OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (ONS)
SUBSEQUENT LICENSE RENEWAL APPLICATION (SLRA)
SAFETY REVIEW

REQUESTS FOR CONFIRMATION OF INFORMATION – SET #1

Regulatory Basis:

Part 54 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," is designed to elicit application information that will enable the U.S. Nuclear Regulatory Commission (NRC) staff to perform an adequate safety review and the Commission to make the necessary findings. Reliability of application information is important and advanced by requirements that license applications be submitted in writing under oath or affirmation and that information provided to the NRC by a license renewal applicant or required to be maintained by NRC regulations be complete and accurate in all material respects. Information that must be submitted in writing under oath or affirmation includes the technical information required under 10 CFR 54.21(a) related to assessment of the aging effects on structures, systems, and components subject to an aging management review. Thus, both the general submission requirements for license renewal applications and the specific technical application information requirements require that submission of information material to NRC's safety findings (see 10 CFR 54.29 standards for issuance of a renewed license) be submitted by an applicant as part of the application.

Background:

By letter dated June 7, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21158A193), as supplemented by letter dated October 22, 2021 (ADAMS Accession No. ML21295A035), Duke Energy Carolinas, LLC (Duke Energy) submitted to the U.S. Nuclear Regulatory Commission (NRC or staff) an application to renew the Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station (ONS), Units 1, 2, and 3. Duke Energy submitted the application pursuant to 10 CFR Parts 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," for subsequent license renewal.

Between July 26 and October 8, 2021, the NRC staff conducted audits of Duke Energy's records to confirm information submitted in the ONS subsequent license renewal applicaiotn.

Request:

During the audit, the staff reviewed several documents that contain information which will likely be used in conclusions documented in the Safety Evaluation Report (SER). To the best of the staff's knowledge, this information is not on the docket. Any information used to reach a conclusion in the SER must be included on the docket by the applicant. We request that you submit confirmation that the information gathered from the documents and listed below is correct or provide the associated corrected information.

Requests for Confirmation of Information (RCIs) – SET #1

RCI No.	Description	Duke Energy's Response
B2.1.3-A	Based on the review of SLRA Section B2.1.3 and the audit of the applicant's breakout question responses posted in the ePortal, the staff needs confirmation of the results of the volumetric examinations in accordance with the ASME Code, Section XI, Table IWB-2500-1 for the 60 reactor closure head studs (Examination Category B-G-1, Item No. B6.20) and the 60 threads-in-flange (Examination Category B-G-1, Item No. B6.40) of each Oconee Nuclear Station (ONS) unit from the last required examinations performed for the unit. Confirm that there were no relevant indications or issues identified during the last volumetric examinations performed, as required by the ASME Code, Section XI, Table IWB-2500-1, for the 60 reactor closure head studs (Examination Category B-G-1, Item No. B6.20) and the 60 threads-in-flange (Examination Category B-G-1, Item No. B6.40) of each ONS unit.	
B2.1.3-B	Based on the review of SLRA Section B2.1.3 and the audit of the applicant's aging management program basis document SLR-ONS-AMPR-XI.M3, Rev. 1, the staff noted that the maximum R_C hardness value for the ONS Unit 1 reactor closure head studs is 36 R_C, which is less than the value of 41 R_C below which stress corrosion cracking does not present a concern as reported in SLRA Section B2.1.3. The staff noted that the document does not have the hardness value for ONS Unit 2. The staff also noted that some of the ultimate strength data (as reported in SLR-ONS-AMPR-XI.M3, Rev. 1) of the material heat applicable to the ONS Unit 1 reactor closure head studs were greater than the 170 ksi recommended in the GALL-SLR report. Similarly, some of the ultimate strength data of the material heat applicable to the ONS Unit 2 reactor closure head studs were greater than 170 ksi. Since ultimate strength correlates to hardness, and both material heats of the ONS Units 1 and 2 reactor closure head studs have ultimate strength data greater than 170 ksi, the maximum R_C value of the ONS Unit 2 reactor closure head studs should also be less than 41 R_C below which stress corrosion cracking does not present a concern. Confirm that the maximum R_C hardness value for the ONS Unit 2 reactor closure head studs is less than the value of 41 R_C below which stress corrosion cracking does not present a concern.	
B2.1.34-A	In the SLRA Section B2.1.34, aging management program (AMP) enhancement 14 states that degradation of piles and sheeting are accepted by engineering evaluation or subject to corrective actions, and AMP operating experience 5 states that the sheet piles at the exterior of the condenser circulating water intake structure are within the scope of the inspections performed under this AMP. The staff found that the sheet piles are included as an AMR line item in Table	

	3.5.2-23, but the staff could not locate AMR line item for sheeting. SLRA Table 3.5.1, item 3.5.1-062 states that ONS has no wooden piles; sheeting is in the scope of subsequent license renewal. Confirm that there is no in-scope sheeting at Oconee Nuclear Station, Units 1, 2, and 3.	
4.6.3-A	Based on the review of SLRA Section 4.6.3, the staff noted that footnote 3 of SLRA Table 4.6.3-1 states, “These governing transients include seismic loads.” Based on audit review of Appendix B of SLR-ONS-TLAA-0300, the staff noted that 5 operating basis earthquake (OBE) events of 9 cycles each and 3 OBE events of 9 cycles each were included in the fatigue usage evaluations for the main steam penetration and main feedwater penetration, respectively. Confirm that the fatigue usage evaluation load combinations included 5 allowable OBE events of 9 cycles each for the main steam penetration evaluation, and 3 allowable OBE events of 9 cycles each for the main feedwater penetration evaluation.	
B2.1.8-A	During its audit, the staff reviewed “FAC [Flow-Accelerated Corrosion] Breakout Questions and Responses 9.13.21,” “ONS SLRA FAC Breakout Session Followup,” and “Follow-up Breakout Questions - TRP 17 - FAC (TerryYoder) with ONS Responses Rev 1.” The staff noted that the Software Quality Assurance (SQA) process includes validation and verification for all new software and any changes to existing software (Procedures AD-IT-ALL-0002, “Software Quality Assurance (SQA) Program Administration,” and AD-EG-ALL-1110, “Design Review Requirements”), and will continue to be performed for FAC software through the subsequent period of extended operation as part of the SQA process. The staff noted that software error notification is also addressed in the SQA process (Procedure AD-IT-ALL-0002). The staff noted that the software supplier and industry support organizations provide error notification. In addition, the staff noted that FAC software error notification will continue through the subsequent period of extended operation. Confirm that validation and verification and error notification for FAC software will continue to be performed during the subsequent period of extended operation as part of the SQA process, including error notification from software supplier and industry support organizations.	
B2.1.15-A	Based on the review of Revision 1 of SLR-ONS-AMPR-XI.M26, “Fire Protection AMP [Aging Management Program] Evaluation Report,” the staff noted that inspection results not meeting acceptance criteria are referred to the Oconee Corrective Action Program where the results are subject to trending. During its audit, the staff reviewed “Breakout Questions - TRP 26 - Fire Protection (TerryGavula)_Final responses for ePortal 9-16” and	

	noted that inspection results are provided to the Fire Protection Engineer to review for degradation trends. In addition to trending the inspection results not meeting acceptance criteria that are entered into the Corrective Action Program, confirm that the Oconee Fire Protection program also trends inspection results not entered in the Corrective Action Program to provide for timely detection of aging effects.	
B2.1.16-A	Based on the review of Revision 1 of SLR-ONS-AMPR-XI.M27, “Fire Water System AMP Evaluation Report,” the staff noted that results not meeting acceptance criteria are addressed in the Oconee Corrective Action Program. In addition, the staff noted that Revision 1 of SLR-ONS-AMPR-XI.M27 states that results of flow testing and ultrasonic testing are provided to engineering for evaluation and trending. During its audit, the staff reviewed “Breakout Questions - TRP 27 - Fire Water System (TerryGavula)_Final Responses for ePortal_9-16” and “Follow-up Breakout Questions - TRP 27 - FWS (Terry)_Final Responses for ePortal_9-28,” and noted that results of flow testing and ultrasonic testing are provided to engineering for trending outside of the corrective action program whether or not acceptance criteria are met. In addition, the staff noted that unexpected flushing results are entered into the corrective action program for evaluation and trending. Confirm the following: (1) the Oconee Fire Water System program trends flow test and ultrasonic test results not entered into the Corrective Action Program in addition to trending flow test and ultrasonic test results not meeting acceptance criteria that are entered into the Corrective Action Program, and (2) all unexpected results of flushes are entered into the Corrective Action Program (all flush result trending is performed under the Corrective Action Program).	
3.5.2.2.2.6-A	SLRA Section 3.5.2.2.2.6, “Reactor Vessel [RV] Support Steel Evaluation,” discusses the reactor vessel (RV) structural support assembly grout. Confirm that the RV structural support assembly grout is in scope of SLR and subject to aging management review (AMR).	
3.5.2.2.2.6-B	Based on reviews of SLRA Section 3.5.2.2.1.2, “Reduction of Strength and Modulus Due to Elevated Temperature,” dated June 7, 2021, UFSAR Section 3.8.1.1, “Description of the Containment,” dated December 31, 2019, and audit review of OS-160, “Specification for Concrete for the Reactor Building, Duke Power Company 1-3,” dated April 5, 1973, the ONS concrete “is made with crushed marble aggregate obtained from Blacksburg, South Carolina … The design strengths are 5000 psi at 28 days…”	

	Confirm that the reactor vessel support pedestal and primary shield wall (PSW) concretes have the composition and design strength as described in SLRA Section 3.5.2.2.1.2, UFSAR Section 3.8.1.1, and OS-160.	
3.5.2.2.2.6-C	Based on the audit review of Oconee DWG No. O-69A, "Reactor Building Primary &Secondary Shield E[W]alls Plan at El. 777'+6" & 802'+0" Concrete," Revision 14, and photos provided on the ePortal in response to breakout questions, the staff noted that the PSW liner at the reactor air cavity is made of corrugated steel and anchored to the PSW with 286 3/8"X6" Nelson Studs. Confirm that the current condition of the PSW liner is anchored to the concrete with 286 3/8"X6" Nelson Studs and is not degraded.	
3.5.2.2.2.6-D	The audit review of ASME Section XI, Subsection IWF InService inspection (ISI) results for the Fourth 10-year Inspection Interval of the RV support skirt and anchorage, indicates that although the noted ISIs addressed the same inspections areas in all three ONS Units in similar environments, the examination results differed. The disparate results were due to changes made in the examination coverage requirements of ONS NDE-91, "NDE Procedures Manual, Volume 1, Reporting Coverage During PreService and InService Inspection," Revision 7, dated 2011, during the course of the ISIs, based on an NRC approved relief. Confirm that Oconee's ASME Section XI, Subsection IWF RV support skirt and anchorage disparate ISI results were solely due to changes made in its NDE-91, Revision 7, inspection procedure acceptance criteria consistent with NRC approved relief for extent of inspection coverage.	
3.5.2.2.2.6-E	SLRA Section 3.5.2.2.2.6, "Reactor Vessel Support Skirt and Reactor Cavity Configuration" states "[t]he reactor cavity serves as a biological shield wall." Confirm that: a) The reactor cavity concrete serves as a biological shield wall (aka PSW) and not the cavity itself. b) There is no other material within the reactor cavity air gap that would be considered as part of the biological shield wall.	
3.5.2.2.2.6-F	SLRA Section 3.5.2.2.2.6, "Reactor Vessel Support Skirt and Reactor Cavity Configuration" states, "The reactor consists of reinforced concrete surrounding the reactor vessel..." Confirm that there is a typographical error and the referenced statement should read "The reactor *cavity* consists of reinforced concrete surrounding the reactor vessel..."	

3.5.2.2.2.6-G	Based on the review of SLRA Section 3.5.2.2.2.6 regarding fluence and gamma dose and the audit review of the applicant's breakout presentation titled "FER Item 3.5.2.2.2.6 Meeting – 9/20/2021, 11:00 am" posted in the ePortal, the staff needs confirmation regarding the location of the 5.53E-04 dpa exposure level calculated in ANP-3898NP, Revision 0, in the RPV skirt assembly. The breakout presentation stated that the 5.53E-04 dpa exposure level was calculated at approximately 1.75 inches above the top of the transition forging-to-skirt weld. Section 9.4.4 of ANP-3898NP, Revision 0, states: "[The] projected 72 EFPY dpa is obtained by calculating dpa at the bottom of a RG 1.190 compliant RPV DORT model, which is 17.49 inches above the transition forging-to-RPV skirt weld…and extrapolating dpa from the bottom of the DORT model to the RPV skirt to transition forging weld." Confirm that the elevation of 1.75 inches above the top of the transition forging-to-skirt weld and the elevation to which the dpa from the bottom of the discrete ordinate transport (DORT) model was extrapolated are the same locations.	
3.5.2.2.2.6-H	Based on the review of Section 9.4.4.3 of ANP-3898NP, Revision 0, and the audit review of the applicant's breakout question responses posted in the ePortal, the staff noted that there were no plant-specific measured values of initial nil-ductility temperature (NDT) for the ONS Units 1, 2, and 3 RPV skirt assembly components or Charpy V-Notch absorbed energy values from which initial NDT can be derived for the components. Confirm that there were no plant-specific measured values of initial NDT for the ONS Units 1, 2, and 3 RPV skirt assembly components or Charpy V-Notch absorbed energy values from which initial NDT can be derived for the components, and that therefore generic values from NUREG-1509 and BAW-10046A, Revision 2 were used.	
3.5.2.2.2.6-I	Based on the review of Section 9.4.4.3 of ANP-3898NP, Revision 0, and the audit review of the applicant's breakout question responses posted in the ePortal, the staff noted that the sources of the initial NDT values and corresponding margins of the RPV skirt assembly components were NUREG-1509 and topical report BAW-10046A. Confirm that the initial NDT values and corresponding margins of the RPV skirt assembly components were obtained from NUREG 1509, "Radiation Effects on Reactor Pressure Vessel Supports," 1996 and BAW-10046A, "Methods of Compliance with Fracture Toughness	

	and Operational Requirements of 10 CFR 50, Appendix G," Revision 2.	
3.5.2.2.2.6-J	Based on the review of Section 9.4.1 of ANP-3898NP, Revision 0, and the audit review of document "OISI-0169.10-0050 BASIS DOC, Revision 0, and the applicant's breakout question responses posted in the ePortal, the staff noted that the transition forging-to-skirt welds (WR36) of the ONS Units 1, 2, and 3 RPV skirt assembly fall under the examination requirements of Division 1, Subsection IWF of the ASME Code, Section XI instead of Subsection IWB. Confirm that the transition forging-to-skirt welds (WR36) of the ONS Units 1, 2, and 3 RPV skirt assembly fall under the examination requirements of Division 1, Subsection IWF of the ASME Code, Section XI instead of Subsection IWB.	
3.5.2.2.2.6-K	Based on the review of Section 9.4.3 of ANP-3898NP, Revision 0, and the audit review of the applicant's breakout question responses posted in the ePortal with excerpts of proprietary Framatome Document No. 32-9311203-000, "Oconee Reactor Vessel Skirt Lowest Service Temperature," the staff noted that air circulation (stagnant or otherwise) through the 9.25-inch ventilation holes in the RPV skirt could affect the lowest service temperature value of 139.05°F determined in Section 9.4.3 of ANP-3898NP, especially in the region of the RPV skirt below the 9.25-inch ventilation holes. Confirm that air circulation through the 9.25-inch ventilation holes in the ONS 1, 2, and 3 RPV skirt was considered in determining the lowest service temperature value of 139.05°F and that there is no impact of this air circulation on the value.	