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RA-22-0040
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10 CFR 50.4
10 CFR Part 54

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, Maryland 20852

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Oconee Nuclear Station (ONS), Units 1, 2, and 3
Docket Numbers 50-269, 50-270, 50-287
Renewed License Numbers DPR-38, DPR-47, DPR-55
Subsequent License Renewal Application
Responses to NRC Request for Additional Information Set 3

References:

1. Duke Energy Letter (RA-21-0132) dated June 7, 2021, Application for Subsequent Renewed Operating Licenses, (ADAMS Accession Number ML21158A193)
2. NRC Letter dated July 22, 2021, Oconee Nuclear Station, Units 1, 2, and 3 - Determination of Acceptability and Sufficiency for Docketing, Proposed Review Schedule, and Opportunity for a Hearing Regarding Duke Energy Carolinas' Application for Subsequent License Renewal (ADAMS Accession Number ML21194A245)
3. NRC E-mail dated September 22, 2021, Oconee SLRA - Request for Additional Information B2.1.27-1 (ADAMS Accession Number ML21271A586)
4. Duke Energy Letter (RA-21-0281) dated October 22, 2021, Subsequent License Renewal Application, Response to Request for Additional Information B2.1.27-1 (ADAMS Accession Number ML21295A035)
5. NRC E-mail dated November 23, 2021, Oconee SLRA – Request for Additional Information - Set 1 and Second Round Request for Additional Information RAI B2.1.27-1a (ADAMS Accession Number ML21327A277)
6. Duke Energy Letter (RA-21-0332) dated January 7, 2022, Subsequent License Renewal Application Responses to NRC Request for Additional Information Set 1 and Second Round Request for Additional Information B2.1.27-1a (ADAMS Accession Number ML22010A129)
7. NRC E-mail dated January 11, 2022, Oconee SLRA – Request for Additional Information - Set 2 (ADAMS Accession Numbers ML22012A043 and ML22012A042)
8. Duke Energy Letter (RA-22-0036) dated February 14, 2022, Subsequent License Renewal Application Responses to NRC Request for Additional Information Set 2
9. NRC E-mail dated January 18, 2022, Oconee SLRA – Request for Additional Information Set 3 (ADAMS Accession Numbers ML22019A103 and ML22019A104)

Ladies and Gentlemen:

By letter dated June 7, 2021 (Reference 1), Duke Energy Carolinas, LLC (Duke Energy) submitted an application for the subsequent license renewal of Renewed Facility Operating License Numbers DPR-38, DPR-47, and DPR-55 for the Oconee Nuclear Station (ONS), Units 1, 2, and 3 to the U.S. Nuclear Regulatory Commission (NRC). On July 22, 2021 (Reference 2), the NRC determined that ONS subsequent license renewal application (SLRA) was acceptable and sufficient for docketing. In emails from Angela X. Wu (NRC) to Steve Snider (Duke Energy) dated September 22, 2021, November 23, 2021, and January 11, 2022 (References 3, 5, and 7), the NRC transmitted specific requests for additional information (RAI) to support completion of the Safety Review. The responses (References 4, 6 and 8) were provided to the NRC on October 22, 2021, January 7, 2022, and February 14, 2022. In an email from Angela X. Wu (NRC) to Steve Snider (Duke Energy) dated January 18, 2022 (Reference 9), the NRC transmitted RAI Set 3 also to support completion of the Safety Review. This submittal provides those responses.

Enclosure 1 contains the responses to the RAI information for Set 3. SLRA changes are provided along with the affected SLRA section(s), SLRA page number(s), and SLRA mark-ups in each affected Enclosure 1 attachment. For clarity, deletions are indicated by strikethrough and inserted text by underlined red font. This submittal contains no new or revised regulatory commitments.

Should you have any questions regarding this submittal, please contact Paul Guill at (704) 382-4753 or by email at paul.guill@duke-energy.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 21, 2022.

Sincerely,

Steven M. Snider
Site Vice President
Oconee Nuclear Station

Enclosure:

1. Responses to Requests for Additional Information Set 3

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ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 3

Enclosure 1
Responses to Requests for Additional Information Set 3

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ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 3

ATTACHMENT 1
RAI 4.7.4-1

Enclosure 1, Attachment 1

RAI 4.7.4-1:

Regulatory Basis:

Pursuant to 10 CFR 54.21(c), the SLRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant shall demonstrate that (i) the analyses remain valid for the period of extended operation; (ii) the analyses have been projected to the end of the period of extended operation; or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation

Background:

SLRA Section 4.7.4 (Leak-Before-Break of Reactor Coolant System Piping) identifies the potential for thermal aging of the RCS piping and fatigue crack growth as the aging effects that must be addressed for SLR.

In the TLAA for the second aspect of the LBB evaluation (flaw stability analysis), the applicant stated that the flaw stability analysis performed using the lower bound NUREG/CR-6177 fracture toughness curves for the period of extended operation was updated to consider the most recent test data per NUREG/CR-4513, Revision 2. The analysis was performed on the Oconee Units 2 and 3 discharge and suction nozzles of the reactor coolant pump casings which were the most limiting material and location of the 28-inch cold leg pipe. The revised analysis appears to demonstrate the acceptability of LBB for the reactor coolant pump cast austenitic stainless steel discharge and suction nozzles for SLR for 80 years of operation.

Issue:

The applicant indicated that it dispositioned the analysis identified in the TLAA in accordance with 10 CFR 54.21(c)(1)(i) ("The analyses remains valid for the period of extended operation.").

Request:

Since the flaw stability analysis was updated as identified in the TLAA, clarify whether the TLAA should be dispositioned in accordance with 10 CFR 54.21(c)(1)(ii) ("The analyses have been projected to the end of the period of extended operation.").

Response to RAI 4.7.4-1:

Since the flaw stability analysis was updated as identified in SLRA Section 4.7.4, revision of the SLRA is required to clarify that the Leak-Before-Break of Reactor Coolant System Piping TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(ii), "The analyses have been projected to the end of the period of extended operation."

Revisions to SLRA Table 4.1.4-1 and Section 4.7.4, included below, provides the clarification of the Leak-Before-Break of Reactor Coolant System Piping TLAA disposition.

SLRA Revisions:

SLRA Table 4.1.4-1 (page 4-6) is revised as follows:

Table 4.1.4-1: Oconee Time-Limited Aging Analyses Categories and Dispositions

TLAA CATEGORY	ANALYSIS	DISPOSITION (Note 1)	SECTION
OTHER PLANT SPECIFIC TLAAs	Reactor Vessel Internals - Loss of Fracture Toughness due to Neutron Embrittlement	(iii)	4.7.1.1
	Reactor Vessel Internals - Flow Induced Vibration Endurance Limits	(ii)	4.7.1.2
	Reactor Vessel Internals - Irradiation Embrittlement	(ii)	4.7.1.3
	Reactor Pressure Vessel Underclad Cracking	(ii)	4.7.2
	Reactor Coolant Pump Flywheel Fatigue Crack Growth Analyses	(ii)	4.7.3
	Leak-Before-Break Analysis for Reactor Coolant System Piping	~~(i)~~ (ii)	4.7.4
	Crane Load Cycle Limits	(i)	4.7.5

Note 1:
(i) Validation: The analyses remain valid for the SPEO.
(ii) Projection: The analyses have been projected to the end of the SPEO.
(iii) Aging Management: The effects of aging on the intended function(s) will be adequately managed for the SPEO.

SLRA Section 4.7.4 (page 4-127) is revised as follows:

TLAA Disposition: ~~10 CFR 54.21(c)(1)(i)~~ 10 CFR 54.21(c)(1)(ii):

The generic LBB analysis for the B&W operating plants reported in BAW-1847, Revision 1, ~~remains valid for~~ has been projected to the end of the SPEO in accordance with ~~10 CFR 54.21(c)(1)(i)~~ 10 CFR 54.21(c)(1)(ii) for the ONS. Reduction of fracture toughness of the reactor coolant pump discharge and suction nozzles was determined to be acceptable for SPEO based on the update to the flaw stability analysis described above. Although the LBB analysis ~~remains valid~~ has been projected to the end of the SPEO, the transient cycles will be monitored and management by the *Fatigue Monitoring* AMP (B3.1, GALL-SLR X.M1) provides additional assurance for the SPEO.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 3

ATTACHMENT 2
RAI 3.5.2.2.2.1-1

Enclosure 1, Attachment 2

RAI 3.5.2.2.2.1-1:

Regulatory Basis:

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(3) requires the applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. As described in the SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:

Per the SRP-SLR, cracking due to expansion from reaction with aggregates could occur in inaccessible concrete areas of Groups 1-5 and 7-9 structures, and Group 6 structures. The related SRP-SLR Sections 3.5.2.2.2.1, item 2 and 3.5.2.2.2.3, item 2, associated with SRP-SLR Table items 3.5-1-043 and 3.5.1-050, respectively, recommend further evaluation to determine if a plant-specific AMP is required to manage this aging effect.

The corresponding review procedures/criteria in SRP-SLR Sections 3.5.3.2.2.1, item 2 and 3.5.3.2.2.3, item 2, state that a plant-specific evaluation or program is required to manage cracking due to reaction with aggregates if (1) reactivity tests or petrographic examinations of concrete samples identify reaction with aggregates or (2) accessible concrete exhibits visual indications of aggregate reactions, such as "map" or "patterned" cracking, alkali-silica gel exudations, surface staining, expansion causing structural deformation, relative movement or displacement, or misalignment/distortion of attached components.

The "detection of aging effects" program element of GALL-SLR report AMP XI.S6 includes guidance for inaccessible, below-grade concrete structural elements. For plants with nonaggressive groundwater/soil (pH > 5.5, chlorides < 500 ppm, and sulfates < 1,500 ppm), the program recommends: (a) evaluating the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas and (b) examining representative samples of the exposed portions of the below-grade concrete, when excavated for any reason.

SLRA Section 3.5.2.2.2.1, item 2, related to AMR item 3.5.1-043 states, in part, "Inspection for Alkali-Silica Reaction [ASR] has been incorporated into the Structures Monitoring program. Structures Monitoring Inspections include examination for pattern cracking with darkened crack edges, water ingress, and misalignment inspections.… Review of plant OE has not identified evidence of ASR in below-grade inaccessible concrete areas of ONS groups 1–5 and 7–9 structures."

SLRA Section 3.5.2.2.2.3, item 2, related to AMR item 3.5.1-050 states, in part, "Plant operating experience has not identified any indications of ASR for the concrete structures at ONS, except for at the Keowee dam spillway.… Inspection for ASR has been incorporated into the Structures Monitoring program.… ONS ASR operating experience is limited to the Keowee dam spillway and is not indicative of the potential for ASR in other concrete structures at ONS. Therefore, a plant-specific AMP or plant-specific enhancements to Structures Monitoring AMP for inaccessible areas are not required to manage cracking due to expansion from reaction with aggregates."

The staff noted that ASR progression at the Keowee dam spillway is monitored by the Dam Safety Surveillance and Monitoring Plan, which is implemented to meet the guidelines of FERC and is evaluated as part of the FERC Part 12D inspection, and is outside the scope of the NRC staff's review.

SLRA Section B2.1.33 states that the Structures Monitoring AMP is an enhanced program that will be consistent with the recommendations provided in Section XI.S6, Structures Monitoring program of NUREG-2191 with the enhancements. Additionally, SLRA Table 3.5-1 states that AMR items 3.5.1-043 and 3.5.1-050 are applicable and consistent with NUREG-2191.

<u>Issue</u>:

1. Based on information in SLRA Section B2.1.33 and audit of related program basis and implementing documents, it is not clear to the staff how inspection for ASR has been incorporated into the Structures Monitoring Program.
2. It is unclear to the staff how the applicant determined that a plant-specific evaluation or program is not required.
3. The staff notes that both SLRA Sections 3.5.2.2.2.1, item 2, and 3.5.2.2.2.3, item 2, appear to lack the information for how the applicant determined that there are no indications of ASR in below-grade inaccessible concrete areas of ONS structures (except for at the Keowee dam spillway). It appears that no provisions for evaluating acceptability of inaccessible areas are in the Structures Monitoring program. Therefore, it is not clear to the staff how the referenced aging effect will be adequately managed during the SPEO in inaccessible areas consistent with GALL-SLR recommendations, as claimed. Further, the SLRA does not appear to provide any supporting justification for the statement that the ONS ASR operating experience is not indicative of the potential for ASR in other concrete structures at ONS.

<u>Request</u>:

1. Clarify how inspection for ASR has been incorporated into the SLRA B2.1.33 Structures Monitoring Program as claimed in the SLRA.
2. Clarify, with supporting information, how it was determined that a plant-specific evaluation or program is not necessary.
3. With regard to the evaluation of applicable SLRA AMR items 3.5.1-047 and 3.5.1-050, clarify how the aging effects of cracking due to expansion from the reaction with aggregates, such as ASR, will be managed during the SPEO in inaccessible areas of concrete components for Groups 1–5 and 7–9 structures, and Group 6 structures (except the concrete components managed by the FERC Inspections of the Keowee Hydro Station) consistent with the GALL-SLR recommendations as claimed in the SLRA. Additionally, provide supporting justification for the statement in the SLRA that ONS ASR operating experience is limited to the Keowee dam spillway and is not indicative of the potential for ASR in other concrete structures at ONS.
4. Update the SLRA as necessary consistent with responses to requests above.

Response to RAI 3.5.2.2.2.1-1:

Response to Request 1

ONS *Structures Monitoring* program procedures require that concrete be examined for spalling, cracking, delaminations, honeycombs, water in-leakage, chemical leaching, peeling paint, discoloration, loss of material or any other change in material property that would be noted by visual inspection. Further, Enhancement 8 for Structures Monitoring program (SLRA Appendix A2.33 and B2.1.33) states:

> *Provide inspection and evaluation criteria for structural concrete using quantitative second tier criteria of Chapter 5 in ACI 349.3R. The program will be enhanced to explicitly mention the changes in material properties of increase in porosity and permeability, and loss of strength and pattern cracking with darkened edges.*

The patterned cracking with darkened edges is an indicator of potential ASR and the program will be enhanced to explicitly add this. The enhancement identified, along with the current programmatic requirements for concrete show that the *Structures Monitoring* program will monitor for ASR.

Response to Request 2

SLRA Section 3.5.2.2.2.3 discusses ASR being identified at Keowee Hydro Station. As part of the ONS response to Information Notice (IN) 11-20, *"Concrete Degradation By Alkali-Silica Reaction"*, ONS performed a review to determine if conditions exist that promote ASR. The review included identifying the following items:

- The type of aggregate (reactive or non-reactive) used in QA concrete,
- Testing method used to determine reactivity of aggregate, and
- Patterned cracking with darkened edges indicative of ASR induced degradation.

ONS identified the type of aggregate used in QA concrete to be non-reactive per ASTM C 33. ASTM C 289 (Potential Reactivity of Aggregates) was historically used to determine reactivity of aggregate. Recently, ONS specifications have been revised to use ASTM C1260 for testing of potential reactivity of aggregates in lieu of ASTM C 289 due to potential inaccuracies present in the ASTM C 289 testing method as described in NRC Information Notice 2011-20. ONS found no visual evidence of patterned cracking with darkened edges that would imply ASR degradation, with the exception of the ASR previously identified at Keowee Hydro Station on the spillway wing walls.

ONS also had industry individuals familiar with the OE from Seabrook perform a walkdown and review photos to identify potential ASR. No indications of pattern cracking with darkened edges were identified signifying ASR degradation. It was determined that no additional testing was warranted to identify ASR.

The *Structures Monitoring* program continues to monitor plant structures for ASR. The inspections performed have not identified any locations of patterned cracking with darkened edges indicating possible ASR degradation. Thus, a plant specific program is not necessary to manage for potential ASR.

The ASR progression at the Keowee dam spillway is monitored by the Dam Safety Surveillance and Monitoring Plan, which is implemented to meet the guidelines of FERC and is evaluated as part of the FERC Part 12D inspection.

Response to Request 3

Aging effects of concrete cracking due to expansion from the reaction with aggregates is addressed between several SLRA AMR items. SLRA AMR item 3.5.1-047 is associated with an aging effect of increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation and is not of focus here. SLRA AMR item 3.5.1-043 is associated with cracking due to expansion from reaction with aggregates. Also, SLRA AMR item 3.5.1-050 is associated with inaccessible areas that may experience this aging effect. This response will address these two line items.

Response to RAI B2.1.33-3 provided in Duke Energy Letter (RA-22-0036) dated February 14, 2022, Oconee added enhancement 22 to the *Structures Monitoring* (B2.1.33) program:

> *"Revise program procedures to specify that evaluation of inspection results includes consideration of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding)."*

SLRA Section 3.5.2.2.2.1, item 3.5.1-043 and Section 3.5.2.2.2.3, item 3.5.1-050, will be revised to include acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding) and will be enhanced to describe steps taken by ONS to determine if ASR is happening to other concrete structures onsite.

In response to IN 11-20 Concrete Degradation By Alkali-Silica Reaction, the plant revisited the concrete structures on site that are exposed to water and found that there had been no visual evidence/ pattern cracking indicative of ASR. Based on the findings from the review performed in response to IN 11-20 and subsequent inspections performed that found no visual indication of evidence/ pattern cracking has been identified that is indicative of ASR, it was determined that the ONS ASR operating experience is not indicative of the potential for ASR in other concrete structures at ONS.

Response to Request 4

The SLRA is updated as shown below.

SLRA Revisions:

Section 3.5.2.2.2.1 (page 3-1314) is revised as follows:

Section 3.5.2.2.2.1

Evaluation

[3.5.1-043] – Inspection for Alkali-Silica Reaction has been incorporated, through enhancement, into the *Structures Monitoring* (B2.1.33) program. Structures Monitoring Inspections include examination for pattern cracking with darkened crack edges, water ingress, and misalignment inspections. Alkali-Silica Reaction inspection results are evaluated by the responsible engineer to identify changes that could be indicative of Alkali-Silica Reaction development (pattern cracking with darkened edges). Such indications will be entered into the corrective action program. The Structures Monitoring program also includes consideration of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding). Review of plant OE has not identified evidence of Alkali-Silica Reaction in below-grade inaccessible concrete

areas of ONS groups 1–5 and 7–9 structures. A review performed based on Information Notice 11-20, *Concrete Degradation By Alkali-Silica Reaction*, determined that the concrete structures at ONS were built with non-reactive aggregates per ASTM C33 and testing of reactivity of aggregates has been enhanced to ASTM C1260 standards. Therefore, a plant-specific AMP for inaccessible areas is not required to manage cracking due to expansion from reaction with aggregates.

Section 3.5.2.2.2.3 (pages 3-1316 and 3-1317) is revised as follows:

Section 3.5.2.2.2.3

Evaluation

[3.5.1-050] – Plant operating experience has not identified any indications of Alkali-Silica Reaction for the concrete structures at ONS, except for at the Keowee dam spillway. During 2001 and 2003, gate binding problems were experienced at Gates 4 and 1, respectively. The binding was resolved by recessing the guide plates into the piers to increase the clearances. A threshold of 0.75 inches of additional spillway end wall movement was established as part of the Potential Failure Modes Analysis, which is performed during each FERC Part 12D inspection. In 2008, the movement threshold on the spillway abutment walls was exceeded. The concrete was then evaluated through drilling, sampling, borehole imaging, compression testing, petrographic testing, in-situ stress testing, biaxial modulus testing, and stress calculation. The investigations found that no significant deterioration of the spillway concrete had occurred. As there are no reliable methods to stop or retard the Alkali-Silica Reaction process, the most viable solution was determined to be monitor and adjust the gates as needed to maintain sufficient operating clearance.

The information obtained from the Alkali-Silica Reaction evaluations has provided anticipated rates of progression that is used in conjunction with visual inspections to guide the implementation schedule for the plant modification of the gates. This is surveyed and monitored by the Dam Safety Surveillance and Monitoring Plan (DSSMP), which is utilized to meet the guidelines of FERC and is evaluated as part of the FERC Part 12D inspection. As stated in NUREG-2191 XI.S7, for dam inspection and maintenance, programs under the regulatory jurisdiction of the FERC or the U.S. Army Corps of Engineers (USACE), continued through the SPEO, are adequate for the purpose of aging management.

Inspection for indications of Alkali-Silica Reaction (pattern cracking with darkened edges) has been incorporated into the *Structures Monitoring* (B2.1.33) program. Changes that could be indicative of Alkali-Silica Reaction development will be entered into the corrective action program. Therefore, a plant-specific aging management for inaccessible areas is not required to manage cracking due to expansion from reaction with aggregates. The Structures Monitoring program also includes consideration of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding).

ONS Alkali-Silica Reaction operating experience is limited to the Keowee dam spillway and is not indicative of the potential for Alkali-Silica Reaction in other concrete structures at ONS. A review performed based on Information Notice 11-20, *Concrete Degradation By Alkali-Silica Reaction*, determined that the concrete structures at ONS were built with non-reactive aggregates per ASTM C33 and testing of reactivity of aggregates has been enhanced to ASTM C1260 standards. Therefore, a plant-specific AMP or plant-specific enhancements to *Structures Monitoring* (B2.1.33) AMP for inaccessible areas are not required to manage cracking due to expansion from reaction with aggregates.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 3

ATTACHMENT 3
RAI 3.5.2.2.2.1-2

Enclosure 1, Attachment 3

RAI 3.5.2.2.2.1-2:

Regulatory Basis:

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(3) requires the applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. As described in the SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:

Per the SRP-SLR, increase in porosity and permeability due to leaching of calcium hydroxide and carbonation could occur in below-grade inaccessible concrete areas of Groups 1-5 and 7-9 structures, and Group 6 structures. The related SRP-SLR Sections 3.5.2.2.2.1, item 4 and 3.5.2.2.2.3, item 3 associated with SRP-SLR Table 3.5-1, Items 3.5.1-047 and 3.5.1-051, respectively, recommend further evaluation of this aging effect in inaccessible concrete areas of these groups of structures to determine if a plant-specific program or enhancement is required to manage the aging effect. The corresponding SRP-SLR review criteria in SRP-SLR Sections 3.5.2.2.2.1, item 4 and 3.5.2.2.2.3, item 3 state, in part, that a plant-specific program is not required for the reinforced concrete structures exposed to flowing water if (1) there is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.

The “detection of aging effects” program element of GALL-SLR report AMP XI.S6 provides recommendations for inaccessible, below-grade concrete structural elements. For plants with nonaggressive groundwater/soil (pH > 5.5, chlorides < 500 ppm, and sulfates < 1,500 ppm), the program recommends: (a) evaluating the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas and (b) examining representative samples of the exposed portions of the below-grade concrete, when excavated for any reason.

SLRA Sections 3.5.2.2.2.1, item 4, and 3.5.2.2.2.3, item 3, as amended by ONS SLRA Supplement 2 dated November 11, 2021 (ADAMS Accession Number ML21315A012), states that review of plant OE identified instances of aging effects related to increase in porosity and permeability due to leaching of calcium hydroxide and carbonation in accessible areas, however, the structural integrity of the components was not impacted, and the conditions will be monitored during future inspections. The applicant concluded that a plant-specific AMP is not required.

SLRA Section B2.1.33 states that the Structures Monitoring AMP is an enhanced program that will be consistent with the recommendations provided in Section XI.S6, Structures Monitoring program of NUREG-2191 with the enhancements. Additionally, SLRA Table 3.5-1 states that AMR items 3.5.1-047 and 3.5.1-051 are applicable and consistent with NUREG-2191.

Issue:

1. It is unclear from information in the SLRA what evaluation of the observed operating experience has been performed to conclude that the structural integrity of the components was not impacted regarding operating experience of increase in porosity and permeability due to leaching of calcium hydroxide and carbonation in accessible areas, and how the evaluation determined that the

observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure in support of the conclusion made.

2. It appears that no provisions for evaluating acceptability of inaccessible areas are in the Structures Monitoring program. Therefore, it is not clear to the staff how the referenced aging effect will be adequately managed during the SPEO in inaccessible areas consistent with GALL-SLR recommendations, as claimed.

Request:

1. Regarding SLRA Sections 3.5.2.2.2.1, item 4, and 3.5.2.2.2.3, item 3, describe what evaluation of the observed operating experience has been performed to conclude that the structural integrity of the components was not impacted, and how the evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure to conclude that a plant-specific AMP is not necessary.
2. With regard to the evaluation of applicable SLRA AMR items 3.5.1-047 and 3.5.1-051, explain or clarify how the aging effects of increase in porosity and permeability, and loss of strength due to leaching of calcium hydroxide and carbonation will be managed during the SPEO in inaccessible areas of concrete components for Groups 1–5 and 7–9 structures, and Group 6 structures (except the concrete components managed by the FERC Inspections of the Keowee Hydro Station), consistent with the GALL-SLR recommendations as claimed. Update the SLRA accordingly, as necessary.

Response to RAI 3.5.2.2.2.1-2:

Response to Request 1

As a result of the five-year civil inspections performed in accordance with the *Structures Monitoring* program, OE related to leaching of calcium hydroxide and carbonation was found in several locations and evaluated for increase in porosity and permeability and loss of strength.

In order for observed leaching to impact the structural integrity of components, two conditions are necessary: (1) the structural concrete members must be exposed to flowing liquid, ponding, or hydraulic pressure in order to be susceptible to leaching of calcium hydroxide, and (2) the concrete must contain defects such as cracks, voids, and improperly prepared construction joints in order to provide a mechanism for entry of water.

The evaluations of the observed concrete leaching at Oconee determined that the locations were not exposed to flowing liquid, ponding, or hydraulic pressure. Additionally, the concrete is a dense, low permeability concrete less susceptible to leaching. Thus, the evaluation determined that the observed leaching did not meet the criteria necessary to adversely impact the structural integrity and therefore would not result in a loss of intended function of the associated concrete structure.

To reinforce the results of this initial evaluation, a subsequent five-year *Structures Monitoring* program inspection of the observed areas with leaching did not find any change in the condition from the prior inspection. Thus, a plant specific program is not necessary to manage for potential increase in porosity and permeability due to leaching of calcium hydroxide and carbonation.

Response to Request 2

An enhancement was added to the *Structures Monitoring* program in response to RAI B2.1.33-3 that revises the program procedures to specify that evaluation of inspection results includes consideration of

the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding).

This enhancement ensures that the aging effects of increase in porosity and permeability, and loss of strength due to leaching of calcium hydroxide and carbonation will be managed during the SPEO in inaccessible areas of concrete components for Groups 1–5 and 7–9 structures, and Group 6 structures (except the concrete components managed by the FERC Inspections of the Keowee Hydro Station) are consistent with the GALL-SLR. The condition of accessible concrete is used as an indicator for the condition of the inaccessible concrete and provides reasonable assurance that degradation of inaccessible concrete will be detected before a loss of intended function.

SLRA Section 3.5.2.2.2.1, item 3.5.1-047 and Section 3.5.2.2.2.3, item 3.5.1-051, will be revised to include acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding). The sections will also be updated to specify that the *Structures Monitoring* program manages aging effects related to leaching of calcium hydroxide and carbonation through an enhancement.

SLRA Revisions:

SLRA Section 3.5.2.2.2.1 (page 3-1314, as revised in Supplement 2 [ADAMS Accession Number ML21315A012]), is revised as follows:

Evaluation

[3.5.1-047]– Reinforced concrete structures at ONS were designed, constructed, and inspected in accordance with ACI and ASTM standards, which provide for a good quality, dense, well- cured, and low permeability concrete. Procedural controls ensured quality throughout the batching, mixing, and placement processes. Review of plant OE identified instances of aging effects related to increase in porosity and permeability due to leaching of calcium hydroxide and carbonation in accessible areas, however it was determined the structural integrity of the components was not impacted and the conditions will be monitored during future inspections. The *Structures Monitoring* (B2.1.33) program ~~confirms the absence of~~ manages the aging effects related to leaching of calcium hydroxide and carbonation. The Structures Monitoring program also includes consideration of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding). Therefore, a plant-specific AMP or plant-specific enhancements to the *Structures Monitoring* (B2.1.33) program for inaccessible areas to manage the aging effects of increase in porosity and permeability due to leaching of calcium hydroxide and carbonation are not required in below-grade inaccessible concrete areas of ONS groups 1–5 and 7–9 structures.

SLRA Section 3.5.2.2.2.3 (page 3-1317, as revised in Supplement 2 [ADAMS Accession Number ML21315A012]), is revised as follows:

Evaluation

[3.5.1-051] – Reinforced concrete structures at ONS were designed, constructed, and inspected in accordance with ACI and ASTM standards, which provide for a good quality, dense, well- cured, and

low permeability concrete. Procedural controls ensured quality throughout the batching, mixing, and placement processes. The *Structures Monitoring* (B2.1.33) program, which includes Group 6 structures, identifies and manages cracks in the concrete structures and manages the aging effects related to leaching of calcium hydroxide and carbonation. The Structures Monitoring program also includes consideration of the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas (e.g. exposed to groundwater/soil environment, structural components covered by metal siding). Review of plant OE identified instances of aging effects related to increase in porosity and permeability due to leaching of calcium hydroxide and carbonation in accessible areas, however it was determined the structural integrity of the components was not impacted and the conditions will be monitored during future inspections. Therefore, a plant-specific AMP or plant-specific enhancements to the *Structures Monitoring* (B2.1.33) program for inaccessible areas to manage the aging effects of increase in porosity and permeability due to leaching of calcium hydroxide and carbonation are not required in Group 6 structures.

ENCLOSURE 1

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
SUBSEQUENT LICENSE RENEWAL APPLICATION
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
SET 3

ATTACHMENT 4
RAI 3.5.2.2.2.2-1

Enclosure 1, Attachment 4

RAI 3.5.2.2.2.2-1:

Regulatory Basis:

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(3) requires the applicant to demonstrate that the effects of aging for each structure and component identified in 10 CFR 54.21(a)(1) will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. As described in the SRP-SLR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-SLR Report when evaluation of the matter in the GALL-SLR Report applies to the plant.

Background:

SLRA Section 3.5.2.2.2.2 states that the operating experience (OE) “identified no issues related to elevated temperatures affecting concrete structures. SLRA Section 3.5.2.2.2.2 also states, “The ONS spent fuel pool (SFP) cooling systems are designed to maintain the pool bulk water temperature below 150 °F for normal heat loads. The spent fuel pools have an actual operating limit of 205 °F (abnormal case). Analyses have been performed to ensure that seismic and structural integrity of the pool liner, supporting concrete, and fuel racks are not compromised at this temperature limit.” The SLRA concludes that a “plant-specific aging management program to manage the effects of reduction of strength and modulus due to elevated temperature is not required.”

UFSAR Section 3.8.4.4, states that the SFP walls were analyzed for thermal loads in accordance with methods presented in ACI 505. Under normal conditions, the interior wall temperature was 150 °F and the maximum calculated thermal stress was 996 psi for concrete and 11,410 psi for reinforcing steel. After prolonged outage of the cooling system, the interior wall temperature could reach 212 °F and the maximum calculated thermal stress was 1681 psi for concrete and 25,600 psi for reinforcing steel.

Calculation OSC-929 on ePortal, “Reanalysis of Unit 1+2 Fuel Pool for Poison Spent Fuel Rack Loadings,” Revision 3, indicates that the maximum pool temperature is 150 °F at normal operation condition, and maximum pool temperature is 255 °F under an accident condition with poison racks.

Calculation OSC-1028 on ePortal, “Reanalysis of Unit 3 Spent Fuel Pool Liner Plate for Poison Racks,” Revision 8, indicates the maximum pool temperature is 150 °F for normal operation condition, and maximum pool temperature is 255 °F for case III.

Section 3.5.2.2.2.2 of the SRP-SLR states concrete temperatures under normal operation or any other long-term period should not exceed 66 °C (150 °F) except for local areas, which are allowed to have increased temperatures not to exceed 93 °C (200 °F) and recommends a further evaluation and a plant-specific program if any portion of the safety-related and other concrete structures exceeds the specified temperature limits. It concludes that higher temperatures may be allowed if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.

Issue:

The SLRA, UFSAR, and calculations OSC-929 and OSC-1028 present different maximum pool temperatures for abnormal cases at 205 °F, 212 °F, and 255 °F, respectively. It is unclear what the maximum pool temperatures are for abnormal cases at ONS Units 1, 2 and 3.

It is unclear how the code acceptance criteria were met for these load combinations including thermal load used for the SFP walls/supporting concrete, pool liner, and fuel racks at the maximum SFP water temperature for abnormal cases at ONS Units 1, 2 and 3.

Request:

1. Clarify the maximum pool temperatures for abnormal cases at ONS Units 1, 2 and 3.
2. Describe, with quantitative results and code allowable limits, how the structural acceptance criteria were met for the SFP walls/supporting concrete for the controlling load combinations including thermal loads at the maximum SFP water temperatures for abnormal cases at ONS Units 1, 2 and 3.
3. Describe, with quantitative results and code allowable limits, how the structural acceptance criteria were met for the pool liner for the controlling load combinations including thermal loads at the maximum SFP water temperatures for abnormal cases at ONS Units 1, 2 and 3.
4. Describe, with quantitative results and code allowable limits, how the structural acceptance criteria were met for the fuel racks for the controlling load combinations including thermal loads at the maximum SFP water temperatures for abnormal cases at ONS Units 1, 2 and 3.
5. Update SLRA Section 3.5.2.2.2.2 accordingly based on the responses above.

Response to RAI 3.5.2.2.2.2-1:

Response to Request 1:

During normal operations, operating procedures maintain spent fuel pool temperature less than 150°F. If the spent fuel pool temperature exceeds limits, operating procedures requires restoration of cooling and if necessary, establishment of supplemental spent fuel pool cooling. The spent fuel pool temperature under normal operation is maintained less than 150°F.

Spent fuel pool temperatures (greater than 150°F) may occur either for short durations which are managed by operating procedures or infrequently during accident conditions (i.e. loss of spent fuel pool cooling and abnormal heat load situations as defined in UFSAR 9.1.3.1) which are addressed in the design. These various other maximum spent fuel pool temperatures referenced in the SLRA, UFSAR and design calculations, such as in Calculations OSC-0929, OSC-1028 and UFSAR Section 3.8.4.4, represent design criteria and analytical assumptions used in the design of various systems, structures, components and in accident analysis. These maximum temperature conditions do not define the environment that could lead to normal, long term aging effects of the spent fuel pool racks, liner and concrete.

Response to Requests 2, 3, and 4

Conditions as a result of an accident/event do not define the environment that could lead to long term aging effects of the spent fuel pool racks, liner and concrete. That environment is defined here as the pool bulk temperatures which are actively maintained by plant operations to below 150°F.

As described in NEI 17-01, Section 4.2.1, *"To determine [aging effects requiring management], the applicant should consider and address the materials, environment, and stressors that are associated with each [structure or component] or commodity group under review."* The central issue of this RAI, as it pertains to the detrimental effects of aging as discussed in SLRA Section 3.5.2.2.2.2, is whether the spent fuel pool concrete temperature is maintained above 150°F. As established in the response to Request 1, the spent fuel pool temperature is maintained below 150°F and therefore, a plant-specific evaluation of the concrete is not required.

Response to Request 5

Based on the above response to Request 1, SLRA Section 3.5.2.2.2.2 requires an update. The update will remove unnecessary information and provide additional clarification to demonstrate that the spent fuel pool concrete is maintained below 150°F.

SLRA Revisions:

SLRA Section 3.5.2.2.2.2 (page 3-1315) is revised as follows:

3.5.2.2.2.2 Reduction of Strength and Modulus Due to Elevated Temperature

Evaluation

[3.5.1-048] – UFSAR Section 9.1.3 states that the ONS spent fuel pool cooling systems are designed to maintain the pool bulk water temperature below 150°F for normal heat loads. This pool bulk water temperature defines the environment for consideration of long term aging of the spent fuel pool components, including the liner and concrete. The normal practice at Oconee is to limit the maximum pool temperature to 150°F. This limit is maintained by plant procedures. When a loss of spent fuel pool cooling occurs, procedures require restoring cooling and, if necessary, establishing supplemental cooling. Spent fuel pool temperature is monitored in the control room and alarms are initiated when the pool temperature reaches 150°F. Therefore, the normal concrete temperature will be maintained below 150°F. ~~The spent fuel pools have an actual operating limit of 205°F (abnormal case). Analyses have been performed to ensure that seismic and structural integrity of the pool liner, supporting concrete, and fuel racks are not compromised at this temperature limit.~~