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ONS-2015-088

July 15, 2015

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

10 CFR 50.55a

Duke Energy Carolinas, LLC (Duke Energy)
Oconee Nuclear Station, Units 1, 2 and 3
Docket Numbers 50-269, 50-270, and 50-287
Renewed License Numbers DPR-38 DPR-47 and DPR-50

Subject: Fourth Ten-Year Inservice Inspection Plan, Relief Request No. 15-ON-004,
Limited Visual Examinations from 1EOC27, 2EOC26 and 3EOC27 Outages

Pursuant to 10 CFR 50.55a(g)(5)(iii), Duke Energy hereby requests NRC approval of the following relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1998 Edition with 2000 Addenda.

The attached Relief Request applies to limited visual examinations performed on the Reactor Vessel Supports during Unit 1, EOC27, Unit 2, EOC26 and Unit 3, EOC27 outages. The relief request details are provided as an enclosure to this letter.

This submittal document contains no regulatory commitments.

If there are any questions or further information is needed you may contact David Haile at (864) 873-4742.

Sincerely,

Scott L. Batson
Vice President
Oconee Nuclear Station

Enclosure

Relief Request Serial #15-ON-004:
Limited visual examinations per 10 CFR 50.55a(g)(5)(iii) for Unit 1, 2 and 3,
Fourth Inservice Inspection Interval

A047
NRB

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cc (with enclosure):

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Oconee Nuclear Station

Enclosure to ONS-2015-088

**Duke Energy Carolinas, LLC
Oconee Nuclear Station, Units 1, 2, and 3**

Relief Request Serial #15-ON-004:

**Limited visual examinations per
10 CFR 50.55a(g)(5)(iii) for Unit 1, 2 and 3,
Fourth Inservice Inspection Interval**

1.0 ASME Code Component(s) Affected

Relief is requested pursuant to 10 CFR 50.55a(g)(5)(iii) for the following component supports:

- Unit 1 Reactor Pressure Vessel Support Skirt (RPVSS), Reactor Coolant System, Component I.D. No. 1-RPV-WR36, Summary Number O1.F1.40.0001 and ASME Code Class 1.
- Unit 2 Reactor Pressure Vessel Support Skirt (RPVSS), Reactor Coolant System, Component I.D. No. 2-RPV-WR36, Summary Number O2.F1.40.0001 and ASME Code Class 1.
- Unit 3 Reactor Pressure Vessel Support Skirt (RPVSS), Reactor Coolant System, Component I.D. No. 3-RPV-WR36, Summary Number O2.F1.40.0001 and ASME Code Class 1.

2.0 Applicable Code Edition and Addenda

ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition through the 2000 Addenda.

3.0 Applicable Code Requirement

IWF-2500, Table IWF-2500-1, Examination Category F-A, Item Number F1.40. Figure IWF-1300-1(c) applies to the RPV Support Skirt Examination Boundary. A VT-3 visual examination of 100% of the support within the examination boundary is required once each inspection interval.

Relief is requested from the requirement to perform a VT-3 visual examination of 100% of the RPVSS within the examination boundary.

4.0 Impracticability of Compliance (Reason for Request)

Duke Energy considers it impractical to perform a VT-3 visual examination on 100% of the interior and exterior surfaces of the Reactor Vessel Support Skirt within the Figure IWF-1300-1(c) boundary due to the existence of insulation panels that partially obstruct the visual examination surfaces. Although these insulation panels can be removed, compliance with the Code requirement to visually examine 100% of the support skirt surfaces is considered impractical due to the excessive personnel radiation dose (estimated) to remove and reinstall these insulation panels. In order to remove the bottom row of mirror insulation needed to access the entire exterior surface of the support skirt, removal of additional rows of mirror insulation is necessary because they are supported by the bottom row of insulation. Previous operating experience at Oconee Nuclear Station has shown that upper rows of these insulation panels can slip when the supporting row of insulation is removed, creating a potential personnel safety risk, as well as incurring additional unplanned radiation exposure associated with any insulation repair or replacement.

In addition, a small portion of the skirt exterior and interior surfaces is inaccessible for visual examination due to the existence of permanent insulation support attachments. Compliance with the Code requirement to visually examine 100% of the support skirt surfaces at these support attachment locations is considered impractical because the

insulation support attachments are not designed to be removed and would have to be re-designed in order to improve the examination coverage.

Those portions of the support skirt surfaces that were visually examined, as well as those surfaces that are considered impractical to examine, are described below and reference sketch #1.

RPVSS outside diameter area; 1) Examined Surface D-E, 2) Portions of surface A-D accessible through removable inspection panels.

RPVSS inside diameter area: 1) Examined surface C-G, 2) Area B-C inaccessible not examined.

5.0 Proposed Alternative and Basis for Use

For each Unit, in lieu of the requirement to perform a VT-3 visual examination of 100% of the support within the examination boundary, a VT-3 visual examination was performed on accessible surfaces of the Reactor Vessel Support Skirt shown in Attachment A Sketch #1. The calculated coverage is approximately 66.5% of the support surface areas within the examination boundary specified in Figure IWF-1300-1(c).

The VT-3 visual examinations were performed using personnel, equipment, and procedures qualified in accordance with the ASME Code, Section XI, 1998 Edition with the 2000 Addenda.

There were no unacceptable conditions or indications detected during these examinations.

6.0 Justification for Granting Relief

The Reactor Vessel Support Skirts were designed, fabricated, tested and inspected in accordance with ASME Section III, 1965 Edition through Summer 1967 Addendum. The support skirts meet the Class 1 criteria for stress and fatigue defined in this Code. The Oconee Reactor Vessel Support Skirts were designed with significant margin and included consideration of normal operating loads plus pipe rupture loads plus the maximum hypothetical seismic loads.

Oconee has a program supporting the NRC GL 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants that provides additional assurance that loss of material of the Reactor Vessel Support Skirt is being managed. Plant procedures require that leakage is tracked and affected components found with boric acid corrosion to be entered into corrective action program and action taken to address.

Duke Energy believes that the proposed alternative is sufficient to confirm the general mechanical and structural condition of the support skirts. All of the anchorages were examined and there were no adverse conditions detected (e.g., loose or missing parts or fasteners). Because the bottom of the support skirts were examined without obstruction, any boric acid leakage and potential resulting corrosion would have been detected had it occurred.

Duke Energy has examined the Reactor Vessel Skirt support to the extent possible utilizing approved examination techniques and equipment. Based on the acceptable examination results for the surfaces examined, coverage completed by the visual

examination and the Reactor Vessel Support design margin it is Duke's position that the visual examination provides a reasonable assurance of continued support structural integrity.

7.0 Burden Caused by Compliance

- The minimum estimated radiation dose associated with obtaining the maximum available coverage on the exterior surfaces of the support skirt is approximately 282.4 rem for each Unit. This is the lowest dose and only access time periods for this very high radiation area.
- The minimum estimated radiation dose associated with obtaining the maximum available coverage on the interior surfaces of the support skirt is approximately 1100 mrem for each Unit.

Note that even if the above radiation dose is incurred, there will still be some portions of the RPVSS skirt interior and exterior surfaces that cannot be examined because of permanent attachments to the support skirt to which the insulation is attached. For these surfaces, the burden caused by compliance would require a redesign of the insulation supports. The estimated radiation dose associated with the redesign and replacement of these insulation supports has not been estimated.

8.0 Duration of Proposed Alternative

The proposed alternative is requested for the 4th Inservice Inspection Interval for Oconee Units 1, 2, and 3.

- The 4th Inservice Inspection Interval for Oconee Unit 1 began on 01/01/2004 and ended on July 15, 2014.
- The 4th Inservice Inspection Interval for Oconee Unit 2 began on 09/09/2004 and ended on July 15, 2014.
- The 4th Inservice Inspection Interval for Oconee Unit 3 began on 01/02/2005 and ended on July 15, 2014.

Attachment A

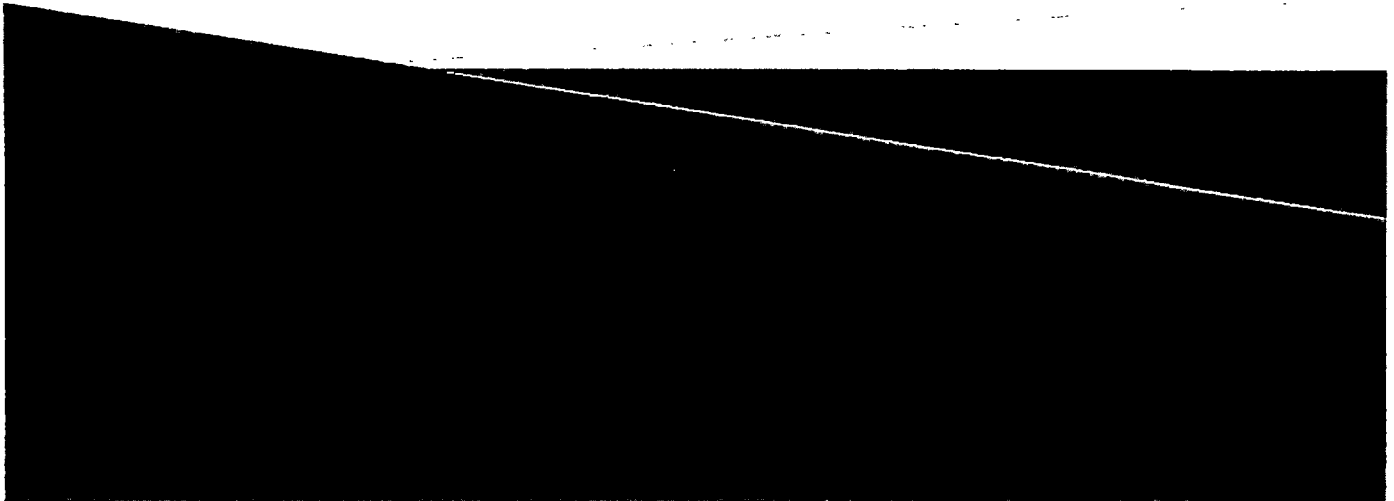
Relief Request

15-ON-004

1EOC-27, 2EOC-26 3EOC-27

VT Limited Coverage.

Sketch 1, OM-201-1480, OM-201-3153
and OM-2201-0575. Typical all Units.



IWF EXAMINATION BOUNDARY: A-D-E-F-G-C-B

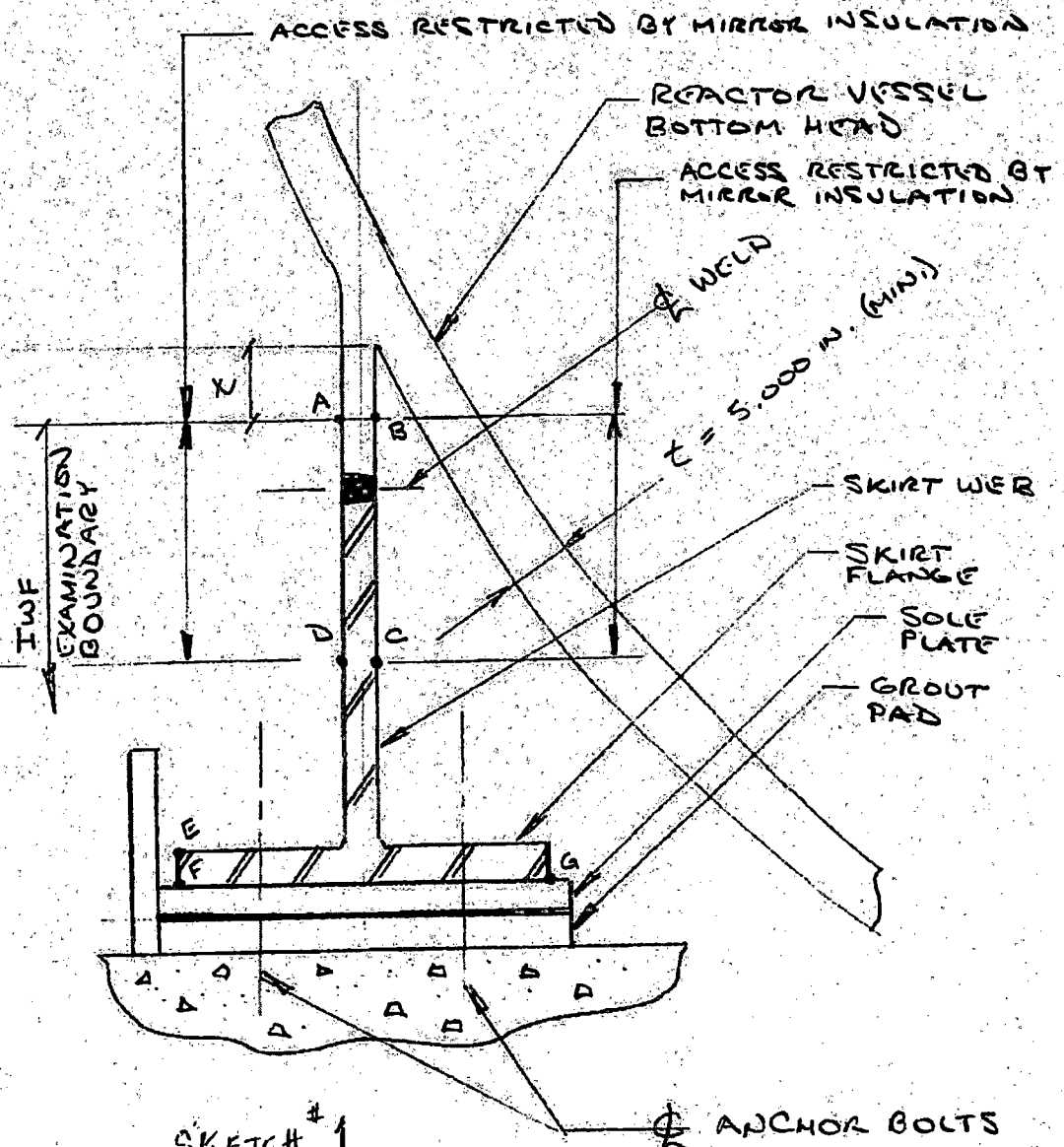
SKIRT O.D. SURFACES EXAMINED:

1. SURFACE D-E
2. PORTIONS OF SURFACE A-D ACCESSIBLE THROUGH (2) 9" X 24" REMOVABLE INSPECTION PANELS

SKIRT I.D. SURFACES EXAMINED:

1. SURFACE C-G (EXAMINATION PERFORMED THROUGH SKIRT W.E.B. OPENINGS (NOT SHOWN).

MULTIPLE MIRROR INSULATION PANEL RINGS
SUPPORTED BY SKIRT ATTACHMENT



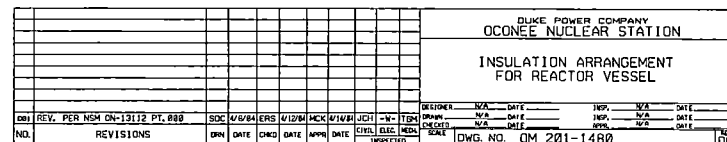
SKETCH #1

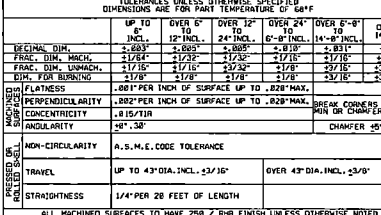
SECTION THROUGH SUPPORT SKIRT

(NOT TO SCALE)

(TYP. - UNITS 1, 2, AND 3)

ATTACHMENT A
Pg 1 of 4





D 18 — HALF SIZE
TYPICAL 2 LOCATIONS

