March 2, 2001

Mr. W. R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNIT 3 RE: THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN REQUEST FOR RELIEF NO. 00-003 (TAC NO. MA9675)

Dear Mr. McCollum:

By letter dated August 1, 2000, Duke Energy Corporation requested relief for Oconee Nuclear Station, Unit 3, from volumetric examination of essentially 100 percent (greater than 90 percent in accordance with Code Case N-460) of the volume as required by the ASME Code, Section XI, for the Class 1 and 2 welds identified in the submittal. The code-required examination was deemed impractical due to component configuration that allows only single-sided access for ultrasonic examination. Therefore, the staff grants the requested relief pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval. The staff's evaluation is enclosed.

Sincerely,

/RA/

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

Docket No. 50-287

Enclosure: As stated

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

THIRD TEN-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN

REQUEST FOR RELIEF NO. 00-003

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNIT 3

DOCKET NO. 50-287

INTRODUCTION

The inservice inspection of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 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 preservice 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 applicable ASME Section XI Code, for Oconee, Unit 3 third 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.

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.

By letter dated August 1, 2000, Duke Energy Corporation (Duke), licensee for the Oconee Nuclear Station, submitted Request for Relief No. 00-003 for the third 10-year inservice inspection interval of Oconee, Unit 3. The request pertains to relief from the volumetric examination of essentially 100 percent of the volume as required by the ASME Code, Section XI, for the Class 1 and 2 welds identified in the relief request, and greater than 90 percent as required by Code Case N-460.

DISCUSSION

System/Component for which Relief is Requested

	Component	ID Number	Item Number
a.	Decay Heat Exchanger Nozzle-to-pipe welds	3-53A-18-11 3-PHA-17	B05.130.001 B05.130.002
b.	Reactor Coolant Pump 3A1 outlet nozzle-to-safe end	3-PDA1-1	B09.011.017
c.	Valve 3HP-27 to elbow	3-51A-66-40	C05.021.050
d.	Valve 3HP-130 to pipe	3-51A-87-54A	C05.021.064

Code Requirement

ASME Code, Section XI, 1989 Edition, in examination categories B-F (Pressure Retaining Dissimilar Metal Welds), B-J (Pressure Retaining Welds in Piping) and C-F-1 (Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping) requires essentially 100 percent volumetric examination coverage of the above welds.

Code Case N-460, which has been approved for use by NRC in Regulatory Guide 1.147, allows credit for full volume coverage of welds if it can be shown that greater than 90 percent of the required volume has been examined.

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. Due to existing geometry, physical barriers, and weld material, the licensee has determined that obtaining greater than 90 percent coverage of the volume as required by Code Case N-460 is impractical.

Licensee's Basis for Relief

Decay Heat Exchanger Nozzle-to-Pipe welds (3-53A-18-11 and 3-PHA-17) are limited to 75 percent coverage of the required volume because of the nozzle taper. In order to achieve more coverage, the nozzles would have to be re-designed to eliminate the taper. The subject welds were examined to the maximum extent practical using ultrasonic techniques in accordance with the requirements of ASME Section XI, Appendix III of the 1989 Edition.

Reactor Coolant Pump 3A1 Outlet Nozzle to Safe End weld (3-PDA1-1), Valve 3HP-27 to Elbow weld (3-51A-66-40), and Valve 3HP-130 to Pipe weld (3-51A-87-54A) are limited to 62.5 percent coverage of the required volume of each weld due to single-sided access for examination as a result of component configuration.

Current ultrasonic technology is not capable of reliably detecting or sizing flaws on the far side of austenitic weld configurations common to each of the above welds. The licensee has demonstrated that the best available techniques were applied through the Performance Demonstration Initiative (PDI). The PDI Performance Demonstration Qualification Summary for austenitic piping certifies that examinations from one side are a "best effort." Therefore, coverage on the far side of the weld is not claimed. The subject welds were examined to the maximum extent practical using ultrasonic techniques qualified in accordance with the requirements of Supplement 2 to Appendix VIII of ASME Section XI, 1992 Edition with the 1993 Addenda as modified by the PDI.

<u>Alternate Examinations or Testing</u> (as stated)

> The use of radiography as an alternate volumetric examination of the welds/components referenced in this request is not a viable option. Restrictions to performing radiography are primarily due to inability to access the inside of the components to place film or position a radiographic source.
>
> Duke Energy proposes to use the pressure test and VT-2 visual examination to compliment the limited examination coverage. The Code requires (reference Table IWB-2500-1, Item Number B15.20) that a system leakage test be performed after <u>each</u> refueling outage. Additionally a system hydrostatic test (reference Table IWB-2500-1, Item Number B15.21) is required once during each 10-year inspection interval. These tests require a VT-2 visual examination for evidence of leakage. This testing will provide adequate assurance of pressure boundary integrity.
>
> In addition to the above Code required examinations (volumetric and pressure test), there are other activities which provide a high level of confidence that, in the unlikely case that leakage did occur through these welds, it would be detected and isolated. Specifically, leakage from these welds would be detected by monitoring of the Reactor Coolant System (RCS), which is performed daily under procedure PT/1,2,3/A/0600/10, "RCS Leakage." This RCS leakage monitoring is a requirement of the Technical Specification 3.4.13, "RCS Operational Leakage." Leakage is also evaluated in accordance with this Technical Specification. The leakage could be detected through several methods. Technical Specification 3.4.15, RCS "Leakage Detection Instrumentation," requires the containment normal sump level indication, in combination with a particulate (RIA-47) or gaseous radioactivity monitor (RIA-49). These monitors are sensitive to low leak rates; are capable of detecting any fission products in the coolant and will make these monitors more sensitive to coolant leakage. In addition to the radiation monitors, a level indicator in the Reactor Building normal sump also monitors leakage. Other checks are the RCS mass balance calculation and level in the Letdown Storage Tank.

> Duke Energy has examined the welds/components referenced in this request to the maximum extent possible utilizing the latest in examination techniques and equipment. Duke Energy will continue to perform ultrasonic examination of all welds/components identified in Section I of this request to the maximum extent practical, within the limits of original design and construction. Future examinations will be in accordance with the requirements of ASME Section XI, 1995 Edition with the 1996 Addenda, Appendix VIII as modified by 10CFR50.55a(b)(2)(xiv, xv, and xvi) and Code Case N-460. This will provide reasonable assurance of weld/component integrity. Thus, an acceptable level of quality and safety will have been achieved, and allowing relief from the aforementioned Code requirements will not endanger public health and safety.

EVALUATION

The staff has evaluated the information provided by the licensee in support of the volumetric examinations of the subject welds performed during the third 10-year inservice inspection interval. For the subject welds, ultrasonic scanning in the axial direction could be performed from only one side of the weld due to component configuration that prevents scanning from the tapered surface on the other side of the weld. Each weld has a stainless steel component on the far side that can not be scanned from the same side due to the taper on the examination surface. However, the licensee's best-effort examination with single-sided access achieved volumetric coverages of the welds ranging from 62.5 to 75 percent. Code Case N-460, which was approved for use by NRC in Regulatory Guide 1.147, allows credit for full volume coverage if it can be shown that more than 90 percent of the required volume has been examined.

The staff has determined that the examination coverage of the subject welds was reduced due to component configuration that restricts scanning from both sides of the weld, allowing only single-sided access. Therefore, based on access limitations, it is impractical to meet the Code coverage requirements. In order to meet the Code requirements, the components would have to be redesigned, fabricated, and installed in the systems, which would impose a significant burden on the licensee. The results of the examinations performed did not identify any rejectable indications. The staff agrees that if there were any service-induced flaws existing in the welds and/or in the base metal adjacent to the welds, the examination of the accessible weld volumes would at least detect a portion of it with a high degree of confidence. Therefore, the staff has determined that the licensee's limited examination of the welds provide reasonable assurance of structural integrity of the subject welds.

CONCLUSION

Based on the above discussion, the staff has concluded that compliance with the Code requirements for volumetric coverage of the subject welds is impractical due to component configuration and that the proposed alternatives provide adequate assurance of the structural integrity of the subject welds. The staff has also determined that if the Code requirements were to be imposed on the licensee, the components must be redesigned, which would impose significant burden on the licensee. The staff believes that the examination coverages of the accessible weld volume provide reasonable assurance of structural integrity of the subject welds. Therefore, the relief is granted for the subject welds pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval for the Oconee Nuclear Station, Unit 3. This grant of relief is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest given due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

Principal Contributor: P. Patnaik

Date: March 2, 2001

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