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Serial: RA-21-0219 10 CFR 50.55(a)
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United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

OCONEE NUCLEAR STATION, UNITS NO. 1, 2 AND 3
DOCKET NO. 50-269, 270 AND 287 / RENEWED LICENSE NO. DPR-38, DPR-47 AND DPR-55

SUBJECT: Response to Request for Additional Information for Request for Alternative for Implementation of Extended Reactor Vessel Inservice Inspection Intervals for Oconee Units 1, 2, and 3

REFERENCES:

1. Duke Energy Letter RA-20-0328, *Request for Alternative for Implementation of Extended Reactor Vessel Inservice Inspection Intervals for Oconee Units 1, 2, and 3,* dated January 19, 2021 (ADAMS Accession No. ML21019A276).
2. NRC Email from Zackary Stone to Arthur Zaremba, *Oconee Nuclear Station, Units 1, 2, and 3 - Request for Additional Information RE: Alternative for ISI RPV Weld Examination from 10 to 20 years (EPID-L-2021-LLR-0004)*, dated July 08, 2021 (ADAMS Accession No. ML21190A017).

Ladies and Gentlemen:

By letter dated January 19, 2021 (Reference 1), Duke Energy Carolinas, LLC (Duke Energy) submitted Relief Request RA-20-0328, "*Request for Alternative for Implementation of Extended Reactor Vessel Inservice Inspection Intervals for Oconee Units 1, 2, and 3,*" to the NRC. Specifically, Duke Energy requested NRC authorization to defer performance of American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Division 1, Section XI (ASME Section XI Code), category B-A and B-D required volumetric examinations until the sixth 10-year inservice inspection (ISI) interval for each unit.

By email dated July 8, 2021 (Reference 2), the NRC requested additional information required to complete its review. Enclosure 1 to this letter provides Duke Energy's response to the request. The response to Request for Additional Information (RAI) number 4 contains information that is proprietary to Framatome Inc. and is requested to be withheld from public access under 10 CFR 2.390. Enclosure 2 contains the full response to RAI number 4 with proprietary information included. Enclosure 3 contains a non-proprietary response to RAI number 4 that may be released to the public. An affidavit is included in Enclosure 4 attesting to the proprietary nature of Enclosure 2.

This document contains no new regulatory commitments.

Should you have any questions concerning this letter, or require additional information, please contact Art Zaremba, Manager – Nuclear Fleet Licensing, at 980-373-2062.

Sincerely,

Steven M. Snider
Vice President
Oconee Nuclear Station

Enclosures:

1. Response to Request for Additional Information
2. Proprietary Response to RAI 4
3. Non-Proprietary Response to RAI 4
4. Affidavit Pursuant to 10 CFR 2.390 (Framatome Inc.)

cc : L. Dudes, Regional Administrator USNRC Region II
J. Nadel, USNRC Senior Resident Inspector – ONS
S. Williams, NRR Project Manager – ONS

Enclosure 1

Duke Energy Carolinas, LLC

Oconee Nuclear Station, Units 1, 2, & 3

Response to Request for Additional Information

Request for Additional Information (RAI) 1

Table 3 in RA-20-0328 calculates through-wall cracking frequency (TWCF) of reactor vessel beltline components based on methodology found in WCAP-16168-NP-A, Revision 3 (ADAMS Accession No. ML11306A084), and using inputs which include the neutron fluence at 60 years of operation. It is not apparent in the submittal whether this fluence considered the increased thermal power of the Measurement Uncertainty Uprate (MUR) request which was submitted on February 19, 2020 and supplemented by letters dated April 6, 2020, July 23, 2020, and August 17, 2020, (ADAMS Accession Nos. ML20050D379, ML20097E117, ML20205L403, and ML20230A127) and which was approved on January 26, 2021 (ADAMS Accession No. ML20335A001).

Request

Please clarify whether the fluence inputs listed in Table 3 of RA-20-0328 were calculated considering the increased thermal power of the MUR request which was approved by the NRC staff on January 26, 2021.

Duke Energy Response to RAI 1:

The projected fluence values used as inputs to calculate the TWCF in Oconee Nuclear Station (ONS) Relief Request RA-20-0328 (Reference 1) were calculated based on Measurement Uncertainty Uprate (MUR) impacts beginning in Cycle 30 for ONS Unit 1, Cycle 29 for ONS Unit 2, and Cycle 29 for ONS Unit 3. The assumed fluence impact of a 1.7% MUR was bounded using a 2% increase in fluence rate to calculate the projected fluence values at the end of 60 years of operation used for the ONS RA-20-0328 submittal.

RAI 1 References

1. Relief Request RA-20-0328, “Request for Alternative for Implementation of Extended Reactor Vessel Inservice Inspection Intervals for Oconee Units 1, 2, and 3,” dated January 19, 2021 (ADAMS Accession No. ML21019A276).

RAI 2

Table 3 in RA-20-0328 calculates TWCF for reactor vessel components in the beltline. The NRC staff notes that no inlet nozzles, outlet nozzles, or nozzle to nozzle belt forging welds are included in Table 3. It also appears that beltline materials include all plate, forging and weld materials if the neutron fluence is greater than or equal to $1x10^{17}$ n/cm^2 (E>1 MeV). It is not apparent whether the inlet nozzles, outlet nozzles, and nozzle to nozzle belt forging welds will be above or below this fluence threshold at the end of 60 years of operation.

Request

Please clarify if the inlet nozzles, outlet nozzles, and nozzle to nozzle belt forging welds will be above or below the fluence threshold of $1x10^{17}$ n/cm^2 (E>1 MeV) at the end of 60 years of operation. If any of these materials will be above the fluence threshold of $1x10^{17}$ n/cm^2 (E>1 MeV) at the end of 60 years of operation, please calculate the TWCF for these materials as required by the methodology found in WCAP-16168-NP-A, Revision 3, and add the information for these materials to Table 3.

Duke Energy Response to RAI 2:

The ONS Units 1, 2, and 3 wetted surface of the inlet and outlet nozzle to nozzle beltline forging welds are above the fluence threshold of $1x10^{17}$ n/cm^2 (E>1 MeV) at the end of 60 years of operation (54 Effective Full Power Years [EFPY]) as shown in Table 1. Similarly, the wetted surface of the postulated flaw location for the outlet nozzles is above the fluence threshold of $1x10^{17}$ n/cm^2 (E>1 MeV) at the end of 60 years of operation (54 EFPY); however, the wetted surface of the postulated flaw location for the inlet nozzles is below this threshold. Although the fluence values of these materials exceed the $1x10^{17}$ n/cm^2 (E>1 MeV) threshold, the methodology found in WCAP-16168-NP-A, Revision 3 (Reference 1) does not require the calculation of TWCF for these materials, as described below. The following paragraphs are provided as justification that the nozzle-to-shell welds and nozzle forgings are not the limiting locations within the reactor vessel and no further analysis of these materials is required relative to the impact of different inspection intervals on the frequency of reactor vessel failure.

The technical basis for the risk-informed extension of the reactor vessel inservice inspection (ISI) interval is provided in WCAP-16168-NP-A, Revision 3. This report has been reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC). The NRC safety evaluation (SE) is attached to the front of WCAP-16168-NP-A, Revision 3. A description of the engineering evaluation is provided in Section 3.2.1 of the SE, which includes discussion of the limiting location for reactor vessel failure based on a deterministic fracture mechanics analysis of the impact of flaws in several reactor vessel regions, including the full penetration nozzle-to-vessel welds. The reactor vessel regions that were considered are shown in Section 3.1, specifically Figure 3-1 and Figure 3-2, of WCAP-16168-NP-A, Revision 3 which includes locations such as the closure head to flange weld, lower shell transition, beltline area, and other areas of the reactor vessel. Note that the full penetration nozzle-to-vessel welds are reviewed in detail as the response to RAI 5 on WCAP-16168-NP, which is attached to Appendix N of WCAP-16168-NP-A, Revision 3.

At the time that WCAP-16168-NP-A, Revision 3 was developed, the “beltline” area was considered to be the cylindrical region of the reactor vessel directly adjacent to the active core. This definition is confirmed by the breakdown and evaluation of the various portions of the reactor vessel with varying geometries in Figures 3-1 and 3-2 of WCAP-16168-NP-A, Revision 3. Thus, the term “beltline” in the WCAP-16168-NP-A, Revision 3 methodology was not intended to imply materials with a projected fluence value of greater than 1×10^{17} n/cm^2 (E>1 MeV) at the end of the licensed period of operation, which is the definition commonly utilized for this term today. Instead, this term referred to the cylindrical area of the reactor vessel directly adjacent to the active core.

The deterministic fracture mechanics analysis in WCAP-16168-NP-A, Revision 3 for the various geometrical regions was performed based on a comparison of ratios of ASME Code allowable stress intensity factors values ($K_{Iallowable}/K_{Iapplied}$). As determined in Figure 3-1, Figure 3-2, and the response to RAI 5 (which is attached to Appendix N of WCAP-16168-NP-A, Revision 3), the most limiting location in the reactor vessel is the beltline (i.e., the cylindrical region directly adjacent to the active core), which has the lowest code allowable stress intensity factor margin ratio of approximately 0.5. The response to RAI 5 demonstrates that the nozzle-to-vessel welds (i.e., the nozzle to nozzle beltline forging welds) have a limiting margin ratio of 0.98, which is more favorable than the beltline region ratio. Thus, as concluded in WCAP-16168-NP-A, Revision 3, the nozzle-to-vessel welds have a margin ratio that is approximately 2 times larger than the reactor vessel beltline region and are not the most limiting region of the reactor vessel.

As discussed in Section 3.2.1 of the SE located at the front of WCAP-16168-NP-A, Revision 3, the results from the deterministic fracture mechanics analysis were consistent with the assumptions utilized in the Pressurized Thermal Shock (PTS) Risk Study, specifically NUREG-1806 (Reference 2) and NUREG-1874 (Reference 3). The PTS Risk Study concluded that the limiting reactor vessel region was the beltline region. Since the reactor vessel beltline region has the lowest margin to failure, the NRC staff concluded that the beltline region is the most limiting location and the beltline location (i.e., the cylindrical region directly adjacent to the active core) can be used to determine the impact of different inspection intervals on the frequency of Reactor Vessel (RV) failure. Based on the NRC staff’s conclusions in Section 3.2.1 of the SE in WCAP-16168-NP-A, Revision 3 and historical precedent that nozzle materials have not been analyzed in any similar relief requests based on WCAP-16168-NP-A, Revision 3, the TWCF is not calculated for the nozzle forgings nor the inlet and outlet nozzle to nozzle beltline forging welds.

Additionally, it should be noted that the SE for WCAP-16168-NP-A, Revision 3 indicates in Section 3.4 that TWCF must be estimated using the methodology in NUREG-1874; however, NUREG-1874 does not provide a methodology to analyze the nozzle materials, which are neither traditional “axial” nor “circumferential” weld materials as described in NUREG-1874. This observation further reinforces that the WCAP-16168-NP-A, Revision 3 methodology was setup to evaluate the most limiting region (i.e., the cylindrical shell beltline region) of the reactor vessel.

More recently, a separate reactor vessel nozzle deterministic fracture mechanics evaluation was completed in PWROG-15109-NP-A (Reference 4), which compared the structural integrity of the nozzle as compared to the beltline region for pressure-temperature (P-T) limits. This report has been reviewed and approved by the NRC. The NRC SE is attached to the front of PWROG-15109-NP-A. Based on the Pressurized Water Reactor Owners’ Group’s (PWROG) justification and the

NRC staff's independent assessment, the NRC staff found that P-T limit curves for inlet and outlet nozzle corners will not be more limiting than P-T limit curves developed for the shell (and associated welds) of the beltline region for neutron fluence values less than 4.28 x 10^{17} n/cm^2 (E > 1 MeV) in the nozzle region. Furthermore, this report indicates that operating U.S. PWR reactor vessel nozzle forgings would not be expected to embrittle at a fluence below 4.28 x 10^{17} n/cm^2 (E > 1 MeV). Since the ONS nozzle forging fluence values are substantially less than the 4.28 x 10^{17} n/cm^2 threshold, the nozzle forgings would not experience neutron embrittlement and would not be limiting compared to the beltline region. This report provides further evidence that the beltline region is limiting and bounds the nozzles based on a deterministic fracture mechanics analysis and additional TWCF calculations of nozzle materials is not necessary.

Based on the previous paragraphs, it is not necessary to calculate TWCF for the nozzle-to-shell welds and nozzle forgings. The nozzle-to-shell welds and nozzle forgings are not the limiting locations within the reactor vessel and further analysis of these materials to determine the impact of different inspection intervals on the frequency of reactor vessel failure is not necessary which is consistent with the NRC staff-approved WCAP-16168-NP-A, Revision 3 methodology and the conclusions of PWROG-15109-NP-A.

Table 1: Projected Fast (E > 1 MeV) Fluence at Wetted Surface

Reactor Vessel Location	54 EFPY Fluence with MUR, n/cm^2		
	Oconee Unit 1	Oconee Unit 2	Oconee Unit 3
Inlet Nozzle Forging to Nozzle Beltline Forging Weld	1.07E+17	1.03E+17	1.08E+17
Outlet Nozzle Forging to Nozzle Beltline Forging Weld	2.39E+17	2.28E+17	2.41E+17
Inlet Nozzle	7.20E+16	6.89E+16	7.50E+16
Outlet Nozzle	1.24E+17	1.19E+17	1.26E+17

RAI 2 References

1. Westinghouse Report, WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," October 2011. (ADAMS Accession No. ML11306A084)
2. NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Summary Report," May 2006. (ADAMS Accession Nos. ML072830076, ML072830081, and ML072820691)
3. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," March 2007. (ADAMS Accession No. ML070860156)
4. PWR Owners Group Report, PWROG-15109-NP-A, Revision 0, "PWR Pressure Vessel Nozzle Appendix G Evaluation," January 2020. (ADAMS Accession No. ML20024E573)

<u>RAI 3</u>

Table 2 in RA-20-0328 states that the licensee has performed four volumetric ISI examinations of the RPV pressure retaining welds. The licensee identifies that the following numbers of indications were detected within the inner 1/10th or inner 1 inch of the RPV wall thickness.

Unit 1: 10 indications
Unit 2: 4 indications
Unit 3: 5 indications

It is not apparent whether the fourth volumetric ISI inspections of the welds containing these indications were the first inspections that revealed evidence of flaw indications or re-inspections of the welds containing the flaw indications. Additional information relative to the risk-based assessments of these flaw indications is needed to confirm that any potential growth of the flaws is bounded by the fatigue flaw growth assumptions and values used in the WCAP-16168-NP-A, Rev. 3, methodology.

<u>Request</u>

Please confirm whether the fourth volumetric ISI inspections were the first ISI inspections that detected the flaw indications and whether there is any site-specific flaw growth data for the flaw indications evaluated in Table 2 of RA-20-0328. If site-specific flaw growth data for the flaw indications is available, please identify the limiting site-specific flaw growth value that was calculated for the flaws evaluated in Table 2.

<u>Duke Energy Response to RAI 3:</u>

<u>Unit 1</u>

The ten (10) flaw indications in Table 2 of RA-20-0328 (Reference 1) are a subset of the total 39 flaw indications that were identified in the beltline and extended beltline region of the RV during the last 4^{th} Interval ISI examination. All indications were evaluated as acceptable per ASME Section XI Table IWB-3510-1 (Reference 2).

Four of the ten indications detected within the inner 1/10th or inner 1 inch of the Reactor Pressure Vessel (RPV) wall thickness evaluated in Table 2 of Relief Request RA-20-0328 were recorded during the previous 3^{rd} Interval ISI examination. All four recorded indications were characterized as fabrication flaws and were evaluated as acceptable per ASME Section XI Table IWB-3510-1. The 2012 (4^{th} Interval) inspections were the first ISI examinations that detected the remaining (6) flaw indications within the inner 1/10th or inner 1 inch of the RPV wall thickness evaluated in Table 2 of Relief Request RA-20-0328. There is no site-specific flaw growth data for any of the ten flaw indications evaluated in Table 2 of RA-20-0328, since these flaw indications are all indicative of fabrication flaws typical of small slag inclusions and all were evaluated as acceptable per ASME Section XI Table IWB-3510-1. Therefore, any potential growth of the flaws is bounded by the fatigue flaw growth assumptions and values used in the WCAP-16168-NP-A, Revision 3 (Reference 3), methodology.

Unit 2

The four (4) flaw indications in Table 2 of RA-20-0328 (Reference 1) are a subset of the total 24 flaw indications that were identified in the beltline and extended beltline region of the RV during the last 4th Interval ISI examination. All indications were evaluated as acceptable per ASME Section XI Table IWB-3510-1 (Reference 2).

One of the four indications detected within the inner 1/10th or inner 1 inch of the RPV wall thickness evaluated in Table 2 of Relief Request RA-20-0328 was recorded during the previous 3rd Interval ISI examination. The recorded indication was characterized as a fabrication flaw and was evaluated as acceptable per ASME Section XI Table IWB-3510-1. The 2013 (4th Interval) inspections were the first ISI examinations that detected the remaining (3) flaw indications within the inner 1/10th or inner 1 inch of the RPV wall thickness evaluated in Table 2 of Relief Request RA-20-0328. There is no site-specific flaw growth data for any of the four flaw indications evaluated in Table 2 of RA-20-0328, since these flaw indications are indicative of fabrication flaws typical of small slag inclusions and all were evaluated as acceptable per ASME Section XI Table IWB-3510-1. Therefore, any potential growth of the flaws is bounded by the fatigue flaw growth assumptions and values used in the WCAP-16168-NP-A, Revision 3 (Reference 3), methodology.

Unit 3

The five (5) flaw indications in Table 2 of RA-20-0328 (Reference 1) are a subset of the total 47 flaw indications that were identified in the beltline and extended beltline region of the RV during the last 4th Interval ISI examination. All indications were evaluated as acceptable per ASME Section XI Table IWB-3510-1 (Reference 2).

The 2014 inspections were the first ISI examinations that detected the (5) flaw indications within the inner 1/10th or inner 1 inch of the RPV wall thickness evaluated in Table 2 of Relief Request RA-20-0328. There is no site-specific flaw growth data for the five flaw indications evaluated in Table 2 of RA-20-0328, since these flaw indications are indicative of fabrication flaws typical of small slag inclusions and were evaluated as acceptable per ASME Section XI Table IWB-3510-1. Therefore, any potential growth of the flaws is bounded by the fatigue flaw growth assumptions and values used in the WCAP-16168-NP-A, Revision 3 (Reference 3), methodology.

RAI 3 References

1. Relief Request RA-20-0328, “Request for Alternative for Implementation of Extended Reactor Vessel Inservice Inspection Intervals for Oconee Units 1, 2, and 3,” dated January 19, 2021 (ADAMS Accession No. ML21019A276).

2. ASME Boiler and Pressure Vessel Code, Section XI, 2007 Edition through 2008 Addenda, American Society of Mechanical Engineers, New York.

3. Westinghouse Report, WCAP-16168-NP-A, Revision 3, “Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval,” October 2011. (ADAMS) Accession No. ML11306A084)

<u>RAI 4</u>

Attachment 1 to the submittal showed a calculation summary that indicates that crack growth based on design transients (for 20 years) was bounded by crack growth based on 12 heat up/cool down cycles per year (for 20 years). Attachment 1 further specified cooldown, power change, and power loading as the top three design transients which contributed to ~90% of crack growth. However, the number of cycles assumed for the design transients was not stated.

Additional information on the design transients contributing most to crack growth is necessary to determine that the crack growth based on design transients is bounded by the crack growth based on the postulated 12 heat up/cool down cycles per year (Design Transients 1A and 1B respectively).

<u>Request</u>

Please identify the number of cycles assumed for Design Transients 1B (cooldown 8% to 0%), 2A (power change (0% to 15%), 2B (power change (15% to 0%), and 3 (power loading 8% to 100%).

<u>Duke Energy Response to RAI 4:</u>

See Enclosure 2 (Proprietary) and Enclosure 3 (Non-Proprietary) for Duke Energy's response to RAI 4. An affidavit pursuant to 10 CFR 2.390 is provided in Enclosure 4.

Enclosure 2

Proprietary Response to RAI 4

Framatome Document Number

ANP-3952, Revision 0, Q1P, Revision 0:

Response to NRC Request for Additional Information Related to Duke Energy's Proposed Alternative Request No. RA-20-0328, for Deferral of RPV ISI Volumetric Examination Until the 6th 10-year ISI Interval for Oconee Units 1, 2 and 3 – (PWROG-20003-NP Rev. 1)

Enclosure 3

Non-Proprietary Response to RAI 4

Framatome Document Number

ANP-3952, Revision 0, Q1NP, Revision 0:

Response to NRC Request for Additional Information Related to Duke Energy's Proposed Alternative Request No. RA-20-0328, for Deferral of RPV ISI Volumetric Examination Until the 6th 10-year ISI Interval for Oconee Units 1, 2 and 3 – (PWROG-20003-NP Rev. 1)

ANP-3952
Revision 0, Q1NP
Revision 0

Response to NRC Request for Additional Information Related to Duke Energy's Proposed Alternative Request No. RA-20-0328, for Deferral of RPV ISI Volumetric Examination Until the 6th 10-year ISI Interval for Oconee Units 1, 2 and 3 – (PWROG-20003-NP Rev. 1)

Technical Report

August 2021

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

Nomenclature

Acronym	Definition
ASME	American Society of the Mechanical Engineers
NRC	Nuclear Regulatory Commission
ONS	Oconee Nuclear Station
PWROG	Pressurized Water Reactor Owners Group
RAI	Request for Additional Information
RPV	Reactor Pressure Vessel
SLR	Subsequent License Renewal
UFSAR	Updated Final Safety Analysis Report
US	United States

1.0 INTRODUCTION

Duke Energy Carolinas (Duke Energy, or the licensee) provided a request for an alternative from the requirements of the American Society of the Mechanical Engineers (ASME) Boiler and Pressure Vessel, Division 1, Section XI (henceforth ASME Section XI) for the Oconee Nuclear Station (ONS) Units 1, 2, and 3 to the United States (US) Nuclear Regulatory Commission (NRC) staff in Reference 1. Duke Energy's proposed alternative to the Code in Relief Request No. RA-20-0328, requests NRC staff authorization to eliminate the performance of the inservice inspection (ISI) volumetric examinations that are required to be performed on pressure retaining welds in the heads, flanges, and shells of the reactor pressure vessel (RPV) and on associated RPV-to-nozzle welds and nozzle inside radius locations (i.e., ASME Code Section XI Category B-A and B-D required examinations) during the ASME-defined fifth 10-Year ISI interval for Oconee Units 1, 2 and 3. The licensee requests staff authorization to defer the performance of these volumetric inspections until the sixth 10-year ISI interval for each unit, by no later than 2034, based on a risk-informed methodology in WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," (Reference 2).

Duke Energy's submittal is based on the Pressurized Water Reactor Owners Group (PWROG) Non-Proprietary Report PWROG-20003-NP Revision 1, "Proposed 10 CFR 50.55a Relief Requests for Implementation of Extended Reactor Intervals for Oconee Units 1, 2 and 3," (Reference 3). This report was performed jointly by Westinghouse who performed the probabilistic fracture mechanics analysis, and Framatome Inc. who performed the deterministic fatigue crack growth analysis (Attachment 1 in References 1 and 2). The US NRC has issued Requests for Additional Information (RAIs) on this submittal in Reference 4. This report provides Framatome Inc.'s response to RAI 4.

2.0 REQUESTS FOR ADDITIONAL INFORMATION AND RESPONSES

The NRC RAI is reproduced from Reference 4 in Section 2.1. The Framatome Inc. response is provided in Section 2.1.1.

2.1 *RAI 4*

Attachment 1 to the submittal showed a calculation summary that indicates that crack growth based on design transients (for 20 years) was bounded by crack growth based on 12 heat up/cool down cycles per year (for 20 years). Attachment 1 further specified cooldown, power change, and power loading as the top three design transients which contributed to ~90% of crack growth. However, the number of cycles assumed for the design transients was not stated.

Additional information on the design transients contributing most to crack growth is necessary to determine that the crack growth based on design transients is bounded by the crack growth based on the postulated 12 heat up/cool down cycles per year (Design Transients 1A and 1B respectively).

<u>Request</u>

Please identify the number of cycles assumed for Design Transients 1B (cooldown 8% to 0%), 2A (power change (0% to 15%), 2B (power change (15% to 0%), and 3 (power loading 8% to 100%).

2.1.1 Response to RAI-4

The number of cycles assumed for the 20-year period for Design Transient 1B (cooldown 8% to 0%), 2A (power change (0% to 15%)), and 2B (power change (15% to 0%)) are [ ] cycles, [ ] cycles and [ ] cycles, respectively. The cycles associated with the above transients represent a conservative estimate over the 20 year interval based on the 80 year projected number of cycles that are provided in the Oconee Nuclear Station, Units 1, 2, and 3 – Subsequent License Renewal Application (Accession Number ML21558A193, Reference 5), Part 1 of 8 (Accession Number ML21558A194, Reference 6), Table 4.3.1.1 (pdf Page 1978). There are no 80-year projected number of cycles for transient 3 as this transient is excluded from logging.

For Design Transient 3 (power loading (8% to 100%)), the number of transient cycles assumed for the 20-year period are 4,500 cycles. The cycles associated with this transient is consistent with Table 5-2 of the Updated Final Safety Analysis Report (UFSAR, Reference 7) for the Oconee Nuclear Station that reports the design basis cycles for Design Transient 3 as 18,000 cycles. These cycles (18,000) are assumed to be conservatively applicable for the 80-year SLR period.

3.0 REFERENCES

1. Duke Energy letter to the U.S. NRC, Serial: RA-20-0328, "Request for Alternative for Implementation of Extended Reactor Vessel Inservice Inspection Intervals for Oconee Units 1, 2, and 3," dated January 19, 2021 (ADAMS Accession No. ML21019A276).
2. Westinghouse Non-Proprietary Class 3 Report WCAP-16168-NP-A Revision 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," October 2011 (ADAMS Accession No. ML11306A084).
3. PWROG Non-Proprietary Report, PWROG-20003-NP Revision 1, "Proposed 10 CFR 50.55a Relief Requests for Implementation of Extended Reactor Vessel Inservice Inspection Intervals for Oconee Units 1, 2 and 3," November 2020.
4. Request for Additional Information by the Office of Nuclear Reactor Regulation Proposed Alternative Request No. RA-20-0328 Regarding Fifth and Sixth Inservice Inspection Program Intervals Oconee Units 1, 2, and 3 Duke Energy Carolinas, LLC Docket Nos. 50-269, 50-270, 50-287 EPID: L-2021-LLR-0004, July 8, 2021.
5. Oconee Nuclear Station, Units 1, 2, and 3, Application for Subsequent Renewed Operating Licenses (Accession Number ML21158A193).
6. Oconee Nuclear Station, Units 1, 2, and 3, Application for Subsequent Renewed Operating Licenses, Part 1 of 8 (2504 pages, Accession Number ML21158A194).
7. Oconee Nuclear Station, Updated Final Safety Analysis Report (UFSAR), Chapter 5, Table 5-2, "Transient Cycles for RCS Components Except Pressurizer Surge Line," Revision 27, dated 31 December 2004.

Enclosure 4

Affidavit Pursuant to 10 CFR 2.390

A F F I D A V I T

1. My name is Philip A. Opsal. I am Manager, Product Licensing for Framatome Inc. (formally known as AREVA Inc.), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the following document referred to herein as "Document":

> Framatome Document ANP-3952, Revision 0, Q1P, Revision 0, "Response to NRC Request for Additional Information Related to Duke Energy's Proposed Alternative Request No. RA-20-0328, for Deferral of RPV ISI Volumetric Examination Until the 6th 10-year ISI Interval for Oconee Units 1, 2 and 3 – (PWROG-20003-NP Rev. 1), Technical Report

Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be

withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) “Trade secrets and commercial or financial information.”

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome’s research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in these Documents is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome’s policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on July 29, 2021.

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Philip A. Opsal