July 19, 2006

Mr. Dhiaa Jamil
Vice President Catawba Nuclear Station
Duke Power Company LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNIT 1, REQUEST FOR RELIEF 05-CN-001, LIMITED WELD EXAMINATIONS DURING END-OF-CYCLE 14 REFUELING OUTAGE (TAC NOS. MC6268, MC9211, MC9212, MC9213, MC9214, MC9215, and MC9216)

Dear Mr. Jamil:

By letter dated February 17, 2005, as supplemented November 28, 2005, Duke Power Company LLC, the licensee, submitted a request for relief, Relief Request No. 05-CN-001, from the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code* (Code), Section XI, 1989 edition requirement pertaining to limited weld examination coverage between the end of operating cycle 13 and the end of operating cycle 14 during the second 10-year inservice inspection (ISI) interval at Catawba Nuclear Station, Unit 1 (Catawba 1). The licensee already performed the scheduled second 10-year interval on the referenced welds and components resulting in limited volumetric and visual coverages. As a result, the licensee has proposed that no alternate examinations or testing will be performed during the end of operating cycle 14 to compensate for the limited ultrasonic examination coverage.

The enclosed Safety Evaluation contains the Nuclear Regulatory Commission (NRC) staff's evaluation and conclusions. Based on the information provided in the relief request, the NRC staff has determined that it is impractical for the welds identified to be examined to the extent required by the ASME Code at Catawba 1 for items 3 through 7 of Table 1 (attached to the Safety Evaluation) concerning reactor pressure vessel supports and containment spray system (NS) valve-to-pipe weld. It is further concluded that reasonable assurance of structural integrity is provided by the examinations that were, and will be, performed by the licensee for items 3 through 7.

Therefore, relief is granted and alternatives are imposed pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(g)(6)(i) for the second 10-year ISI interval at Catawba 1 for items 3 through 7 of Table 1. The relief granted and alternative examinations imposed for items 3 through 7 of Table 1 are authorized by law and will not endanger life, property, or the common defense and security, and are 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. All other requirements of ASME Code, Section XI, for which relief has not been specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

However, for items 1 and 2 of Table 1 concerning the residual heat removal (RHR) heat exchanger nozzle-to-shell welds, the NRC staff has determined that the licensee has failed to meet ASME Code examination requirements, irrespective of the limited accessibility caused by the RHR heat exchanger nozzle-to-shell weld geometries. Therefore, the relief is denied for items 1 and 2 of Table 1 concerning the RHR heat exchanger nozzle-to-shell welds.

Sincerely,

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-413

Enclosure:
Safety Evaluation

cc w/encl: See next page

However, for items 1 and 2 of Table 1 concerning the residual heat removal (RHR) heat exchanger nozzle-to-shell welds, the NRC staff has determined that the licensee has failed to meet ASME Code examination requirements, irrespective of the limited accessibility caused by the RHR heat exchanger nozzle-to-shell weld geometries. Therefore, the relief is denied for items 1 and 2 of Table 1 concerning the RHR heat exchanger nozzle-to-shell welds.

Sincerely,

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-413

Enclosure:
Safety Evaluation

cc w/encl: See next page

<u>DISTRIBUTION:</u>

Public	LPL2-1 R/F	RidsNrrPMSLingam (hard copy)
RidsNrrPMJStang (hard copy)	TMcLellan, NRR	RidsAcrsAcnwMailCenter
RidsRgn2MailCenter (MErnstes)	RidsOgcRp	RidsNrrCvib(MMitchell)
RidsNrrLAMO'Brien (hard copy)	BSingal, DORL DPR	RidsNrrDorlLpc(EMarinos)
S Lee, EDO RGN II	T Chan, NRR	EReichelt, NRR

ADAMS Accession No: ML061680010 * SE input dated NRR-028

OFFICE	LPL2-1/PM	LPL2-1/LA	NRR/CVIB/BC	NRR/CPNB/BC	OGC	LPL2-1/BC
NAME	JStang	MO'Brien	MMitchell*	TChan*		EMarinos
DATE	/ /06	/ /06	02/17/06	03/08/06	/ /06	/ /06

OFFICIAL AGENCY RECORD

cc:

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

Ms. Lisa F. Vaughn
Duke Energy Corporation
Mail Code - PB05E
422 South Church Street
P.O. Box 1244
Charlotte, North Carolina 28201-1244

Ms. Anne Cottingham, Esquire
Winston and Strawn LLP
1700 L Street, NW
Washington, DC 20006

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

NCEM REP Program Manager
4713 Mail Service Center
Raleigh, North Carolina 27699-4713

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

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

Mr. Henry Porter, Assistant Director
Division of Waste Management
Bureau of Land and Waste Management
Dept. of Health and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201-1708

Mr. R.L. Gill, Jr., Manager
Nuclear Regulatory Issues
and Industry Affairs
Duke Energy Corporation
526 South Church Street
Mail Stop EC05P
Charlotte, North Carolina 28202

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

Mr. Peter R. Harden, IV, Vice President
Customer Relations and Sales
Westinghouse Electric Company
6000 Fairview Road
12th Floor
Charlotte, North Carolina 28210

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

cc:

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

Mr. Henry Barron
Group Vice President, Nuclear Generation
and Chief Nuclear Officer
P.O. Box 1006-EC07H
Charlotte, NC 28201-1006

Diane Curran
Harmon, Curran, Spielberg &
Eisenberg, LLP
1726 M Street, NW
Suite 600
Washington, DC 20036

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF NO. 05-CN-001

CATAWBA NUCLEAR STATION, UNIT 1

DUKE POWER COMPANY LLC

DOCKET NO. 50-413

1.0 INTRODUCTION

By letter dated February 17, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML050610019), Duke Power Company LLC, the licensee, submitted Request for Relief 05-CN-001 from requirements of the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code* (Code), Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*. In response to a Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI), the licensee provided further information in a letter dated November 28, 2005 (ADAMS Accession No. ML060410118). This request was submitted as part of the inservice inspection (ISI) program for the second 10-year ISI interval at Catawba Nuclear Station, Unit 1 (Catawba 1).

The NRC staff, with technical assistance from its contractor, the Pacific Northwest National Laboratory (PNNL), has reviewed and evaluated the information provided by the licensee. Table 1, attached to this safety evaluation, summarizes the disposition of relief requests for items 1 through 7. Relief Request for items 1 and 2 pertains to a reduced examination coverage of the residual heat removal (RHR) heat exchanger nozzle-to-shell welds. Relief Request for items 3 through 6 pertains to a reduced examination of the surface of the reactor pressure vessel supports. Relief Request for item 7 pertains to a reduced examination of a containment spray system (NS) valve 1NS1B-to-pipe circumferential weld.

2.0 REGULATORY REQUIREMENTS

Inservice inspection (ISI) of the ASME 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), Part 50, Section 50.55a(g), except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). Section 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 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) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The ASME Code of Record for the Catawba 1 second 10-year ISI program, which began on June 29, 1995, is the 1989 edition of Section XI of the ASME Code, with no addenda. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

3.0 TECHNICAL EVALUATION

The information provided by Duke Power Company LLC in support of the request for relief from ASME Code requirements has been evaluated and the basis for disposition is documented below. For clarity, the request has been evaluated in several parts according to ASME Code Examination Category.

3.1 Request for Relief 05-CN-001 (TAC Nos. MC6268 and MC9211), Examination Category C-B, Item No. C2.21, Pressure Retaining Nozzle Welds in Vessels, Residual Heat Removal Heat Exchanger (items 1 and 2)

3.1.1 ASME Code Requirement

Examination Category C-B, Item C 2.21, requires 100-percent volumetric and surface examinations, as defined by Figures IWC-2500-4(a) or (b), of the length of Class 2 nozzle-to-shell (or -head) full penetration welds. ASME Code Case – 460, *Alternative Examination Coverage for Class 1 and Class 2 Welds*, as an alternative approved for use by the NRC in Regulatory Guide (RG) 1.147, Revision 14, *Inservice Inspection Code Case Acceptability* (RG 1.147), states that a reduction in examination coverage due to part geometry or interference for any Class 1 and 2 weld is acceptable provided that the reduction is less than 10-percent, i.e., greater than 90-percent examination coverage is obtained.

3.1.2 Licensee's ASME Code Relief Request

In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from examining 100 percent of the ASME Code-required inspection volume(s) shown in Figures IWC-2500-4(a) or (b), as applicable, for inlet and outlet nozzle-to-shell welds 1ARHRHX-5-A and 1ARHRHX-5-B on the RHR system heat exchanger 1A.

3.1.3 Licensee's Basis for Relief Request

During the ultrasonic examination of the RHR or (ND System) heat exchanger nozzle-to-shell welds 1ARHRHX-5-A and 1ARHRHX-5-B, greater than 90-percent coverage of the required examination volumes was not achieved. The examination coverages were limited to

14.25-percent because of the nozzle-to-shell weld design shown in Attachments 1 through 5[1] within Attachments "A" and "B" of this relief request. The weld joins a 13⅞-inch outside diameter nozzle to a vessel shell. The weld length is approximately 50 inches. Both materials are SA-240 F304 austenitic stainless steel. The nozzle configuration is a "set-on" type, which is not shown in ASME Section XI, Figure IWC-2500-4.

he percentage of coverage reported represents the aggregate coverage of all scans performed on the weld. Scans parallel to the weld axis using 45E shear waves covered 25 percent of the required examination volume in two opposite directions. Scans perpendicular to the weld axis using 70E refracted longitudinal waves covered 7 percent of the required volume from the nozzle side of the weld only. Weld joint geometry does not permit examination of the required volume from the vessel head side nor does it allow circumferential scanning for more than 25 percent of the weld length. In order to obtain more coverage, the weld would have to be re-designed, which is impractical.

3.1.4 Licensee's Proposed Alternative Examination

The scheduled 10-year Code examination was performed on the referenced areas/welds and resulted in the noted limited coverage of the required ultrasonic volume and visual coverages. No alternate examinations or testing is planned for the areas/welds during the current inspection interval.

3.1.5 Response to Request for Additional Information

The statement in Section VIII, Paragraph F of the February 17, 2005, submittal, "weld joins two carbon steel components using compatible weld materials", was in error. Welds 1ARHRHX-5-A and 1ARHRHX-5-B are nozzle (SA240 Type 304)-to-shell (SA240 Type 304) welds on the residual heat removal 1A heat exchanger. The welds join two austenitic base materials at an offset juncture of the 42-1/4" ID shell to 14" inlet and outlet nozzles. These materials are 18 Cr-8Ni stainless steels and a) have a high corrosion resistance with low contribution of corrosion products to the coolant, b) have good mechanical properties, and c) are highly weldable. Very few service induced problems with stainless steel in a pressurized water reactor (PWR) primary system applications have been observed in operating plants. There has been limited susceptibility to stress corrosion cracking (SCC) due to chloride contamination and cracking in stagnant borated systems. However, chemistry limits on chlorides, fluorides, sulfides, and dissolved oxygen are controlled by Selected Licensee Commitments (SLC) and other administrative procedures at Catawba 1 to ensure that any favorable conditions for SCC are precluded. Additionally, controls on welding filler material consistent with Regulatory Guide 1.31 also have served to limit the susceptibility of these welds to SCC. These lines are flushed quarterly during periodic testing of the RHR train during normal operation; thus, the concern with SCC of stagnant borated systems is not significant. No other known degradation mechanisms are applicable to this material at this particular location within the system.

1 Drawings, sketches and inspection data sheets in the licensee's submittal are not included in this report.

A liquid penetrant examination was performed on the subject welds. 100-percent coverage of the welds was obtained. No recordable indications were found. These welds had no limitations.

The examinations were performed only from the nozzle side of the weld for the following reasons:

- The base material thickness on the vessel shell side of the weld is 0.9 inch. The base material thickness on the nozzle side of the weld is 0.375 inch. The basic calibration block listed in the outage plan and required by Appendix III was suitable only for examination from the nozzle side of the weld. A calibration block suitable for the vessel shell side was not available. The differences in material thickness and the weld joint geometry were unknown until the start of the outage. A calibration block of the appropriate thickness could not be obtained within the period for performing the examinations.

- Given the geometric conditions on the vessel shell side of the weld, standard manual examination techniques would have achieved limited additional coverage in the axial direction only in the 90° and 270° quadrants. These examinations were performed in November 2003. The licensee purchased Phased Array ultrasonic equipment in 2005, which will enhance the capability for achieving greater coverage of this weld along the entire length in the axial direction during future examinations.

- These welds were not previously examined either for pre-service or prior inservice inspections and there were no existing ultrasonic examination records, because the 1974 Edition, with Summer 1975 Addenda for the pre-service inspection and the 1980 Addenda for the first inspection interval of ASME Section XI for Catawba 1 did not require a volumetric examination for these welds.

3.1.6 Evaluation

The ASME Code requires that 100-percent volumetric and surface examinations of the entire length of nozzle-to-shell welds be examined for selected Class 2 pressure retaining vessels. The licensee has requested relief from examining 100-percent of the ASME Code-required volume, as shown in Figures IWC 2550-4(a) or (b), for two nozzle-to-shell welds on the RHR system Heat Exchanger 1A, based on the design geometry.

The inlet and outlet nozzles on the subject RHR heat exchanger consist of a “set-on” design, which is not specifically depicted in Figures IWC-2500-4(a) or (b). This type of configuration is common for certain vessels, and the ASME Code provides a clearer depiction of a similar nozzle design in the requirements for Class 1 nozzle-to-vessel welds, i.e., Figure IWB-2500-7(c). The 14-inch diameter RHR heat exchanger inlet and outlet nozzles are joined to the 42-inch diameter cylindrical shell in a manner that produces a varied cross-sectional profile along the length of these full penetration nozzle-to-shell welds. This configuration causes significant challenges when attempting to volumetrically examine the welds, as is shown by the relatively poor (approximately 14 percent) coverage obtained by the licensee.

The licensee ultrasonically examined this weld from the nozzle side only. However, the ASME Code requires that examinations be performed from both the nozzle and shell sides of full

penetration welds in this RHR heat exchanger, when accessible. When asked in an NRC RAI about examining accessible portions of these welds from the shell side, the licensee stated that no calibration block was available, and that the weld joint geometry and varied thicknesses of the shell and nozzle were unknown prior to the start of the last refueling outage in the interval. The licensee claimed that these unknown parameters did not allow a calibration block for the 0.9-inch thick shell side to be obtained within the outage period. The nozzle base material is approximately 0.375 inch in thickness. This lack of knowledge pertaining to the RHR shell and nozzles, and the existence of appropriate calibration blocks, neither justifies or provides a reasonable basis for not meeting the ASME Code requirement to examine the nozzle-to-shell welds from both accessible sides. These examinations occurred at the end of the subject interval, therefore, the licensee had nearly 10 years to determine what preparations would be necessary for the examinations required by the ASME Code, and these component examinations are required to be listed in the Catawba 1 ISI Program Plan, which is due at the beginning of the inspection interval.

Additionally, the licensee appears to have mis-interpreted the ASME Code volumetric coverage requirement. The volumes depicted in Figures IWC-2500-4(a) and (b), albeit the design shown does not exactly match the Catawba 1 RHR heat exchanger nozzle-to-vessel welds, exhibits the volume to be inspected to include the weld and parent material for 1/4-inch beyond the fusion zone, extending from the inside surface to 1/3*t*, where *t* is the thickness of the *shell* base material. This is essentially the lower one-third of the weld and base material. Since the shell is 0.9 inch in thickness, the upper boundary of the ASME Code volume should be 0.3 inches from the inside surface of the weld/parent materials. Cross-sectional coverage sketches[2] submitted by the licensee indicate the top boundary of the inspection volume was 0.125 inches from the inside surface of the heat exchanger instead of 0.3 inches, as required by ASME Code. Apparently, the licensee incorrectly applied the 1/3*t* requirement to the *nozzle* base material (0.375-inch thickness), resulting in this 0.125-inch upper boundary. Therefore, the lower one-third of the weld and accompanying base material would not have been thoroughly examined, even if the geometry of the nozzle-to-shell welds would have allowed complete accessibility.

It is concluded that the licensee has failed to meet ASME Code examination requirements, irrespective of the limited accessibility caused by the RHR heat exchanger nozzle-to-shell weld geometries. Therefore, items 1 and 2 of Request for Relief 05-CN-001, concerning the RHR heat exchanger nozzle-to-shell welds, the requested relief is denied.

3.2 Request for Relief 05-CN-001 (TAC Nos. MC9212, MC9213, MC9214, MC9215), Examination Category F-A, Item No. F1.40, Supports, Reactor Pressure Vessel Nozzle Supports (items 3 through 6)

3.2.1 ASME Code Requirement

For components other than piping, Examination Category F-A, Item 1.40, requires that a 100-percent visual VT-3 examination of structural supports be performed in accordance with Figure IWF-1300-1 during each inspection interval.

---

2 Licensee submitted sketches of the weld cross-sections and volumetric coverages are not included in this report.

3.2.2 Licensee's ASME Code Relief Request

In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from performing 100-percent visual VT-3 inspections on the examination boundary described in Figure IWF-1300-1 for reactor pressure vessel (RPV) supports located at the inlet and outlet nozzles. The supports have been designated by the licensee as 1-RPV-SUPPORTS A, B, C, AND D.

3.2.3 Licensee's Basis for Relief Request

During the visual examination of reactor vessel supports I-RPV-SUPPORT A, 1-RPV-SUPPORT B, 1-RPV-SUPPORT C and 1-RPV-SUPPORT D complete examination of the supports surfaces could not be achieved. The interface between the concrete structure and the reactor vessel precludes access for a visual inspection of the reactor vessel side of the support. An area 71 inches long and 36 inches high on each support was not examined because of the limited clearance between the support and the reactor vessel shell insulation. The reactor vessel supports rest on the concrete shield wall and are flush with the edge of the shield wall. The support extends approximately 23" down from the bottom of the nozzle forging to the concrete primary shield wall below. On each side of the nozzle forging, the support extends approximately 10" to the concrete primary shield wall enclosing the vessel. The gap between the shield wall and the reactor vessel insulation is only 2 inches. Consequently, there is insufficient access for cameras or mirrors to examine this area. In order to examine the area the insulation would have to be removed from the reactor vessel.

The requested relief is associated with the four reactor vessel supports (ID. Nos. RPV-SUPPORT A, RPV-SUPPORT B, RPV-SUPPORT C and RPV-SUPPORT D). These supports are located on alternating nozzles around the reactor vessel. Two supports are associated with the hot leg nozzles and 2 supports with the cold leg nozzles. Each support is located immediately adjacent to the reactor vessel and provides restraint in the lateral and vertical directions through an integral nozzle forging attached to the bottom of the piping. Under normal operating conditions, deadweight, pressure and thermal loads are transmitted from the reactor coolant loop components to the building structure. Transmitted forces resulting from design loads produce generally compressive stresses in the support.

The interface between the concrete structure and the reactor vessel precludes access for a visual inspection of the reactor vessel side of the support. As a result, the inspection was limited to the outboard side of the support. The justification for this limited inspection is that the supports are made of carbon steel (SA516, Grade 70, SA36) with a partial coating to protect the external surfaces from corrosion. The only significant degradation mechanism for the supports is corrosion. Other potential failure modes including thermal fatigue, high cycle fatigue due to vibration, stress corrosion cracking, creep, galvanic corrosion and erosion/corrosion may be eliminated from consideration based on material selection, operating stresses, environmental factors and operating experience. As for corrosion, there was no evidence of significant corrosion during the limited visual inspection. There was some very minor degradation of the coating at isolated locations, yet there was no wastage of the underlying steel. For these locations, the areas surrounding each support are normally dry. The only period when the support steel may be subjected to moisture is during refueling operations where leakage past the refueling cavity seals/nozzle inspection port seals may occur. During these periods, water temperature is generally low (approximately 100 EF) and boron concentration is relatively high. Even under these conditions, the corrosion rate of

uncoated carbon steel material subjected to water with a boron concentration of 2500 ppm and a temperature of 100 EF is approximately 0.007 in/year[3]. This small corrosion rate will not significantly affect the structural integrity of the supports. Based on these reasons, the limited inspection of the reactor vessel supports is justified.

3.2.4 Licensee's Proposed Alternative Examination

The scheduled 10-year code examination was performed on the referenced areas/welds and resulted in the noted limited coverage of the required ultrasonic volume and visual coverages. No alternate examinations or testing is planned for the areas/welds during the current inspection interval.

3.2.5 Response to Request for Additional Information

There were no significant deposits or accumulations of boron residues on support surfaces noted during the inspection. There were no obstructions due to leakage that precluded a visual inspection. There were some minor boron residue trails running down the walls of the nozzle inspection ports (sand boxes) but these thin translucent films did not affect visibility of support condition.

Approximately 50 percent of the support was examined. A pole camera was inserted in each of the four nozzle inspection ports (sand boxes) from above. The entire support was inspected from the outboard side. However, there is no access to the inboard side of the support.

There is no direct access to the vessel side of these supports with the refueling cavity seal removed. The drawing shows the general configuration and the very limited space (approximately 2") between the reactor vessel and the concrete building structure. The vessel insulation in the gap between the reactor vessel and the building structure limits the access for a borescope or other optical device through the refueling cavity seal gap. Although a small gap is present, there remain two horizontal offsets and the nozzle insulation that preclude an effective visual inspection of this side of the support. The vertical distance (8 to 10 feet below the flange) combined with the horizontal offsets and existing vessel and nozzle insulation prohibit an effective remote visual examination.

3.2.6 Evaluation

The ASME Code requires that a visual VT-3 examination of the reactor pressure vessel (RPV) component supports be performed during each inspection interval. The VT-3 examination must include welded and mechanical connections to the building structure and the RPV, clearances of guides and stops, accessible sliding surfaces, and all other surfaces of the structural support members. However, design and placement of the RPV supports at Catawba 1 precludes a visual VT-3 examination over the entire surfaces of the supports. In order to perform the ASME Code-required examinations, design and access modifications would need to be made to the biological shield wall, RPV insulation, and the subject supports. Therefore, it is impractical to

3 The licensee cited EPRI Report 1000975, Boric Acid Corrosion Guidebook, Revision 1, November 2001.

perform the ASME Code-required visual VT-3 examinations over the entire surfaces of the supports.

The RPV supports at Catawba 1 are located below four of the eight inlet/outlet RPV nozzles, i.e., these supports are on alternating cold leg and hot leg nozzles around the circumference of the RPV. The design includes a built-up region, that is an integral part of each nozzle forging, resting on the remaining structural steel plate elements of the support. The support structure is flush with, and rests on, the biological shield wall below each respective nozzle. These supports are accessible for visual VT-3 examination from the nozzle inspection ports (sand boxes), which only allow the outboard side of the supports to be examined. The inboard side of these supports cannot be accessed by direct or remote means due to the limited annular spaces and offset ledges between the outside surface of the reactor vessel, RPV insulation, and the biological shield wall.

The supports are made of carbon steel (SA516, Grade 70, and SA36), and the most likely type of degradation would be corrosion. For this reason, the supports are partially coated to resist corrosive attack. During the subject visual VT-3 examinations, no evidence of significant corrosion was noted on the accessible outboard surfaces of these supports. The licensee obtained approximately 50-percent surface coverage of each support by using a pole camera inserted in each of the sand boxes from above.

Based on the impracticality of examining the entire surfaces of the subject RPV support structures, the 50-percent coverage obtained, and the absence of any corrosion noted, it is reasonable to conclude that significant degradation, if existing, would have been detected by the examinations performed. Therefore, regarding the RPV supports, it is recommended that, pursuant to 10 CFR 50.55a(g)(6)(i), relief be granted for Items 3 through 6 of Request for Relief 05-CN-001.

3.3 Request for Relief 05-CN-001 (TAC No. MC9216), Examination Category C-F-1, Item No. C5.11, Containment Spray System (NS) Valve 1NS1B-to-Pipe Circumferential Weld (item 7)

3.3.1 Code Requirement

ASME Code, Section XI, 1989 edition, in examination category C-F-1 (Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping) requires essentially 100-percent volumetric examination of the above weld.

3.3.2 Code Requirement from Which Relief is Requested

Relief is requested from the requirement to examine essentially 100 percent of the required volume specified in the ASME Code, Section XI, 1989 edition, examination category C-F-1. Due to existing piping/valve geometry, interferences, and existing examination technology, the ultrasonic examination coverage did not meet the essentially 100-percent examination coverage requirements of the ASME Code.

3.3.3 Licensee's Basis for Relief

The licensee stated that during the ultrasonic examination of the Containment Spray weld 1NS6-25, greater than 90-percent coverage of the required examination volume from two axial and two circumferential directions was not achieved. Coverage was limited to 31.50 percent because of single sided access limitations. The licensee stated that they do not take credit for far-sided examination coverage when the sound beam must pass through austenitic weld metal. A best effort examination using 60E refracted longitudinal waves and 70E shear waves covered the weld metal while a 60E shear wave covered the near side base material. A 45E shear wave probe covered 100 percent of the pipe base material and ½ of the weld in two opposite directions. The slope of the transition and the weld width prevented coverage on the valve side. The aggregate coverage from all scans equals 31.50 percent of the required volume. The licensee stated that in order to achieve greater than 90-percent coverage in two axial and two circumferential directions, the valve would have to be re-designed to allow scanning from both sides of the weld, which is impractical. The licensee stated that there were no recordable indications found during the inspection of this weld.

3.3.4 Justification for Granting Relief

The licensee stated that Weld ID. No. 1NS6-25 is located on the containment spray pump side of valve 1NS1B, which is the NS pump 1B suction from containment sump isolation valve. The physical location in the plant is on the 522 elevation in room 103, which is the 1B NS pump room. During normal operations, this valve is closed and the pipe is filled with stagnant refueling water storage tank (FWST) water at relatively low pressure (FWST static head). The licensee stated that the only time flow would be forced through this pipe would be during a loss-of-coolant accident (LOCA) inside containment where the 1B NS pump would be required to take suction from the emergency core cooling system (ECCS) sump inside containment.

The licensee stated that given the location of this valve and the boron concentration of the water in the system, there are several in-place programs that would quickly identify a through-wall leak on this valve. The licensee stated that the valve is located in an area in the plant that is visited twice per day by operators when the auxiliary building rounds are performed. Since the water is highly borated, a small leak would result in noticeable boron accumulation in the immediate area. The licensee stated that if a significant leak were to develop, the leakage is routed to a floor drain in room 103 that is then routed to the ND/NS sump on the 522 elevation. Leakage into this sump is monitored by radwaste chemistry on an ongoing basis. The licensee concluded that a significant leak causing an increase in the sump pump-down frequency would be identified.

4.0 <u>NRC STAFF EVALUATION</u>

The NRC staff, with technical assistance from its contractor, PNNL, has reviewed and evaluated the information provided by the licensee in its letter dated February 17, 2005, which proposed its Second 10-Year Interval Inservice Inspection Program Plan Request for Relief 05-CN-001. In response to an NRC request for additional information, the licensee provided additional information in its letter dated November 25, 2005. The NRC staff adopts the evaluations and recommendations for granting or denying relief based on Sections 3.1 and 3.2 of this Safety Evaluation (SE) provided by PNNL. Attachment 1 (Table 1) to this SE lists each relief request and the status of approval.

For Request for Relief 05-CN-001, Items 1 and 2 (TAC Nos. MC6268, and MC9211), the ASME Code, Section XI, Table IWC-2500-1 Examination Category C-B, Item C2.21, Pressure Retaining Nozzle Welds in Vessels, Residual Heat Removal Heat Exchanger, requires 100% volumetric and surface examinations, as defined by Figures IWC-2500-4(a) or (b), of the length of Class 2 nozzle-to-shell (or -head) full penetration welds. The NRC staff has determined that the licensee has failed to meet ASME Code requirements relative to proper examination volumes and performance of the examinations from either side of the welds, irrespective of the limited accessibility caused by the Residual Heat Removal System (RHR) heat exchanger nozzle-to-shell weld geometries. Therefore, the NRC staff determined that for Items 1 and 2 of Request for Relief 05-CN-001, concerning the RHR heat exchanger nozzle-to-shell welds, relief is denied. For a detailed discussion of this denial, see Section 3.1 of this SE.

For Request for Relief 05-CN-001, Items 3, 4, 5, and 6 (TAC Nos. MC9212, MC9213, MC9214, and MC9215), the ASME Code, Section XI, Table IWF-2500-1, Examination Category F-A, Item F1.40, Supports, RPV Nozzle Supports requires that for components other than piping a 100-percent visual VT-3 examination of structural supports be performed in accordance with Figure IWF-1300-1 during each inspection interval. The NRC staff determined that the design and placement of the RPV supports at Catawba 1 precludes a visual VT-3 examination over the entire surfaces of the supports. In order for the licensee to perform the ASME Code-required examinations, design and access modifications would need to be made to the biological shield wall, RPV insulation, and the subject supports. Therefore, the NRC staff determined that it is impractical to perform the ASME Code-required visual VT-3 examinations over the entire surfaces of the supports and to require the licensee to perform the ASME Code-required examinations would be a significant burden.

The licensee obtained approximately 50-percent surface coverage of each support by using a pole camera inserted in each of the sand boxes from above. Furthermore, since the supports are made of carbon steel and the most likely type of degradation would be corrosion, the supports are partially coated to resist corrosive attack. During the subject VT-3 visual examinations, no evidence of significant corrosion was noted on the accessible outboard surfaces of these supports. Based on the above, the NRC staff determined that the 50-percent visual coverage obtained, and the absence of any corrosion found during the ASME Code-required inspections, provides reasonable assurance of structural integrity of the subject supports. The NRC staff further determined that it is reasonable to conclude that significant degradation, if existing, would have been detected by the examinations performed for items 3 through 6 of Request for Relief 05-CN-001.

For Request for Relief 05-CN-001, Item 7, the NRC staff has evaluated the information provided by the licensee in support of the volumetric examinations of the subject weld performed during the End of Cycle 14 Refueling Outage during the second 10-year ISI. For the subject weld, ultrasonic scanning from two axial and two circumferential directions could not achieve greater than 90-percent coverage. Ultrasonic examination could be performed from only one side of the weld due to the component configuration and geometry. The licensee's best effort examination with single-sided access achieved volumetric coverages of the weld equaled 31.50 percent. Additionally, in response to a RAI dated November 28, 2005, the licensee stated that a surface examination was performed on the welds and the results were acceptable. The licensee also stated that the ultrasonic examination of the valve-to-pipe weld was performed using personnel, procedures, and equipment qualified in accordance with ASME Code, Section XI, Appendix VIII, Supplement 2, 1995 edition through the 1996 addenda.

The NRC staff has determined that the examination coverage of the subject weld was reduced due to component configuration and geometry which restricted scanning from both sides of the weld, allowing only single-sided access. In order to meet the ASME Code requirements, the valve would have to be redesigned, fabricated, and installed in the system, which would impose a significant burden on the licensee. The NRC staff has determined that the licensee's limited examination coverage of the weld provides reasonable assurance of structural integrity. Based on the access limitations, it is impractical for the licensee to meet the Code coverage requirements.

5.0 CONCLUSIONS

For the full penetration nozzle-to-shell welds on the RHR system heat exchanger 1A, as described in Items 1 and 2 of Request for Relief 005-CN-001, the NRC staff concludes that the licensee has failed to meet ASME Code requirements relative to proper examination volumes and performance of the examinations from either side of the welds, as accessible. Therefore, Request for Relief 005-CN-001, Items 1 and 2, are denied.

The NRC staff concludes that the ASME Code visual VT-3 examination coverage requirements are impractical for the RPV nozzle supports listed in Items 3 through 6 of Request for Relief 05-CN-001 and to require the licensee to perform the ASME Code examinations would be a significant burden. Furthermore, the licensee obtained 50-percent coverage on the subject RPV nozzle supports, and if significant service-induced degradation were occurring in the subject components, there is reasonable assurance that evidence of it would have been detected. Thus, the inspections performed provide reasonable assurance of the structural integrity of the subject RPV nozzle supports. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), for Items 3 through 6 of Request for Relief 005-CN-001 relief is granted for the second 10-year interval at Catawba 1, which concluded on June 28, 2005.

The NRC staff has reviewed the licensee's submittal and has concluded that compliance with the Code requirements for volumetric coverage of the valve 1NS1B-to–pipe circumferential weld is impractical due to component configuration. The NRC staff has also determined that if the Code requirements were to be imposed on the licensee, the component must be redesigned, which would impose significant burden on the licensee. The NRC staff finds that the examination coverage of the accessible weld volume as complemented by the additional examinations performed by the licensee, provide reasonable assurance of structural integrity of the subject welds.

The NRC staff has determined that granting relief for items 3 through 7 of Request for Relief 05-CN-001 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. 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.

Principal Contributors: T. McLellan
E. Reichelt

Date: July 19, 2006

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

TABLE 1

SUMMARY OF RELIEF REQUESTS

Relief Request Number	TLR RR Sec.	System or Component	Exam. Category	Item No.	Volume or Area to be Examined	Required Method	Licensee Prop Alternativ
05-CN-001 Item 1	3.1	RHR Heat Exchanger	C-B	C2.21	100% of full penetration nozzle-to-shell welds (inlet nozzle)	Volumetric and Surface	Use achieved 15 volumetric cover
05-CN-001 Item 2	3.1	RHR Heat Exchanger	C-B	C2.21	100% of full penetration nozzle-to-shell welds (outlet nozzle)	Volumetric and Surface	Use achieved 15 volumetric cover
05-CN-001 Item 3	3.2	RPV Support A	F-A	F1.40	100% of support surfaces	Visual VT-3	Use achieved 50 coverage on out side
05-CN-001 Item 4	3.2	RPV Support B	F-A	F1.40	100% of support surfaces	Visual VT-3	Use achieved 50 coverage on out side
05-CN-001 Item 5	3.2	RPV Support C	F-A	F1.40	100% of support surfaces	Visual VT-3	Use achieved 50 coverage on out side
05-CN-001 Item 6	3.2	RPV Support D	F-A	F1.40	100% of support surfaces	Visual VT-3	Use achieved 50 coverage on out side
05-CN-001 Item 7	3.3	1NS6-25	C-F-1	C5.11	100% of valve-to-pipe circumferential weld	Volumetric and surface	Use achieved 31 volumetric cover

ATTACHMENT TO SAFETY EVALUATION