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SOUTHERN COMPANY

9/18/15

Docket Nos.: 52-025
52-026

ND-15-1742
10 CFR 50.90
10 CFR 52.63

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Southern Nuclear Operating Company
Vogtle Electric Generating Plant Units 3 and 4
Request for License Amendment and Exemption:
CA04 Structural Module ITAAC Dimensions Change (LAR-15-015)

Ladies and Gentlemen:

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC), the licensee for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, requests an amendment to Combined License (COL) Numbers NPF-91 and NPF-92, for VEGP Units 3 and 4, respectively. The requested amendment requires changes to related plant-specific Tier 1 information, with corresponding changes to the associated COL Appendix C information. The amendment also requires involved changes to the Updated Final Safety Analysis Report (UFSAR) in the form of departures from the incorporated plant-specific Design Control Document (PS-DCD) Tier 2 information. Pursuant to the provisions of 10 CFR 52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is also requested for the plant-specific DCD Tier 1 material departures.

The proposed departures consist of changes to a plant-specific Tier 1 (and COL Appendix C) table and UFSAR text related to changes to the concrete wall thickness tolerances for four Nuclear Island walls.

Enclosure 1 provides the description, technical evaluation, regulatory evaluation (including the Significant Hazards Consideration determination), and environmental considerations for the proposed changes in the License Amendment Request (LAR). Enclosure 2 provides the background and supporting basis for the requested exemption. Enclosure 3 identifies the requested changes and provides markups depicting the requested changes to the plant-specific Tier 1 table and Tier 2 information.

The changes proposed in this License Amendment Request are consistent in technical content with LAR 14-07, submitted by South Carolina Electric & Gas Company (SCE&G) on September 25, 2014 as supplemented on March 13, 2015, (ADAMS Accession Nos. ML14268A388 and ML15072A306 respectively) with the exception of one alternative plant-specific change. The NRC approved the SCE&G LAR 14-07 on August 24, 2015 (ADAMS Accession No. ML15216A071).

This letter contains no regulatory commitments.

SNC requests staff approval of this license amendment and exemption by November 9, 2015, to support concrete placement from Elevation 83'-0" to Elevation 87'-6". Delayed approval of this licensing request could result in delay of the associated construction activity and subsequent dependent construction activities. SNC expects to implement the proposed amendment (through incorporation into the licensing basis documents; e.g., the UFSAR, COL Appendix C, and the plant-specific Tier 1 table) within 30 days of the approval of the requested changes.

In accordance with 10 CFR 50.91, SNC is notifying the State of Georgia of this LAR by transmitting a copy of this letter and enclosures to the designated State Official.

Should you have any questions, please contact Mr. Jason Redd at (205) 992-6435.

Mr. Brian H. Whitley states that: he is the Regulatory Affairs Director of Southern Nuclear Operating Company; he is authorized to execute this oath on behalf of Southern Nuclear Operating Company; and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

Brian H. Whitley

Brian H. Whitley

LISA MYRICK SPEARS NOTARY PUBLIC STATE OF ALABAMA

BHW/PTR/ljs

Sworn to and subscribed before me this 18th *day of* September, *2015*

Notary Public: Lisa Myrick Spears

My commission expires: June 18, 2019

Enclosures:
1) Request for License Amendment Regarding CA04 Structural Module ITAAC Dimension Change (LAR-15-015)
2) Exemption Request: CA04 Structural Module ITAAC Dimension Change (LAR-15-015)
3) Proposed Changes to the Licensing Basis Documents (LAR-15-015)

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Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-15-1742

Enclosure 1

Request for License Amendment Regarding

CA04 Structural Module ITAAC Dimension Change

(LAR-15-015)

(13 pages, including this cover page)

Table of Contents

1. Summary Description

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC), the licensee for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, requests an amendment to Combined License (COL) Numbers NPF-91 and NPF-92, for VEGP Units 3 and 4, respectively.

The proposed changes would revise the COLs by increasing the concrete wall thickness tolerances of four Nuclear Island walls identified in COL Appendix C and associated plant-specific Design Control Document (DCD) Tier 1 Table 3.3-1, "Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building." The proposed changes also include a revision of the Updated Final Safety Analysis Report (UFSAR) in the form of a departure from the incorporated plant-specific DCD Tier 2 information in Subsection 3.8.3.6.1 as discussed further below.

This enclosure requests approval of the license amendment necessary to implement the Tier 2 and COL Appendix C changes. Enclosure 2 requests the exemption necessary to implement the involved changes to the plant-specific Tier 1 information.

2. Detailed Description

This license amendment request (LAR) proposes to revise COL Appendix C and associated plant-specific DCD Tier 1 Table 3.3-1, "Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building" for the concrete thickness tolerance for four walls from ±1" to ±1-5/8". In addition, this LAR provides the associated Tier 2 changes to identify the departure from the American Concrete Institute (ACI) 117 code requirements for the four walls discussed below. The need for this proposed change was identified during a survey performed of installed modules where it was identified that the tolerance specified in COL Appendix C was not met in a portion of one wall and there were possible inconsistencies with the underlying design construction tolerances.

The nuclear island structures, consisting of the containment vessel, containment internal structures, shield building, and auxiliary building are founded on the 6-foot-thick, cast-in-place, reinforced concrete basemat foundation. The primary functions of the nuclear island structures are to provide support, protection, and separation for the seismic Category I mechanical and electrical equipment located in the nuclear island. The nuclear island structures are structurally designed to meet seismic Category I requirements as defined in Regulatory Guide 1.29.

The nuclear island structures provide protection for the safety-related equipment from the consequences of either a postulated internal or external event. The nuclear island structures are designed to withstand the effects of natural phenomena such as hurricanes, floods, tornados, tsunamis, and earthquakes without loss of capability to perform safety functions. The nuclear island structures are designed to withstand the effects of postulated internal events such as fires and flooding without loss of capability to perform safety functions.

The design and construction of the shield building is not affected by this activity. The design of the shield building structural wall modules is described in UFSAR Subsection 3.8.4.5.5

As discussed in UFSAR Subsection 3.8.3.1, the containment internal structures are those concrete and steel structures inside, but not a part of, the containment pressure boundary that

support the reactor coolant system components and related piping systems and equipment. The concrete and steel structures also provide radiation shielding. The containment internal structures consist of the primary shield wall, reactor cavity, secondary shield walls, in-containment refueling water storage tank (IRWST), refueling cavity walls, operating floor, intermediate floors, and various platforms. This activity involves structural modules found inside of containment which are shown on UFSAR Figure 3.8.3-1. Plant north on UFSAR Figure 3.8.3-1 is on the right side for the plan views.

This proposed change refers to the tolerance for the concrete wall thicknesses for the containment internal structural modules CA04, CA01, and CB65. The CA04 module forms the reactor vessel cavity, and the walls of the CA01 module comprise the central walls of the containment internal structures including the two steam generator compartments and the refueling canal. Finally, the CB65 module is used in creating the walls of the reactor coolant drain tank room (termed the RCDT Room).

Adherence to the wall thickness definitions provided in COL Appendix C Table 3.3-1, along with its associated ITAAC, provide assurance that the final plant construction complies with the design presented in the UFSAR by providing concrete thicknesses for selected walls. Note 2 to this table currently specifies that the concrete thicknesses for these walls have tolerances of ±1".

The first four walls listed in this table, each adjacent to the reactor vessel cavity, are unique in that they are formed by placing concrete between two separate modules, utilizing the fabrication and assembly, and placement tolerances from both modules. The four walls affected by the proposed change in concrete thickness tolerances are located as follows:

(1) Shield Wall between Reactor Vessel Cavity and RCDT Room. This wall has a nominal thickness of 3'-0" and is shown in more detail (but not identified) in UFSAR Figure 3.8.3-1 (Sheet 1 of 7) between the CA04 module and the CB65 module. No changes are proposed to UFSAR Figure 3.8.3-1 (Sheet 1 of 7).
(2) West Reactor Vessel Cavity Wall. This wall has a nominal thickness of 7'-6" and can be seen on UFSAR Figure 3.8.3-1 (Sheet 3 of 7) between the west side of the CA04 Module and the CA01 Module. No changes are proposed to UFSAR Figure 3.8.3-1 (Sheet 3 of 7).
(3) North Reactor Vessel Cavity Wall. This wall has a nominal thickness of 9'-0" and can be seen on UFSAR Figure 3.8.3-1 (Sheet 3 of 7) between the north side of the CA04 Module and the CA01 Module. No changes are proposed to UFSAR Figure 3.8.3-1 (Sheet 3 of 7).
(4) East Reactor Vessel Cavity Wall. This wall has a nominal thickness of 7'-6" and can be seen on UFSAR Figure 3.8.3-1 (Sheet 3 of 7) between the east side of the CA04 Module and the CA01 Module. No changes are proposed to UFSAR Figure 3.8.3-1 (Sheet 3 of 7).

As noted in UFSAR Subsection 3.8.3.6.1, structural module tolerances conform to the requirements of applicable sections of ACI 117, AWS D1.1, and AISC N690, and UFSAR Subsection 3.8.4.4.1 requires the design and analysis procedures are in accordance with ACI 349-01. The design documentation conforming to these requirements provides installation

tolerances of ±1/2" and where wall thickness measurements are not taken at internal wall module stiffeners, fabrication wall thickness tolerances of ±1/2". At locations of welded pieces, including stiffeners, and datum points, total fabrication and assembly tolerances of 1/8" are utilized. These tolerances, derived from the ACI 349-01 standards, are used in establishing the total design tolerance for the individual walls.

A review was performed of the total wall tolerances, and it was identified that the total concrete thickness tolerance after fabrication and placement exceeded the ±1" concrete thickness tolerance specified in COL Appendix C and ACI 349-01 due to the issue of thickness measurement tolerance stack-up. Furthermore, ACI 117 Section 4.5, "Deviation from Cross-Sectional Dimensions," requires that for walls of this thickness, the tolerances used shall be plus 1" and minus 3/4" (+1"/-3/4"), a tighter tolerance than specified in COL Appendix C.

Using the concrete thickness tolerances mentioned above, the following total stack-up concrete thickness tolerance was found for the CA04/CA01 wall. Measurements to check ITAAC compliance are taken at stiffener locations on CA04; at these locations, the fabrication and assembly tolerance of ±1/8" applies. However, the corresponding locations on CA01 for measuring ITAAC compliance are not located at stiffeners on the CA01 module. The CA01 measurement locations are in between stiffeners, so the fabrication and assembly tolerance of ±1/2" applies.

<u>Total Tolerance ("Stack-Up") Consideration</u>

	Fabrication + Assembly Tolerance	**Installation Tolerance**	**Total Individual Wall Tolerance**
CA04 Wall	± 1/8"	± 1/2"	± 5/8"
CA01 Wall	± 1/2"	± 1/2"	± 1"
Total CA04/CA01 Wall Tolerance	**± 5/8"**	**± 1"**	**± 1-5/8"**

By using the concrete thickness tolerances mentioned above, the following total concrete thickness tolerance was found for the CA04/CB65 wall where measurements are taken between wall module stiffeners.

	Fabrication + Assembly Tolerance	**Installation Tolerance**	**Total Individual Wall Tolerance**
CA04 Wall	± 1/8"	± 1/2"	± 5/8"
CB65 Wall	± 1/8"	± 1/2"	± 5/8"
Total CA04/CB65 Wall Tolerance	**± 1/4"**	**± 1"**	**± 1-1/4"**

After placement of the CA04 and CB65 structural modules at Vogtle Unit 3, a survey of the module indicated that the fabrication tolerances for the module were not met, exceeding the COL Appendix C tolerance in one area of approximately 60 square inches. In order to address this deviation, this LAR proposes to increase the tolerance of this wall to utilize the same tolerance of ±1-5/8" that is proposed for the CA04/CA01 walls.

Licensing Basis Change Descriptions

To address these issues, it is proposed that the concrete thickness tolerance specified in COL Appendix C and associated plant-specific DCD Tier 1 Table 3.3-1 for both the CA04/CA01 and CA04/CB65 walls be increased to ±1-5/8" to accommodate the combined fabrication and assembly tolerance plus the installation tolerance of the adjacent modules by adding an exception to Note 2 and adding one note to Table 3.3-1. Note 2 is proposed to be modified to state that the tolerance is as stated in the note except as noted elsewhere.

The added note to Table 3.3-1, labeled Note 10, is footnoted to the concrete thickness in the four affected walls, the first four items in Table 3.3-1. This note specifies that this concrete thickness has a construction tolerance of ±1-5/8".

In order to address the departure from the requirements of ACI 117 for the four walls identified in this LAR, changes to UFSAR Tier 2 Subsection 3.8.3.6.1 are proposed to identify that in locations around the reactor vessel cavity where the concrete is placed between portions of unconnected modules or between a module and a left in place form, the tolerance for the wall thickness may be increased over those in ACI 117. The proposed change also identifies that these walls have been evaluated against the requirements of ACI 349-01 reinforcement requirements.

3. Technical Evaluation

The basis for the safety classification of the AP1000 structures and associated modules is found in its conformance to industry codes and standards that are appropriate for its intended function. Specifically, the codes listed in UFSAR Subsection 3.8.3.2 detail the key requirements for containment internal structures including the three modules affected by this activity (CA04, CA01, and CB65). As stated above, UFSAR Subsection 3.8.3.6.1 specifies that the tolerance for fabrication, assembly, and installation of structural modules conforms to the requirements of ACI 117, AWS D1.1, and AISC N690, and UFSAR Subsection 3.8.4.4.1 requires that the design and analysis procedures are in accordance with ACI 349-01. For the affected walls, the concrete thickness tolerances listed in COL Appendix C Table 3.3-1 do not utilize the complete allowance specified in these codes with the exception of ACI 349-01 and ACI 117.

The proposed installation tolerance of ±1-5/8" between the CA04/CB65 wall is greater than the ±1" specified in ACI 349-01 Section 7.5.2.1 and the +1"/-3/4" found in ACI 117. To determine the impact and the margin, an assessment was performed of the affected areas. Figure 1 depicts the CA04/CB65 cross-section and shows the d value which is the distance from the extreme compression fiber to the centroid of the rebar per ACI 349-01. The d value is the governing parameter for the design in regards of determining the rebar demand and spacing.

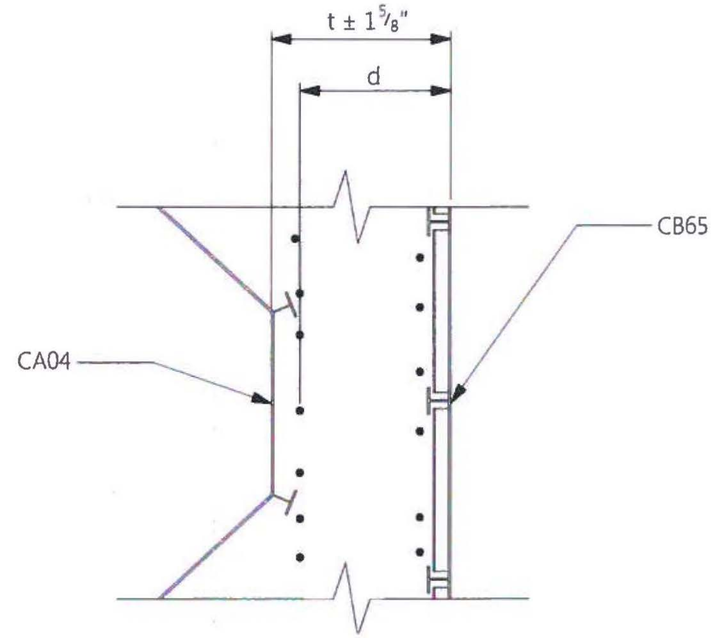

Figure 1: Concrete Cross-Section between CA04 and CB65 (not to scale)

In Figure 1, the thickness of the wall and the proposed thickness tolerance change requested for the CA04/CB65 wall is shown. For evaluation of the impact and the margin of the wall design variation, the wall thickness and corresponding d value was set to ±1-5/8", which is greater than ACI 117 tolerances on cross-sectional dimensions. As a result of the assessment, it was found that the minimum margin for vertical reinforcement is 47.9%, for horizontal reinforcement 54.8% and for shear 61.3%. Since the thickness of the wall also influences the d value, an assessment of the shear reinforcement spacing was performed which found the provided spacing sufficient.

Similarly, the CA04/CA01 walls were examined to determine the impact of increasing allowable thickness tolerance by an additional ±5/8" beyond the ACI 349-01 requirements and an additional +5/8" and -7/8" beyond the ACI 117 requirements. These walls are 7'-6" to 9'-0" thick and because of their size, are designed as mass concrete structures, resulting in an evaluation of the volume of concrete in lieu of evaluating the adequacy of the existing reinforcement. For evaluation of the impact of the thickness differences between CA01 and CA04 the variation of the concrete thickness was set to ±1-5/8". This assessment conservatively assumed the worst case tolerances in all directions of the CA04 module instead of only the affected walls. The analysis found that there was a potential for a 1.3% decrease to the area of the surrounding concrete and that this decrease is insignificant and negligible in the structural analyses.

Therefore, including consideration of the increased concrete thickness tolerance for these four walls, the functions of the walls are shown to be maintained and the walls continue to meet their design functions. Consequently, the deviations from the ACI 349-01 and ACI 117 codes are acceptable.

Because the increased concrete thickness tolerances either maintain conformance with the applicable construction codes or sufficient margin exists to justify the deviation, the walls, while

increased in thickness tolerance from what is shown in COL Appendix C, continue to perform their design basis functions in accordance with the underlying safety analyses. The containment internal structures design criteria and requirements are unchanged by this activity. Thus, the proposed changes do not alter the expected response of the structure to seismic events or the structural analysis of the containment internal structures.

As stated earlier, these concrete wall thickness tolerances for the CA04/CA01 walls were previously assumed in the design for the fabrication, assembly, and installation of the containment internal structures but the discrepancy between the total wall tolerance and the COL Appendix C tolerance was not previously identified. Additionally, the deviation from the tolerance in the CA04/CB65 wall is in a localized area, not affecting the overall wall design. Accordingly, there is no impact on the physical layout of other internal structures or system components. The impact to the walls' effectiveness in providing radiation shielding was also examined, and there were no adverse effects because the radiation source terms were conservatively selected to envelope plant operating conditions. Consequently, this method accounts for tolerances and small perturbations in the as-built configuration of the plant are not expected to impact the bounding conclusions of the radiation analysis.

The activity does not alter the fire loads found in any adjacent fire zones and areas as no equipment is added or removed by the activity. The proposed changes do not affect any function or feature use for the prevention and mitigation of accidents or their safety analyses. The proposed changes do not involve nor interface with any SSC accident initiator or initiating sequence of events related to the accidents evaluated in the plant-specific DCD or UFSAR. The proposed changes do not affect the radiological source terms (i.e., amounts and types of radioactive materials released, their release rates and release durations) used in the accident analyses. The thickness of these concrete walls around the reactor vessel is also not used as an input to the PRA, and therefore, there is no PRA impact as a result of the tolerance change.

No system or design function or equipment qualification is affected by the proposed changes. The changes do not result in a new failure mode, malfunction or sequence of events that could affect a radioactive material barrier or safety-related equipment. The proposed changes do not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that would result in significant fuel cladding failures.

The proposed changes associated with this license amendment request do not affect the containment, control, channeling, monitoring, processing or releasing of radioactive and non-radioactive materials. The types and quantities of expected effluents are not changed, and no effluent release path is affected by the proposed changes. Therefore, radioactive or non-radioactive material effluents are not affected by the proposed changes.

Plant radiation zones (as described in UFSAR Section 12.3), controls under 10 CFR Part 20, and expected amounts and types of radioactive materials are not affected by the proposed changes. The increased wall tolerance was also examined with respect to the walls' effectiveness in providing radiation shielding, and no adverse impacts were identified. Therefore, individual and cumulative radiation exposures do not change.

<u>Summary</u>

Ultimately, the proposed change increases the tolerance found in concrete thickness for four COL Appendix C identified walls. Because conformance is either maintained with the applicable codes and standards identified in the underlying Tier 2 information or sufficient margin exists to justify the deviation, there are no adverse impacts from the expected responses of the containment internal structures. Therefore, the above proposed changes would not adversely affect any safety-related equipment or function, design function, radioactive material barrier, or safety analysis.

4. Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 52.98(f) requires NRC approval for any modification to, addition to, or deletion from the terms and conditions of a COL. This activity involves a departure from COL Appendix C Inspections, Tests, Analyses and Acceptance Criteria information, and a corresponding change to plant-specific DCD Tier 1; therefore, this activity requires a proposed amendment to the COL as well as Tier 2 information. Accordingly, NRC approval is required prior to making the plant-specific changes in this license amendment request.

10 CFR 52, Appendix D, Section VIII.B.5.a allows an applicant or licensee who references this appendix to depart from Tier 2 information, without prior NRC approval, unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2* information, or the Technical Specifications, or requires a license amendment under paragraphs B.5.b or B.5.c of the section. This change involves a revision to plant-specific Tier 1 information (and corresponding COL Appendix C information), and thus requires NRC approval for the Tier 1 and involved Tier 2 departures.

10 CFR 50, Appendix A, General Design Criterion (GDC) 1, *Quality standards and records*, requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. By continuing to follow the guidelines of the NRC Regulatory Guides and industry standards, the requirements of GDC 1 have been maintained.

10 CFR 50, Appendix A, GDC 2, *Design bases for protection against natural phenomena*, requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. Because there is no change to the expected responses to natural phenomena, and the walls, even with the increased tolerance continue to be able to respond to the same design basis earthquake, there are no changes to the conformance with GDC 2.

10 CFR 50, Appendix A, GDC 4, *Environmental and dynamic effects design bases*, requires that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. The increased tolerances do not alter the environmental conditions associated with normal operation, and because the same design criteria are used

before and after the change, the containment internal structures continue to be able to withstand similar conditions.

4.2 Precedent

This proposed change is substantially similar to the NRC-approved V.C. Summer Nuclear Station Units 2 and 3 Amendment No. 29 (ML15216A071). That amendment request, LAR 14-07, was submitted on September 25, 2014 (ML14268A388) and supplemented on March 13, 2015 (ML15072A306). The amendment made changes to the first four entries in COL Appendix C and plant-specific Tier 1 Tables 3.3-1, increasing the tolerance for the CA04/CB65 wall from ±1 to ±1-1/4" and the three CA04/CA01 walls from ±1" to ±1-5/8". While this amendment request proposes to make the same change for the three CA04/CA01 walls, this request alternatively proposes to increase the wall tolerance for the CA04/CB65 wall (Shield Wall between Reactor Vessel Cavity and RCDT Room) to ±1-5/8", matching the tolerance for the other entries. This request utilizes the same method of demonstrating adequate reinforcement for the affected walls and ensuring no adverse effects result from the increased tolerance.

4.3 Significant Hazards Consideration Determination

The proposed changes would revise the Combined Licenses (COLs) by increasing the tolerances listed for four concrete thicknesses in COL Appendix C Table 3.3-1, "Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building," from ±1" to ±1-5/8". In addition, the changes include an update to Tier 2 Subsection 3.8.3.6.1 to address the exceeded ACI 117 tolerance for the four affected walls.

An evaluation to determine whether or not a significant hazards consideration is involved with the proposed amendment was completed by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

4.3.1 Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

As indicated in the Updated Final Safety Analysis Report, Subsection 3.8.3.1, the containment internal structures and associated modules support the reactor coolant system components and related piping systems and equipment. The increase in tolerance associated with the concrete thickness of four of these containment internal structure walls and the deviation from American Concrete Institute (ACI) 117 do not involve any accident initiating components or events, thus leaving the probabilities of an accident unaltered. The increased tolerance does not adversely affect any safety-related structures or equipment nor does the increased tolerance reduce the effectiveness of a radioactive material barrier. Thus, the proposed changes would not affect any safety-related accident mitigating function served by the containment internal structures.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

4.3.2 Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed tolerance increases and the code deviation from ACI 117 do not change the performance of the affected containment internal structures. As demonstrated by the continued conformance to the other applicable codes and standards governing the design of the structures, the walls with an increased concrete thickness tolerance continue to withstand the same effects as previously evaluated. There is no change to the design function of the affected modules and walls, and no new failure mechanisms are identified as the same types of accidents are presented to the walls before and after the change.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

4.3.3 Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The proposed change to increase the concrete thickness tolerance for four walls identified in COL Appendix C Table 3.3-1 does not alter any design function, design analysis, or safety analysis input or result, and sufficient margin exists to justify departure from the ACI 117 requirements for the four affected walls. As such, because the system continues to respond to design basis accidents in the same manner as before without any changes to the expected response of the structure, no safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the proposed changes. Accordingly, no safety margin is reduced by the increase of the wall concrete thickness tolerance.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Pursuant to 10 CFR 50.92, the requested change does not involve a Significant Hazards Consideration.

5. Environmental Considerations

The details of the proposed changes are provided in Sections 2 and 3 of this license amendment request.

The proposed changes would revise the Combined Licenses (COLs) by increasing the concrete thickness tolerance for certain walls noted in COL Appendix C Table 3.3-1 along with the corresponding plant-specific DCD Tier 1 change. In addition, the changes include an update to Tier 2 Subsection 3.8.3.6.1 to address the exceeded ACI 117 tolerance for the four walls discussed in this LAR.

This review has determined the proposed change requires an amendment to the COL. However, a review of the anticipated construction and operational effects of the requested amendment has determined the requested amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), in that:

(i) *There is no significant hazards consideration.*

As documented in Section 4.3, Significant Hazards Consideration Determination, of this license amendment request, an evaluation was completed to determine whether or not a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment." The Significant Hazards Consideration determined that (1) the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) the proposed amendment does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

(ii) *There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.*

The proposed changes in the requested amendment increase the concrete thickness tolerance found in four walls internal to the containment structure. The proposed changes are unrelated to any aspect of plant construction or operation that would introduce any change to effluent types (e.g., effluents containing chemicals or biocides, sanitary system effluents, and other effluents), or affect any plant radiological or non-radiological effluent release quantities. Furthermore, the proposed changes do not affect any effluent release path or diminish the functionality of any design or operational features that are credited with controlling the release of effluents during plant operation. Therefore, it is concluded that the proposed amendment does not involve a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite.

(iii) *There is no significant increase in individual or cumulative occupational radiation exposure.*

The proposed changes increase the tolerances for four wall concrete thicknesses inside containment. Plant radiation zones (addressed in UFSAR Section 12.3) are not affected, and controls under 10 CFR Part 20 preclude a significant increase in occupational radiation exposure. The increased wall tolerance was also examined with respect to the walls' effectiveness in providing radiation shielding, and no adverse impacts were identified. Therefore, the proposed amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above review of the requested amendment, it has been determined that anticipated construction and operational effects of the proposed amendment do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the requested amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment of the proposed amendment and exemption is not required.

Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-15-1742

Enclosure 2

Exemption Request:

CA04 Structural Module ITAAC Dimension Change (LAR-15-015)

(8 pages, including this cover page)

1.0 PURPOSE

Southern Nuclear Operating Company (the Licensee) requests a permanent exemption from the provisions of 10 CFR 52, Appendix D, Section III.B, *Design Certification Rule for the AP1000 Design, Scope and Contents*, to allow a departure from elements of the certification information in Tier 1 of the generic AP1000 Design Control Document (DCD). The regulation, 10 CFR 52, Appendix D, Section III.B, requires an applicant or licensee referencing Appendix D to 10 CFR Part 52 to incorporate by reference and comply with the requirements of Appendix D, including certified information in DCD Tier 1. The Tier 1 information for which a plant-specific departure and exemption is being requested includes changes related to the design details of containment internal structural modules.

This request for exemption provides the technical and regulatory basis to demonstrate that 10 CFR 52.63, §52.7, and §50.12 requirements are met and will apply the requirements of 10 CFR 52, Appendix D, Section VIII.A.4 to allow departures from generic Tier 1 information due to the proposed additions to the non-system based design descriptions and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) as identified in updated Note 2, and new Note 10 in Table 3.3-1, "Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building."

Modular construction techniques are used extensively in the containment internal structures. Subassemblies are initially fabricated both offsite and onsite. Module assembly consists of combining the subassemblies into structural modules after which they are installed in the plant. Design finalization of the modules, fabrication experience in the shop environment, and the comparison of text and figures presented in the Updated Final Safety Analysis Report (UFSAR) have identified the need for proposed design changes to certain concrete wall thickness tolerances relevant to the fabrication, assembly, and erection of the structural module walls identified in plant-specific DCD Tier 1 Table 3.3-1.

2.0 BACKGROUND

The Licensee is the holder of Combined License Nos. NPF-91 and NPF-92, which authorize construction and operation of two Westinghouse Electric Company AP1000 nuclear plants, named Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively.

Plant-specific DCD Tier 1 Table 3.3-1 identifies the general tolerance of ±1" as a tolerance acceptable to each module. However, the consideration of module fabrication and assembly tolerances in addition to module installation tolerances results in an increased tolerance for certain concrete wall thicknesses than is permitted by the total ±1". Because of this higher design tolerance, there is a possibility that the final tolerance of the completed walls will exceed the ±1" tolerance specified in Tier 1.

The change activity is to update Note 2 and to add Note 10 to Tier 1 Table 3.3-1 that provides revised tolerances for the concrete wall thicknesses for the following containment building internal structures:

1. Shield Wall between Reactor Vessel Cavity and Reactor Coolant Drain Tank (RCDT) Room
2. West Reactor Vessel Cavity Wall
3. North Reactor Vessel Cavity Wall
4. East Reactor Vessel Cavity Wall

The concrete wall thickness tolerance is proposed to be increased from ±1" to ±1-5/8" (i.e., Note 10) for the above four walls.

3.0 TECHNICAL JUSTIFICATION OF ACCEPTABILITY

UFSAR Subsection 3.8.3.6.1 specifies that the tolerance for fabrication, assembly, and installation of structural modules conforms to the requirements of ACI 117, AWS D1.1, and AISC-N690, and UFSAR Subsection 3.8.4.4.1 requires that the design and analysis procedures are in accordance with ACI 349-01. For the affected walls, the concrete thickness tolerances listed in Tier 1 Table 3.3-1 do not utilize the complete allowance specified in these codes with the exception of ACI 349-01 and ACI 117. Because conformance is either maintained with the applicable codes and standards identified in the underlying Tier 2 information or sufficient margin exists to justify the deviation, the increase in concrete wall thickness tolerance for the above identified walls is found to be acceptable.

The proposed installation tolerance of ±1-5/8" between the CA04/CB65 wall is greater than the ±1" specified in ACI 349-01 Section 7.5.2.1 and the +1"/-3/4" found in ACI 117. For evaluation of the impact and the margin of the wall design variation, the wall thickness and corresponding d value was set to ±1-5/8", which is greater than ACI 117 tolerances on cross-sectional dimensions. As a result of the assessment, it was found that the minimum margin for vertical reinforcement is 47.9%, for horizontal reinforcement 54.8% and for shear 61.3%. Since the thickness of the wall also influences the d value, an assessment of the shear reinforcement spacing was performed which found the provided spacing sufficient.

Similarly, the CA04/CA01 walls were examined to determine the impact of the extra ±5/8" beyond the ACI 349-01 requirements and +5/8" and -7/8" beyond the ACI 117 requirements. Those walls were examined and the review found that there was a potential for a minor, 1.3% decrease to the area of the surrounding concrete, however this change in tolerance is negligible and has no significant impact on the in place structural analyses or qualifications.

Despite increasing the concrete thickness tolerance of these four walls, the functions of the walls are maintained and the walls are able to perform adequately. The walls therefore continue to perform their design basis functions in accordance with the

underlying safety analyses. The containment internal structures design criteria and requirements are unchanged by this activity.

Detailed technical justification supporting this request for exemption is provided in Section 3 of the associated License Amendment Request in Enclosure 1 of this letter.

4.0 JUSTIFICATION OF EXEMPTION

10 CFR Part 52, Appendix D, Section VIII.A.4 and 10 CFR 52.63(b)(1) govern the issuance of exemptions from elements of the certified design information for AP1000 nuclear power plants. Since SNC has identified changes to the Tier 1 information as discussed in Enclosure 1 of the accompanying License Amendment Request, an exemption from the certified design information in Tier 1 is needed.

10 CFR Part 52, Appendix D, and 10 CFR 50.12, §52.7, and §52.63 state that the NRC may grant exemptions from the requirements of the regulations provided six conditions are met: 1) the exemption is authorized by law [§50.12(a)(1)]; 2) the exemption will not present an undue risk to the health and safety of the public [§50.12(a)(1)]; 3) the exemption is consistent with the common defense and security [§50.12(a)(1)]; 4) special circumstances are present [§50.12(a)(2)]; 5) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption [§52.63(b)(1)]; and 6) the design change will not result in a significant decrease in the level of safety [Part 52, App. D, VIII.A.4].

The requested exemption satisfies the criteria for granting specific exemptions, as described below.

1. **This exemption is authorized by law**

The NRC has authority under 10 CFR 52.63, §52.7, and §50.12 to grant exemptions from the requirements of NRC regulations. Specifically, 10 CFR 50.12 and §52.7 state that the NRC may grant exemptions from the requirements of 10 CFR Part 52 upon a proper showing. No law exists that would preclude the changes covered by this exemption request. Additionally, granting of the proposed exemption does not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations.

Accordingly, this requested exemption is "authorized by law," as required by 10 CFR 50.12(a)(1).

2. **This exemption will not present an undue risk to the health and safety of the public**

   The proposed exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would allow changes to elements of the plant-specific Tier 1 DCD to depart from the AP1000 certified (Tier 1) design information. The plant-specific DCD Tier 1 will continue to reflect the approved licensing basis for VEGP Units 3 and 4, and will maintain a consistent level of detail with that which is currently provided elsewhere in Tier 1 of the DCD. Therefore, the affected plant-specific DCD Tier 1 ITAAC will continue to serve its required purpose.

   The revised tolerances for the concrete wall thicknesses discussed in Section 2, do not represent any adverse impact to their design functions or the systems, structures and components therein and will continue to protect the health and safety of the public in the same manner. The revised tolerances do not introduce any new industrial, chemical, or radiological hazards that would represent a public health or safety risk, nor do they modify or remove any design or operational controls or safeguards intended to mitigate any existing on-site hazards. Furthermore, the proposed changes would not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that would result in fuel cladding failures. Accordingly, these changes do not present an undue risk from any existing or proposed equipment or systems.

   Therefore, the requested exemption from 10 CFR 52, Appendix D, Section III.B would not present an undue risk to the health and safety of the public.

3. **The exemption is consistent with the common defense and security**

   The requested exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would allow the licensee to depart from elements of the plant specific DCD Tier 1 design information. The proposed exemption does not alter the design, function, or operation of any structures or plant equipment that is necessary to maintain a safe and secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures.

   Therefore, the requested exemption is consistent with the common defense and security.

4. **Special circumstances are present**

   10 CFR 50.12(a)(2) lists six "special circumstances" for which an exemption may be granted. Pursuant to the regulation, it is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request. The requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

   The rule under consideration in this request for exemption is 10 CFR 52, Appendix D, Section III.B, which requires that a licensee referencing the AP1000 Design Certification Rule (10 CFR Part 52, Appendix D) shall incorporate by reference and comply with the requirements of Appendix D, including Tier 1 information. The VEGP Units 3 and 4

COLs reference the AP1000 Design Certification Rule and incorporate by reference the requirements of 10 CFR Part 52, Appendix D, including Tier 1 information. The underlying purpose of Appendix D, Section III.B is to describe and define the scope and contents of the AP1000 design certification, and to require compliance with the design certification information in Appendix D.

The proposed exemption would revise the concrete wall thickness tolerances for four walls in the Nuclear Island.

The proposed revised tolerances for the concrete wall thicknesses, discussed in Section 2, maintain the design margins of the internal containment structures. This change does not impact the ability of any structures, systems, or components to perform their functions or negatively impact safety. Accordingly, this exemption from the certification information will enable the Licensee to safely construct and operate the AP1000 facility consistent with the design certified by the NRC in 10 CFR 52, Appendix D.

Therefore, special circumstances are present, because application of the current generic certified design information in Tier 1 as required by 10 CFR Part 52, Appendix D, Section III.B, in the particular circumstances discussed in this request is not necessary to achieve the underlying purpose of the rule.

5. **The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption.**

Based on the nature of the changes to the plant-specific Tier 1 information and the understanding that these changes support the design function of the internal containment structures, it is expected that this exemption will be requested by other AP1000 licensees and applicants. However, a review of the reduction in standardization resulting the departure from the standard DCD determined that even if other AP1000 licensees and applicants do not request this same departure, the special circumstances will continue to outweigh any decrease in safety from the reduction in standardization because the key design functions of the structures associated with this request will continue to be maintained. Furthermore, the justification provided in the license amendment request and this exemption request and the associated mark-ups demonstrate that there is a limited change from the standard information provided in the generic AP1000 DCD, which is offset by the special circumstances identified above.

Therefore, the special circumstances associated with the requested exemption outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption.

6. **The design change will not result in a significant decrease in the level of safety.**

The exemption revises the plant-specific DCD Tier 1 information by revising tolerances for the concrete wall thicknesses as discussed in Section 2. The changes to the tolerances do not change the design requirements of the nuclear island structures. Because these functions continue to be met, there is no reduction in the level of safety.

5.0 RISK ASSESSMENT

A risk assessment was not determined to be applicable to address the acceptability of this proposal.

6.0 PRECEDENT EXEMPTIONS

The changes proposed in this exemption are consistent in technical content with LAR 14-07 and exemption, submitted by South Carolina Electric & Gas Company (SCE&G) on September 25, 2014 as supplemented on March 13, 2015, (ADAMS Accession Nos. ML14268A388 and ML15072A306 respectively) with the exception of one alternative plant-specific change. The NRC approved the SCE&G LAR 14-07 on August 24, 2015 (ADAMS Accession No. ML15216A071).

7.0 ENVIRONMENTAL CONSIDERATION

The Licensee requests a departure from elements of the certified information in Tier 1 of the generic AP1000 DCD. The Licensee has determined that the proposed departure would require a permanent exemption from the requirements of 10 CFR 52, Appendix D, Section III.B, *Design Certification Rule for the AP1000 Design, Scope and Contents*, with respect to installation or use of facility components located within the restricted area, as defined in 10 CFR Part 20, or which changes an inspection or a surveillance requirement; however, the Licensee evaluation of the proposed exemption has determined that the proposed exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.25(c)(9).

Based on the above review of the proposed exemption, the Licensee has determined that the proposed activity does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment of the proposed exemption is not required.

Specific details of the environmental considerations supporting this request for exemption are provided in Section 5 of the associated License Amendment Request provided in Enclosure 1 of this letter.

8.0 CONCLUSION

The proposed changes to Tier 1 are necessary to update Note 2, and add new Note 10 in plant-specific Tier 1 Table 3.3-1 to reflect the discussed tolerance revisions for the concrete wall thicknesses. The exemption request meets the requirements of 10 CFR 52.63, *Finality of design certifications*, 10 CFR 52.7, *Specific exemptions*, 10 CFR 50.12, *Specific exemptions*, and 10 CFR 52 Appendix D, *Design Certification Rule for the AP1000.* Specifically, the exemption request meets the criteria of 10 CFR 50.12(a)(1) in

that the request is authorized by law, presents no undue risk to public health and safety, and is consistent with the common defense and security. Furthermore, approval of this request does not result in a significant decrease in the level of safety, satisfies the underlying purpose of the AP1000 Design Certification Rule, and does not present a significant decrease in safety as a result of a reduction in standardization.

9.0 REFERENCES

None

Southern Nuclear Operating Company

Vogtle Electric Generating Plant Units 3 and 4

ND-15-1742

Enclosure 3

Proposed Changes to the Licensing Basis Documents

(LAR-15-015)

(3 pages, including this cover page)

Revise COL Appendix C Table 3.3-1 and corresponding Plant-Specific Tier 1 Table 3.3-1 as shown below:

Table 3.3-1
Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building[1]

Wall or Section Description	Column Lines[7]	Floor Elevation or Elevation Range[7][8]	Concrete Thickness[2][3][4][5][9]	Applicable Radiation Shielding Wall (Yes/No)
Containment Building Internal Structure				
Shield Wall between Reactor Vessel Cavity and RCDT Room	E-W wall parallel with column line 7 (Inside face is 3'-0" north of column line 7. Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity.)	From 71'-6" to 83'-0"	3'-0"[10]	Yes
West Reactor Vessel Cavity Wall	N-S wall parallel with column line N (Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity).	From 83'-0" to 98'-0"	7'-6"[10]	Yes
North Reactor Vessel Cavity Wall	E-W wall parallel with column line 7 (Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity).	From 83'-0" to 98'-0"	9'-0"[10]	Yes
East Reactor Vessel Cavity Wall	N-S wall parallel with column line N (Width of wall section with stated thickness is defined by inside wall of reactor vessel cavity.)	From 83'-0" to 98'-0"	7'-6"[10]	Yes

Notes:

2. These wall (and floor) thicknesses have a construction tolerance of ±1 inch, except as noted and for exterior walls below grade where the tolerance is +12 inches, -1 inch. These tolerances are not applicable to the nuclear island basemat.

10. This wall thickness has a tolerance of ±1-5/8 inch.

UFSAR Subsection 3.8.3.6.1 "Fabrication, Erection, and Construction of Structural Modules" Fourth Paragraph is revised as shown below:

Tolerances for fabrication, assembly and erection of the structural modules conform to the requirements of section 4 of ACI-117, applicable sections of AWS D1.1, and sections Q1.23 and Q1.25 of AISC-N690. In walls around the reactor vessel cavity, where the concrete is placed between portions of unconnected modules or between a module and a left-in-place form, the tolerance for the wall thickness may be increased over those in ACI 117. These walls have been evaluated against ACI 349-01 reinforcement design requirements and demonstrated sufficient margin to accommodate the increased tolerance. Tolerances for shear stud spacing requirements are identified on Figure 3.8.3-8, Sheet 1 and conform to AWS D1.1, Paragraph 7.4.5.