld. To complete the examinations to the extent required by the Code, design modifications would be necessary. Imposition of the Code requirements would result in a considerable burden on the licensee.

The licensee has completed a significant portion (75%) of the Code-required volumetric examination and 100% of the surface examination. Based upon the portion of the weld examined volumetrically, and the 100% surface examination coverage, it is concluded that significant patterns of degradation would have been detected, and reasonable assurance of the structural integrity of the circumferential weld is provided.

Considering the impracticality of meeting the Code requirements for the subject examination area, and the reasonable assurance provided by the examination that was completed, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i) for the subject weld.

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 1.

The INEEL staff has reviewed the licensee’s submittals and concludes that the Code requirements are impractical to meet for the issues contained in Relief Request Serial No. 99-02 Part A-E. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

Catawba Nuclear Station

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PD II-1 DOCUMENT COVER PAGE

DOCUMENT NAME: C:\CRLF6316.wpd

ORIGINATOR: C. PATEL

SECRETARY: CHERI NAGEL

SUBJECT: SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN REQUEST FOR RELIEF NO. 99-02 (PARTS A, B, C, D, AND E) FOR CATAWBA NUCLEAR STATION, UNIT 1 (TAC NO. MA6316)

DATE: July 12, 2000

*******ROUTING LIST*******

NAME	DATE
1. C. PATEL	/ /00
2. C. HAWES	/ /00
3. R. EMCH	/ /00
4. _____	/ /00
5. _____	/ /00
6. _____	/ /00

7. **Secretary - dispatch - 08-H-12**

❒ return incoming/background to licensing assistant/project manager