

July 2, 2003

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

**Subject: Duke Energy Corporation
Oconee Nuclear Station, Unit 3 - Docket No. 50-287
Third Ten-year Inservice Inspection Interval
Request for Alternate (Relief Request No. 03-004)**

Duke Energy Corporation (Duke) hereby submits a Request for Alternate for the Oconee Unit 3 third 10-year reactor vessel examination per 10 CFR 50.55a(g)(5)(iii). This examination is planned for the next Unit 3 refueling outage scheduled for the fall of 2004.

The reactor pressure vessel nozzle inner radius sections must be volumetrically examined per ASME Section XI, Appendix VIII, 1995 Edition through the 1996 Addenda. Duke uses the Performance Demonstration Initiative (PDI) to qualify vendor and in-house ultrasonic test personnel. However, the PDI test set for nozzle inner radius qualification from the inside surface does not adequately model the B&W 177 reactor vessel core flood nozzle. In addition, design features (permanently installed flow restrictors) prevent full coverage of the examination surface.

Therefore, Duke proposes to use a remote enhanced VT-1 (visual) examination as described in the attachment as an alternate to the specified volumetric examination. Approval is requested per 10 CFR 50.55a(a)(3)(i) because the alternative provides an acceptable level of quality and safety.

This request is similar to requests submitted by Florida Power and Light for Turkey Point Units 3 and 4, and Public Service Electric and Gas for Salem Units 1 and 2.

Please direct any questions to R. P. Todd at (864) 885-3418.

Very truly yours,



R. A. Jones,
Site Vice President,
Oconee Nuclear Station

Attachment: ISI Relief Request 03-04

A047

July 2, 2003

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xc(w/o att):

M. C. Shannon
Senior NRC Resident Inspector
Oconee Nuclear Station

DUKE ENERGY CORPORATION
Oconee Nuclear Station Unit 3

10-YEAR INTERVAL REQUEST FOR RELIEF NO. 03-004

Pursuant to 10CFR50.55a (a) (3) (i), Duke Energy Corporation (Duke) proposes an alternative to the requirements of ASME Section XI, Table IWB-2500-1, Examination Category B-D, Item No. B3.100, Nozzle Inner Radius Sections.

I. System/Components for Which the Alternative Applies:

Category B-D Full Penetration Welds of Nozzles in Vessels

Item Numbers:

B03.100.007 Core Flood Nozzle Inner Radius, 3-RPV-WR54

B03.100.008 Core Flood Nozzle Inner Radius, 3-RPV-WR54A

II. Code Requirement: ASME Section XI, Table IWB-2500-1 Examination Category B-D, Full Penetration Welds of Nozzles In Vessels, Volumetric Examination, Figure IWB-2500-7(b), Examination Volume M-N-O-P of ASME Section XI, 1989 Edition with no addenda.

III. Code Requirement for Which the Alternative is Requested:
Relief is requested to perform a remote VT-1 examination in lieu of the volumetric examination required in Table IWB-2500-1 Examination Category B-D, Full Penetration Welds of Nozzles In Vessels, Item No. B3.100, Nozzle Inner Radius Sections.

IV. Basis for Relief: Pursuant to 10CFR50.55a (a) (3) (i), Relief is requested to perform a remote VT-1 examination in lieu of the required volumetric examination on the basis that the proposed alternative provides an acceptable level of quality and safety.

V. Alternative Method for Ultrasonic Examination:
Duke Energy Corporation proposes to use a remote VT-1 examination of surface M-N as shown in ASME Section XI, Figure IWB-2500-7(b) of the 1989 Edition with no addenda. The remote VT-1 equipment will have sufficient magnification and sensitivity to resolve a 0.001 inch wire in lieu of the sensitivity required for an ultrasonic examination. The examination results will be evaluated in accordance with ASME Section XI, IWB-

3140, 1989 Edition with no addenda. Crack-like surface flaws exceeding the acceptance criteria of Table IWB-3512-1 are unacceptable for continued service unless the reactor vessel meets the requirements of IWB-3142.2, IWB-2142.3 or IWB-3142.4. When applying Table IWB-3512-1 criteria, crack depth will be assumed to be equal to one-half of the measured crack length.

It must be noted that because of the permanent attachment of flow guides in the core flood nozzles, full coverage of the examination surface will not be possible (See attached drawing). During ultrasonic examinations performed in the first and second Inspection Intervals 50% coverage of the examination volume was achieved. Relief from the volumetric examination coverage requirements was sought under Oconee Relief Request 94-01 submitted by letter of April 4, 1994 and supplements dated April 14, 1994, and March 16, 1995. Relief was granted by NRC in a letter of June 12, 1995. The VT-1 examination coverage is expected be no less than the ultrasonic examination.

VI. Justification for the Granting of Relief:

The Core Flood (CF) piping from each of the two RV Nozzles to the first upstream check valve is Duke Class A (ASME Code Class 1). The Oconee Unit 3 core flood nozzles in the reactor pressure vessel are made of forged SA-508 ferritic steel with stainless steel cladding. The piping is stagnant during all normal and upset operating conditions. These piping components have been analyzed to Section III of the 1983 Edition of the ASME B&PV Code. The results of this analysis show that the CF piping components meet all Code requirements and allowable stresses.

The primary degradation mode in reactor vessel nozzles is thermal fatigue. The thermal fatigue cumulative usage factor (including the effects of all applicable thermal cycling events and thermal stratification) is less than 0.033 (the Code limit is 1.00) for all Duke Class A piping components. Therefore, thermal fatigue degradation is unlikely.

Thermal fatigue in this application may produce hairline surface indications along the nozzle's inner radius section. The intent of the Code requirement for an ultrasonic examination is that the examination would detect such surface indications.

These nozzles were nondestructively examined during fabrication and subsequently examined inservice twice using the ultrasonic method. There were no examination findings and no flaws were detected in any of the Oconee reactor pressure vessel nozzles.

According to a NRC memorandum^[1], the NRC staff has indicated that an ultrasonic examination could be replaced by an enhanced VT-1 visual examination for the proposed nozzle inspections. Subsequent to that memorandum, the NRC granted requests similar to this request to Florida Power and Light for Turkey Point Units 3 & 4 (submitted May 6, 2002 and approved August 15, 2002) and Public Service Electric and Gas for Salem Units 1 & 2 (submitted February 11, 2002 and approved March 21, 2002).

Like Florida Power and Light and Public Service Electric and Gas, Duke proposes to use high magnification cameras to give 1-mil resolution capability for the remote VT-1 examination of the accessible portion of the nozzle inner radius section surface area. With this resolution, it is highly likely that Duke would detect and disposition flaws using the allowable flaw length criteria in Table IWB-3512-1 of the ASME Code, Section XI, for the disposition of any linear flaws. Therefore, the proposed alternative provides reasonable assurance of structural integrity.

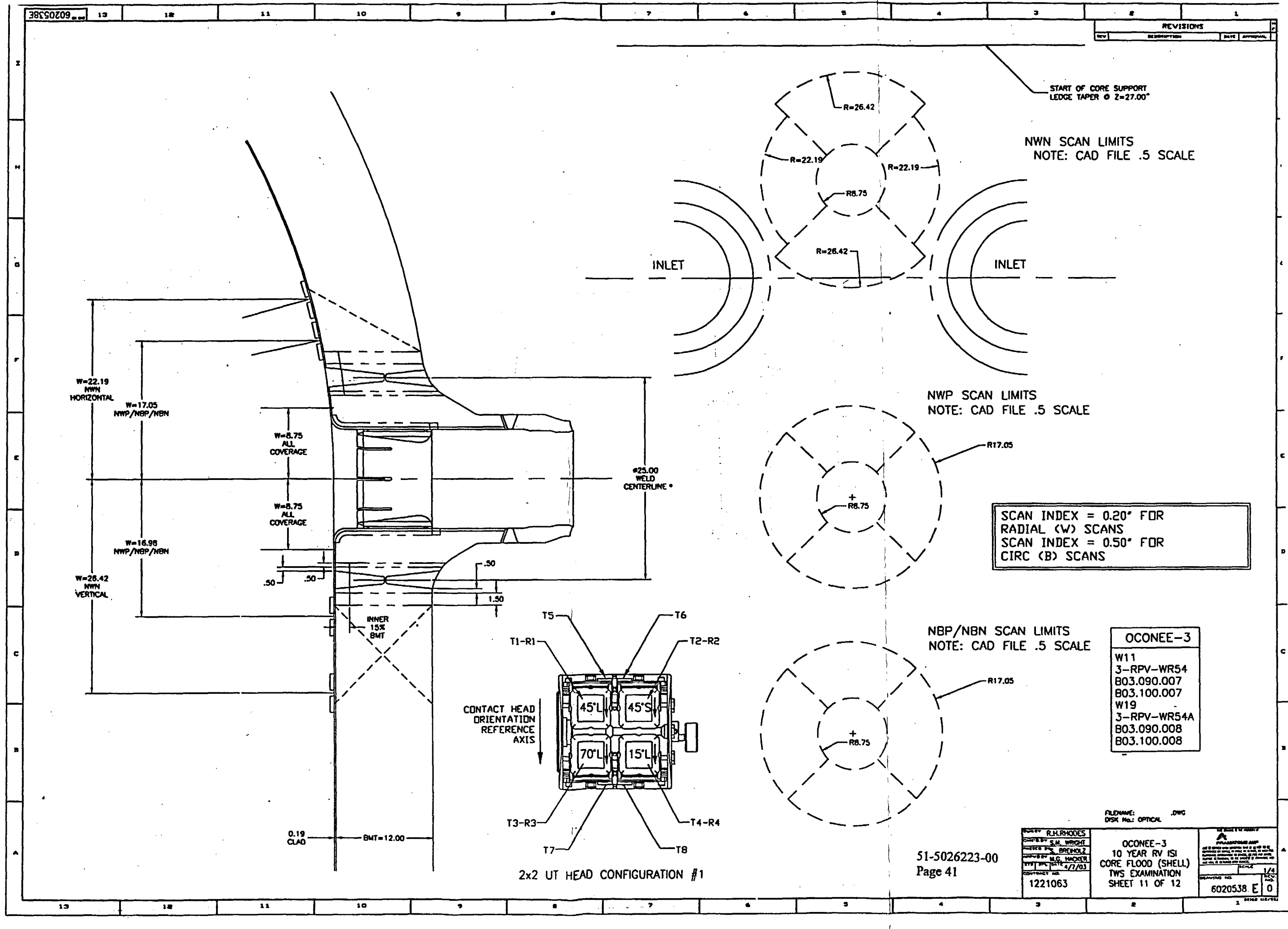
VII. Implementation: Duke Energy will perform the remote VT-1 examination of the core flood nozzle inner radii in conjunction with the ONS 3rd 10 year reactor pressure vessel examination. The ONS 3rd 10 year interval started 7-15-1994, and ends 12-16-2004.

Sponsored By James J. Mc Gille III Date 5-29-03

Approved By R. Kevin Rhayne Date 5/29/03

¹NRC Memorandum from K.R. Wichman to W.H. Bateman dated May 25, 2000; Subject: The Third Meeting with the Industry to Discuss the Elimination of RPV Inner Radius Inspection (ADAMS Accession No. ML003718630).

REVISIONS		
REV	DESCRIPTION	DATE



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COVERAGE

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ALL
COVERAGE

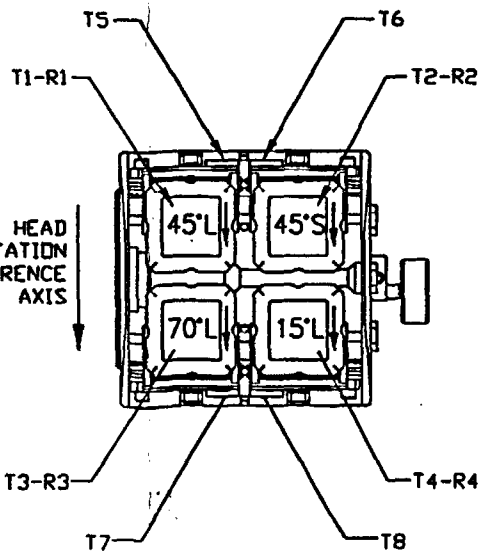
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CONTACT HEAD
ORIENTATION
REFERENCE
AXIS



2x2 UT HEAD CONFIGURATION #1

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CLAD

BMT=12.00