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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF THE SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PLAN

REQUESTS FOR RELIEF NO. 97-01 AND NO. 97-02

DUKE ENERGY CORPORATION

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

The Technical Specifications for Catawba Nuclear Station, Units 1 and 2, state that the inservice inspection and testing of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (ASME Code) and applicable Addenda as required by Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a(g), except where specific written relief has been granted by the Commission 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 difficulties without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in 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) on the date 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable edition of the ASME Code, Section XI, for Catawba Nuclear Station, Units 1 and 2, during the second 10-year inservice inspection (ISI) interval, is the 1989 Edition. 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.

Enclosure

Pursuant to 10 CFR 50.55a(g)(5), if the licensee determines that conformance with an examination requirement of Section XI of the ASME Code is not practical for its facility, information shall be submitted to the Commission in support of that determination and a request made for relief from the ASME Code requirement. After evaluation of the determination, pursuant to 10 CFR 50.55a(g)(6)(i), the Commission may grant relief and may impose alternative requirements that are determined to be authorized by law, 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.

Pursuant to 10 CFR 50.55a(g)(6)(ii)(A), the Commission revoked all previous reliefs granted to licensees for the extent of volumetric examinations of reactor vessel shell welds, as specified in Section XI, Division 1, of the ASME Code. The Commission further required that all licensees augment their reactor vessel examination by implementing, as part of the ISI interval in effect on September 8, 1992, the Item B1.10 requirements (examine essentially 100% of the volume of each shell weld) of the 1989 Edition of the ASME Code.

Under 10 CFR 50.55a(g)(6)(ii)(A)(4), licensees may satisfy the augmented requirements by performing the ASME Section XI reactor vessel shell weld examinations scheduled for implementation during ISI intervals in effect on September 8, 1992. As a result, the licensee is required to submit both an alternative to 10 CFR 50.55a(g)(6)(ii)(A) and a request for relief pursuant to 10 CFR 50.55a(g)(5)(iii), or a proposed alternative pursuant to 10 CFR 50.55a(3), for the same welds when the licensee obtains less than the required coverage (essentially 100%) during the examinations.

Additionally, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5), a licensee who makes a determination that it is unable to completely satisfy the requirements for the augmented reactor vessel shell weld examination specified in 10 CFR 50.55a(g)(6)(ii)(A) shall submit information to the Commission to support the determination and shall propose an alternative to the examination requirements that would provide an acceptable level of quality and safety. The licensee may use the proposed alternative when authorized by the Director of the Office of Nuclear Reactor Regulation.

In a letter dated February 18, 1997, and a subsequent response on September 2, 1997, to the staff's request for additional information, Duke Energy Corporation (DEC, the licensee) submitted to the NRC its alternatives to the augmented examination of the reactor vessel shell welds to be conducted pursuant to 10 CFR 50.55a(g)(6)(ii)(A) for Catawba, Units 1 and 2, during the second 10-year interval. The licensee's alternative plan includes examination of all reactor vessel shell welds to the maximum extent practical. The staff has reviewed and evaluated the licensee's request for alternative, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5) and the supporting information for Catawba, Units 1 and 2.

2.0 DISCUSSION

Examination Requirement:

Section 50.55a(g)(6)(ii)(A)(2) states that all licensees shall augment their reactor vessel examinations by implementing the examination requirements for reactor pressure vessel (RPV) shell welds specified in Item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of Subsection IWB of the 1989 Edition of Section XI, Division I, of the ASME Code, subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(A)(3) and (4). For the purpose of this augmented examination, essentially 100% as used in Table IWB-2500-1 means more than 90% of the examination volume for each weld. Additionally, 10 CFR 50.55a(g)(6)(ii)(A)(5) requires licensees that are unable to completely satisfy the augmented RPV shell weld examination requirement to submit information to the NRC to support the determination, and propose an alternative to the examination requirements that would provide an acceptable level of quality and safety.

Licensee's Request For Relief:

Reactor vessel shell welds specified in Item B1.10 of Examination Category B-A of the ASME Code, Section XI, 1989 Edition that did not receive "essentially 100%" examination.

Unit 1 Reactor Vessel

Head-to-Shell Circumferential Weld

ID Number	Item Number
1RPV-W03	B01.001.001

Shell-to-Nozzle Belt Circumferential Weld

ID Number	Item Number
1RPV-W06	B01.001.004

Unit 2 Reactor Vessel

Lower Head-to-Shell Circumferential Weld

ID Number	Item Number
2RPV-101-141	B01.011.001

Lower Shell Longitudinal Seams

ID Number	Item Number
2RPV-101-142A	B01.012.007
2RPV-101-142B	B01.012.008
2RPV-101-142C	B01.012.009

Licensee's Basis For Requesting Alternative:

During the ultrasonic examination of the welds, the minimum 90% coverage requirement of ASME Section XI, 1980 Edition through Winter 1981 Addenda, clarified by Code Case N-460, could not be obtained due to part geometry and physical barriers. A combination of multiple angles and UT techniques was used to obtain the maximum coverage possible. Although the coverage requirements of ASME Section XI could not be met, the amount of coverage obtained for these examinations provides an acceptable level of quality and integrity. Based on these evaluations, the limited coverage will in no way endanger the health and safety of the general public.

These welds were examined to the maximum extent practical in accordance with ASME Section V, Article 4, 1980 Edition with Winter 1981 Addenda and the additional requirements of Regulatory Guide 1.150.

No additional examinations will be required.

Licensee's Alternate Examinations:

The licensee states that the use of radiography as an alternate volumetric examination method is not practical due to component thickness and configurations. Other restrictions making radiography impractical, are physical barriers prohibiting access for placement of source, film, number bands, etc.

The licensee will continue to use the most current ultrasonic techniques available for future examination of the welds for which relief is being requested. The licensee believes that the limited examination is the best available and will continue to perform an ultrasonic examination of all reactor vessel welds to the maximum extent practical in accordance with the requirements of ASME Section V, Article 4, 1989 Edition and Regulatory Guide 1.150, Revision 1, Appendix A.

3.0 EVALUATION

The staff has evaluated the alternatives proposed by the licensee for the volumetric examination of the above-mentioned reactor vessel shell welds in regard to the following factors.

- o Physical constraints at each weld that limits the examination coverage
- o Maximum extent of volumetric coverage obtained with the existing constraints
- o Supplementing inner diameter examination with examination from outside
- o Results of previous vessel examinations
- o Detect presence of degradation mechanism, if any, from the examination
- o Effect of neutron irradiation on the subject welds as of the second 10-year inspection interval

The licensee performed a best-effort examination of the above welds in both reactor vessels from the inside surface. In the Unit 1 reactor vessel, a volumetric coverage of 44% and 48% for the lower head-to-shell weld and the shell-to-nozzle belt weld respectively, was obtained. The limited scan was due to geometric configuration of the welds. All other welds specified in Item B1.10 of Examination Category B A, of Unit 1 reactor vessel met the Code examination requirement. In the Unit 2 reactor vessel, the lower head-to-shell weld received a volumetric coverage of 57% and three lower shell longitudinal seams received volumetric coverage of 81%, each due to physical obstructions from the core guide lugs. All other welds specified in Item B1.10 of Examination Category B-A, of Unit 2 reactor vessel met the Code examination requirement. In response to staff's request for additional information dated April 17, 1997, the licensee stated that in both units there is insufficient clearance between the reactor vessel wall and concrete to perform an examination from the outside surface for the shell-to-nozzle belt weld in Unit 1 and three lower shell longitudinal seams in Unit 2. However, the volumetric examination coverage of the lower head-to-shell weld in both units can be increased by an outer diameter examination with high man-rem penalty.

The licensee's response to the staff's request for additional information states that the preservice inspection of the welds for which an alternative is being requested did not indicate any recordable flaw in the unexamined volume of each weld. Furthermore, the shell-to-nozzle belt weld and the lower head-to-shell weld, being located outside the vessel beltline region, have not experienced neutron embrittlement, which would have adversely affected the fracture toughness of the welds and the heat-affected zones. With the number of transients below the design basis, the probability of origination and growth of a flaw to unacceptable dimensions is extremely small within the second 10-year interval. Moreover, the extent of volumetric coverage should have detected the presence of such a flaw. The three lower shell longitudinal welds of Unit 2, located within the vessel beltline region received volumetric examination coverage of 81%. The staff has determined that flaws of unacceptable dimensions caused due to any degradation mechanism, if present, would have been detected with reasonable confidence with such volumetric coverage. Therefore, the licensee's proposed alternative provides an acceptable level of quality and safety.

4.0 CONCLUSION

The staff has reviewed the licensee's submittals and concludes that the licensee has maximized examination coverage for the reactor vessel welds and that service-induced degradation, if present, would have been detected. Compliance with the code results in hardship or unusual difficulty without a compensating increase in the level of quality and safety in that the licensee's proposed alternative contained in Requests for Relief No. 97-01 and 97-02 provides an acceptable level of quality and safety. Therefore, the licensee's proposed alternative is authorized pursuant to the provisions of 10 CFR 50.55a(g)(6)(ii)(A)(5) and 50.55a(3)(ii) for Catawba, Units 1 and 2, during the second 10-year interval.

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